

KOREAN RADIOACTIVE WASTE SOCIETY

2017 추계학술논문요약집

| 일시 |

2017. 10. 18(수) ~ 10. 20(금)

| 장소 |

창원컨벤션센터

Abstracts of Proceedings of the
Korean Radioactive Waste Society
AUTUMN 2017



사단 한국방사성폐기물학회
법인 Korean Radioactive Waste Society

KOREAN RADIOACTIVE WASTE SOCIETY

2017 추계학술논문요약집

| 일시 |

2017. 10. 18(수) ~ 10. 20(금)

| 장소 |

창원컨벤션센터

Abstracts of Proceedings of the
Korean Radioactive Waste Society
AUTUMN 2017



사단 한국방사성폐기물학회
법인 Korean Radioactive Waste Society

Contents

KOREAN
RADIOACTIVE WASTE
SOCIETY

1분과 핵주기정책 및 핵비확산(Oral)

09:00~09:40	Radioactive Waste Management Policy in Korea - Experiences, Challenges, and the Way Forward	19
Jae Hak Cheong [Kyung Hee Univ.]		
09:40~10:00	Investigation of Legal and Regulatory Frameworks of Waste Management System for Very Low Level Waste	21
Ji Young Song, Jin Ho Park, Jong Hwa Kim, and Kwang Pyo Kim [Kyung Hee Univ.]		
10:00~10:20	Current Issues and Improvement Plans for Management and Disposal of Small Quantity Nuclear Material	23
Junghee Son, Chul Heo, and Seungmin Lee [KINAC]		
10:40~11:00	Pyro Technical Data and Information Management	25
Jeong-Guk Kim, Won-Il Ko, Jeong-Hoe Ku, and Hyo-On Nam [KAERI]		
11:00~11:20	Progress of Design of Containment and Surveillance System for Pyroprocessing Safeguards	27
Se-Hwan Park, Dae-Yong Song, Seong-Kyu Ahn, Bong Young Kim, and Jeong-Hoe Ku [KAERI]		
11:20~11:40	Fast Neutron Detection for Spent PWR Fuel in Dry Storage	29
Ju Young Jeon, Jung Youn Choi, and Heejun Chung [KINAC], Taehoon Jeon [Orbitech Co., Ltd], Jin Ho Park and Kwang Pyo Kim [KyungHee Univ.]		
11:40~12:00	Design Information Analysis of TRU Fuel Manufacturing Facility	31
Min Su KIM [KINAC]		

1분과 핵주기정책 및 핵비확산(Poster)

P0101	Study on Efficiency Improvement of Strategic Item Classification Through Intelligent System	35
Seung-hyo Yang, Sung-ho Yoon, and DongHoon Shin [KINAC]		
P0102	Development of Export License Review System for Strategic Trade Control	37
Sung-ho Yoon, Seung-hyo Yang, and DongHoon Shin [KINAC]		
P0103	Identification System Concept Design for Illegal Transfer of Strategic Items	39
DongHoon Shin, Seung-hyo Yang, and Sung-ho Yoon [KINAC]		

P0104	Development of Virtual Training Software for Bulk Handling Facilities Based on the IAEA Training Course	41
	Chul Heo [KINAC]	
P0105	The Application of ICP (Inner Compliance Program) at KAERI	43
	I.C. Kim, S.H. Lee, B.D. Lee, H.J. Kim, H.S. Kim, and J.A. Jung [KAERI]	
P0106	Essential Factors of Decommissioning Strategies for Korean Nuclear Industry	45
	Hyejin Jung, Taesik Yun, Young gook Kim, and Kwangho Jo [KHNP-CRI]	

2분과 사용후핵연료관리 (Oral)

09:00~09:20	Reaction Rate of Porous UO ₂ Pellets for Electrolytic Reduction	49
	Min Ku Jeon, Eun-Young Choi, Sung-Wook Kim, Sang-Kwon Lee, and Jeong Lee [KAERI]	
09:20~09:40	Analysis of Material Behaviors of Integrated Operation Model of Pyroprocessing and the Phased Approach Strategy of Optimized Mass Flow	51
	Hun Suk Im, Hong Jang, Hyo Jik Lee, Ju Ho Lee, Min Ku Jeon, Chang Hwa Lee, Gha-Young Kim, and Hwan-Seo Park [KAERI]	
09:40~10:00	Needs and Goals of Pyroprocessing Automation Verification Mockup	53
	Dongseok Ryu, Jongkwang Lee, Sungwook Kim, Sungjai Lee, Jin-Mok Hur, and Hyojik Lee [KAERI]	
10:00~10:20	Modeling of the Thermal Conductivity of Mo-ZrO ₂ Based Composites for Inert Matrix Fuel	55
	Qusai M. Mistarihi and Ho Jin Ryu [KAIST], Vivek Raj[IITK]	
10:40~11:00	Spent Nuclear Fuel Safety Evaluation Methodology (SSEM) for Storage and Transportation	57
	Y.K. Kim and S.K. Lee [KEPCO-NF], J. S. Noh [ACT], T.W. Kim [Hanyang Univ.]	
11:00~11:20	Structural Integrity Assessment of Spent Nuclear Fuel Rod Under Drop Accident Condition	59
	Sanghoon Lee [Keimyung Univ.], Belal Almomani and Dongchan Jang [KAIST]	
11:20~11:40	Establishment of the Design and Testing Methods of Lid Welds as the Confinement Boundary for Spent Nuclear Fuel Storage Canister	61
	Chang-Yeal Baeg and Chun-Hyung Cho [KORAD]	
11:40~12:00	Advanced Nuclear Fuel Cycle Train Program in Tohoku University: Experiments on Solid States and Solution Chemistry	63
	Geonyong Lee and Kwangheon Park [Kyunghee Univ.]	

P0201	Theoretical Analysis for Direct Conversion of U_3O_8 Into UCl_3 as an Alternative to Oxide Reduction Reaction	67
	Min Ku Jeon, Jin-Mok Hur, and Do-Hee Ahn [KAERI]	
P0202	Selection of Surrogate Materials for Oxide Reduction of Used Nuclear Fuel Using Carbon Anode	69
	Min Ku Jeon, Sung-Wook Kim, Sang-Kwon Lee, and Eun-Young Choi [KAERI]	
P0203	Residual Salt Separation of Reaction Products in Pyroprocessing	71
	Sung-Wook Kim, Jin-Mok Hur, Dong Hyun Heo, Sun-Seok Hong, and Eun-Young Choi [KAERI]	
P0204	Effect of Cl_2 on Electrodeposition Behavior in Electrowinning Process	73
	Si Hyung Kim, Taek-Jin Kim, Gha-Young Kim, Jun-Bo Shim, Seungwoo Paek, and Sung-Jai Lee [KAERI]	
P0205	Measurement of Evaporation Rates for Lanthanum and Neodymium Chlorides	74
	S.W. Kwon, Y.S. Lee, J.H. Jung, J.H. Chang, S.H. Kim, and S.J. Lee [KAERI]	
P0206	Modification of Tablet Press for Compaction of U_3O_8 Powder in Hot Cell	75
	Jae-Won Lee, Do-Yoon Lee, Young-Soon Lee, Sang-Chae Jeon, Ju-Ho Lee, and Yung-Zun Cho [KAERI]	
P0207	A Rough Estimation of the Repository Footprint of HLW for the Used Fuel Management Option	77
	Do-Hee Ahn, Heui-Joo Choi, Hyo On Nam, and Eun-Young Choi [KAERI]	
P0208	Maintenance of Remote Handling Equipment in ACPF	79
	SungHyun Kim and Eun-Yong Choi [KAERI]	
P0209	Development of Kr Adsorption Process in Pyroprocessing	81
	Seok-Min Hong, Ahreum Han, Jae Hwan Yang, and Yung-Zun Cho [KAERI]	
P0210	Electrochemical Behavior of Tm Ion and Bi Ion in LiCl-KCl Molten Salt	83
	Beom-Kyu Kim, Hwa Jeong Han, Ji Hye Park, Won Ki Kim, and Byung Gi Park [Soonchunhyang Univ.]	
P0211	Evaluate of the Spent Nuclear Fuel Rod Model Using the Monte Carlo Simulation	85
	Hwa Jeong Han, Beom-Kyu Kim, Ji Hye Park, Won Ki Kim, and Byung Gi Park [Soonchunhyang Univ.]	
P0212	Design of Furnace Tray for Improved UO_2 Fragment Oxidation Rate	87
	JuHo Lee, Yung-Zun Cho, Sang-Chae Jeon, and Young-Soon Lee [KAERI]	

P0213	The Fabrication and Characteristics of Gd ₂ O ₃ Particle Reinforced Aluminum Matrix Composite	89
	Tae Gyu Lee and Ho Jin Ryu [KAIST]	
P0214	Technical Issues of Remote Assembler for TRU Fuel Assembly	91
	Young-Ho Lee, Sang-Gyu Park, Ki-ho Kim, Jeong-Yong Park, and Chan-Bock Lee [KAERI]	
P0215	Thermal Conductivity Modeling of U-TRU-RE-Zr Metal Fuel for PGSFR	93
	Byoung Oon Lee, Chan Bock Lee, June Hyung Kim, Jin Sik Cheon, Jeong Yong Park, and Sung Ho Kim [KAERI]	
P0216	Re-Fabrication of U-Zr Alloy System Fuel Slugs Recycling Fuel Scraps	95
	Ki-Hwan Kim, Young-Mo Ko, Seok-Jin Oh, and Jeong-Yong Park [KAERI] Seong-Jun Ha and Seung-Uk Mun [CNU]	
P0217	Improvement of Spent Fuel Inspection System	97
	Mu-gap Shin and Young-hwan Kim [KEPCO E&C]	
P0218	Estimation of On-Site Dry Storage Capacity Needed Based on Current Nuclear Policy in Korea	99
	Hong Jang and Hyo On Nam [KAERI]	
P0219	Line Balancing of an Integrated Pyroprocessing Using IOMP	101
	Doyeon Kim, Hyo jik Lee, Hunsuk Im, and Hong Jang [KAERI]	
P0220	Application of the Technical Safety Requirements for the Fuel Cycle Facility	103
	Woo Jin Jo, Seok-Jun Seo, Siwan Noh, and Seung Nam Yu [KAERI]	
P0221	Pressure Drop of U-Shape Bellows at Argon Circulation System	105
	Ji-sun Lee, Byungsuk Park and Jaehoon Lim [KAERI]	
P0222	Conceptualization of Dry Room for Verification of Pyroprocessing Equipment	107
	Jaehoon Lim, Byungsuk Park, Dongsuk Ryu, Hyo Jik Lee, and Seung Nam Yu [KAERI]	
P0223	A Consideration on the Dehumidification Methods for Designing the Dry Room	109
	Byungsuk Park, Jaehoon Lim, Dongseok Ryu, Hyo jik Lee, and Seung nam Yu [KAERI]	
P0224	Enforced Risk Criteria for SF Recycling Facilities in US and Its Availability in Korea	111
	Gilsung You, Seok jun Seo, Siwan Noh, Hongrae Jeon, Hyo jik Lee, Woo jin Jo, Hyunsook Im, Hohee Lee, Wonmyung Choung, Seung nam Yu, Jaehoon Lim, and Jeonghoe Ku [KAERI]	
P0225	Structural Evaluation Under Accident Condition for Concrete Storage Cask of Spent Fuels	113
	Tae-Chul Moon and Chun-Hyung Cho [KORAD]	

P0226	Requirements for the Transportation of Spent Nuclear Fuel (SNF) in Terms of Fuel Integrity and Data Needed According to	115
	J. S. Noh [ACT], Y.K. Kim [KEPCO-NF], T.W. Kim [Hanyang Univ.]	
P0227	Thermal Analysis of KORAD21 CASK Using COMSOL MULTIPHYSICS	117
	Sunwoo Park and Kwangheon Park [Kyung Hee Univ.]	
P0228	Considerations of Drying System Design for the Dry Storage of Spent Nuclear Fuel	119
	Kyung-Wook Shin, Byeong-Mok Park, Jae-Hyun Han, Geon-Hui Lee, Gyung-Sun Chae, and Jae-Seok Park [SAE-AN Eng.]	
P0229	Thermal Expansion Analysis of Concrete Spent Nuclear Fuel Dry Storage Developed in Korea	121
	Wooyong Choi and Kwangheon Park [Kyung Hee Univ.]	
P0230	A Study on Drop Analysis of the Spent Nuclear Fuel Metal Cask (OASIS-32D)	123
	Hye Jin Lim, Jang Won Lee, Jung Gyu Kim, Ki Kwang Sung, and Hyun Min Kim [KEPCO E&C]	
P0231	Seismic Evaluation for a Fuel Assembly in Storage Rack of Spent Nuclear Fuel Pool	125
	Sunghwan Chung and Kiyong Kim [KHNP-CRI]	
P0232	Electro-Deposition of Cr Layer to Prevent Fuel Cladding Chemical Interaction in Sodium Fast Reactor (SFR)	127
	Sunghwan Yeo, Koo Hye Yeon, and Junhwan Kim [KAERI]	

3분과 고준위폐기물처분 (Oral)

09:00~09:40	Direction of Radioactive Waste Management at the Time of Energy Transition	
	Byoung-Sub Han [KFEM]	
09:40~10:00	Application of Thermo-Hydro-Mechanical Damage Model on Triaxial Test of Granite	131
	Jin-Seop Kim, Jung-Woo Kim, and Geon-Young Kim [KAERI], Jun-Seo Jeon [KAIST]	
10:00~10:20	Proposal on Regulatory Guidelines for the Assessment of Human Intrusion Into a Radwaste Repository	133
	Dong-Keun Cho, Jung-Woo Kim, Jongtae Jeong, Heui-Joo Choi, Jong-Youl Lee, Jae-Won Lee, and Min-Hoon Baik [KAERI], Hyosub Kim [Hanyang Univ.]	
10:40~11:00	Integrated Risk-Weighted Radioactivity Flux Ratios as Performance Indicators of Defense-in-Depth for HLW Disposal System	135
	Chan Woo Jeong, Jinyong Park, Eun Jin Seo, and Sung Il Kim [KINS]	

11:00~11:20	Spectroscopic Studies on Pu(III) Hydrolysis Under Reducing and Elevated Temperatures Conditions	137
	Hye-Ryun Cho, Hee Kyung Kim, Euo Chang Jung, and Wansik Cha [KAERI]	
11:20~11:40	Investigation of Present-Day Stress States at KURT Site	139
	Yeonguk Jo, Kyung-Woo Park, Sung-Hoon Ji, and Changsoo Lee [KAERI]	
11:40~12:00	Long-Term Corrosion Behavior of Canister Materials in KURT Environment ...	141
	Minsoo Lee, Heui-Joo Choi, and Sung-Hoon Ji [KAERI]	

3분과 고준위폐기물처분 (Poster)

P0301	Recent Techniques for Measuring the SWRC of Unsaturated Compacted Bentonite Buffer	145
	Jaw Owan Lee and Geon-Young Kim [KAERI]	
P0302	Preliminary Study on Paleostress Reconstruction Around KURT Area	147
	Soolim Jung and Sung-Hoon Ji [KAERI]	
P0303	Biogeochemical Alteration of KJ-II Bentonite Under Alkaline-Reducing Conditions	149
	Hyo-Jin Seo, Seung Yeop Lee, Jae-Kwang Lee, and Wan Hyoung Cho [KAERI]	
P0304	Statistical Approach of the Geochemical Mobility and Behavior of Natural Uranium in KURT Groundwater	151
	JiHye Jeong, Jang-Soon Kwon, Yong-Kwon Koh, and Jae-Kwang Lee [KAERI]	
P0305	A Preliminary Deep Borehole Disposal Concept for Cs-Wastes From Pyro-Processing	153
	Jong-Youl Lee, Heui-Joo Choi, In-Young Kim, and Dong-Keun Cho [KAERI]	
P0306	Investigation on Regulator's Role for Confidence Building of Stakeholders in Radioactive Waste Disposal	155
	Kyongwon Han, Jihyun Kim, and Sang Hoon Park [RadWorks Corp.], Jinyong Park and Yohan Kim [KINS]	
P0307	Concept Development of Drift Storage-and-Disposal of Cs-Wastes	157
	Heui-Joo Choi, Jong-Youl Lee, In-Young Kim, and Dong-Keun Cho [KAERI]	
P0308	Effect of Initial Nuclide Concentration on the Purification of LiCl Waste Salt by Layer-Melt Crystallization Process	159
	Tae-Kyo Lee, Jung-Hoon Choi, Ki Rak Lee, Seung-Youb Han, and Hwan-Seo Park [KAERI]	
P0309	Melt-Crystallization Monitoring System for the Purification of LiCl Salt Waste	161
	Jung-Hoon Choi, Tae-Kyo Lee, Ki Rak Lee, Seung-Youb Han, Na-Young Kim, Seon-Ah Jang, and Hwan-Seo Park [KAERI]	

4분과 방사성폐기물관리 (Oral)

09:00~09:20	Removal of Iodide and Iodate Using Layered Double Hydroxides (LDHs) Under Aqueous Condition	165
	Jaehyuk Kang and Wooyong Um [POSTECH]	
09:20~09:40	Concentration Averaging Option of NPP Decommissioning Low Level Radioactive Waste	167
	Sun-Kee Lee, Heon Kim, and Chang-Lak Kim [KINGS]	
09:40~10:00	Decontamination Technology for Low Activity Spent Resin From PWR's	169
	Young-Soo Jo, Hyun-Ki Kim, Jun Heo, Se-Ban Lee, and Jin-Hyeon Sung [HuvisWater Co.]	
10:00~10:20	Development of Treatment Technology of Non-Disposal Radwastes for the Increase of Their Volume at Disposal Site	171
	MinYoung Jung, SangHyun Lim, and Jongsoon Song [Chosun Univ.]	
10:40~11:00	Carbon-14 Inventory Optimization Process for LILW Silo Repositories in a Normal Scenario by Adopting Triangular Distribution Function	173
	Sol-Chan Han and Jong-II Yun [KAIST]	
11:00~11:20	Development of C-14 Stripping Method From Waste Resin Generated From HWR via Microwave Treatment: Part 2, Demonstration Equipment for C-14 Stripping	175
	Ki Rak Lee, Hwan-Seo Park, Jung-Hoon Choi, and Seung-Youb Han [KAERI] Hyeon-Oh Park [Sunkwang T&S]	
11:20~11:40	Chemical Pretreatment of Dry Active Waste for Determination of Radionuclides	177
	Jaesik Hwang, Simon Oh, Kwang-soon Choi, and Hong Joo Ahn [KAERI]	
11:40~12:00	Conceptual Design for Kori Unit 1 Reactor Vessel and Internal Segmentation Process	179
	Young Hwan Hwang, Seokju Hwang, Joo-Young Yoon, and Cheon-Woo Kim [KHNP-CRI]	

4분과 방사성폐기물관리 (Poster)

P0401	Analysis of Data Field of Radioactive Waste Management for Development of Decommissioning Waste Management Program	183
	Jin Ho Park, Ji Young Song, Hyun Woo Kim, Kang Hyun Song, and Kwang Pyo Kim [Kyung Hee Univ.]	
P0402	Radioactive Waste Treatment for Release of Korea Research Reactor Site ..	185
	Seung-Geon An, Sang-bok Lee, Tae-ryong Kim, Jung-soo Park, Yeo-jib Choi, and Seong-jin Cho [Sunkwang T&S]	

P0403	Rainfall Infiltration Analysis for the Near-Surface Radioactive Waste Disposal Facility	187
	Jin-Kyu Choi and Han-Jeong Na [KEPCO E&C]	
P0404	Cutting Force Test of Cutting Blade Modules for Slitter Design	189
	Young-Hwan Kim, Yung-Zun Cho, and Young-Soon Lee [KAERI]	
P0405	Immobilization of Radioactive Waste Generated From Primary System Decontamination Using Geopolymer Waste Form	191
	Jinmo Ahn, Won-Seok Kim, and Wooyong Um [POSTECH], Seon-Byeong Kim and Wangkyu Choi [KAERI]	
P0406	The Improvement for Operational Convenience of Solid Waste Management System	193
	Young-hwan Kim, Chang-kyun Oh, and Mu-gap Shin [KEPCO E&C]	
P0407	Review on Regulations and Practices of Periodic Safety Review for LILW Repositories	195
	Jinyong Park, Eunjin Seo, Hojin Lee, and Sangmyeon Ahn [KINS], Jaehak Cheong [Kyung Hee Univ.]	
P0408	Study on Mechanical Strength of External Wall Coating Material for Cold Crucible Induction Melter	197
	Young Hwan Hwang, Seokju Hwang, Mi-Hyun Lee, and Cheon-Woo Kim [KHNP-CRI]	
P0409	Trapping Characteristics of Cesium Iodide by Aluminosilicate Filters	199
	Jae Hwan Yang, Joo Young Yoon, Ju Ho Lee, Seok-Min Hong, and Yung-Zun Cho [KAERI]	
P0410	Semi-Quantitative Analysis of Electron Probe Micro-Analyzer for Zr-2.5Nb Pressure Tube Material	201
	Y. S. Choo, Y. J. Kim, B. O. Yoo, S. J. Baik, Y. G. Jin, K.S. Heo, H. M. Kim, and S. B. Ahn [KAERI]	
P0411	Study on Bubbler Position Optimization of Cold Crucible Induction Melter ...	203
	Seokju Hwang, Young Hwan Hwang, and Cheon-Woo Kim [KHNP-CRI]	
P0412	Spent Nuclear Fuel Storage Rack Sampling and Process Development for Evaluation of the Radionuclide Inventory	205
	Kyung-Ho Kim, Bong-Ki Ko, and Young-Ku Choi [Sunkwang T&S], Dae-Sung Kim [KHNP]	
P0413	Development of C-14 Stripping Method From Waste Resin Generated From HWR via Microwave Treatment: Part 1, Characterization of Anion and Cation Surrogate Waste Resin With Microwave Treatment	207
	Ki Rak Lee, Hwan-Seo Park, Jung-Hoon Choi, and Seung-Youb Han [KAERI], Hyeon-Oh Park [Sunkwang T&S]	
P0414	Ultra-Low Temperature, Non-Conventional Sintering of Iodate Substituted Hydroxyapatite (IO-HAP) for the Immobilization of Radioiodine	209
	Muhmood ul Hassan and Ho Jin Ryu [KAIST]	

P0415	Heavy Water Reactor (HWR) Spent Resin Mixture Processor Development ...211 Hyeon-Oh Park, Kyu-Tae Park, Jung-Min Yoo, and Young-Ku Choi [Sunkwang T&S], Yeo-jin Kim [KHNP]
P0416	The Removal of Uranium From Aqueous Effluents via Meta-Ankoleite Formation 213 Richard I. Foster, Kwang-Wook Kim, Min-Jeong Kim, Maeng-Kyo Oh, and Keun-Young Lee [KAERI]
P0417	Fabrication of Graphene Fiber Fabrics as Sorbents for Radioactive Iodine Gas 215 Minseok Lee and Ho Jin Ryu [KAIST]
P0418	An Evaluation of the Glass-Ceramic Solidification Characteristics of Uranium Catalyst Waste by Green Body Pressure 217 Hyun-hee Sung, Kwang-Wook Kim, Keun-Young Lee, Jimin Kim, and Bum Kyoung Seo [KAERI]
P0419	Effects on SS304 Contents to Corrosion Properties of ZIRLO-SS304-INCONEL718-NM Alloy for Long-Term Disposal 219 Seung-Youb Han, Seon-Ah Jang, and Hwan-Seo Park [KAERI]
P0420	Behavior of Microstructure and Corrosion Properties of Zirlo-Cr-Si-NM Metal Waste Form Alloys by Change of Si Content 221 Seon-Ah Jang, Seung-Youb Han, and Hwan-Seo Park [KAERI]
P0421	Study on Cs ⁺ Separation by Using Ionic Liquid-Solid Extraction System 223 Seung-Yeon Cho, Jung-Weon Choi, Hyo-Jeong Lee, Won-Zin Oh, and Sang-June Choi [Kyungpook Natl. Univ.]
P0422	Study on Optimization of Volume Reduction Treatment Process of Domestic Industrial Radioactive Waste 225 Yongho Hong, Heekyung Kim, Hyoungmin Park, Eunsuk Choi, and Dongchul Kim [ACT]
P0423	Preliminary Assessment of Potential Risk From Inadvertent Recycling of Disposed Activated Metal Waste and Its Implications 227 Wang Hyeon Lee, Ki Nam Kwon, and Jae Hak Cheong [Kyung Hee Univ.]
P0424	Feasibility Study on the Development of Position Sensitive System Based on Plastic Scintillating Fiber for Detecting the Leak Position of Radioactive Waste Drum 229 Young Beom Song, Hye Jin Kim, Sang Hun Shin, and Bongsoo Lee [Chung-Ang Univ.]
P0425	A Conservative Approach to Estimate the Gaseous Radionuclide Migration in LILW Repository 231 Jung-Woo Kim, Jaewon Lee, Ji-Hun Ryu, Jongtae Jeong, and Dong-Keun Cho [KAERI], Jeheon Bang [KINS]

09:00~09:20	Laser Cutting Performances of Thick Stainless Steel Plates With a High-Power Fiber Laser for Dismantling of Nuclear Power Plants 235
	Jae Sung Shin, Seong Yong Oh, Hyunmin Park, Sangwoo Seon, Chin-Man Chung, Taek-Soo Kim, Lim Lee, and Jonghwan Lee [KAERI]
09:20~09:40	The Preparation of NiFe ₂ O ₄ Thin Film Prepared by E-beam Evaporation Method for Simulated Specimen in PWR System 237
	Mansoo Choi, Junyoung Jung, Jungsoon Park, Wangkyu Choi, Sangyoon Park, Hui-Jun Won, and Seon-Byeong Kim [KAERI]
09:40~10:00	A Study of Waste Treatment in SP-HyBRID Decontamination Process Using On-Line Monitoring 239
	J. Y. Jung and S. J. Park [Chungnam Natl. Univ.], S. Y. Park, H. J. Won, W. K. Choi, S. B. Kim, and J. K. Moon [KAERI]
10:00~10:20	Reversibly Cross-Linked and Strippable Hydrogels for the Removal of Radioactive Cesium From Surface 241
	Hee-Man Yang, Chan Woo Park, Kune-Woo Lee, Ilgook Kim, and In-Ho Yoon [KAERI]
10:40~11:00	Cesium Removal From Hydrobiotite by Cationic Polyelectrolyte and Hydrogen Peroxide 243
	Ilgook Kim, Bo Hyun Kim, Chan Woo Park, In-Ho Yoon, Hee-Man Yang, and Kune-Woo Lee [KAERI]
11:00~11:20	Weathering Effects on Fractionation Change of Cesium in Soil 245
	Sang-Min Park, Seunghee Kang, and Kitae Baek [Chonbuk Natl. Univ.]
11:20~11:40	Derivation of DCGLs for the Surface Soil of Kori-1 NPP by Using RESRAD Probabilistic Analysis 247
	Ji Hyang Byon, Sang June Park, and Seokyoung Ahn [Pusan Natl. Univ.]
11:40~12:00	Development of Measurement and Segregation System for Radioactive Soil 249
	Jae-Mun Han, Sang-Chul Lee, Min-Jung Kim, and Jin-Hee Kim [HYUNDAI E&C]

P0501	Considerations in Unit Cost Factor Development for Nuclear Power Plant Decommissioning Project 253
	Wanil Jung, Myung Duck Yang, and Young Jin Yang [KEPCO E&C]
P0502	Laser Cutting Experiment of Thick Stainless Steel for Nuclear Decommissioning: Effect of Nozzle Types 255
	Seong Y. Oh, Jae Sung Shin, Sangwoo Seon, Taek Soo Kim, Hyunmin Park, Lim Lee, Chin-Man Chung, and Jonghwan Lee [KAERI]

P0503	Accident Case Analysis for Safety Assessment of Decommissioning	257
	Kyungmin Kim, Jaeyong Lee, and Yong-soo Kim [Hanyang Univ.], Je-keun Jun [UjuEnertech Co.,]	
P0504	Preliminary Evaluation of DCGLs for Site Release After Decommissioning of Korean Research Reactor 1 & 2	259
	Tae Gwan Do, Jin Ho Park, Kwang Pyo Kim, and Geun Ho Kim [Kyung Hee Univ]	
P0505	pH and Electric Conductivity of Cement Solidification of Metal Hydroxide Waste	261
	Daeseo Koo, Hyun-Hee Sung, Sang Bum Hong, and Bum-Kyoung Seo [KAERI]	
P0506	Polymeric Nanofiltration Membranes for Desalination & Nuclide Separation	263
	Hyung-Ju Kim, Keun-Young Lee, and Bum-Kyoung Seo [KAERI]	
P0507	Development of 3D Programs for Efficient Radiation Dose to Data Management	265
	SangBok Lee, Seung-gun An, Yeo-jib Choi, Jeong-soo Park, Tae-ryong Kim, and Seung-jin Cho [Sunkwang T&S]	
P0508	Extraction of Actinide and Lanthanide From Nitric Acid Medium Using Ionic Liquid	267
	Iksoo Kim, Dongyong Chung, Keunyoung Lee, and Bum-Kyoung Seo [KAERI]	
P0509	Case Study of Interview for HSA at Decommissioned NPPs in the US	269
	Donghee Lee [KHNP-CRI]	
P0510	Segmentation and Packaging Plan for the Reactor Pressure Vessel	271
	Hyun-Jae Yoo and Byung-Sik Lee [Dankook Univ.]	
P0511	Study on Induction Melter Technology to Treat Decommissioning Metallic Radwaste Generated From Kori Unit 1	273
	Sung-Hoon Hong, Seokju Hwang, and Cheon-Woo Kim [KHNP-CRI]	
P0512	Magnetic Clay Microspheres Prepared by Spray-Drying for Adsorption of Cs	275
	Chan Woo Park, Hee-Man Yang, and In-Ho Yoon [KAERI], Gyeong Jun Chung and Juri Hwang [Chungnam Natl. Univ.]	
P0513	Development of Milling Machine to Decontamination Radioactive Metal Waste	277
	Dong-Kyun Ko, Geon-Hwa Lee, Tae-Heon Kim, Sung-Jun Hong, and Eui-Dong Lee [Hana Nuclear Power Eng.]	
P0514	Case Study on Safety Assessment Approach in Yankee and Trojan Decommissioning Plan	279
	Kwangho Jo, Taesik Yun, Donghee Lee, Younggook Kim, Hyejin Jung, and Jae Yong Oh [KHNP-CRI]	

P0515	An Analysis on Internal Exposure Evaluation Code for Radioactive Aerosols	281
	Hakyun Lee, Sunll Kim, and Jongsoon Song [Chosun Univ.]	
P0516	An Analysis of Characteristics of Radioactive Aerosol by Cutting Methods Available for Decommissioning of Nuclear Power Plants	283
	Sunll Kim, Hakyun Lee, and Jongsoon Song [Chosun Univ.]	
P0517	Preliminary Considerations for Preparing a Domestic Decommissioning Radiological Environmental Impact Assessment	285
	Sang-Ho Lee and Chang-Lak Kim [KINGS]	
P0518	Preliminary Study on Environment Mapping for Nuclear Reactor Dismantling	287
	Sungmoon Joo, Jonghwan Lee, Ikjune Kim, Dongjun Hyun, Shinyoung Kang, Kwanseong Jeong, and Byungseon Choi [KAERI]	
P0519	Design Characteristics for Ease of Dismantling of Research Reactor	289
	Kook-Nam Park, Seung-Kook Park, Yun-Dong Choi, and Young-San Choi [KAERI] Do-Hyung Kim [ORION E&C]	
P0520	Study on the Precipitation Behavior of Metal Ions in the Chemical Decontamination Waste by Using Carbonates	291
	Jin-Hee Kim, Hyun-Kyu Lee, June-Hyun Kim, Won-Zin Oh, and Sang-June Choi [Kyung pook Natl. Univ.]	
P0521	Computational Model of Copper Catalyzed Metal Oxide Dissolution in Reducing Acidic Solution	293
	Seon-Byeong Kim, Hui-Jun Won, Sang-Yoon Park, Mansoo Choi, Wangkyu Choi, and Bum-Kyoung Seo [KAERI]	
P0522	Development of Waste Characterization System for the Life-Cycle of Decommissioning Nuclear Facilities	295
	Inhye Hahm, Daeseo Koo, Kook-Nam Park, Seung-Kook Park, Sang Bum Hong, and Bum-Kyoung Seo [KAERI]	
P0523	Correlations Between Safety Assessment and WDFs in NPP Decommissioning	297
	Jae Yong Oh, Kwangho Jo, Ji-hwan Yu, and Young gook Kim [KHNP-CRI]	
P0524	Dismantlement of Surveillance Capsule for Hanul Unit 3	299
	Kisoo Heo, Boungok Yoo, Seungjae Baik, Gilsoo Kim, and Youngjun Kim [KAERI]	
P0525	Consideration of Treatment and Disposal for Decommissioning Waste	301
	Young gook Kim, Taesik Yun, Hyejin Jung, Kwangho Jo, and Jae Yong Oh [KHNP-CRI]	

6분과 방사선환경 및 안전 (Oral)

09:00~09:40	Aerial Radiation Measurements as an Effective Means of Collecting Emergency Information	305
	Byoung-Jik Kim [KINS]	
09:40~10:00	Beta-Ray and Gamma-Ray Detection Technique for the Soil Surface Using Plastic Scintillators	307
	Jun Woo Bae, Uk Jae Lee, and Hee Reyoung Kim [UNIST]	
10:00~10:20	The Real-Time Monitoring of Dose Rate and Radioactivity Using an Environmental Radiation Monitor Based on the NaI(Tl) Detector	309
	Young-Yong Ji, Chang-Jong Kim, Kyo-Sun Lim, and Wanno Lee [KAERI] Hyon-Sock Chang and Taehyung Lim [SI Detection Co. Ltd.]	
10:40~11:00	Design Requirements of a Large-Area Radiation Measuring Instrument for Decommissioning of Nuclear Power Plants	311
	Kidoo Kang, Yuho Won, Moonhyung Cho, and Jaeun Lee [KHNP-CRI]	
11:00~11:20	Radioactivity Characteristics and Safety Management of Steel Scarp	313
	Yeongsuk Go, Eun Lee, and Suk K. Sim [en2t], Sun Dong Ju, and Sang-uk Oh [KOFONS]	
11:20~11:40	Radiological Assessment of Spent Resin Treatment Device for Disposal Safety	315
	Uk Jae Lee, Woo Nyun Choi, and Hee Reyoung Kim [UNIST]	
11:40~12:00	Inhalation Dose Assessment at Phosphate and Potassium Industries	317
	Jung Hwan Jang, Seung Woo Ji, Jin Hee Jeon, Kwang Pyo Kim, and Jae Kook Lee [KyungHee Univ.]	

6분과 방사선환경 및 안전 (Poster)

P0601	Improved PRIDE Access Control System	321
	Seonho Noh, Youngkuk Jang, Hui-seok Gang, and Ilje Cho [KAERI]	
P0602	Tritium Inventory Estimation for HCCR-TBS at PD-1 Phase	323
	Hyung Gon Jin, Dong Won Lee, Jae Sung Yoon, Suk Kwon Kim, Eo Hwak Lee, Seong Dae Park, Dong Jun Kim, and Yunjeong Hong [KAERI], Seungyon Cho [NFRI]	
P0603	External Dose Assessment to Workers in a NORM Industry Applying External Dose Rate Map	325
	Seung Woo Ji, Jung Hwan Jang, Do Yeon Kim, Jeong Hwan Yoon, Kwang Pyo Kim, and Jeong Hwan Yoon [KyungHee Univ.]	
P0604	Improvement of Remote Control of Storage Vault Lid O-Ring Replacement in ACPF	327
	Yunmock Jung, Jongkwang Lee, Sunseok Hong, and Eunyong Choi [KAERI]	

P0605	Analysis of Natural Radionuclides Concentration in Jeju Atmospheric PM ₁₀ Aerosols During the Period of Asian Dust in 2015	329
	Chung-Hun Han, Duk-Woo Kim, and Jae-Woo Park [Jeju Natl. Univ.]	
P0606	Enhanced Maintenance of the Supply Piping for L-Ar	331
	Youngkuk Jang, Seonho Noh, Hui-seok Kang, and Ilje Cho [KAERI]	
P0607	Study on Nuclear Test Scheme and Test Results for Safety Neutron Measurement System in Research Reactor	333
	Sanghoon Bae, Young-ki Kim, and Hanju Cha [KAERI]	
P0608	Analysis of Off-Site Dose on the Public Due to the Discharge of Gaseous Tritium From Wolsong NPPs	335
	Dae-Sun Son, Jong kuk Lee, Kwan-hee Lee, Sung-il Kim, and Sangmyeon Ahn [KINS]	

7분과 방사화학 (Oral)

09:00~09:20	Liquid-Liquid Extraction Combined With Capillary Electrophoresis for the Determination of Rare Earth Elements in a High Saline Matrix (LiCl-KCl Molten Salt)	339
	Jihye Kim and Tae-Hong Park [KAERI], Kihwan Choi [KRISS]	
09:20~09:40	Development of Spectrophotometric On-Line Monitoring System for Groundwater Near Silo Repository	341
	Tae-Hyeong Kim and Hong Joo Ahn [KAERI], Sol-Chan Han and Jong-Il Yun [KAIST]	
09:40~10:00	Kinetic and Mechanistic Studies on Phosphoester Cleavage by Tetravalent Uranium Species	343
	Hyejin Cho and Wansik Cha [KAERI]	
10:00~10:20	UV-Vis Absorption Properties of Uranyl-Nicotinic Acid Complexes in Aqueous Medium	345
	Seonggyu Choi and Jong-Il Yun [KAIST]	
10:40~11:00	Variation of Compositions at the Interface Between U _{0.97} Th _{0.03} O ₂ and Zr From High-Temperature Annealing	347
	Young-Sang Youn, Jeongmook Lee, and Jandee Kim [KAERI], Jong-Yun Kim [UST]	
11:00~11:20	Electrochemical Behaviors of Nickel Hexacyanoferrate-Graphene Oxide Nanocomposite for Cesium Ion Separation	349
	Dongchul Choi, Sang-Eun Bae, and Tae-Hong Park [KAERI]	
11:20~11:40	A Study on Chemical Separation of Pm-147 From RI Wastes	351
	Jung Bo Yoo, Sang Ho Lim, Junhyuck Kim, Hong Joo Ahn, and Kwang Yong Jee [KAERI]	
11:40~12:00	pH Dependence of Cs(I) and Sr(II) Coprecipitation With Fe(III) in Aqueous Solutions	353
	TaeJun Kim and Minsik Kim [UST], Jei-Won Yeon [KAERI]	

P0701	Physicochemical Properties of Trivalent Rare Earth Elements-Doped UO_2 ...357 Jandee Kim, Jeongmook Lee, Young-Sang Youn, and Jong-Goo Kim [KAERI], Seohyeon Park and Jeongmi Park [Chungnam Natl. Univ.], Jong-Yun Kim and Sang Ho Lim [UST]
P0702	Separation of Niobium From Nickel-Chromium-Iron Containing Nickel-Based Alloys 359 Kyungwon Suh and Youngkyong Ha [KAERI]
P0703	Development of an In-Hotcell Bromine Method for the Determination of Metallic Contents of Electrolytic Reduction Products in the Pyroprocessing of Spent Nuclear Fuel 361 Young-Hwan Cho, Ji-Hye Kim, and Tae-Hong Park [KAERI]
P0704	Electrochemical Dissolution of Platinum in a Chloride-Based Molten Salts ...363 Suhee Choi, Sang-Eun Bae, and Tae-Hong Park [KAERI]
P0705	Chemical Properties of Hypo-Stoichiometric $\text{U}_{1-y}\text{Nd}_y\text{O}_{2-x}$ Produced by Neodymium Doping Into UO_2 365 Jeongmi Park, Seohyeon Park, and Choong Kyun Rhee [Chungnam Natl. Univ.], Young-Sang Youn, Jeongmook Lee, Jandee Kim, and Sang Ho Lim [KAERI]
P0706	Structural and Electrochemical Properties of Non-Stoichiometric UO_{2-x} Doped With Trivalent Rare Earth 367 Seohyeon Park, Jeongmi Park, and Choong Kyun Rhee [Chungnam Natl. Univ.], Jandee Kim, Jeongmook Lee, Young-Sang Youn, and Sang Ho Lim [KAERI]
P0707	Study on Dissolution of Glass for Measurement of Activity of α , β , and γ Emitting Nuclides in Vitrification Radioactive Waste Sample 369 Kwang-Soon Choi, Jaesik Hwang, Simon Oh, Young-Bok Kim, and Hong Joo Ahn [KAERI]
P0708	Electrical Conductivities of $\text{UCl}_3\text{-LiCl-KCl}$ Molten Salts Measured by Using a Short Time Interval and Multiple-Potential Step Technique 371 Sang-Heon Lee, Jong-Yun Kim, Sang-Eun Bae, and Tae-Hong Park [KAERI]
P0709	Burnup Determination of Irradiated Sintered Solid and Annular Pellets by Neodymium Monitor Methods Based on Isotope Dilution Mass Spectrometric Measurements 373 Jung Suk Kim, Byungman Kang, Yang-Soon Park, Jai Il Park, and Yeong Keong Ha [KAERI]
P0710	Determination of ^{129}I in Simulated Radwastes Using Anion-Exchange Resin 375 Hwa Sung Yoo, Ke Chon Choi, Keon Ho Kim, and Hong Joo Ahn [KAERI]
P0711	Comparison of ^{55}Fe Recovery Results by Gravimetric and Spectrometric Analysis 377 Hyeon Jeong Oh, Jung Bo Yoo, Hong Joo Ahn, and Kwang Yong Jee[KAERI]

1분과

핵주기정책 및 핵비확산 (Oral)



Radioactive Waste Management Policy in Korea – Experiences, Challenges, and the Way Forward

Jae Hak Cheong

Kyung Hee University, 1732 Deokyoungdaero, Giheung-gu, Yongin, Gyeonggi-do, Republic of Korea

jhccheong@khu.ac.kr

1. Introduction

The first national radioactive waste management (RWM) policy and strategy was established in 1984, and then the RWM policy and strategy have been revised several times until recently. Though some of the objectives in the RWM policy and strategy have been attained, many of them have not been successfully implemented due to oppositions from residents and misconducts in communication with interested parties. However, no specific studies on the advantages and disadvantages of the Korean government's past policy framework for RWM have been openly reported. Accordingly, fundamental principles for establishing the RWM policy and strategy are reviewed, the past experiences in Korea are analyzed, and then the desirable future direction of the RWM policy framework is proposed in this paper.

2. Principles of RWM Policy Framework

2.1 Policy and Strategy

RWM policy is a series of goals and/or requirements for safe management of radioactive waste (RW), which allocates clear roles and responsibilities [1]. The RWM policy is established by the government and is codified in the national legislative system in some countries.

On the other hand, RWM strategy is a set of measures to achieve the goals of the RWM policy and can be developed by the government, national RWM agency, or RW producers either in governmental or civil sectors.

Table 1 shows and compares the RWM policy and strategy for each country of interest.

Table 1. RWM Policy and/or Strategy in Selected Countries

Country	RWM Policy and Strategy
USA	· Nuclear Waste Policy Act of 1982
	· DOE, Strategy for the Management and Disposal of Used Nuclear Fuel, 2013.
Canada	· Low-Level Radioactive Waste Policy Amendments Act of 1985
	· Radioactive Waste Policy Framework (1996)
France	· National Plan for the Management of Radioactive Materials and Waste (PNGMDR)
Spain	· General Radioactive Waste Management Plan
Korea	· Basic Plan for RWM (LILW in 2015 and HLW in 2016)
	· Implementation Plan for RWM (LILW in 2017)

In a few countries such as France and Spain, the national plan for RWM involving a combined strategy and policy is established. In addition, the policy and strategy for RWM is codified in the national legislative system in USA.

2.2 Elements of RWM Policy

Main elements to be considered in establishing a RWM policy are reportedly known to be as follows [1]:

- Allocation of responsibilities of ministries, regulatory authority, operators of RWM facilities, etc.;
- Provisions of resources,
- Safety and security objectives;
- Minimization of RW
- Position on export and import of RW;
- Plan for spent fuel management (SFM);
- Plan for RWM (i.e. disused sealed sources and other types of RW);
- Naturally occurring radioactive material (NORM);

- and
- Public information and engagement.

Regarding export and import of RW, for instance, Korea has been accepting the disused sealed sources exported to overseas if the importing country has an intention to return it back to the Korean manufacturer. However, returning back of the imported disused sealed sources to the exporting country to reduce the domestic inventory of RW is not a mandatory option in Korea yet.

3. Gap Analysis

The first RWM policy of Korean government was established as “RWM Measures” in 1984 after deliberation by the former Atomic Energy Commission. There have been a variety of revisions of the national policy and strategy for the last thirty-six years, and now the latest version of the policy and strategy for RWM are shown in Table 1.

Through gap analysis between the principles of RWM policy and the present framework for RWM policy and strategy in Korea, a few gaps were identified as:

- Unclear involvement of competent ministries in the official policy making process;
- Insufficient consideration of identification of RW streams to be potentially non-compliant to the waste acceptance criteria (WAC);
- Unclear plan to minimize national inventory of each type of RW; and
- Unclear process to inform and engage the Public in establishing, implementing, and assessing the performance of the LILW management policy.

4. Conclusion

The points identified in the gap analysis would be potential action items to be considered to enhance the effectiveness of the RWM policy and strategy in Korea as:

- Officially inviting competent ministries, interested organizations, and stakeholders in establishing and revising the RWM policy;

- Covering all types of RW streams in Korea and shed light on the potential compliance of those with WAC;
- Considering a position on obligatory returning of the disused sealed sources back to the overseas manufactures; and
- Communicating more with the public in establishing, implementing, and assessing the performance of the RWM policy.

REFERENCES

- [1] IAEA, Policies and Strategies for Radioactive Waste Management, IAEA Nuclear Energy Series No. NW-G-1.1, 2009.
- [2] H.J. Kim, A Study on Establishment of Support Systems to AEC and AEC's Expert Committee, KAERI/RR-2596/2004, 2005.

Investigation of Legal and Regulatory Frameworks of Waste Management System for Very Low Level Waste

Ji Young Song, Jin Ho Park, Jong Hwa Kim, and Kwang Pyo Kim*

Kyung Hee University, 1732 Deokyoung-daero, Giheung-gu, Yongin-si, Gyeonggi-do, Republic of Korea

*kpkim@khu.ac.kr

1. Introduction

The Korean government determined permanent shutdown of Kori 1 unit in 2017. As a result of decommissioning, large amounts of decommissioning waste will be generated. The proportions of the wastes by waste categorization were reported to be 4.2% for intermediate level waste (ILW), 28.7% for low level waste (LLW), and 67.1% for very low level waste (VLLW) [1]. The half of VLLW is estimated to be clearance waste after the delay and decay process.

Article 6 and 9 of Radioactive Waste Control Act in Korea require management for transport, storage, processing, and disposal of radioactive waste. Consequently, the government should consider a countermeasure for transport, packaging, and acceptance criteria of radioactive waste. Due to high generation volume of VLLW resulting from nuclear power plant (NPP) decommissioning, waste management system for VLLW is necessary. However, there is no such waste management system for NPP decommissioning in Korea.

Final objective of this study is to develop a waste management system. As a beginning step for the objective, we investigated legal and regulatory framework of waste management system for VLLW.

2. Radioactive Waste Management System

To manage radioactive waste, cooperation of at least 3 organizations is required. They include NPP licensee, radioactive waste disposal agency, and the government.

General radioactive waste management system is given in Fig. 1 [2]. If decommissioning wastes are generated to NPP licensee, the wastes are classified by generation and classification of the radioactive wastes (See Fig. 2). The classified wastes are processed based on the radioactive waste properties. For example, combustibility radioactive waste may be processed by incineration. After treatment, waste is inserted in a container. It is radioactive waste management system for NPP licensee.

Packaged waste is inspected by legal and regulatory requirements of transport, packaging, and disposal. Legal and regulatory requirements are documented by the national acts. The government provides legal and regulatory requirements for transport, packaging, and disposal of the radioactive wastes. It is radioactive waste management system for the government.

Moving to disposal site, waste is stored and disposed following radioactive and non-radioactive characteristics. It is radioactive waste management system for radioactive waste disposal agency.

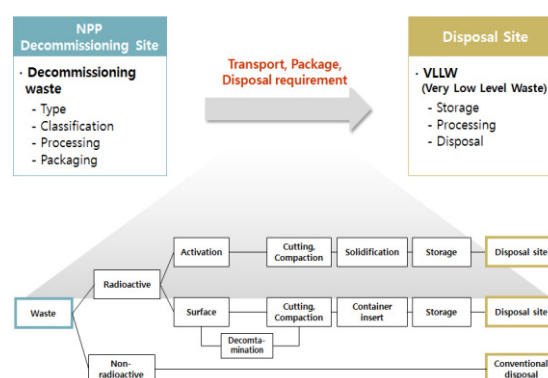


Fig. 1. Radioactive waste management system.

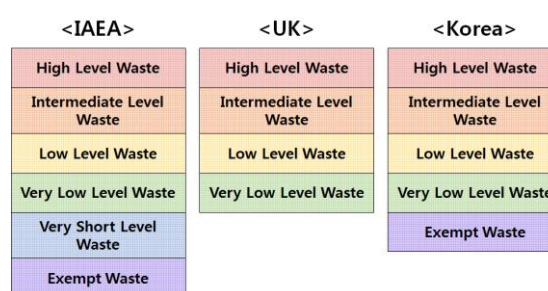


Fig. 2. Radioactive waste classification of IAEA, UK, and Korea.

3. Legal and Regulatory Frameworks of Radioactive Waste Management System

In this study, we investigated legal and regulatory frameworks of waste management system for VLLW of IAEA and UK.

3.1 IAEA

3.1.1 Transport and storage criteria. Waste packages are designed and fabricated to have sufficient mechanical strength to bear loads and to comply with following requirements for waste handling, stacking, transport and storage.

- Weight, volume or dimension limits
- Stress and corrosion resistance
- Mechanical properties etc.

3.1.2 Waste Acceptance criteria. Waste acceptance requirements are qualitative or quantitative criteria to safely storage or disposal. General requirement are initially defined (1) national radioactive waste disposal policy, (2) general information on the type and quantities of waste package, (3) the availability or certain sites. General radiological criteria are established in national regulations. Dose rate restrictions can be achieved when using standard package and waste containers by limiting the activity of certain radionuclides. Non-radiological criteria are based on chemical or biological hazards. For example, explosives or products capable of strong exothermic reactions should be limited.

3.2 UK

3.2.1 Transport and storage criteria. The acceptable range of waste packaging for disposal includes 210 L drum and soft-side packaging. The acceptable range of waste transport include 1/3 height, 2/3 height and full height ISO container. Radiation level on the waste package shall be controlled to less than 5 $\mu\text{Sv/h}$ at contact and less than 2.5 $\mu\text{Sv/h}$ at a distance of 1 meter.

3.2.2 Waste Acceptance criteria. Radioactivity limit for disposal is given Table 1. Most of radionuclide shall be as low as 4 MBq/ton and tritium as low as 40 MBq/ton. In case of metallic waste, radionuclide concentration shall be as low as 200 MBq/ton. External non-fixed contamination levels shall be not more than 0.4 Bq/cm² for all alpha-emitting radionuclide and 4 Bq/cm² for all other radionuclides averaged over an area of 300 cm².

For physical and chemical properties, waste shall not contain reactive metal, explosive materials, corrosive materials, liquid with flashpoint less than

21 °C, strong oxidizing agents etc. Therefore, chemical materials affecting human and environment shall not be contained. In addition, surface of package shall not be contaminated chemically [3].

Table 1. The radioactivity limit for disposal in the UK

Radionuclide	Specific Activity (MBq/ton)	
	Acceptable	Likely to be Accepted
All	< 4	< 200
Tritium	< 40	< 200

4. Conclusion

We investigated legal and regulatory frameworks of waste management system for VLLW of IAEA, UK. Suggested comprehensive information are given in the literature. In addition, detailed numeric values are provided for transport, package, and disposal. Based on this information, we will develop a waste management system for NPP decommissioning. It will contribute the establishment of overall management system in accordance with the characteristics of decommissioning wastes.

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. 20171510300580).

REFERENCES

- [1] KEITI, "Trend of NPP Decommissioning waste management", Korea Environmental Industry Technology Institute, Konetic report 2016-138, 2016.
- [2] IAEA, "Disposal Aspects of Low and Intermediate Level Decommissioning Waste", International Atomic Energy Agency, IAEA-TECDOC-1572, 2007.
- [3] LLW Repository Ltd, "Waste Acceptance Criteria-Very Low Level Waste Disposal", WSC-WAC-VER 3.0, 2012.

Current Issues and Improvement Plans for Management and Disposal of Small Quantity Nuclear Material

Junghee Son*, Chul Heo, and Seungmin Lee

Korea institute of Nuclear Nonproliferation and Control, 1534, Yuseong-daero, Yuseong-gu, Daejeon, Rep. of Korea

*jhson@kinac.re.kr

1. Introduction

Small quantity nuclear material are not subject to use permit under the Nuclear Safety Act, but are subject to national accounting management and reporting under the Korea-IAEA safeguards agreement. Therefore, small quantity of nuclear material need to be managed at the national level. However, due to lack of relevant regulations and the smallness of the use agencies, practical management is difficult. In particular, there are no clear procedures or standards for the disposal of small quantity nuclear material, which may cause various problems in the future.

In this paper, we analyze the current system related to the disposal of small quantity nuclear material and propose a plan for the effective disposal of nuclear materials.

2. Related regulations and current issues

2.1 Scope and related obligations of small quantities of nuclear material

Small quantity nuclear material belong to the nuclear fuel material, and also belong to specific nuclear material, as defined by the Nuclear Safety Act. It is also a material exempted from the use permit in accordance with the law.

On the other hand, in the Additional Protocol to the IAEA Safeguards Agreement, there is no lower limit of the quantity of nuclear material that is exempted from the safeguards measure unlike the domestic law.

Therefore, nuclear materials used for nuclear purposes in Korea are subject to national safeguard regardless of whether they are included in the current law or not, and it is obliged to provide IAEA with information on the amount and use thereof and to receive inspections.

Most of the materials referred to as small nuclear materials in Korea are depleted uranium used radiation shielding for radiation projector.

In recent years, it has been used in various forms such as tracers, standard sources and chemical samples, and its usage and kinds are increasing.

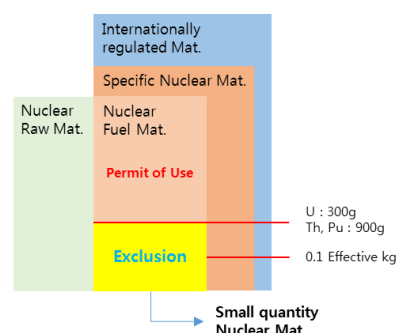


Fig. 1. Scope & definition of small quantity nuclear material.

2.2 Domestic criteria for disposal of nuclear fuel Material

Detailed criteria for radionuclide concentration related to radioactive waste disposal are presented by the Nuclear Safety and Security Commission (NSSC).

According to the NSSC, if the concentration of radionuclide in the waste is less than its self disposal limit, it is regarded as general industrial waste, not radioactive waste, and can be incinerated, buried or recycled. If it exceeds the self disposal limit, it should be classified as a radioactive waste and disposed of at an approved disposal facility.

Table 1. Self disposal limit level for each radionuclide

Radionuclide	Specific Activity (Bq/g)
Th-229, U-232, Pu-238, Pu-239, Pu-240, Pu-242, Pu-244	0.1
Pu-236	1
U-230b, U-236, Np-240, U-241	10
Pu-234, Pu-235, Pu-237 U-231, U-237, U-239, U-240,	100
Th-226, Pu-243	1,000

However, at present, small quantity nuclear material are not included in the scope of disposal because of the operation plan of domestic middle and low level disposal facilities. Therefore, there is no way to dispose of small quantities of nuclear material in Korea.

Because of this, small quantity nuclear materials, which are no longer used, are kept by the use agencies themselves, and there is a possibility of

unauthorized disposal and arbitrary disposal due to the burden of management and the smallness of the facility.

2.3 IAEA Safe Guard criteria for nuclear fuel Material

The criteria for judging the release of regulation on nuclear fuel materials are different between domestic laws and IAEA agreements.

In domestic law, the waste that meets the concentration criteria of radionuclide, which is judged to have no risk based on the radiation hazard, is defined as the object of release of regulation. The IAEA determines whether or not a safe guard termination is based on whether or not the subject material can be used for nuclear activities.

Therefore, even if it is subject to own disposal under the Nuclear Safety Act, there is a possibility it is still subject to safeguards from the viewpoint of the IAEA. In such cases, it will have administrative burdens such as reporting and inspections.

Also, it is difficult to know the quantitative level for safe guard termination by the agreement. And its applicability should also be judged by the IAEA.

Table 2. Safeguards termination article under the ROK-IAEA Agreement

Article 11 TERMINATION OF SAFEGUARDS Consumption or dilution of nuclear material

- nuclear material upon determination by the Agency that the material has been consumed, or has been diluted in such a way that it is no longer usable for any nuclear activity relevant
- or has become practically irrecoverable.

from the point of view of safeguards

3. Improvement Plans

3.1 IAEA safeguards agreement reflected in domestic waste management system

It is necessary to reflect the IAEA safeguards agreement in the domestic waste management system in order to minimize the difference between the radioactive waste management standard in the Nuclear Safety Act and the IAEA safeguards agreement.

Therefore, it is necessary to establish an institutional apparatus that can confirm whether or not the activities are violating the IAEA safeguards agreement when disposing small quantity nuclear materials. In addition, through consultation with the IAEA, it will be possible to improve institutional efficiency by clarifying the criteria for quantitative safeguards termination and incorporating the IAEA safeguards agreement into the domestic waste management system.

3.2 Establishment of national disposal and management system for small quantity nuclear material

Currently, the radioactive waste disposal facility operation plan does not include nuclear material in the disposal of the first stage disposal facility.

Future plans for the disposal of nuclear materials will need to be made in consultation with relevant ministries and agencies in order to include management plans for small quantities of nuclear material in the future radioactive waste management plan.

4. Conclusion

At present, radioactive waste related standards have been established in view of safety. It would not be possible to challenge this approach because it is universal and reasonable to exclude materials that are considered to have no significant risk from safety as a criterion for radioactive waste regulation.

However, in Korea, where nuclear nonproliferation and state control of nuclear material are obligatory under the IAEA safeguards agreement, approaching regulations only from a safety standpoint is problematic. This causes difficulties in carrying out systematic management at the national level.

To minimize these administrative difficulties, it is necessary to introduce a safe guard perspective within the current regulatory framework.

And there is a need for a reasonable and realistic way to do this, and this requires discussions between relevant agencies and government departments.

REFERENCES

- [1] Choi Young Myung and others, Overseas case study on implementation of bilateral nuclear cooperation agreement and management of small quantity nuclear material, Alumni Network of Science & Technology Institutes, 2017.
- [2] Baek Min Ho and others, A Study on Institutional Improvement Plan for State Management of Small Quantity Nuclear Material, Korea Institute of Nuclear Nonproliferation And Control, 2014.

Pyro Technical Data and Information Management

Jeong-Guk Kim*, Won-Il Ko, Jeong-Hoe Ku, and Hyo-On Nam

Nuclear Fuel Cycle Technology Lab., KAERI, 989-111 Daedeok-daero, Yuseong-gu, Daejeon, 34057

*jungkim@kaeri.re.kr

1. Introduction

Two main management areas needed to successfully carry out a complex project with long project term, many stakeholders and success targets are systems engineering (SE) and project management [1]. Pyro project is an example of complex project, which develops process and durable equipment under high radioactivity and high corrosive environment, design of equipment and facility, and also involves joint research to verify technical and economic feasibilities, and nuclear nonproliferation acceptance.

In this article, we reviewed the technical management related standards in a view of systems engineering (SE), and then suggested the necessary process, a technical data and information management.

2. Systems Engineering (SE) and Technical Management Processes

2.1 Systems Engineering (SE) and Project Management

According to ISO/IEC/IEEE-15288:2015 standard [2], “systems” are defined as man-made, created and utilized to provide products or services in defined environments for the benefit of users and other stakeholders. And, “systems engineering (SE)” is an interdisciplinary approach and means to enable the realization of successful systems, and is an integral part of project management in that it plans and guides the engineering effort [3].

2.2 SE Standards

Typically, the following typical standards: ANSI/EIA-632 [4] and ISO-15288 [2], are widely known, and only ISO-15288 is accepted as the Korean Industry Standard, KS-X-ISO/IEC-15288 [5].

ISO-15288 was first enacted in 2002 and has been upgraded in 2009 and 2015. The ISO-15288:2015 consists of 30 processes in 4 groups: agreement, organizational project-enabling, technical management, and technical processes. The technical management process group is composed of project planning process, project assessment and control process, decision management process, risk management process, configuration management process, measurement process, and quality assurance process.

Another best practice in SE effort is NASA/SP-2007-6105 Rev1, “NASA Systems Engineering Handbook” [1], which has been applied to the development and implementation of large and small NASA programs and projects. The NASA’s SE consists of 17 processes in 3 groups: system design, product realization, and technical management processes. Here, the technical management processes, which is composed of technical planning, requirement management, interface management, technical risk management, configuration management, technical data management, technical assessment, and decision analysis, play an important role to manage and control the systems between system design and product realization.

3. Pyro Technical Data and Information

Management

The review of above SEs shows that an important function in technology management is to manage the resources or technical performance (including technical information, data and configuration) of the project. In addition, decision making, risk management, assessment and measurement are also important technology management processes. In the current state of the pyro project, which focuses on the development of pyro technology and equipment, an 'integrated technical data management' with the function to manage pyro-related technical data and information, assess and measure the pyro technology, and maintain the product consistency is essential. On the other hand, quality assurance and risk management should be promoted independently at commercialization stage.

4. Conclusion

The technical management processes were reviewed by the SE standards and handbooks. The technology management processes that can be applied from the pyro technology development phase were analyzed to be assessment and control, configuration management, measurement, technical data management and QA. Because QA has been already applied, an 'integrated technical data management' with the function to manage pyro-related technical data and information, assess and measure the pyro technology, and maintain the product consistency is needed for the present pyro project.

REFERENCES

- [1] NASA/SP-2007-6105 Rev1, "Systems Engineering Handbook", US National Aeronautics and Space Administration (2007).
- [2] ISO/IEC/IEEE-15288, "Systems and software engineering - System life cycle processes", International Standard Organization (2015).
- [3] A. Kossiakoff, et al., "Systems Engineering Principles and Practice", 2nd Ed., John Wiley & Sons, Inc., Hoboken, New Jersey (2011).
- [4] ANSI/EIA-632, "Processes for Engineering a System", Electronic Industries Alliance, Approved Jan. 7, 1999 (1998).
- [5] KA X ISO/IEC/15288:2009, "Information Technology – Systems and software engineering - System life cycle processes", Korean Agency for Technology and Standards, <http://www.kats.go.kr> (2009).

Progress of Design of Containment and Surveillance System for Pyroprocessing Safeguards

Se-Hwan Park *, Dae-Yong Song, Bong Young Kim, Seong-Kyu Ahn, and Jeong-Hoe Ku

Korea Atomic Energy Research Institute, Yuseong-gu, Daejeon, 34057, Korea

*Corresponding author: ex-spark@kaeri.re.kr

1. Introduction

Pyroprocessing has been developed for the reuse of the spent fuel at Korea Atomic Energy Research Institute (KAERI). Since the effective and efficient safeguards in the pyroprocessing facility is essential to realize the pyroprocessing facility in Korea, the safeguards technologies has been developed and the safeguards system has been designed in parallel with the process technology development.

Nuclear material accountancy (NMA) plays most important role in safeguards of the nuclear facility. KAERI has designed the conceptual safeguards systems, especially based on NMA, of Reference Engineering-scale Pyroprocessing Facility (REPF), and conceptually designed Korea Advanced Pyroprocessing Facility (KAPF). Based on the calculation of the Material Unaccounted For (MUF) and MUF uncertainty of the NMA systems, the NMA system is shown to be effective in detecting the diversion of the nuclear material in the pyroprocessing facilities.

Containment and Surveillance (C/S) is also important in the safeguards of the nuclear facility, especially in the pyroprocessing facility. When KAERI developed the safeguards systems, the C/S system was also designed to enhance the safeguardability in the pyroprocessing facility.

In the present work, a method to upgraded the system requirement of C/S system is proposed and the C/S system is being upgraded.

2. C/S system requirement development

The C/S system concepts for the safeguards in reprocessing plants were developed in the previous works [1, 2, 3]. A containment boundary concept was introduced in their work, and all the nuclear materials were usually assumed to be positioned inside the containment boundary. Nuclear materials were transferred through penetrations, which were placed in the containment boundary. The penetration was the position where both the nuclear material diversions and the detection of nuclear material diversion occurred. Diversion detection probabilities or diversion detection thresholds in the penetrations were assigned to assess the effectiveness of the C/S system.

In the present work, the C/S equipment, which can be applied to the C/S system of pyroprocessing facility, were selected at first. They are the neutron monitor, gamma monitor, seal and surveillance camera. The C/S equipment can be divided into 2 groups. The first group contains the quantitative equipment, such as neutron monitor, and gamma monitor, the performance of which can be expressed quantitatively. The second group contains the qualitative ones, such as seal and surveillance camera, the performance of which can be represented qualitatively. The performance of the C/S system is evaluated only considering the quantitative C/S equipment. The sensitive diversion paths are extracted. After that, the C/S system performance is

revised including the qualitative C/S equipment.

The process materials are transferred through the transfer points such as transfer tunnel, transfer port or transfer hatch. The transfer point corresponds to the penetration in the previous work [1]. The each hot cell of the pyroprocessing facility correspond to the containment boundary [1]. The diversion materials are also transferred through the transfer points. The transfer points are divided into two groups. The first group includes the transfer points, where the process material or diversion material are transferred between the hot cells, and the second group includes the transfer points, where the process material or diversion material transferred outside the facility.

The quantitative C/S equipment are assumed to be placed to the first group transfer points, and the neutron yield, gamma dose rate and mass of transfer material are calculated for the three cases, 1) process material transfer, 2) diversion material transfer, and 3) the simultaneous transfer of process material and diversion material. The calculation is based on pre-conceptually designed KAPF facility model [4]. From the calculation, the difference of neutron yield and gamma dose rate per mass between the diversion case and non-diversion case are obtained. Transfer points can be categorized based on the easiness of diversion detection with the quantitative C/S equipment.

After the categorization of the transfer points and analyze the effectiveness of the quantitative C/S equipment, the effectiveness of the qualitative C/S equipment are included to the design the C/S system of the pyroprocessing facility.

3. Summary

C/S places important role to meet the IAEA safeguards goal efficiently in the pyroprocessing

facility. The system requirement of C/S system is developed by including the containment boundary and penetration concept. The effect of the quantitative C/S equipment is assessed and the penetration points are categorized depending on the diversion detection easiness with the quantitative C/S equipment. The C/S system is upgraded including the qualitative C/S equipment. This work will help to validate the efficiency of C/S system in the pyroprocessing safeguards.

Acknowledgement

This work was supported by a National Research Foundation of Korea (NRF) grant funded by the Korean government's Ministry of Science and ICT (MSIT) (No. NRF-2017M2A8A5015084).

REFERENCES

- [1] M. E. Bleck et al., "Containment/Surveillance Concepts for International Safeguards in Reprocessing Plants", SAND80-1390C (1980).
- [2] J. N. Cooley et al., "Assessing the Credibility of Diverting through Containment Penetrations", 1980 Annual Meeting of INMM (1980).
- [3] L. B. Ellwein et al., "A Methodology for Evaluation of Containment and Surveillance Safeguards System Performance", Journal of Nuclear Material Management, XI(1), 28 (1982).
- [4] H. J. Lee et al "Development of Pyroprocessing Facility Design Technology", KAERI/RR-4165/2016 (2016).

Fast Neutron Detection for Spent PWR Fuel in Dry Storage

Ju Young Jeon¹, Taehoon Jeon³, Jin Ho Park², Jung Youn Choi¹, Kwang Pyo Kim², and Heejun Chung^{1*}

¹Korea Institute of Nuclear Nonproliferation and Control, 1534, Yuseong-daero, Yuseong-gu, Daejeon, Rep. of Korea

²Kyung Hee University, 1732 Deokyoung-daero, Giheung-gu, Yongin-si, Gyeonggi-do, Republic of Korea

³Orbitech Co., Ltd, 1130, Beoman-ro, Geumcheon-gu, Seoul, Republic of Korea

*Corresponding author:hjchung2@kinac.re.kr

1. Introduction

Nuclear safeguards is of fundamental importance for any country operating nuclear power plants. The management of spent nuclear fuel (SNF) recently became important concern in S. Korea due to nuclear-free energy policy and out of pool storage capacity. These would lead to require the accelerated plan how to secure SNF in dry storage.

SNF in dry storage is notoriously difficult to accurately verify with nondestructive methods, so it is necessary to develop nondestructive techniques for ensuring and verifying the integrity of spent nuclear fuel.

Several techniques based on gamma and thermal neutron detection have been developed and applied for the monitoring and certification of SNF. However, there are very limited technologies available due to the influence of shielding materials and interference of background radiation.

Therefore, Korea Institute of Nuclear Nonproliferation and Control (KINAC) currently proposed a novel SNF monitoring technique using He-4 scintillation fast neutron detectors.

2. Methods and Results

2.1 Proposed Method via Fast Neutron Measurement

The ultimate goal of this approach is to verify integrity of SNF in dry storage via comparing the computational and experimental results [Fig. 1].

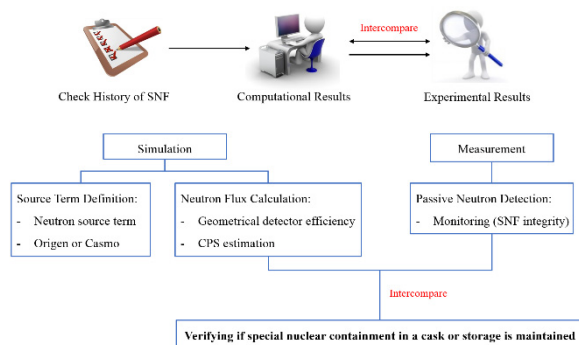


Fig. 1. Logical pathway of the proposed method.

2.2 Neutron Flux Evaluation

In order to evaluate neutron flux on the surface of a dry cask or storage, two separated steps are typically required such as; 1) source term evaluation via ORIGEN or CASMO code and 2) neutron flux calculation via MCNP code.

These two steps are so complicated and time consuming since it is not easy to write inputs, analyze outputs, and transfer data for each code without some specialized knowledge. The present authors have thus developed a new algorithm, coupled ORIGEN-ARP and MCNP, and the graphical user interface (GUI). It was designed for that a user can easily communicate with the GUI [1].

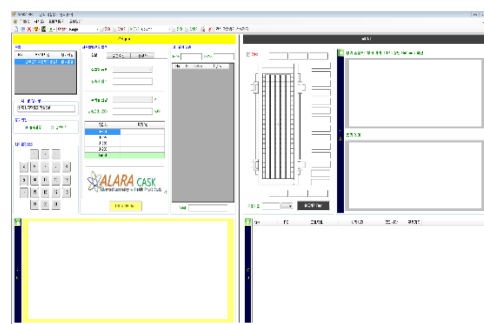


Fig. 2. GUI coupled ORIGEN-ARP and MCNP.

Through this GUI, the estimated neutron source term can be automatically transferred into MCNP inputs. The final outputs can also be graphically presented like Fig. 3.

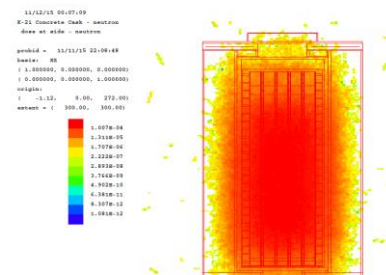


Fig. 3. Graphical result presented by the developed GUI.

Design Information Analysis of TRU Fuel Manufacturing Facility

Min Su KIM*

Korea Institute of Nuclear Nonproliferation and Control, 1534 Yuseong-daero, Yuseong-gu, Daejeon, Rep. of Korea

* kms@kinac.re.kr

1. Introduction

According to the decision of the Atomic Energy Commission of Korea government, Transuranic (TRU) fuel manufacturing facility (TFMF) capable of 1.8 t-HM/yr will be constructed by 2025 together with technology development of nuclear fuel using TRU material recovered from pyro-processing of spent fuel [1]. However, the IAEA safeguards approach for fuel fabrication facility using TRU material is not developed yet. In this paper, to facilitate the development of safeguards approach for TFMF, safeguards design information of TFMF is analyzed and material balance area (MBA) and key measurement points (KMP) are suggested.

2. Safeguards Design Information of TFMF

2.1 General information

TFMF is 1.8 t-HM/yr scale fuel fabrication facility to supply TRU (U-TRU-RE-Zr) metallic fuel to Prototype Generation IV Sodium-cooled Fast Reactor (PGSFR) of 150MWe. Fabrication processes are composed of mainly fuel slug, fuel rod, fuel assembly and scrap recycle process. A simplified flow diagram of TFMF is shown in Figure 1[2].

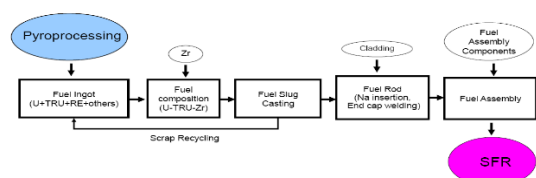


Fig. 1. Flow diagram of TRU fuel manufacturing facility.

2.2 Process descriptions

Fuel slug fabrication consists of induction melting, injection casting, demolding and shearing of fuel slug. Injection casting method is adopted to fabricate TRU metallic fuel slug of PGSFR. Injection casting is

known to be the most cost efficient, capable of mass production and suitable for remote control [3].

Fuel rod fabrication process is composed of hardware production and fuel slug loading part. The fuel rod is fabricated with an upper end cap, a cladding, a lower end cap, and wire [3]. Design parameters of fuel rod are shown in Table 1 [4].

Table 1. Fuel rod design parameters

Fuel rod	Design Para.
OD/thickness of cladding	7.4 mm / 0.5 mm
Wire diameter	0.95 mm
Fuel rod length	2,240 mm
Fuel active length	900 mm
Composition	U-TRU-RE-Zr
Fuel slug diameter	5.54 mm

In fuel rod fabrication process, sodium which has high thermal conductivity is inserted into the jacket and placed between fuel and cladding. Sodium reduces central temperature of fuel rod. Then, fuel rods are moved into the fuel assembly process and assembled with the other fuel components such as duct, nose piece, fuel handling socket, reflector. Fuel assembly consists of 217 fuel rods and weighs 290 kg. the schematic diagram of the fuel rod and assembly is shown in figure 2 [5].

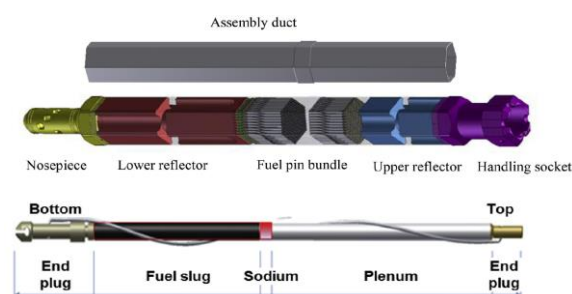


Fig. 2. Schematic of fuel rod and fuel subassembly.

Scrap is generated mostly from fuel slug fabrication process because injection casting method has a drawback of low yield of up to 55% [6]. Therefore, scrap recovery process is essential to TFMF.

2.3 Material descriptions

Nuclear materials which TFMF mainly handles are U and U/TRU metal ingot as feed, fuel slug and rod as intermediate product, and fuel assembly as final product. Recycled U-TRU-Zr can be used as feed material. U oxide material to dilute TRU is not considered and only U and TRU metal ingot recovered from pyro-processing facility is used. TRU material is handled in a shielded hot cell remotely because it contains chemically active lanthanide impurities [3]. Nuclear materials of fuel casing and part of fuel rod process are handled in Ar hot cell. Once welding of fuel rod which prevent leakage of TRU and sodium, is completed then material is moved to air hot cell. Feed and product material are stored in air hot-cell.

Low hold up is expected because nuclear material is not disclosed to outside other than fuel slug process and shearing cutting technique that prevent disperse of dust is used during ingot and fuel slug cutting. However, large amount of waste including nuclear material is expected because 700 quartz molds and 10 crucibles per 1 assembly are used.

3. MBA and KMPs

Determination of MBA and KMPs is next step for development of safeguards approach TFMF. In this section, MBA and KMPs of TFMF are suggested based on the design information.

3.1 Material balance area

Main point for MBA determination is whether more than one MBA is necessary. Process of TFMF is relatively simple compare to oxide fuel fabrication facility. And shipper/receiver difference (SRD) can be provided by identifying a SRD KMP within a MBA. So, single MBA is suggested for TFMF as shown in figure 3. This MBA suggestion is based on the assumption that TFMF and other related facilities such as pyro-process facility and PGSFR are distributed. If TFMF is grouped with other facilities, then separating storage area from single MBA can be considered.

3.2 Key measurement points

KMPs refer to a location where nuclear material appears in such a form that it may be measured to

determine material flow or inventory. 8 inventory KMPs and 5 flow KMPs are suggested as shown in figure 3.

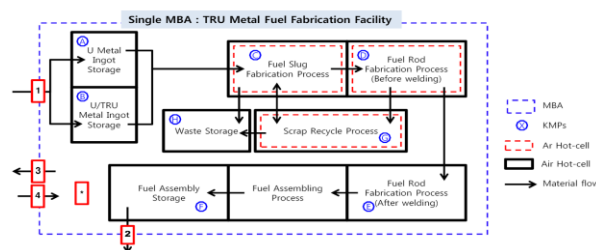


Fig. 3. Suggested MBA and KMPs for TFMF.

4. Conclusion

As construction of TRU fuel manufacturing facilities is planned, safeguards design information of TFMF is analyzed. MBA and KMPs are also suggested based on the design information. Although TFMF is still preliminary conceptual design stage, there is need to identify technical challenges and develop safeguards measures to safeguards TFMF.

Acknowledgement

This work was supported by the National Research Foundation of Korea (NRF), Republic of Korea (Grand No. NRF-2015M2A8A4047118).

REFERENCES

- [1] 6th Atomic Energy Commission, Road map for technology development of future nuclear energy system and demonstration strategy, 2016.
- [2] C. B. Lee, et al., "Status of SFR Metal Fuel Development", International Conference on Fast Reactors and Related Fuel Cycle, March 4-7, 2013, France.
- [3] C. B. Lee, et al., "Metal Fuel Development and Verification for Prototype Generation IV Sodium-Cooled Fast Reactor", Nuclear Engineering and Technology, 48 (2016) 1096-1108.
- [4] J. Y. Park, "Development of SFR fuel fabrication process" KINAC technical seminar, 2017.
- [5] Jaewoon Yoo, et al., "Overall System Description and Safety Characteristics of Prototype Gen IV Sodium Cooled Fast Reactor in Korea" Nuclear Engineering and Technology, 48 (2016) 1059-1070.
- [6] K.H. Kim, et al., "Fabrication and Characterization of U-10wt.%Zr-RE Fuel Slugs Recycling of Metallic Fuel Scraps", 6th Asia-Pacific Symposium on Radiochemistry, September 17-22, 2017, Jeju, Korea.

1분과

핵주기정책 및 핵비확산 (Poster)



Study on Efficiency Improvement of Strategic Item Classification Through Intelligent System

Seung-hyo Yang, Sung-ho Yoon, and DongHoon Shin *

Korea Institute of Nuclear Nonproliferation and Control (KINAC), 1534 Yuseong-daero, Yuseong-gu, Daejeon, Korea

*nucleo@kinac.re.kr

1. Introduction

Korea has strictly adhered to procedures such as Classification and Export License according to the Foreign Trade Act [1] on strategic items. However, there are various problems such as delay in processing and inconsistency of results when a limited reviewer processes Classification to determine whether it is strategic item for thousands of goods and technologies included in a nuclear power plant.

Intelligent System for Strategic Item Classification (ISSIC) applying latest artificial intelligence technology such as expert system, text mining, and deep learning technique has developed in order to process more quickly, minimize deviation of the results and ensure consistency in similar cases.

2. System structure and development results

2.1 Overall system configuration

ISSIC consists of Classification Expert System to systemize the know-how of the Classification reviewer, Similar Case Searching System to search for similar cases using text and images included in the application documents, Open Data Search System to automatically collect open documents on the web and compares the similarity with the application documents to determine whether they are open or not and Export Control Semantic Search System to provide one-stop searching service of various export control related materials, as shown in Fig. 1.



Fig. 1. Main screen.

2.2 Classification Expert System (CES)

Expert system is a system that solves problems similar to experts by systematizing expert knowledge, experience and know-how of experts. In order to develop this system, know-how of Classification review was analyzed through interviews with reviewers who had many years of experience.

As a result of the analysis, the examination of Classification was derived in the following 4 steps. (1) Search for export control items related to application technology (2) Confirmation of exclusion of strategic technology according to relevant laws (3) Review of special examination requirements (4) Analysis of type and key factor of application technology [2]. By implementing the analyzed review procedure using web programming, CES can be easily accessed by a web browser. This system induces the implementation of the same Classification procedure for each reviewer and makes it possible for beginners to review at the level of experienced reviewers.

It is possible to trace the whole classification process by database and to automatically generate the draft of the review report, which can shorten the report writing time by at least 30% with this system.

2.3 Similar Case Searching System (SCSS)

Applicants frequently apply for a Strategic Item Classification of similar documents repeatedly in order to export thousands of documents contained in a nuclear power plant. Maintaining the consistency of the classification results of similar documents is very important for securing the reliability of the classification. Until now, the scope of the search for similar cases was very limited because the application documents can be found based on document file names.

SCSS has developed to search similar documents with texts and images included in application documents as well as document names. SCSS automatically extracts key words related to nuclear and export control from the

application documents and calculates frequency of key words. The similarity of the document is calculated by using the TF-IDF (Term Frequency - Inverse Document Frequency) method, which is often used in recent data mining techniques.

In addition, image similarity comparison function using Hog and Deep Learning has developed to compare similarity of images. Based on the HoG algorithm, black and white images such as design drawings are used to extract specific values of an image and calculate the similarity using the k-NN method [3]. The color-based image extracts a specific value of the image using the deep network-based residual network (ResNet) [4] and calculates the similarity by the k-NN method.

SCSS allows the search for similar documents based on the contents of 16,000 documents submitted during the period, and the search time is about 3 minutes.

2.4 Open Data Search System (ODSS)

The domestic law and the international export control system apply the principle to do not control the technology already open to the public. Therefore, the technology that has already been released is not Strategic Item and no export license is required. It is not easy to confirm whether or not the application document is released except for the documents prepared for the publication of the papers and patents.

ODSS has developed, which collects publicly available nuclear data and compares similarities and confirms whether they are open to the public. The public data collection was implemented based on Google Custom Search. The keywords and site information needed for the search were selected and applied through the interview method with the classification experts.

ODDS shows search results within about 5 minutes among a total of 24,191 document files and 18,883 web-based publications collected as of August 2017.

2.5 Export Control Semantic Search System

There are many types of data that should be examined for Classification such as export control codes, nuclear terminology information, Denial Lists, national nuclear nonproliferation status, and past review reports. Each data should be searched by keyword. Because there are many abbreviations and synonyms, the search time is delayed and the search

range is limited due to iterative search.

First, export control database of export control related domestic and foreign data accumulated over 10 years has been created. In addition, a semantic search function is constructed that automatically analyzes the synonyms related to the input keywords by applying the nuclear export control ontology and searches all the export control related data at once. ODDS is expected to reduce the processing time of classification considerably because it can search all export control related data in real time at once.

3. Conclusion

With the development of ISSIC, processing time of classification has been drastically shortened through real-time similar case search function and the consistency of classification result has been improved through the expert system that systemizes the examination know - how. This will support the timely export of nuclear power plants and improve the credibility of Korea's nuclear nonproliferation through the implementation of thorough strategic export controls

ACKNOWLEDGEMENT

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. 1305014-0517-SB110)

REFERENCES

- [1] The National Assembly of the Republic of Korea, "Foreign Trade Act", Act No. 13838, Jan.27, 2016
- [2] Seung-hyo Yang, Jae-woong Tae, Dong-hoon Shin, "A Study on the Systematization of Classification Process for NSG Trigger List Items", Transaction of Korean Nuclear Society Spring Meeting, May.30-31, 2013.
- [3] Altman, N. S., "An introduction to kernel and nearest-neighbor nonparametric regression". The American Statistician, 46 (3): 175-185, 1992.
- [4] K. He, X. Zhang, S. Ren, J. Sun, "Deep Residual Learning for Image Recognition", arXiv: 1512.03385v1, Dec.10, 2015.

Development of Export License Review System for Strategic Trade Control

Sung-ho Yoon, Seung-hyo Yang, and DongHoon Shin*

Korea Institute of Nuclear Nonproliferation and Control, Yuseongdae-ro 1534, Yuseong-gu, Daejeon, Republic of Korea

*nucleo@kinac.re.kr

1. Introduction

Nuclear items listed on Nuclear Suppliers Group (NSG) Guideline Part 1 are controlled by domestic trade law of each member country [1]. In South Korea, Nuclear Safety and Security Commission (NSSC) is responsible for the strategic trade control of nuclear items. And it is supported by the Korean Institute of Nuclear Nonproliferation and Control (KINAC) which provides technical review and develops policy.

Recently, the volume of strategic technology is increasing rapidly. To review them in limited resources, KINAC developed efficient export license assistant system called the Intelligent Export Control Review System (IXCRS) [2]. This paper describes main features of export license system in IXCRS and direction of future research.

2. Export License

2.1 General information

An export license is issued after verifying item types, quantity, end-users and usage of strategic goods. There are several main review points for strategic trade control.

Table 1. Review list for export license

No	Review points
1	Application documents
2	Item characteristic / purpose
3	Peaceful end use
4	Retransfer possibility
5	Denial country / recipient
6	Reliability of country / recipient

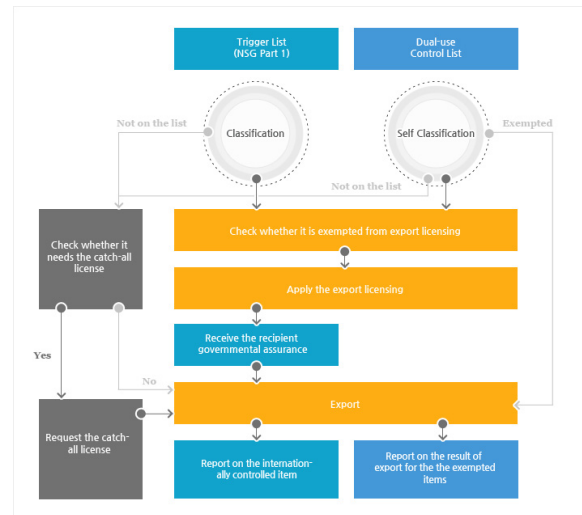


Fig. 1. Export license flow chart.

2.2 Process of export license

To apply export license, applicants should get classification result. If that result is correspond to nuclear strategic item, applicants must get permission before export item. Fig. 1 shows the process of export license.

3. Export License system

This system assists technical reviewers to review efficiently and prohibit mistake. Following shows major steps for export license review in system.

3.1 Step 1: check application documents

In this step, reviewer checks the application documents whether acceptable or not in basis on domestic law.

Application documents information			
	Application documents type	Submit	Remarks
1	Contract, agreement, a deed of contract	☐	
2	Technical particular for strategic item (Annex 5)	☐	
3	Documents for performance and use of item	☐	
4	Import confirmation document for strategic item (Annex 2)	☐	
5	Exporter's pledge (Annex 3)	☐	
6	End user pledge (Annex 2.2)	☒	
7	The others	☒	
Check			

Fig. 2. Screen for application documents information.

3.2 Step 2: check recipient

In this step, reviewer checks the recipient information by searching internet homepage and google map. Usually, Major nuclear related companies can be found in this way. After that, system automatically provides searching result of denial list in international sanctions. An import /export records of recipient are also considered for detecting abnormal trade.

Importer information

Importer	Bob
Address	Broadway 2nd, NewYork
End user	Bob Brother
Address	Broadway 1st, NewYork

Denial list searching

Input

Importer	Bob
Address	Broadway 2nd, NewYork
End user	Bob Brother
Address	Broadway 1st, NewYork

Search

Matching Results

Database Name	Matching Status	Matching Contents	More >>
NSG Denial Notification	Yes	Bob ~~~~	More >>
NSG Complementary Information	No	N/A	More >>
The Denied Persons List	No	N/A	More >>
Debarred Parties List	No	N/A	More >>
The Entity List	No	N/A	More >>
Unverified List	No	N/A	More >>
Specialty Designated Nationals List	No	N/A	More >>
DOD Debarred List	No	N/A	More >>
EU Denial Information	No	N/A	More >>

<< Previous Next >>

Fig. 3. Screen for recipient information.

Previous import record

Related import record for Importer(End user)

Importer	Bob
End user	Bob Brother
Address	Broadway 1st, NewYork

Search

No.	Registration No.	Name	Result	Approval
1	20111203_XX1	Test1	Strategic	Approved
2	20111203_XX2	Test2	Non-Strategic	Uncalled
3	20111203_XX3	Test3	Strategic	Denied
4	20111203_XX4	Test4	Non-Strategic	Uncalled

<< Previous Next >>

Fig. 4. Screen for recipient import/export information.

3.3 Step 3: check importing country

In this step, reviewer checks the importing country information refer to the participating status of sanctions, regimes, treaties, and agreements etc. Using these information, reviewer can save time to search and reduce mistakes.

Import country information

Import country: United States of America (USA)

Country characteristics

NPT		IAEA		Physical protection agreement		
Sign	Statement	Comprehensive Safeguard Agreement	Additional Protocol	Sign	entrust	Statement
1968.07.01	1970.03.05	1980.12.09(Statement)	2009.01.06(Statement)	1980.03.03	1982.12.13	1987.02.06

CTBT	Export Control Regimen				CWC (Statement)	BWC (Ratification)	Agreement with FOC	Nuclear weapon	Remarks
	WA	NSG	MITR	AG					
1996.09.24 (Approved) X	O	O	O	O	O (1997.04.29)	O (1975.03.26)	O (1972.11.24)	O	

<< Previous Next >>

Fig. 5. Screen for importing country information.

4. Conclusion

This study introduced main features of the export license review system. Now, it is on practical use by connected to NEPS (Nuclear Export/Import control System). Compared with before, the time required for review is quite decreased. And information for technical review can be easily acquired which make review easier.

However, there still remain a few things to improve for this system. First, there is no function to update information automatically. It makes system user to update the latest information by manual. Second, lack of accumulated data. To analysis recipient and item information, a lot of background data are needed.

In further research, we will apply a professional trade database to the system to improve verification performance. And using improved system, we will try to find illegal trade evidences.

ACKNOWLEDGEMENT

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. 1305014-0517-SB110)

REFERENCES

- [1] The National Assembly of the Republic of Korea, "Foreign Trade Act", Act No. 13838, Jan.27, 2016
- [2] Dong-hoon Shin, "Applications of Artificial Intelligence in the Export Control Domain", SAI Intelligent Systems Conference 2016, Sep 20-22, 2016, London, UK.

Identification System Concept Design for Illegal Transfer of Strategic Items

DongHoon Shin*, Seung-hyo Yang, and Sung-ho Yoon

Korea Institute of Nuclear Nonproliferation And Control, 1534 Yuseong-daero, Yuseong-Gu, Daejeon, Korea

*nucleo@kinac.re.kr

1. Introduction

Strategic Items means goods and technologies that can be used for the development, production, and use of weapons of mass destruction such as nuclear weapons and biological and chemical weapons, etc. The international community manages these strategic items to maintain international security and peace. In general, the management methods of strategic items consist of controlling the export of strategic items, identifying or confirming that the strategic items are illegally exported or transferred, and also conducting strategic items handlers. Among the above-mentioned methods, the method of identifying or confirming on-site that the strategic items are illegally exported or transferred involves raiding the spot according to the intelligence, or confirming whether it is authorized according to the legitimate procedure in the course of customs clearance. However, police, military or customs officers (usually called field officers) involved in this process often lack knowledge of strategic items and it is not easy to identify the illegal transfer of strategic items. Therefore, this study aimed to develop an identification system that can assist the field officers in identifying the strategic items in the field.

2. Concept & Results

2.1 Analyzing On-site Requirements

The identification system to be developed in this study should implement the following functions, according to the on-site requirement.

First, identification system has the function of confirming whether the items found in the field corresponds to the strategic items, managed by the international regimes or domestic law. That is, this function should provide the function of confirming whether one item is a strategic item by analyzing only the appearance of the specific item when the field officer lacks the ability to identify the strategic items.

Second, a function of confirming whether one item is a

strategic item or not should be implemented, when the items are not opened or officers can see only the documents for items' specifications, this function should provide the function of confirming whether one item is a strategic item, through analyzing the text or numerical information, etc., on the specification documents, not analyzing the items' appearance.

The first and second functions are complementary to each other and should be able to provide the user with an indication that the item to be identified is a strategic item, even though one function is not working properly.

The third function is necessary by the situation of on-the-spot execution, and identification system should provide the identified results to the field officers in real time or very short time. In general, the time required to identify strategic items at customs or PSI (Proliferation Security Initiative) is very short, and if the identification system takes a considerable amount of time to determine what the product is, the value of the use of the identification system will be significantly reduced.

2.2 Investigating the Applicable methods

According to the above-mentioned requirements, this study considered the application of two artificial intelligence techniques.

The technique required to implement the first function is a technique that can measure the similarity between the image of the target item and the images of the strategic items by utilizing only the features of the image such as appearance, texture, color, etc. In general, there are technique such as HOG (Histogram of Oriented Gradient), Shape Context, and so on. Recently, Deep Learning technique is appeared [1, 2].

The technique required for the implementation of the second function is a technique for extracting specific texts or keywords on images related to the specification document or clearance report, searching or comparing those text in the database. In general, these technique are called as like searching, semantic searching or similarity retrieval [3].

The above-mentioned techniques are generally known

in the field of artificial intelligence and can be utilized in the construction of identification system.

2.3 Gathering the information and DB Development

Although the selection of artificial intelligence techniques has been completed and these techniques can be implemented in the development of systems, it is inevitable to construct a database in order for this system to operate properly. In order to implement the necessary functions in this study, first of all, it is necessary to construct a database containing text information related to strategic items and images associated with each text.

So, textual information could be collected from laws and explanatory notes related to strategic items, and specialized books related to nuclear power plants. However, in the case of collecting image information, there is not much information published for the nuclear fields, and even if a lot of images related to the nuclear fields are collected in the open source domain, it is not easy to get exactly the same images related to the strategic items that are actually exported. Therefore, it was necessary to acquire a large amount of images for specific products such as reactor vessels, pressurizers, etc., through companies manufacturing and exporting nuclear related products.

2.4 Results

The conceptual design results are shown in the following figure. As mentioned above, this system is composed of a function of comparing the similarity of images and a function of comparing the similarity of texts. First of all, in order to extract images and text, it is necessary to take a picture of the target item. When a photograph is taken, text information and the image features should be extracted from the photograph. And it is necessary to compare the database with such information. If there is no information retrieved, the users need to enter image related information into the database. Through this procedure, this system can be used to confirm or identify the illegal export and the performance of identification system can be improved gradually.



Fig. 1. Concept of Identification System.

3. Conclusion

As mentioned earlier, in order to ensure that export control of strategic items works properly, it is essential to license strategic items' export based on the law as well as confirm or detect the illegal export. Therefore, the conceptual design of the identification system that can confirm or detect the illegal export of strategic items was carried out to support confirming or detecting activities in this study.

As a result of the conceptual design, the algorithms or techniques required for the identification system were image and text similarity techniques and it is confirmed that an accurate and vast amount of database is necessary for each module to operate properly. At the concept design phase, the algorithms for text similarity were chosen, but the image similarity algorithms have not yet been selected due to image's complexity or uncertainty for each algorithms' suitability to our system. But the result of this study can be applied to the development of the identification system, though there will be some modifications. And if the identification system is developed and operated properly based on our concept design, it can contribute to strengthening export control system in ROK.

ACKNOWLEDGEMENT

This work was supported by the Nuclear Safety Research Program, granted financial resource from the Nuclear Safety and Security Commission (NSSC), Republic of Korea. (No. 1305014-0517-SB110)

REFERENCES

- [1] S. J. Pan, and Q. Yang, A survey on transfer learning, IEEE Transactions on Knowledge and Data Engineering, 22(10), 2010.
- [2] Phillip Ian Wilson et al, FACIAL FEATURE DETECTION USING HAAR CLASSIFIERS, JCSC 2006.
- [3] Sholom M. Weiss, Text Mining: Predictive Methods for analyzing Unstructured Information, 2005

Development of Virtual Training Software for Bulk Handling Facilities Based on the IAEA Training Course

Chul HEO*

Korea Institute of Nuclear Non-proliferation and Control, 1534 Yusungdae-ro, Yusung-Gu, Daejeon, Korea

*Corresponding author:hc0228@kinac.re.kr

1. Introduction

ROK has been supporting and contributing total 17 tasks to IAEA in-cash and in-kind. These tasks consist of training, safeguards approaches, Analysis Support and NWAL Coordination, information analysis, Safeguards Information Systems and System Usability. Among them, ROK is focusing on development of virtual training for bulk handling facilities. It would be expected that ROK will continuously support virtual training to IAEA through the ROK-SP[1].

IAEA has difficulties in the inspection training due to the problem of finding the training place, and therefore, has a high interest in the training software module development. Especially, the nuclear fuel processing plant training process is complex and the inspection has to be carried out using different verification equipment for each process, so almost no software development was not carried out[2].

Accordingly, ROK agreed with IAEA on then nuclear fuel processing plant software development and started the development. Especially, IAEA provided a book introducing the internally developed process and inspection process for the nuclear fuel processing plant, and based on this, the training software was developed.

Also, the developer directly participated in the training course and completed the training for the part on the inspection process and training for the effort to develop the software closer to the viewpoint of IAEA.

Based on this, the nuclear fuel processing plant's training zones were divided for each process, and the software was developed based on the core measurement zone for each process.

2. Development of IAEA Inspector Training Software

2.1 The 1st Stage Virtual Training Software in 2012

The first SW simulated the BHF with a 3D perspective model similar to the real one. Inside the facility, ROK were able to move freely to first-person viewpoints, and ROK helped the user to understand by attaching a process diagram.

In addition, the main equipment and the IAEA inspection equipments are expressed in a similar

manner, and the main functions and usage methods are explained.



Fig. 1. The 1st Stage Virtual Training SW.

2.2 The 2nd Stage Virtual Training Software

With the 1st stage completion in Feb. 2012 of the training course software development for the nuclear fuel processing plant, which was newly signed in 2011, ROK proposed a discussion with IAEA to have the discussion meeting with IAEA, and IAEA mentioned as follows for the additional matters and future implementation plan.

IAEA asked if IAEA can add new verification equipment to the verification equipment developed in the current software, and ROK explained that it was possible if ROK had detail pictures and operation manual. Also, IAEA asked to insert 'guided tour' part in the software so that the beginner can look around the overall process at the first running of the software, and ROK answered that it can be added after the development of the simplified model.

IAEA requested to re-configure the software after the development with the system that the trainees can utilize the software after completing the training, and for this, IAEA requested the following additional functions.

- Test module with un-matching facility inventory list and quantity (insert number, figure insertion, etc.) having the trainees find them
- Module with a mission at the beginning and have the trainees complete the mission (Module directing to directly go to a certain process and perform certain activity to understand the process in the procedure of completing the given mission. Also add the description of receiving help from the person in charge of the facility in detail)
- If IAEA provides diversion and misuse scenario, it will be added to the software training evaluation module (hidden facility, etc.)

In the second SW, ROK reflected some of the IAEA requests, and made the following improvements.

- The user can grasp the facility at a glance, thereby improving understanding of the structure. It was made in perspective view so that both outside and inside can be confirmed at the same time.
- Process equipments and inspection equipments are also arranged in a three-dimensional form, so as to confirm equipments in actual facilities.
- In addition, in order to improve the understanding of each process, the process flow diagram and the process equipment are connected to implement the function of activating the equipment in different colors when selecting the equipment in the process flow chart.
- The documents used in the IAEA inspections are also presented through examples to enhance the understanding of the users.

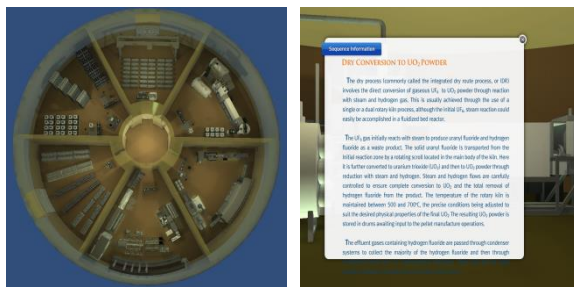


Fig. 2. The 2nd Stage Virtual Training SW.

2.3 The 3rd Stage Virtual Training Software

ROK finished 3rd stage reflecting the result of discussion with IAEA as follows.

- A fly-through tour describing the facility, with animation of the key equipment (process oriented and focused on the nuclear material fuel flow without safeguards)
- A tour of the facility or an interactive walk through where the user can freely explore the facility and learn about safeguards activities or equipment (more than an on-line manual)
- Interactive challenge or simulation that allows the IAEA to “test” the student and incorporate learning objectives in the virtual environment; such that after “playing”, the user is familiar with the key elements because he or she has had to challenge him/herself.

In addition, Some functions are added such as IAEA HW compatibility, Logo modification, Contents, 3D reality improvement, Training & Evaluation mode function improvement, etc at 3rd stage.



Fig. 3. The 3rd Stage Virtual Training SW.

3. Conclusions

ROK sent our training module to IAEA and held the discussion meeting to review additional modifications and future application plan. IAEA highly evaluated this software module and decided to apply to the IAEA inspector training course in the future. Especially, to consider the user convenience, IAEA inspectors tested the demo version and received opinions. ROK promised IAEA to continuously enhance the convenience and add the parts that the lecturer can utilize. In addition, the following improvements will be made in future upgrading work.

- Optimization to increase versatility
- Produced in realistic shape
- User interface improvements with minimal manipulation

REFERENCES

- [1] ROK SP Handbook, ROK, 2013.
- [2] Fuel Fabrication (IAEA Training Manual on the Nuclear Fuel Cycle), IAEA, 2011.

The Application of ICP (Inner Compliance Program) at KAERI

I.C. Kim*, S.H. Lee, B.D. Lee, H.J. Kim, H.S. Kim, and J.A. Jung

Korea Atomic Energy Research Institute, 989-11 Daeduk-Daero, Yousung-Gu, Daejeon, Korea

*ickim@kaeri.re.kr

1. Introduction

To effectively and efficiently control an ITT (Intangible Technologies Transfer), the Korean government encourages the R&D institute and universities obtaining the ICP from the relevant authority, MOTIE. This means that the exporters can control the ITT by themselves, because the exporters know very well the counterparts of the trading and the exporting items and technologies. According to the revised relevant laws and regulations in 2014, the Korean government can issue a license of ICP to R&D institutes and universities. KAERI is an R&D institute sponsored by the Korean government whose main field of R&D is nuclear energy. As of now, most of the export licenses at KAERI have been related to nuclear items and technologies controlled by the NSG part 1. However, KAERI needs to control dual use items such as common materials and equipment, or other fields for export controls of ITT. KAERI has considered introducing an ICP and is cooperating with the relevant authority. At the beginning of the year (2017), KAERI applied for an ICP from the relevant authority, MOTIE. It is very hard for KAERI to prepare all documents, because there is no R&D institute to obtain the ICP except KAIST. KAIST is the first university to obtain ICP from MOTIE. At the time, MOTIE nominated KAIST as the first organization to obtain the ICP with level II and supported KAIST to give a license for ICP. However, KAERI is different from KAIST. KAERI decided to set up an ICP intentionally and is cooperating with the relevant authority. In association with the foreign trade act and decree, there are three types of ICP ("A, AA, and AAA"). To control the technology invented by R&D institutes, the relevant authority can give an AA license for R&D institutes. As of now, there are no R&D institutes to obtain the ICP. However, KAERI is trying to obtain a license from the authority, MOTIE. In an effort to do so, KAERI completed enacting a new self-regulation in 2016, and proceeded to apply for a license of ICP in 2017. The relevant ministry, MOTIE, and KOSTI has deeply cooperated with KAERI to issue an ICP license as the first R&D institute.

2. Requirements to obtain ICP

To establish the ICP, the organization should set up an oversight office, training and education, auditing and monitoring, internal regulations, the procedures of export control, and the CEO's will as well as individual punishments for non-compliance. Based on the applicant's (international trade company or R&D institute/university) characteristics and will, the Korean government issues a license of the ICP. According to the organization's operation approaches and procedures for the exporting items and technologies, there might exist differences in each organization. Even in the case of R&D institutes, there are no exemptions to the export controls of strategic items and technologies. Thus, they need to set up all of the abilities such as self-classification and the catch-all confirmation on the importer. Table 1 shows all requirements to obtain the license of ICP for level 2, 'AA'.

Table 1. Requirements for ICP

Evaluation Indicator	Contents
Organization	Responsibility and organization structure
Regulation	Regulation to control and operate the strategic items
CEO's will	CEO's will to control on the export control
Procedure of classification	Procedures for advance classification and certificate
Training & Education	Education Plan and Implementation related export control
Auditing	Auditing related the plan and implementation records
Documentation	Documentation related export control
Report & punishment	Reporting and punishment related individual

3. Reviews on the application for ICP

At the end of 2016, KOSTI visited KAERI to discuss the best way to obtain the ICP for dual-use items from MOTIE. KOSTI recommended KAERI to apply for the license of ICP for level 2, 'AA'. According to the revised laws and regulations, the R&D institute and university can obtain this level of ICP. In comparison with the trading companies, the license of ICP for level 2, 'AA,' is used for the R&D institute and university. Generally, this organization may export the strategic technologies more often than the strategic items. KAERI has studied the ways to obtain the ICP, and enacted a inner regulation for strategic items and technologies including nuclear materials in 2016. KAERI already set up a Web-based system to manage export controls. However, KAERI decided to develop the current Web-based system KAERI uses, because it does not seem to catch up with the current environment on export controls. To obtain the level 2 ICP, KAERI should finish all requirements shown in table 1. KAERI applied for the license of ICP for level 2, 'AA' on Feb. 2017, and KOSTI made the reviews of that application shown in table 2. Here are the reviews on the application submitted by KAERI.

Table 2. Reviews on the ICP application

Evaluation Indicator	Comments
Organiza-tion and regulation	The suggestion of a clear and concrete organization chart to show the exclusive department for export controls A certificate of education for an exclusive staff on export controls should be given
Regulation	The revision system of the inner regulation should be given
CEO's will	The suggestion of the implementation announcement published by the president
Procedure of classification	The suggestion of a specific procedure for self-notification The DB system to store and manage the results of licensing and exemption should be established
Training & Education	The implementation plan of the education & training should be set
Auditing	The audit should be implemented in every 2 years
Documen-tation	KOSTI confirm the status in the field review
Report & punishment	KOSTI confirm the status in the field review

To establish the ICP for dual-use items controlled by MOTIE, KAERI should supplement some essential requirements. Among them, the most important thing is to set up the procedure of self-classification before obtaining the export license from the authority. KAERI has many workers who can do the self-classification. In fact, they have already worked for the authorities needing counseling from a special expert to give a license for export controls. KAERI is considering building an inner committee to solve this requirement. The best way to set the procedure of self-classification is to make a committee for export control. As aforementioned, KAERI has excellent experts owning the ability for self-classification. However, building a committee is problematic. KAERI cannot pay counseling allowances to the committee members due to the relevant laws and regulations. Other supplements except this problem can be solved sufficiently.

4. Conclusion

To effectively and efficiently control the ITT, KAERI is trying to obtain the license of ICP for level 2, 'AA' from the authority, MOTIE. Once KAERI obtains this license, it can control dual use items and technology except nuclear items based on NSG part 1 by itself. KAERI completed enacting a self-regulation in 2016. and is proceeding to obtain a license of ICP in 2017. This paper shows the results of the review on KAERI's application. However, there seem to be some complements required by the relevant authority. Even though obtaining the ICP for level 2 'AA' is not easy, after complementing the requirements, KAERI is going to obtain this license by means of a deep cooperation with MOTIE and KOSTI.

5. Reference

- [1] IC.Kim 57th INMM Annual Meeting "The introduction of ICP for KAERI export controls" 2016.7
- [2] KOSTI Report "The review report on KAERI's application for ICP" 2017. 3

Essential Factors of Decommissioning Strategies for Korean Nuclear Industry

Hyejin Jung, Taesik Yun, Younggook Kim*, and Kwangho Jo
Decommissioning Technology Team, Central Research Institute, KHNP
70, Yuseong-daero 1312beon-gil, Yuseong-gu, Daejeon, 34101, Korea
*Corresponding author: kim.090909@khnp.co.kr

1. INTRODUCTION

Kori-1 unit came into shutdown on June 18 2017 which is the first commercial nuclear power plant being decommissioned in Korea in accordance with the Korean government's decision in June 2015. Kori-1 has generated 576 MWe electricity capacities since 1978, plus a refurbishment for 10-year extension. This article is prepared to factor out decommissioning strategies mostly appropriate to the decommissioning Kori-1 nuclear power plant.

Terms used to delineate the lifetime of an authorized facility and of the associated licensing process consists of six core stages such as siting, design, construction, commissioning, operation and decommissioning. The term decommissioning implies the administrative and technical actions taken to allow the removal of some or all of the regulatory controls from a facility except for the part of a disposal facility in which the radioactive waste is emplaced [1]. Whole range of each process of decommissioning should be considered throughout the other five stages. The decommissioning process is typically composed of its planning, conducting actions and terminating the authorization.

2. GLOBAL DECOMMISSIONING INDUSTRY

The prospects of the global decommissioning industry could be demonstrated in the Figure 1 assuming the time for immediate and deferred dismantling is 10 and 50 years respectively. At the stage of the introduction, from 1960s to 1980s, the industry was led by the USA whose scale of the market is \$353 million on the yearly basis reached to the accumulation of \$8,109 million. The growth stage after the Chernobyl accident whose scale of the market is \$1,219 million on the yearly basis reached to the accumulation of \$30,485 million. In the matured stage after the Fukushima accident, the scale of the market is \$4,016 million on the yearly basis reached to the accumulation of \$201,186 million.

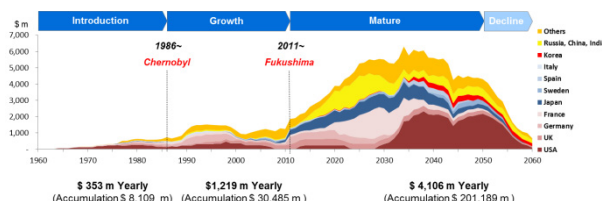


Fig. 1. Global Decommissioning Industry.

It is expected the decommissioning industry would be into the stage of decline after 2020 in nations such as UK, France,

and Italy. Korea, however, is supposed to be into the growth stage from 2030s in the case of one time extension, or 2050s in the case of twice extensions each.

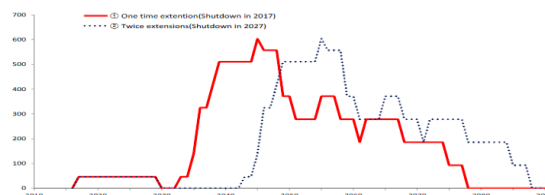


Fig. 2. Scale of Decommissioning in Korea.

Since new government declared the nuclear energy free strategy, more efforts of not only physical and financial source should be prepared from the early stage in a scrutinized manner.

3. DECOMMISSIONING STRATEGY

In order to accomplish successful decommissioning, strategies for decommissioning adopted should be the essential element. Normally, immediate dismantling and deferred dismantling are suggested as the principle consideration for the decommissioning applicable for all nuclear related facilities [1, 2].

Immediate dismantling could be apt to the case that decommissioning actions begin shortly after the permanent shutdown. Equipment and structures, systems and components (SSCs) of a facility containing radioactive material are removed and/or decontaminated to a level that permits the facility to be released from regulatory control for unrestricted /restricted use on its future use. On the other hand, deferred dismantling is an alternative to apply to the case that after removal of the nuclear fuel from the facility, all or part of a facility containing radioactive material is either processed or placed in a safe storage and the facility maintained until it is subsequently decontaminated and/or dismantled. A combination of these two strategies may be considered practicable under the circumstance not to violate safety and environmental requirements, technical availability, local conditions. Entombment is not an option in the case of planned permanent shutdown. It might be considered only under exceptional circumstances such as the Fukushima event [3].

4. IMPACT FACTORS ON STRATEGY [2]

4.1 Legislative and regulatory requirements

Each countries having nuclear power plants under construction or in operation or in consideration of

decommissioning has different regulations governing decommissioning strategies. For instance, the US Nuclear Regulatory Commission (NRC) limits the safe enclosure period up to 60 years, UK, however, allows dismantling to be delayed for Magnox reactors and for them to be kept in a safe enclosure mode for more than 100 years. Such a long period allows levels of radioactive decay so that workers could work on a Magnox reactor without limitation, and also allows the accumulation of decommissioning funds. However, this policy is being reconsidered.

From the perspective of responsibility, a facility's operator has the primary responsibility for all technical and financial measures except for the disposal of radioactive waste which is under the supervision of national regulatory bodies or national agencies.

4.2 National waste management strategies

The most crucial regulatory requirements are related to clearance criteria. International recommendations for exemption and clearance have been issued by IAEA and European commission. These specify radiological concentrations below which material can be considered to be non-radioactive and released from regulatory control. Such protocols are now established in many countries such as Germany, UK, USA, South Korea etc.

4.3 Future use of site

An alternative of decommissioning strategy would rely on the planned future use of the site. In case of lacking of sites for new plant construction, the owner may choose to reuse a site for a new plant. In this case, immediate dismantling may be the good choice. If the plant to be decommissioned is in the multi-plant site, safe enclosure may be the preferred choice. In this case, the necessary security, surveillance and maintenance for the shutdown facility could be provided by the remaining operating facilities. The examples of reuse of the sites decommissioned are such that the turbine building of a decommissioned plant was reused for a fossil fired plant (Fort, St. Vrain, USA), the chinon-1 nuclear power plant in France was converted into a museum, Part of the Greifswald nuclear power plant in Germany is being converted into a biodiesel production facility.

4.4 Radiological factors

While the remaining residual radioactive will present smaller risks compared to that of an operating reactor, it should be taken good care for the workers, the public and the environment during decommissioning. Currently technological breakthroughs in electronics, robotics and remote handling have considerably reduced the hands-on works to highly contaminated area. Hence, it has set aside the importance of radiological factors in selecting a decommissioning strategy.

4.5 Availability of technology and other resources

In general, decommissioning technology is more available in countries with much experience. Such countries have both expertise and experience related to their nuclear technologies

and resources In this sense, technology setting is also one of the important factor to perform a decommissioning.

4.6 Stakeholder considerations

Due to widespread heightened public sensitivity to environmental protection, any waste management or decommissioning decision will typically require thorough public examination and the involvement of many stakeholders. The diversity of relevant social, political, economic, and cultural environments makes it difficult to develop universally applicable guidance. However, we have to find in the experience of other good practices that can be adapted to our own project.

4.7 Decommissioning cost and funding

Whatever choice and decisions are made, it is the responsibility of the plant owner to make financial provisions sufficient to cover the cost of all stages of decommissioning in accordance with pertinent national legislation and funding requirements. Korea has established legal provisions to decommission Kori-1 and accumulate appropriate liability for the reaming facilities.

4.8 Knowledge management

The decommissioning of a nuclear facility should be considered from the earliest stage of its life cycle, and emphasis should be given to eh acquisition and maintenance of all relevant records.

5. CONCLUSION

Technically, decommissioning is a mature industry; many steps and processes are similar to maintenance, storage, and transport procedures experienced during the operation. However, to minimize hazardous and radioactive materials as much as possible produced in process of decommissioning should be the focused issue for the protection of workers, the public and the environment. In order to achieve the successful decommissioning, the impact factor on the strategy should be analyzed and evaluated to optimally apply to Kori-1 project. From my perspective, among eight factor, stakeholder's consideration and spent fuel management are considered the key elements we have to concentrate on to smoothly go ahead for successful decommissioning of Kori-1.

REFERENCES

- [1] ASME(2004), "The decommissioning handbook".
- [2] IAEA(2005), " Selection of decommissioning strategies: Issues and factors, IAEA-TECDOC-1478.
- [3] Daniel et. al (1986), "Guideline for producing commercial nuclear power plant decommissioning cost estimates", AIF/NESP-036.

2분과

사용후핵연료관리 (Oral)



Reaction Rate of Porous UO₂ Pellets for Electrolytic Reduction

Min Ku Jeon*, Eun-Young Choi, Sung-Wook Kim, Sang-Kwon Lee, and Jeong Lee

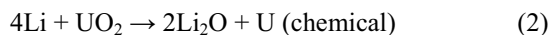
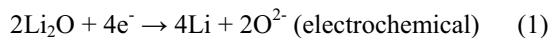
Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, 34059 Daejeon, Republic of Korea

*minku@kaeri.re.kr

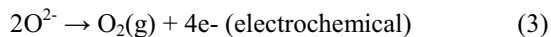
1. Introduction

In order to adopt the previously established metal fuel based pyroprocessing in PWR (Pressurized Water Reactor) oxide fuels, electrolytic oxide reduction (OR) step is essential. The OR process under intensive research at KAERI employs LiCl salt with 1 ~ 3 wt% of Li₂O. It is widely accepted that Li₂O is electrochemically decomposed to produce metallic Li and oxygen gas and then Li reacts with UNF (Used Nuclear Fuel) oxides to produce Li₂O as following equations.

Cathode:



Anode:



This reaction scheme is very attractive because the amount of Li₂O is kept constant until the reactant (UNF) is depleted. However, unlike direct electrochemical reaction, actual reduction rate is determined by the chemical reaction (2), which is not detectable in-situ. In the present study, the reduction reaction rate of porous UO₂ pellets was quantitatively investigated.

2. Experimental

Each experiment was performed with 9 porous UO₂ pellets (average 15.7 g) at 650°C using LiCl salt containing 1.4 ~ 1.5 wt% of Li₂O. The pellets had

cylindrical shape with 7.65 mm (H), 5.74 mm (D), and 8.79 g/cm³ (density). Reaction rate was analyzed by producing samples which undergone various number of charge cycles, where one cycle is composed of 30 minutes of 3.2 V cell potential and 10 minutes of power-off. Largest amount of charge applied was 45,000 C. Degree of reduction was determined by analyzing center-cut image of the samples. The concentration of Li-Li₂O was determined by the HCl titration technique.

3. Results and Discussion

In order for quantitative comparison of the reaction rate, the term ‘degree of reduction’ was defined as following:

$$\alpha = 1 - \left(\frac{r}{r_0}\right)^2 \quad (4)$$

where r_0 is the radius of a whole pellet and r is the radius of remaining oxide part. Measured values for various Q (applied charge) are shown in Fig. 1.

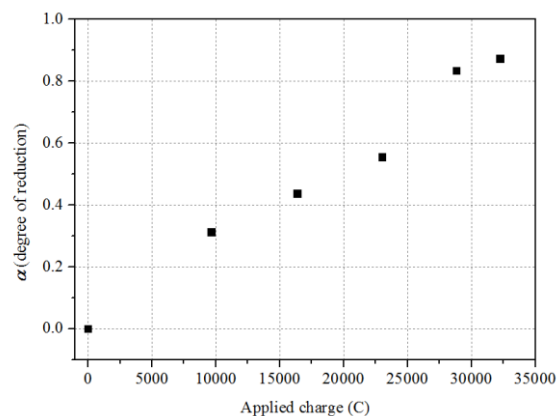


Fig. 1. Degree of reduction as a function of applied charge.

The Sharp-Hancock plot was derived using the data of Fig. 1 to identify the most suitable geometry function [1]. The Sharp-Hancock plotting revealed a slope of 0.88 in the region of $0 \leq \alpha \leq 0.56$. This result suggested reaction models of 1) first-order (slope = 1.00), 2) contracting area (slope = 1.07), and 3) 1-D diffusion (slope = 0.62).

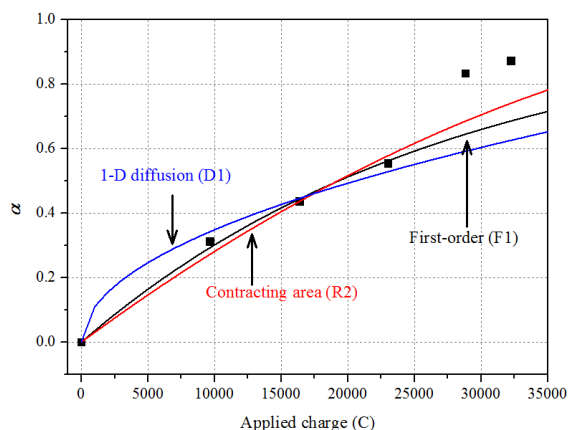


Fig. 2. Fitting results for three geometry functions.

Fitting results for the suggested reaction models are shown in Fig. 2, where the first-order reaction model was identified as the best geometry function to describe the reduction rate of porous UO_2 pellets.

The reduction rate equation of porous UO_2 pellets could be described as:

$$\alpha = 1 - e^{-(3.59 \times 10^{-5} \times Q)} \quad (0 \leq \alpha \leq 0.56) \quad (5)$$

where Q stands for the amount of applied charge (in C) [2].

4. Conclusion

The reduction rate equation for porous UO_2 pellets was derived using repeated experimental data, which revealed that the first-order reaction model is the geometry function to describe this system.

REFERENCES

- [1] J.D. Hancock and J.H. Sharp, "Method of comparing solid-state kinetic data and its application to the decomposition of kaolinite, brucite, and BaCO_3 ", *Journal of the American Ceramic Society*, 55, 74-77 (1972).
- [2] M.K. Jeon, E.-Y. Choi, S.-W. Kim, S.-K. Lee, H.W. Kang, S.S. Hong, J. Lee, J.-M. Hur, S.-C. Jeon, J.H. Lee, Y.-Z. Cho, and D.-H. Ahn, "Electrolytic reduction rate of porous UO_2 pellets", *Korean Journal of Chemical Engineering*, 33(7), 2235-2239 (2016).

Analysis of Material Behaviors of Integrated Operation Model of Pyroprocessing and the Phased Approach Strategy of Optimized Mass Flow

Hun Suk Im^{*}, Hong Jang, Hyo Jik Lee, Ju Ho Lee, Min Ku Jeon, Chang Hwa Lee, Gha-Young Kim,
and Hwan-Seo Park

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

*ih95@karei.re.kr

1. Introduction

Pyroprocessing technology has been developed at Korea Atomic Energy Research Institute (KAERI) as an alternative for management of spent nuclear fuel [1-2]. Each fundamental unit process of pyroprocessing has been developed successfully up to the lab-scale so far [3-4], while integrated process for continuous operations from spent nuclear fuel to waste product has not been achieved completely. Modeling and simulation studies for integrated pyroprocessing have been challenged up to now[5]. However they had limitation in that they had focused into material flow of their own pyroprocessing technique. As the result the integrative studies including mechanical operation condition, safety analysis, and safeguards for nuclear materials has remained at the basic state. Thus the first integrated operation model for pyroprocessing(IOMP) needed to be developed, reflecting various needs of safety, safeguards and mechanical operation conditions. This model implements the current flowsheet of pyroprocessing and covers the up-to-date process operation schedules.

2. IOMP Purpose

The objective of IOMP is that it is supplied on the process designers as an analysis tool for each capacity of unit process, the bottle-neck points, and the distribution of nuclear spent fuel stream for integrated operation and mass flow of pyroprocessing. IOMP has functions for tracking the target nuclear element over the whole integrated process in real time. Hence it is possible to analyze efficiency of the current flowsheet, the distribution of main nuclear materials, and the total operation days. The most important is it will give perception of phased approach strategy for establishment of the integrated

pyroprocessing technology. It is possible to change in-cell structure so that the influence of logistics by 2D-design can be determined.

3. Methods

3.1 System Modeling

This program has been invented with ExtendSim. It consists of model file(.mox), library file(.lib) and EXCEL import file for user-defined conditions. The integrated pyroprocessing consists of spent fuel storage vault, head end cell, oxide reduction cell, electrolytic recovery cell, and waste treatment cell. Above all conceptual diagram of each unit cell was constructed [Fig. 1].

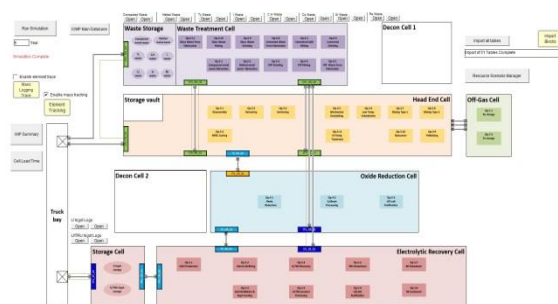


Fig. 1. Conceptual diagram of IOMP.

The next, various operation conditions of each unit process were reflected into the process diagram. There are reference batch size, operation time, the number of operators, user-defined separation ratios, the number of equipment, frequency of breakdown and setup for individual process, mean time between failure, time for inter-cell transfer lock according to hot cell configuration, and the number of utility like robot crane.

3.2 Mass tracking system.

Mass tracking of nuclear materials is very important for the dynamic mass balance and nuclear material accountancy. This program is implemented to trace main target nuclear materials mass in all paths through which the material passes. It counts the mass whenever the material items departure from the start point and arrive at the end point. Hence it gives the insight for distribution of main nuclear materials over the whole integrated processes.

3.3 Process capability analysis.

IOMP has been implemented for analysis of process capability, including process lead time, operation utilizations by standalone state and fully-linked state, total number of batch, and buffer capacity. It is also including the procedure for nuclear material accountancy operation because even though the influence of accountancy operation on the processing ability is small, the increase of the lay-off time for the analysis of the nuclear material samples can be observed.

4. Result and Conclusion

Salt-soluble element can remain in the oxide reduction molten salt. The representative elements as Cesium (Cs), Strontium (Sr) and Barium (Ba) are accumulated in LiCl molten salt bath. These species concentration is very important since it can affect process operation procedures or the quality of waste product due to high heat elements. Fig. 2 shows amount of Sr, Ba in oxide reduction molten salt bath during long-term process time. At the first year the concentration were not saturated. However they were fully-saturated from around 400 batches and reached to the equilibrium state.

Fig. 2 also means that the optimal mass balance and flow should be conducted after the integrated process reaches to the equilibrium state. The current designed integrated operation model is not satisfied to the target operation capacity since it is based on the assumptions of each unit process and does not reflect the interaction between pre- and post-process. It means that various parameters and process scenario should be considered and optimized. The phase approach strategies have to be proposed for solving complex equations for optimization of process operation and material flow through stepwise computer simulation

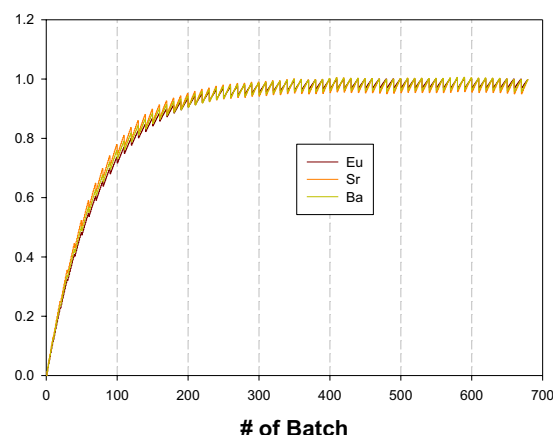


Fig. 2. Behaviors of salt-soluble element in the oxide reduction.

Several scenario of operation model based on the separation ratio, process capacity, WIP capacity and process line balancing can be simulated as well. It will play a significant role for the analysis of process operation, logistics. This work was supported by National Research Foundation of Korea (NRF) (NRF-2017M2A8A5015083).

REFERENCES

- [1] H. S. Lee et al, "Pyroprocessing technology development at KAERI," Nuclear Engineering and Technology, 43, 4, 3171 (2011).
- [2] D. H. Ahn et al, "Technical development of Pyroprocessing for managing spent fuel," News & Information for chemical engineers, 34, 3, 249 (2016).
- [3] J. M. Hur et al, "Electrochemical reduction of uranium oxide in the Li₂O-LiCl molten salt," Seventh information exchange meeting on actinide and fission products partitioning and transmutation, Journal Name, Vol. , p. , Publisher, City, Country (year).
- [4] H. S. Lee et al, "Current status of Pyroprocessing development at KAERI," Science and Technology of Nuclear Installations, 2013, issue numbers, page number (year).
- [5] F. M. LAST, The Title of the Book, page or chapter numbers, Publisher, City (year).

Needs and Goals of Pyroprocessing Automation Verification Mockup

Dongseok Ryu*, Jongkwang Lee, Sungwook Kim, SungJai Lee, Jin-Mok Hur, and Hyojik Lee

Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, 34059 Daejeon, Republic of Korea

*sayryu@kaeri.re.kr

1. Introduction

KAERI (Korea Atomic Energy Research Institute) have extensively studied on the pyroprocessing, and several facilities have been prepared for the relevant experiment, such as PRIDE and ACPF [1, 2]. The facilities mainly focuses on the radioactive experiments. However, troubles comes not from the radioactive treatment, but from the mechanical reliability due to high temperatures or molten salt. The facilities shows limitations and inconveniences to simple and quick test of the equipment design for the mechanical reliabilities. This research propose a new experimental space, PAVM (Pyroprocessing Automation Verification Mockup), dedicated to evaluate the mechanical feasibility of equipment including automation function.

2. Pyroprocessing automation verification mockup

2.1 Needs and goal

The pyroprocessing is composed of dozens of sub-processes, and many pieces of equipment for the sub processes are closely interacted each other. To evaluate each piece of the equipment, argon or dry air environment are required, because the equipment utilizes the molten salt, which is extremely corrosive when it binds to oxygen. A glove box is enough for the small size equipment. A large processing room or cell should be prepared to experiment on the large

size equipment. If radioactive material is investigated in the equipment, the way to shield should be additionally considered. KAERI prepares all spectrums of experimental setups for pyroprocessing, as listed in table 1.

Table 1. Features of facilities in KAERI

No	PRIDE	ACPF	Glove boxes
Radioactive area	O (simfuel)	O (spent fuel)	O
In/export	difficult (argon)	strictly managed	convenient
Removing used equipment	difficult (large size)	prohibited (high level waste)	not easy
Access and operation	difficult (argon)	difficult (radiation)	convenient
Size of equipment	large	small	small

In the existing facilities, many trials and errors were happened during operation, and one important lesson was learned; the most suspending of experiment comes usually from the mechanical failures, not from a processing failure or the radioactive trouble. This means the mechanical reliability of empirical equipment, such as high temperature endurance, corrosion resistance against molten salt, remote operability and automation function, has highest priority in the equipment design. The mechanical reliability should be prepared in prior to the radioactive experiment, and a specialized experimental space are essential to dedicated for the mechanical issue.

To meet the raised new needs, PAVM is proposed to accomplish two goals, as below.

- ◆ Evaluation of mechanical reliability
 - Exclude radioactive test
 - Dedicate to equipment behavior in high temperature, molten salt, high electricity
- ◆ Evaluation of automation function
 - Automated loading/unloading crucibles, automated basket transportation, quick replacing modules and filter cartridges, etc.

2.2 Preliminary requirements

Pyroprocessing is actually done by series of equipment. If small empirical space for two pieces of equipment are available, and if one section from the series process were proved with the previous and the following equipment, and if the section results were successively extended, then the full process can finally be evaluated from the small empirical space.

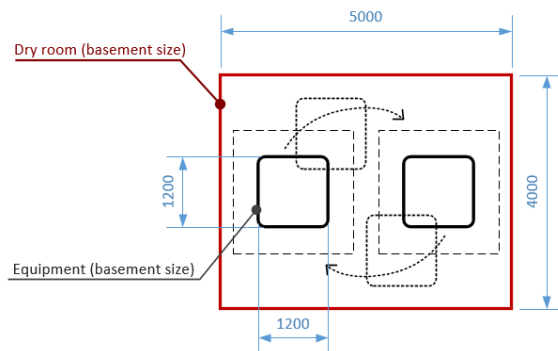


Fig. 1. Sample Figure.

Therefore, the PAVM was sized enough to accommodate two pieces of equipment, as shown in Fig. 1. The basic dimension (1200 mm x 1200 mm x 1500 mm) of equipment was determined in the consideration of 1.5 kg uranium reduction.

The very basic requirements for PAVM were listed below.

- ◆ Dry room
 - Enough space for two pieces of equipment
 - Dry air regulation with dew point -40°C

- Convenient access of human operator
- Air tight, large window, etc.
- ◆ Remote-handling automation system
 - Gantry robot for basket transportation
 - Modular design of sensor and actuator

Fig. 2 shows the conceptual drawing.

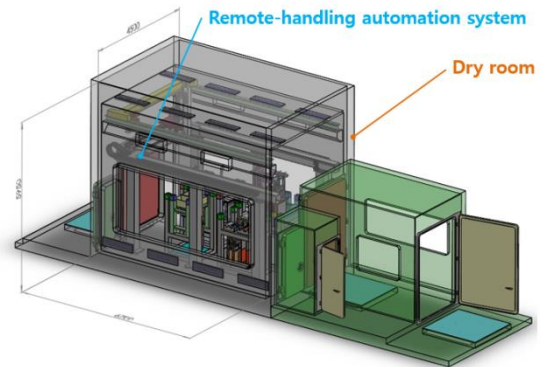


Fig. 2. Conceptual drawing of the PAVM.

3. Conclusion

The pyroprocessing automation verification mockup was proposed to provide experimental space dedicated to the mechanical feasibility of equipment, including automation function. Background of the PAVM was described, and the needs and goals were defined. The basic requirement of the PAVM were shown in this research.

REFERENCES

- [1] Dongseok Ryu, et al. "Development of a Virtual 3D Simulator to Evaluate Remote Operations in PRIDE," Proc. of ANS 2016 Tropical meeting D&RS, pp. 3-6 (2016).
- [2] Dongseok Ryu, et al. "Virtual Verification using 3D Simulator for a Remote Operation Procedures in PRIDE," Proc. of 2016 IPRC, p. 66 (2016)

Modeling of the Thermal Conductivity of Mo-ZrO₂ Composites for Inert Matrix Fuel

Qusai M. Mistarihi¹, Vivek Raj², and Ho Jin Ryu^{*1}

¹Korea Advanced Institute of Science and Technology, 291 Daehak-ro, Yuseong-gu, Daejeon, Korea

²Indian Institute of Technology-Kanpur, Kalyanpur, Kanpur, Uttar Pradesh 208016, India

*corresponding author: hojinryu@kaist.ac.kr

1. Introduction

Inert matrix fuel (IMF), in which the TRUs including Pu and minor actinides (MAs) are dispersed in a neutron transparent matrix, is one of the concepts developed for the management of radioactive spent fuel [1]. ZrO₂ is one of the candidate materials for the matrix of IMF owing to high melting point, low neutron absorption cross section, and good chemical and irradiation stability [2]. However, it has a low thermal conductivity that reduces its performance. The fabrication of Mo reinforced ZrO₂ composites with a low volume fraction of the reinforcement has been recently demonstrated and the effect of the reinforcement microstructure on the thermal conductivity of ZrO₂ has been measured [3]. The objective of this paper to investigate the effect of reinforcement microstructure on the thermal conductivity of ZrO₂ matrix using finite element method (FEM) and analytical modeling and to benchmark the theoretically predicted results with the experimentally measured ones.

2. Thermal conductivity modeling

2.1 Finite element method

Solid models of Mo-ZrO₂ composites with different structures were developed as shown in Fig. 1.

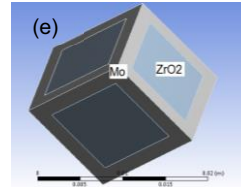
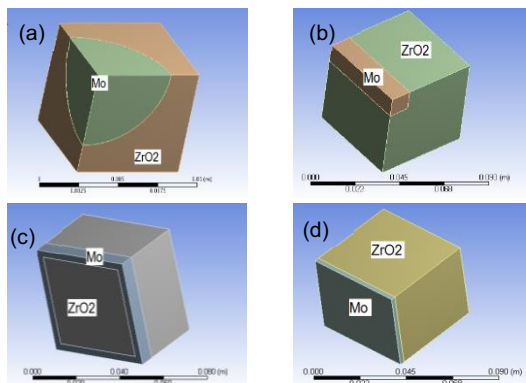


Fig. 1. Solid models of (a) Mo particle- (b) Mo fiber- (c) Mo mesh- (d) Mo sheet-, and (e) 3D interconnected Mo-ZrO₂ composites.

The thermal conductivities of Mo-ZrO₂ composites as function of temperature were calculated using the following equation:

$$K_{eff} = \frac{Q \times L}{A \times \Delta T} \quad (1)$$

where Q is the applied heat flux, L is the length along which the heat flows, A is the cross-sectional area, and ΔT is the temperature difference between the two surfaces. The thermal conductivity was calculated by applying a constant heat flux at one surface and a constant temperature at the opposite surface.

2.2 Analytical modeling

Maxwell model [4] is one of the earliest models that have been developed based on the effective medium theory. The Maxwell model was developed for composites with dilute concentration of uniformly distributed spherical particles (Eq. 2):

$$K_{eff} = K_m \frac{(2 - 2V_r)K_m + (1 + 2V_r)K_r}{(2 + V_r)K_m + (1 - V_r)K_r} \quad (2)$$

where K_r , K_m , and K_{eff} are the thermal conductivities of the reinforcement, matrix, and composite, respectively, and V_r is the volume fraction of the reinforcement.

3. Results and discussions

FEM modeling of the thermal conductivity of Mo

fiber, mesh, sheet-ZrO₂ composites when the heat flux was applied parallel and perpendicular to the direction of the reinforcement is shown in Fig. 2 together with the Maxwell model predictions. When the heat flux was applied parallel to the direction of the reinforcement, the thermal conductivity increased from fiber, mesh, to sheet microstructure due to the increased degree of high thermal conductivity reinforcement. When the heat flux was perpendicular to the direction of the reinforcement, the thermal conductivity of the reinforced ZrO₂ composites from mesh, fiber, to sheet due to the increased density of the reinforcement matrix interfaces.

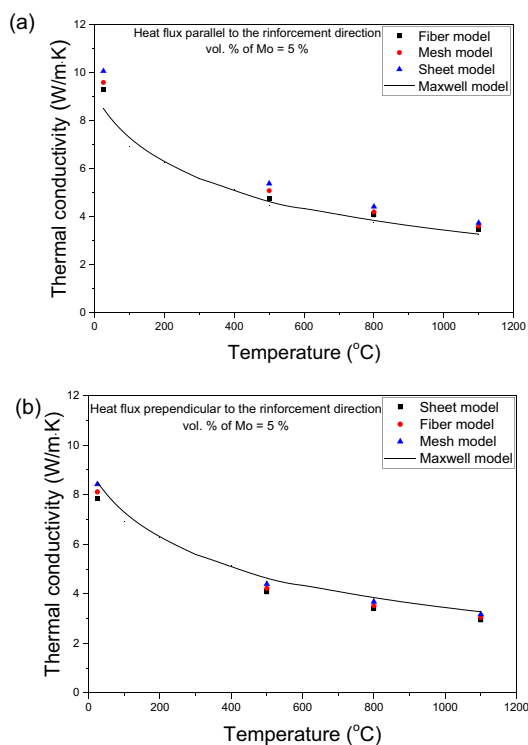


Fig. 2. FEM predicted thermal conductivity of Mo-ZrO₂ composites with different reinforcement structures and Maxwell model predictions.

Fig. 3 Shows a comparison between the experimentally measured thermal conductivity of Mo-ZrO₂ composited with dispersed particle and randomly distributed Mo wire mesh structures with the FEM and Maxwell model predictions. The thermal conductivity of Mo powder-ZrO₂ composites were corrected for the experimentally measured porosity of 99.0% using the modified Maxwell-Eucken correlation [5]. In general, the theoretically predicted thermal conductivity values are in good agreement with the experimentally measured ones.

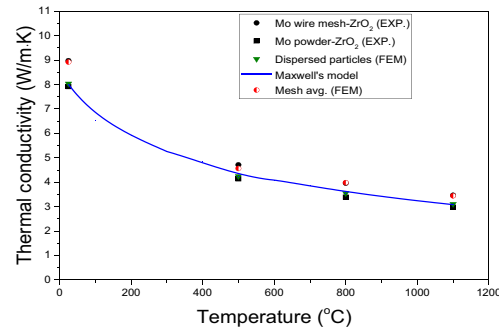


Fig. 3. Comparison between the theoretical predictions and the experimental measurements of Mo-ZrO₂ composites.

4. Conclusion

The FEM modeling of the thermal conductivity of Mo-ZrO₂ composites showed that with increasing the interconnectivity of the reinforcement structure, the thermal conductivity significantly improved due to the higher degree of interconnectivity of high thermal conductivity pathways and the reduced intensity of reinforcement/matrix interfaces. The theoretically predicted thermal conductivities of Mo-ZrO₂ composites were compared with the experimentally results and a good agreement was found.

Acknowledgments

This study was supported by the KUSTAR-KAIST institute at KAIST

REFERENCES

- [1] C. Degueldre, W. Wiesenack, Zirconia Inert Matrix Fuel for Plutonium and Minor Actinides Management in Reactors and as an Ultimate Waste Form, in: 2008.
- [2] C. Hellwig, M. Streit, P. Blair, T. Tverberg, F.C. Klaassen, R.P.C. Schram, F. Vettraino, T. Yamashita, Inert matrix fuel behaviour in test irradiations, *J. Nucl. Mater.* 352 (2006) 291–299. doi:10.1016/j.jnucmat.2006.02.065.
- [3] J. Hui Kim, S.H. Hong, H. Jin Ryu, Enhanced Thermal Conductivities of ZrO₂ Pellets Reinforced with Mo Wires or Powder, in: *Trans. Korean Nucl. Soc. Autumn Meet., Korean Nuclear Society, PyeongChang*, 2014.
- [4] W.D. Kingery, H.K. Bowen, D.R. (Donald R. Uhlmann, *Introduction to ceramics*, Wiley, 1976.
- [5] P.G. Lucuta, H. Matzke, I.J. Hastings, A pragmatic approach to modelling thermal conductivity of irradiated UO₂ fuel: Review and recommendations, *J. Nucl. Mater.* 232 (1996) 166–180. doi:10.1016/S0022-3115(96)00404-7.

Spent Nuclear Fuel Safety Evaluation Methodology (SSEM) for Storage and Transportation

Y.K. Kim^{1*}, J. S. Noh^{2**}, S.K. Lee¹ and T.W. Kim³

¹ KEPCO Nuclear Fuel Co., 242, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

² ACT Co., Ltd., Techno 9 Ro-35, Yuseong-Gu, Daejeon, Korea

³ Hanyang University, 222, Wangsimni-ro, Seongdong-gu, Seoul, Korea

*ykkim@knfc.co.kr, **jsnoh@actbest.com

1. Introduction

When the nuclear fuel is removed from the reactor after around 5 years of irradiation, it is initially stored under water in the nuclear power plant (NPP) pools (Spent Fuel Pool(SFP)) for at least 5 years through 40 years maximum. During the storage in the SFP, the decay heat of SNF shall be decreased enough to be shifted and stored to the dry storage facility.

To transport the SNF from the SFP to the dry storage facility, loading the SNF into the dry storage system should be practiced in SFP followed by drying process with inert gas afterwards. Then the SNF loaded in dry cask would be transported to the dry storage facility and remained until the further process.

Throughout the overall processes, the most important thing is the safety of the system and SNF itself as well to protect public health. In this paper, the new approach to obtain the goal of SNF safety is proposed, so called SSEM.

2. Regulations regarding to Transportation and Storage of SNF

In order to safely transport and store the high level of waste materials, like SNF, it is necessary to comply with the safety regulations strictly. For the regulations, there are a few of nuclear acts in Korea and USNRC Codes & Standards for specific technical details.

The legal requirements and regulations for the designs and operations of the transportation system and storage system/facility are stipulated in USNRC 10CFR71(Transportation), 10CFR72(Storage), and the specific technical guidance are in NUREG-1617(Transportation), NUREG-1536(Storage System), NUREG-1567(Storage Facility).

Detailed requirements on Structural, Thermal, Confinement, Shielding, Criticality, Material(Corrosion, Cladding Integrity/Fuel), Operating Procedure, Acceptance Test & Maintenance, Radiation Protection, Accident Analysis, Technical Specifications and Operating Controls and Limits, Quality Assurance are described in these regulations and guidance.

For the containment review of the transportation

system, fractions of rods that develop cladding breaches were assumed 3 % for normal condition of transport, 100 % for hypothetical accident conditions, respectively. [NUREG 1617]

For the design of dry storage system, it is described in NUREG-1536 that the system shall be evaluated assuming rupture of 1 percent, 10 percent, and 100 percent of the fuel rods for normal, off-normal, and accident-level conditions, respectively. [NUREG 1536 Rev.1]

Technical Specifications for the operations of the Transportation/Storage system/facility designed and manufactured considering these assumptions are required to establish Acceptance Criteria for Functional/Operating Limit, Monitoring Instruments, Limiting Control Setting and Limiting Conditions to protect the safety of the system and the integrity of SNF and to protect the operators from radiation. Also, Acceptance Criteria for the inspection schedule and scope of the structures, systems and components, which are Important To Safety are required. [NUREG 1536 Rev.1]

The licensing processes according to these legal requirements and regulations are presented in Fig. 1, 2. [USNRC Draft Regulatory Issue Summary 2015-XX Considerations in Licensing High Burnup Spent Fuel in Dry Storage and Transportation, XX, 2015]

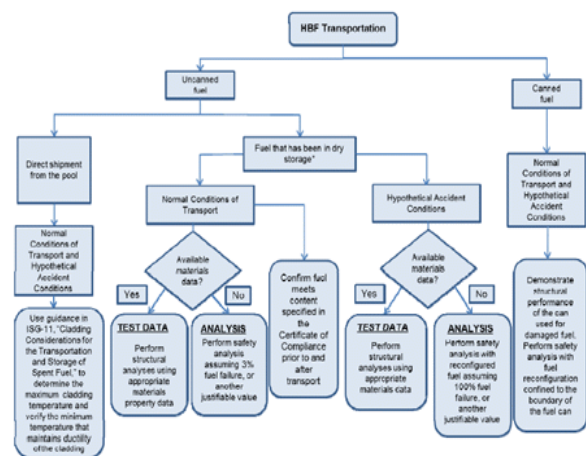


Fig. 1. Transportation Licensing Approach.

In order to transport and dry store the SNF safely, not only the requirements for licensing the system

and facility, but also the requirements for the integrity of SNF should be reviewed.

3. SSEM

In order to meet those regulation requirements and guidance, the vendor and utility shall prepare a plan implementable for the safe management of SNF and need to set up an action plan and methodology.

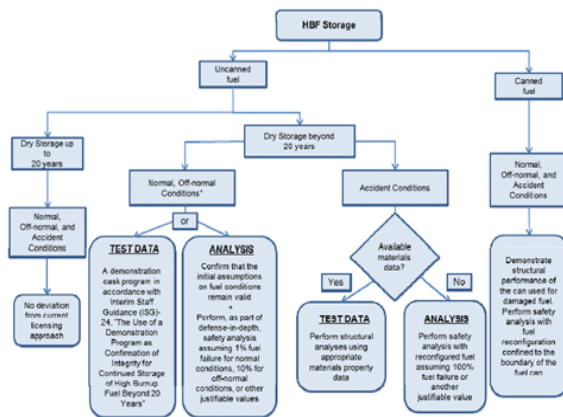


Fig. 2. Storage Licensing Approach.

Without this kind of methodology which could facilitate the reviewing processes with simple and outlined fashion, there could be some kind of confusion in determination of scope of reviewing processes between the licensees and regulatory body. It will help the licensees and regulatory body prepare and review the transportation or storage of SNF in consistency without missing important check points, resulting in a good safety management of SNF.

The main concept of the methodology would be a simplified form of reviewing processes for the transportation or storage of SNF.

In addition to the requirements for licensing the system and facility, limiting conditions for the actual operations of system and facility shall be reviewed.

Limiting conditions for operations to maintain the integrity of the system/facility and fuel itself shall be established before implementing actual storage and transportation works. (NUREG-1536, 1567)

Based on these foundation, there is a need to develop a new methodology (SSEM: Spent Nuclear Fuel Safety Evaluation Methodology), which enables to facilitate the licensing process for the transportation/ Dry Storage of SNF being systematic.

The frame of this methodology mainly relies on the safety check for each campaign of dry storage/ transportation based on the reference SAR, which was prepared using conservative bounding values for key parameters.

This methodology could verify and evaluate whether the each transportation and dry storage campaign meets the regulatory requirements with standard format and processes.

By using this methodology, the licensee can get permission for the transportation and dry storage campaign from the regulatory body by submitting the Spent Fuel Safety Evaluation Report(SSER). The outline of this methodology is illustrated in Fig. 3.

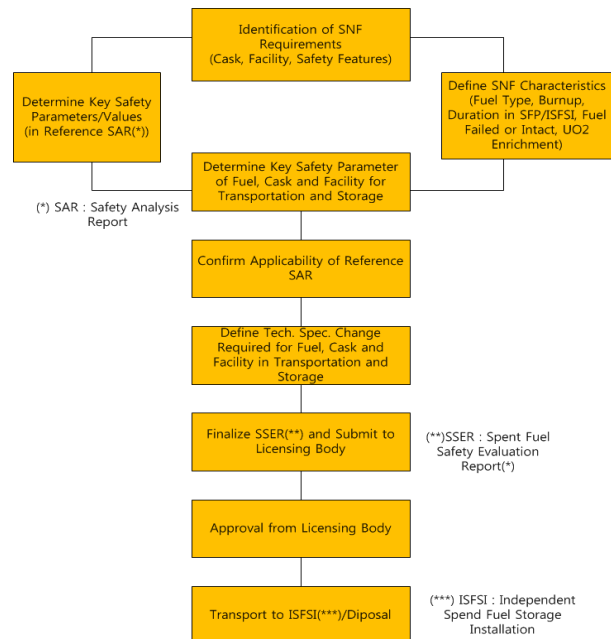


Fig. 3. Spent Fuel Safety Evaluation Methodology (SSEM) Outline.

The SSEM could be a technical foundation in transportation and storage of the SNF with safe and efficiency and regulatory requirements could be met with it too. It seems to be an essential methodology and public reliability on SNF management could be enhanced with it.

4. Summary

New approach to achieve the safety goals in transportation and dry storage of SNF, so called SSEM has been proposed. The main concept of the SSEM is that it simplifies the reviewing processes of each campaign of the transportation or storage of SNF with standard format.

This SSEM could be considered as a model case for assuring public that the SNF be managed safely.

REFERENCES

- [1] NUREG 1536r1 Standard Review Plan for SNF Dry storage systems at a general license facility.
- [2] NUREG 1567 Standard Review Plan for SNF Dry Storage Facilities.
- [3] NUREG 1617 Standard review plan for Transportation for SNF.
- [4] NRC RIS 2015-XX, 'Considerations in licensing HBF in dry storage and transportation'.

Structural Integrity Assessment of Spent Nuclear Fuel Rod Under Drop Accident Condition

Sanghoon Lee^{1*}, Belal Almomani², and Dongchan Jang²

¹Keimyung University, 1095 Dalgubeol-daero, Dalseo-gu, Daegu, Korea

²Korea Advanced Institute of Science and Technology, 291 Daehak-ro, Yuseong-gu, Daejeon, Korea

*shlee1222@kmu.ac.kr

1. Introduction

Wet and dry storages are widely used technologies to safely store the spent nuclear fuel (SNF) for several decades. Nonetheless, transfer SNF from wet to other types of longer term storage may occur multiple times before reprocessing or final disposal, due to accumulation problem or to enhance the safety and security level. This activity would probably make abrupt changes to the environment around the SNF assemblies and may lead to undesirable consequences on the fuel integrity. Moreover, the major requirements of SNF for long-term storage under normal storage conditions are to maintain fuel cladding integrity by complying the regulatory requirements of transportability and retrievability, and to provide adequate cooling and environmental control to prevent fuel degradation as stated by Liu (2015). Thus, technical issues on retrievability and cladding integrity with the concern of storage conditions and the aging effects should be investigated as major obstacles for safe handling, safe transportation operations, and economic management of SNF.

The present research is limited to a single fuel rod elastic-plastic deformations dynamics under vertical (end) and horizontal (side) drop orientations. The sensitivity analysis is performed to determine the relative impact of various parameters relevant to design safety on fuel rod strength/stiffness based on impact loads and resulting strains on fuel cladding, since it is considered as the first barrier against the release of radioactive materials. The parameters for the sensitivity study include material's mechanical properties of the fuel rod, pellet-to-cladding mechanical interaction (PCMI), spacer grid (SG) spring and dimple stiffness, and rod internal pressure (RIP). High burnup fuel with burnup higher than 45 GWd/MTU is considered for this study together with reported storage conditions.

2. FE Model and Storage Condition

2.1 Finite element model of single fuel rod

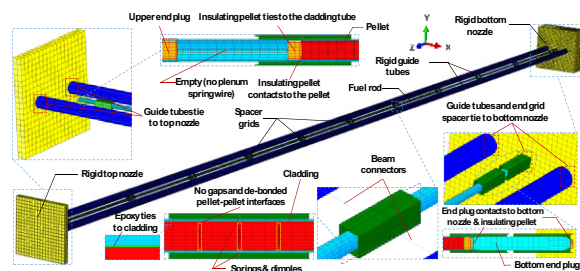


Fig. 1. Fuel rod model-finite element details.

The FE model of a single fuel rod has been developed, including all the structural features, to study the individual behavior of fuel rod under drop condition. The fuel rod was modelled with the actual dimensions of 350 fuel pellets in a 4.094-m-long fuel cladding referred to CE16x16 Plus 7 fuel type, as well as 2 end plugs, 2 insulating pellets, 2 guide tubes, top/bottom nozzles, and 12 SGs (see Fig. 1) were all modelled to investigate the interaction of fuel rod against the supporting components. This model is designed in a way to provide useful results that can be transferred to fuel assembly level model in a cask-loaded condition.

2.2 Storage conditions

The temperature history is an important variable for evaluating SNF behavior during storage and post-storage transportation accidents. The initial temperature, at vacuum drying stage, is somewhat around 400°C but decreases gradually influenced by the radioactive/heat decay. The temperature drops typically from 400°C to 250°C in the first ten years. The neutron fluence (or burnup) is also considered as a significant variable that affects the mechanical properties and failure criteria; particularly fracture toughness of irradiated Zircaloy. The cladding fluence associated with high burnup fuel (HBF) of 45~60 GWd/MTU in a pressurized water reactor

(PWR) is roughly ranged from 7 to $12 \times 10^{25} \text{ n/m}^2$. For discharge burnup in the range of 45-60 GWd/MTU, the average hydrogen concentration is ranged between 200-600 ppm. These variables make a momentous influence on the structural integrity of spent nuclear fuel associated with the material degradation and failure criteria. Therefore, the effect of lower and upper bounds storage conditions within the first ten years has been studied in the sensitivity analysis.

3. Sensitivity Analysis

3.1 Test matrix

The sensitivity analyses are performed for high degree uncertain parameters such as material properties of cladding, pellet-cladding mechanical interaction(PCMI), spacer grid stiffness, and rod internal pressure to demonstrate the impact of each parameter on the resulting stress and strain predictions.

3.2 Cladding material properties

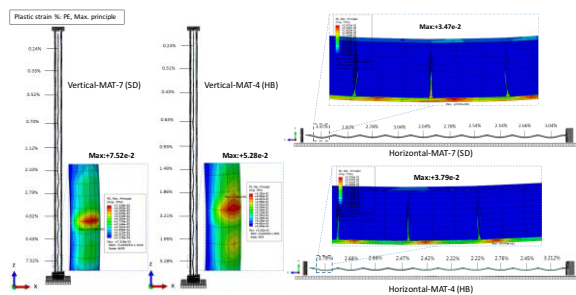


Fig. 2. Plastic strain at vertical and horizontal drop.

In the vertical drop, it appears that the largest lateral displacements occurred between the bottom SGs that near to the impact end. For the horizontal drop, the bending moments along the fuel rod are almost equivalent in both cases. The PE max. principal in the cladding for SD material is 37% higher than the HB material in the vertical drop and 5.1% higher in the horizontal drop.

3.3 PCMI

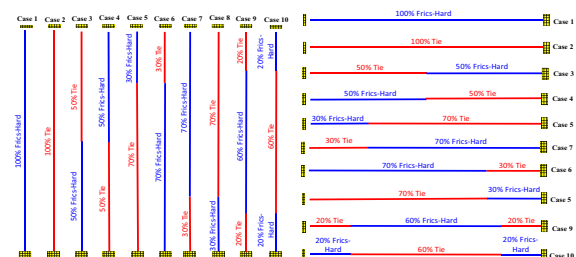


Fig. 3. Detailed fuel assembly analysis model.

Ten generalized interface bonding and de-bonding conditions have been implemented using frictionless hard contact property and surface-based tie constraint (permanently glued surfaces) to provide a fundamental understanding of the interface bonding efficiency for the fuel rod composite system as shown in Fig. 3. It is concluded that the PCMI can highly alter the location and magnitude of maximum plastic strain with a significant effect on fuel rod response under drop condition and thus affects the probability of fuel rod failure.

3.4 SG stiffness and RIP

Three stiffness reduction percentages (25%, 50%, 75%) of fresh springs and dimples have been examined representing the radiation-induced changes. RIP of the lower limit is 6 MPa at temperature 250°C and the upper limit is 7.72 MPa at temperature 400°C was examined.

The PE max. principal difference decreased 14.4%, 35%, and 70% in vertical drop, and decreased 4%, 2.2%, 1.9% in the horizontal drop when the springs and dimples stiffness decreased 25%, 50%, and 75%, respectively. PE max. principal difference decreased 13% in vertical drop, and 1.4% in the horizontal drop when the RIP increased from 6 MPa to 7.72 MPa.

4. Conclusions

This work establishes the basis of an intensive parametric study of uncertain parameters that highly affect the cladding integrity. Sensitivity analysis has been defined to evaluate the impact of fuel rod material properties, PCMI, RIP and SGs stiffness within the first ten years of dry storage period. Through this study, it is revealed that the PCMI bonding condition showed a high influence on the fuel rod behavior and in altering the strain deformation results comparing to other parameters.

REFERENCES

- [1] Liu, Y.Y. Wet storage of spent nuclear fuel, in Safe and Secure Transport and Storage of Radioactive Materials, Elsevier, Oxford, UK (2015).
- [2] Adkins, H. et al. Used Nuclear Fuel Loading and Structural Performance under Normal Conditions of Transport— Demonstration of Approach and Results on Used Fuel Performance Characterization, US DOE, 2013.

Establishment of the Design and Testing Methods of Lid Welds as the Confinement Boundary for Spent Nuclear Fuel Storage Canister

Chang-Yeal Baeg* and Chun-Hyung Cho

Korea Radioactive Waste Agency, 168 Gajeongro, Yuseong-gu, Daejeon, Korea

*baegcy@korad.or.kr

1. Introduction

This paper describes the design and testing methods of lid welds as the confinement boundary for austenitic stainless steel storage canisters. This canister must be designed to protect the release of radioactive materials during the dry storage period [1]. At present, the storage canister developed by KORAD (Korea Radioactive Waste Agency) applies redundant lid and closures that are using the multi-pass weld. We also investigated and analyzed the integrity test methods for the confinement boundary.

2.2 General Requirements

This storage canister must rely on weld integrity to assure continued confinement effectiveness. The redundant welded closures must be helium leakage tested per the method of ANSI N14.5 [2] or exempted from the helium leakage test according to the "large weld exception criteria design" [3]. In order for closure weld to be exempt from the helium leakage test, all of the following conditions are satisfied;

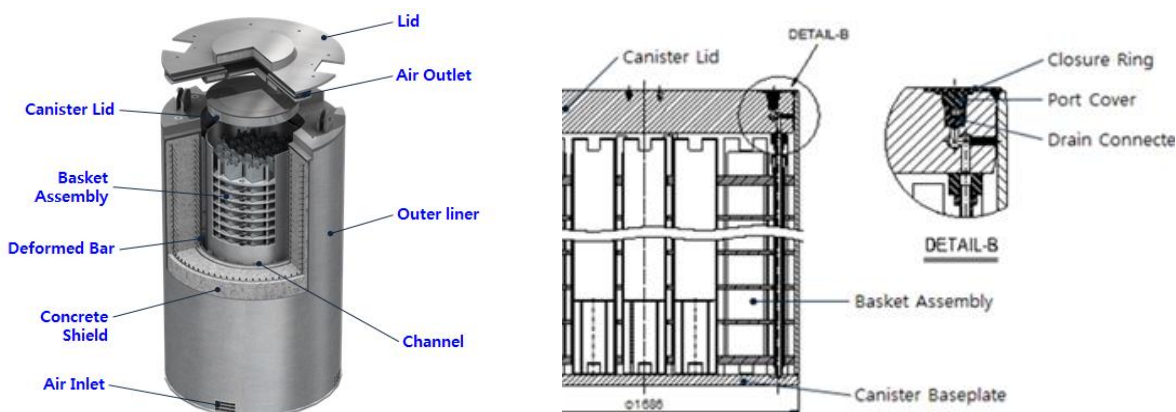


Fig. 1. Conceptual drawing of the concrete storage cask and its canister.

2. Design and testing of lid welds

2.1 Description of storage canister

The storage canister developed by KORAD consists of fuel basket, canister shell, lid and closure plate (Fig. 1). The major components of this canister are made of 2 types austenitic stainless steel (inner parts; SA 204 TP 304 and outer parts; SA 204 TP 316L) considering corrosion and environmental impacts. After spent fuel assemblies are loaded, the canister lid and closure ring plate are sealed by remote welding.

- 1) The weld must be multi-pass, with a minimum weld depth comprised of at least 3 distinct weld layers.
- 2) Each layer of weld may be composed of one or more adjacent weld beads.
- 3) The layer must be complete across the width of the weld joint.
- 4) If only three weld layers comprise the full thickness of the weld, each layer must be PT examined.
- 5) The weld cannot have been executed under conditions where the root pass might have been

subjected to pressurization from the helium fill in the canister itself.

2.3 Test of the Confinement Boundary

This storage canister design meets the "single lid with cover plate" concept (refer to fig. 2). The confinement integrity test should be performed with the following steps;

The first closure boundary (fig. 2, the dotted line marked ①) maintains confinement integrity because one weld meets the large weld exemption criteria and the other (drain and vent port) weld was He leakage tested. Thus, this first closure boundary meets the guidance by ensuring at least one of the two redundant closure boundaries is leakage tested or conforms to the large weld exemption criteria.

The second boundary (fig. 2, the dotted line marked ②) continues through the cover plate to the fillet weld joining the cover plate to the canister lid. The weld joining the cover plate to the canister wall and lid cannot be helium leakage tested since there is no feasible means to do so. Since this second boundary does not meet all the criteria for a confinement boundary.

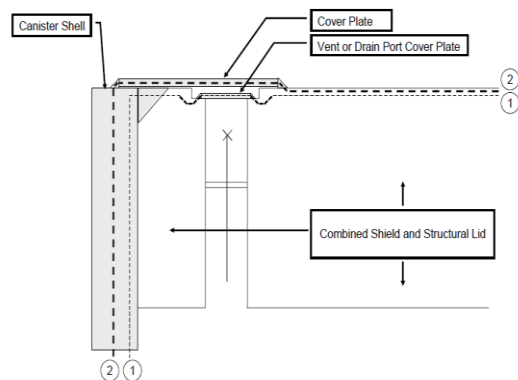


Fig. 2. Configuration of leakage testing a single lid with cover plate design.

3. Conclusion

The storage canister developed by KORAD has redundant lid and closures using the in situ multi-pass weld after fuel loading at the domestic PWR NPPs. The integrity test of the storage canister meets the safety requirements by applying the large weld exception criteria in accordance with the exceptional

provisions of the He leakage test. Other parts of the integrity confirmed the He leakage test.

In this way, we also established and confirmed the confinement integrity test methods for the storage canister.

ACKNOWLEDGEMENT

This work was supported by the Korea Institute of Energy Technology Evaluation and Planning (KETEP) granted financial resource from the Minister of Trade, Industry and Energy, Republic of Korea.

REFERENCES

- [1] U.S. Nuclear Regulatory Commission (NRC), 10CFR72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radio-active Waste and Reactor Related Greater than Class C Waste (2012).
- [2] ANSI N14.5, Leakage tests on packages for shipments (1997).
- [3] U.S. Nuclear Regulatory Commission (NRC), NUREG-1536, Revision 1, Standard Review Plan for Dry Cask Storage Systems (2010).

Advanced Nuclear Fuel Cycle Train Program in Tohoku University: Experiments on Solid States and Solution Chemistry

Geonyong Lee and Kwangheon Park*

Department of Nuclear Engineering, Kyunghee University, 1732, Deogyong-daero, Giheung-gu,
Yongin-si, Gyeonggi-do, Korea

*Corresponding author: kpark@khu.ac.kr

1. Introduction

In this experimental study program in Tohoku University, there are four experiment related with nuclear fuel cycle. First one is the oxidation and reduction process of UO_2 . Second one is to observe microscopic structure of solid solution of UO_2 and ZrO_2 using SEM/EDX and XRD system. Third one is milking process of ^{239}Np from ^{243}Am by solvent extraction using Tri-n-Octylamine (TOA). Last one is solvent extraction in PUREX by the simulated mixed aqueous solution of U, ^{85}Sr and ^{239}Np which is represented minor actinide elements included in the spent nuclear fuel.

2. Experiment procedure

2.1 Procedure of UO_2 Oxidation and Reduction

We conducted two experiment. First, we did experiment which is related about the transition of UO_2 by heating it up. The process is reduction and oxidation (REDOX). After heat treatment. We used XRD to analyze the structure of material. second we used TG-DTA to analyze weight gain and enthalpy of each reaction.

2.2 Procedure of solid solution $\text{Zr}_y\text{U}_{1-y}\text{O}_2$ ($y = 0.05$ and 0.1)

We did experiment which is related about the Formation of Solid Solution $\text{Zr}_y\text{U}_{1-y}\text{O}_2$. Heating up the temperature from R.T to 1400°C using Ar + 10% H_2 gas. In this experiment, we investigate the morphology of $\text{Zr}_y\text{U}_{1-y}\text{O}_2$ by scanning electron microscopy (SEM). In addition, the structure change by the formation of solid solution were examined using X-ray diffraction (XRD).

2.3 Procedure of milking process for extraction of Np-239

The procedure of experiment for milking process can be divided into four stages. The first step is to make the pre-equilibrium. The second step is a extraction of Np-239 to organic phase. The third step is Stripping process, which is back extraction of Np to aqueous phase. The final step is fuming of Np-239 solution.

2.4 Procedure of extraction of uranium in the solution that Contains Np-239 and Sr-85

In this study, we are focusing on the behavior of MAs and FPs during the solvent extraction. ^{239}Np is a MAs and it is extracted with U and Pu in some conditions. so we using 3 different solution containig U, ^{239}Np , ^{85}Sr .

and Solution has various nitric acid concentration as 0.1M, 1.0M, 5.0M, the separation factor is affected by the nitric acid concentration so we calculated the separation factor and find best condition for the extraction of Uranium only, from aqueous phase containig ^{239}Np and ^{85}Sr .

3. Result and discussion

3.1 Result of UO_2 Oxidation and Reduction

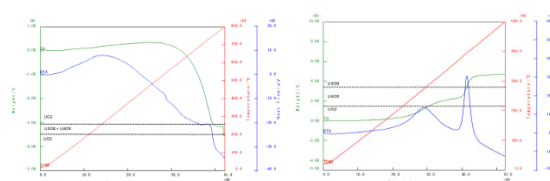


Fig. 1. TG-DTA curve of oxidation and reduction.

DTA line shows two peaks. around the black line. This peak show us its exothermic reaction during oxidation. Additionally, this two peaks and mass gradient is steeply change almost simultaneously. This peak and mass change proved phase transformation

In the reduction process, It is only once that the mass gradient changes steeply. Also, DTA line shows one exothermic peak around the black line. Theoretical mass change of U_3O_8 to U_4O_9 is -2.38wt% and mass change of U_3O_8 to UO_2 is -3.8wt%. In our experiment, there are not flat line around -2.38wt%, In fig. 4, Second flat line of TG is around -3.5wt%. Thus, reduction reaction of U_3O_8 is directly change to UO_2 .

We can calculate the density of UO_2 and U_3O_8 using the measured lattice parameter, volume of unit cell of UO_2 and U_3O_8 is $163.4879 \times 10^{-24} \text{ cm}^3$ and $334.8809 \times 10^{-24} \text{ cm}^3$, As a result of calculation using volume and mass of unit cell, the density of UO_2 and U_3O_8 is 10.9680 g/cm^3 and 8.3501 g/cm^3 . During oxidation of UO_2 , the density decreased by about 30%, which means a volume increase of about 30%. [1]

3.2 Result of solid solution $\text{Zr}_y\text{U}_{1-y}\text{O}_2$ ($y = 0.05$ and 0.1)

According to SEM/EDX images of synthesized $\text{Zr}_y\text{U}_{1-y}\text{O}_2$ solid solution ($y = 0.05$, Figure 2), uranium and zirconium seem to be distributed homogeneously.

However, EDX results were $\text{Zr}_{0.0311}\text{U}_{0.9689}\text{O}_2$ and $\text{Zr}_{0.0537}\text{U}_{0.9463}\text{O}_2$ for $y = 0.05$ and 0.1 , respectively. Determined values (0.0311 and 0.0537) by EDX were less than initial values (0.05 and 0.1). the rate limiting step in the formation of solid solution may be the self

diffusion of U in UO_2 [2]. So we need more annealing time considering self diffusion coefficient of U in UO_2 for formation of solid solution.

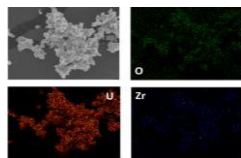


Fig. 2. Result of EDX analysis.

Fig. 2 is a result of lattice parameter. Lattice parameter of $\text{Zr}_y\text{U}_{1-y}\text{O}_2$ were 5.43458 and 5.42422 Å for $y = 0.05$ and 0.1 . The lattice parameter had a decrease tendency as increasing Zr portion. Thus, volume of solid solution is decrease compared with UO_2 . Shrinking of unit cell may be due to the intrusion of Zr

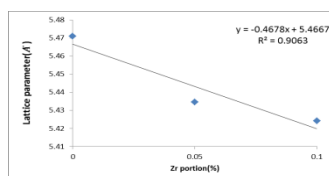


Fig. 3. Lattice parameter of solid solution.

3.3 Result of milking process for extraction of Np-239

The daughter radionuclides can be extracted from parent radionuclides persistently using milking process. For this process, the condition of secular equilibrium should be satisfied. Np239 is extracted from Am243.

During the Np extraction process, Np(IV) and Np(VI) are extracted to organic phase(Tri-Octylamine in xylene) and Am243 remains in aqueous phase(HCl). For Stripping process, the Np back extraction occurs because of the distribution ratio decreases by decreasing the concentration of HCl. Finally for fuming process, nitric acid is used for higher purification

3.4 Result of extraction of uranium in the solution that Contains Np-239 and Sr-85

$$\gamma = \frac{[U]_{org}/[U]_{aq}}{[Np]_{org}/[Np]_{aq}} \quad (1)$$

Separation factor is calculated following equation (1). separation factor means that we can get much more purified uranium during solvent extraction.

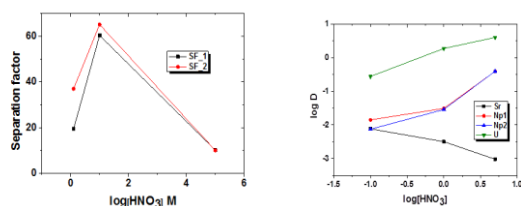


Fig. 4. Separation factor and concentration of nitric acid.

Separation factor is affected to concentration of nitric acid. In the Fig. 4. separation factor has a most high value

at the 1.0M of nitric acid. so we can guess 1.0M of nitric acid is a best condition of PUREX for extract uranium without neptunium. this result means that 1.0M of nitric acid can make a high purification of uranium

4. Conclusion

In the first experiment, we are able to get the sense of phase change of uranium oxide as measuring weight from TG-DTA experiment. When UO_2 is in oxidation to U_3O_8 , it has a change of structure. By using XRD, we can get the structure of $\text{UO}_2, \text{U}_3\text{O}_8$ and the lattice parameter of that. By using the lattice parameter, we can notice that there is a volume expansion during the oxidation process of UO_2 to U_3O_8 .

In the second experiment, By using EDX, we measured a ratio of Zr and U in the solid solution. the measured ratio is less than initial value (0.05 and 0.1). The diffusion coefficient of U is much smaller than O in UO_2 . So U is the rate limiting step in formation of solid solution. we need more annealing time for formation of solid solution. By using XRD, we are able to guess the effect of cation to the lattice parameter. The lattice parameter had a decrease tendency as increasing Zr. Thus Shrinking of unit cell may be due to the intrusion of Zr.

In the third experiment, extraction of Np239 using the milking process is analyzed during the experiment. The chemical reactions are derived during this process. After a series of steps using the TOA. We can get the Np239 from Am234.

At last one, To determined the best condition of PUREX for a high quality of uranium, Neptunium has a low distribution coefficient. as increasing concentration of nitric acid. Neptunium exist Np (IV) and Np (VI) dominant, It make that distribution coefficient is increased. In this study, the best condition for PUREX process is 1.0M of nitric acid. To get more value, need more experiment as a various concentration of nitric acid.

5. Reference

- [1] Reduction Kinetics of Uranium Trioxide to Uranium Dioxide Using Hydrogen Pedro Orrego Alfaro1, José Hernández Torres1, Fernando Puchi Thiele2, World Journal of Nuclear Science and Technology, 2015, 5, 149-156.
- [2] R. Perriot, X.-Y. Liu, C. R. Stanek, and D. A. Andersson, "Diffusion of Zr, Ru, Ce, Y, La, Sr and Ba fission products in UO_2 ." Journal of Nuclear Materials (2015).
- [3] Taylor, P., Nakahara, M., Sano, Y., Koma, Y., & Kamiya, M. (2012). Separation of Actinide Elements by Solvent Extraction Using Centrifugal Contactors in the NEXT Process Separation of Actinide Elements by Solvent Extraction Using Centrifugal Contactors in the NEXT Process. Journal of Nuclear Science and Technology, 44(3), 373-381.

2분과

사용후핵연료관리 (Poster)



Theoretical Analysis for Direct Conversion of U_3O_8 Into UCl_3 as an Alternative to Oxide Reduction Reaction

Min Ku Jeon*, Jin-Mok Hur, and Do-Hee Ahn

Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, 34059 Daejeon, Republic of Korea

*minku@kaeri.re.kr

1. Introduction

Oxide reduction process is considered as a key link which enables application of the metal fuel based pyroprocessing in PWR (Pressurized Water Reactor) fuels. According to the KAERI's pyroprocessing flow diagram, UO_2 fuel is reduced into its metallic form and then U is selectively recovered through electro-refining. And group recovery of U and TRU (TRansUranic) nuclides performed via electro-winning process [1]. Although these processes show promising results, new approaches should be considered for higher efficiency. In this work, an alternative process, so-called MKZP (Min Ku Zirconium Process) was proposed to convert U_3O_8 into UCl_3 directly to simplify the pyroprocessing.

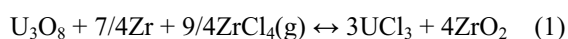
2. Theoretical calculations

The HSC chemistry code (Reaction Equations module) was employed for the theoretical calculations including changes in Gibbs free energy (ΔG) and equilibrium constants (K). The ΔG value helps to identify direction of the reaction, in other words, right-hand reaction is preferred when ΔG is negative, while reverse reaction is preferred when ΔG is positive.

3. Results and Discussion

3.1 Conversion of U and actinide oxides into chlorides

The key reaction of the MKZP is as follows:



The calculation results provided ΔG and K values of -258.0 kcal and 8.62×10^{72} at 500 °C. These values strongly suggest that the formation of UCl_3 is highly preferred thermodynamically. Sakamura et al. published a similar reaction scheme which was performed using UO_2 instead of U_3O_8 and under a LiCl-KCl molten salt environment [2]. Considering pulverization of UO_2 during the oxidative decladding process to produce fine U_3O_8 powder, and low solubility of ZrCl_4 in molten salt (1.3 at% at 500°C), reaction (1) of MKZP is considered to have higher potential for higher reaction rate and scale-up.

Further calculations revealed that the combination of metallic Zr and ZrCl_4 can convert other actinide nuclides such as Np, Pu, Am, and Cm into their chloride forms.

3.2 Conversion of other nuclides into chlorides

Other nuclides of major interest in the pyroprocessing are those with high decay heat such as Sr and Cs. It was theoretically identified that these nuclides prefer their chloride forms in the presence of ZrCl_4 (metallic Zr is not even necessary).

3.3 Zirconium supply

The biggest concern on the MKZP is that a lot of metallic or chloride form of zirconium is consumed by the process. However, 42 wt% of Zr necessary for the MKZP can be fulfilled by cladding hulls which

consists of more than 95wt% of zirconium. And they are easily converted into $ZrCl_4$ by reacting with chlorine gas.

3.4 Process flow diagram

The overall flow diagram is shown in Fig. 1. The MKZP could eliminate complicate pellet production step between the voloxidation and oxide reduction of the pyroprocessing. It is clear that the MKZP

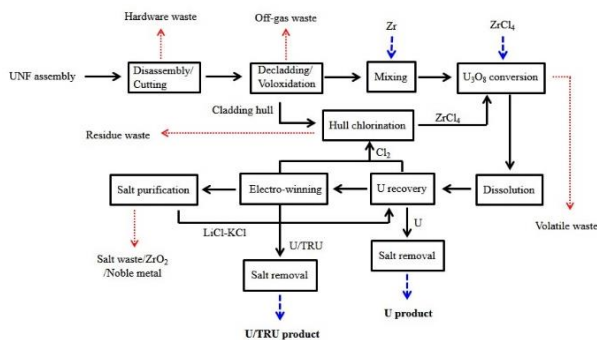


Fig. 1. Process flow diagram of the MKZP.

4. Conclusion

Although experimental study should be performed intensively prior to actual comparison, theoretical calculation results revealed that the MKZP can reduce the number of processes from 16 of pyroprocessing to 11 [3].

REFERENCES

- [1] H. Lee, G.I. Park, K.H. Kang, J.M. Hur, J.G. Kim, D.H. Ahn, Y.Z. Cho, and E.H. Kim, "Pyroprocessing technology development at KAERI", Nuclear Engineering and Technology, 43, 317-328 (2011).
- [2] Y. Sakamura, T. Inoue, T. Iwai, and H. Moriyama, "Chlorination of UO_2 , PuO_2 and rare earth oxides using $ZrCl_4$ in LiCl-KCl eutectic melt", Journal of Nuclear Materials, 340, 39-51 (2005).

- [3] M.K. Jeon, J.M. Hur, and D.-H. Ahn, "Introduction of MKZP (Min Ku Zirconium Process) as an alternative to pyroprocessing for used nuclear fuel management", Journal of Radioanalytical and Nuclear Chemistry, 308, 747-752 (2016).

Selection of Surrogate Materials for Oxide Reduction of Used Nuclear Fuel Using Carbon Anode

Min Ku Jeon*, Sung-Wook Kim, Sang-Kwon Lee, and Eun-Young Choi

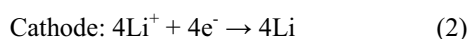
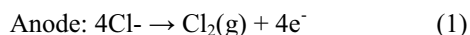
Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, 34059 Daejeon, Republic of Korea

*minku@kaeri.re.kr

1. Introduction

Carbon was recently proposed as a new anode electrode material to replace costly platinum for oxide reduction of used nuclear fuels [1]. Employing carbon sacrificial anode enables operation of the cell at a higher voltage than the platinum case where the anode voltage is limited below platinum oxidation potential.

The oxide reduction reaction with carbon anode occurs via two steps of electrochemical and chemical reactions. First, LiCl molten salt is electrochemically decomposed into metallic Li and chlorine gas as follows:



And then metallic lithium produced in the cathode electrode reacts with UNF as follows:



This consequent reaction mechanism suggests that the surrogate material should have similar properties to U or other nuclides in both electrochemical and chemical aspects.

In this work, theoretical calculations were performed to identify the most appropriate surrogate materials for the carbon anode oxide reduction process. Employing surrogate materials will help to

minimize the amount of radioactive waste produced during the research of oxide reduction process.

2. Theoretical calculations

Reduction potentials of various metal oxides were calculated from Gibbs free energy (ΔG) values using the following equation:

$$E = -\Delta G / zF \quad (4)$$

where E is the reduction potential, z is number of electrons involved in the reaction, F is the Faraday constant. In chemical reaction point of view, ΔG values and equilibrium constants were compared with UO_2 and PuO_2 . These calculations were performed using the HSC chemistry code.

3. Results and Discussion

3.1 Surrogate materials for UO_2

Among various metal oxides, four of Al_2O_3 , TiO , ZrO_2 , and BaO exhibited similar reduction potential with UO_2 as shown in Table 1. Comparing ΔG for chemical reaction with lithium, TiO and ZrO_2 were identified as the best surrogate material to replace UO_2 . In addition, it should be mentioned that metallic Al melts at 660°C (close to operation temperature (650°C) of oxide reduction process) and metallic Ba has high solubility in LiCl salt. Although zirconia is

known to react with Li_2O to produce Li_2ZrO_3 , low Li_2O concentration of the carbon anode system might suppress this reaction [2].

Table 1. Theoretical calculation results for UO_2 surrogate candidates at 650°C

Material	E (V)	ΔG for reaction with Li (kcal/mol)
UO_2	2.396	-6.394
Al_2O_3	2.395	-9.757
TiO	2.350	-5.328
ZrO_2	2.394	-6.591
BaO	2.402	-2.929

3.2 Surrogate materials for PuO_2

Identical approach was employed to identify appropriate material to replace PuO_2 . Among various oxides, CeO_2 and Eu_2O_3 were identified to have similar reduction potential and ΔG for chemical reaction. However, it is well known that Eu_2O_3 exhibits high solubility in LiCl salt leaving CeO_2 as the only candidate for PuO_2 .

Table 2. Theoretical calculation results for PuO_2 surrogate candidates at 650°C

Material	E (V)	ΔG for reaction with Li (kcal/mol)
PuO_2	2.281	-17.03
CeO_2	2.327	-12.747
Eu_2O_3	2.358	-14.85

4. Conclusion

Considering both electrochemical and chemical properties, TiO and ZrO_2 were proposed as the best surrogate materials to replace UO_2 . In the case of PuO_2 , CeO_2 was identified as the most appropriate surrogate material.

REFERENCES

- [1] S.-W. Kim, M.K. Jeon, H.W. Kang, S.-K. Lee, E.-Y. Choi, W. Park, S.-S. Hong, S.-C. Oh, and J.-M. Hur, "Carbon anode with repeatable use of LiCl molten salt for electrolytic reduction in pyroprocessing", *Journal of Radioanalytical and Nuclear Chemistry*, 310, 463-467 (2016).
- [2] W. Park, E.-Y. Choi, S.-W. Kim, S.-C. Jeon, Y.-H. Cho, J.-M. Hur, "Electrolytic reduction of a simulated oxide spent fuel and the fates of representative elements in a Li_2O - LiCl molten salt", *Journal of Nuclear Materials*, 477, 59-66 (2016).

Residual Salt Separation of Reaction Products in Pyroprocessing

Sung-Wook Kim,* Jin-Mok Hur, Dong Hyun Heo, Sun-Seok Hong, and Eun-Young Choi

Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon, Korea

*swkim818@kaeri.re.kr

1. Introduction

Molten salts such as LiCl and LiCl-KCl are used as electrolytes of electrochemical processes (i.e., oxide reduction, electrorefining, electrowinning) in pyroprocessing [1]. Hence, the reaction products after each electrochemical process should contain residual salts. Distillation is the common method to remove the residual salts [2]. The distillation process, however, requires a complex apparatus because it is operated in high temperature condition in vacuum environment to ensure sufficient vapor pressure for evaporating the residual salts. In case of the oxide reduction, such high temperature process should be prohibited because the metal products may be reconverted to the original oxide states owing to chemical reaction at the elevated temperature (e.g., $U + 2Li_2O = UO_2 + 4Li$) [3].

Here, we proposed simple technique using heating gun without any vacuum components to separate the residual salts from the reaction products. Injecting heated gas with sufficiently high flux seems to be promising for the removal of the residual salts.

2. Experimental Setup

Fig. 1 shows photograph of the residual salt separation reactor used in this study. Heating gun is the key component of this reactor, which provides two driving forces of the separation: one is heat and the other is momentum. The heating temperature should be higher than the melting point of the target salts. When the heated gas is injected into the reactor, the residual salts is gradually melted and, at the same time, high momentum of the injection gas particles derives the physical separation of the liquid salt from the reaction products. Gravity is also helpful for the separation, but the momentum of the injection gas should be dominant. The reactor was placed on vertical-type furnace installed inside an Ar-filled glove box. Ar gas inside the glove box was used as

the injection gas and off-gas was also exhausted inside the glove box for circulation.

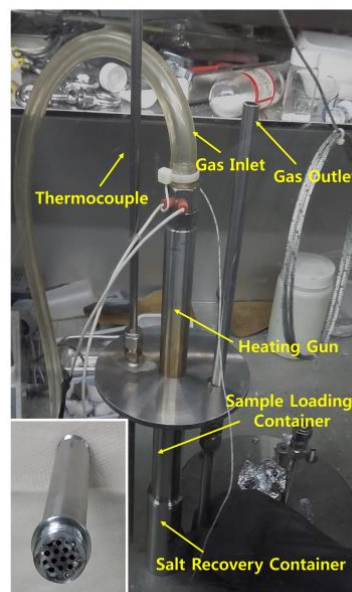


Fig. 1. Configuration of the residual salt separation reactor (inset: sample loading container).

3. Results

LiCl, the electrolyte of the oxide reduction process, was chosen as the salt material. 4 pellets of TiO (6.160 g in total) were introduced into molten LiCl at 650°C and then taken out to form the thin salt layer in their surface. Weight of the TiO pellets after the immersion was 6.371 g, indicating that the salt weight is 0.211 g. The TiO pellets became dark as shown in Fig. 2, suggesting that the chemical and/or physical states of the TiO pellets were somehow changed even after the simple immersion. TiO is known to have yellow, brown, or black color depending on its form. The salt-containing TiO pellets were loaded into the reactor. 1.401 g of additional bulk LiCl (total LiCl weight = 1.612 g) was loaded together into the reactor to clearly determinate the salt separation capability.

Ar gas, with a flow rate of 30 l min^{-1} , was heated to $700\text{--}720^\circ\text{C}$ inside the heating gun to melt LiCl (melting point: $\sim 610^\circ\text{C}$). The reactor was heated to 660°C during the experiment to minimize the heat loss of the heated Ar gas. Ideally the reactor temperature needs to be reduced below the melting point to increase the salt recovery efficiency and thus further optimization is required. The heated gas was injected to the reactor for 9 h.

After the injection, the final weight of the TiO pellets was 6.180 g, which is almost identical to the pristine state. No noticeable frozen LiCl existed on the surface of the TiO pellets and the bulk LiCl was completely removed. These indicate that this technique using the heating gun is viable for the effective removal of LiCl. On the other hand, thin white layer was formed on the surface of the TiO pellets as shown in Fig. 2, which would be due to oxidation of TiO to TiO_2 by residual H_2O and O_2 in the glove box. Indeed, the reactor was severely oxidized after the reaction. It is worth noting that the process temperature for the residual salt separation can be remarkably lowered compared to the common distillation process (e.g., $\sim 800\text{--}900^\circ\text{C}$ for LiCl) [2, 3] using this simple technique.



Fig. 2. (a) Pristine TiO pellets, (b) TiO pellets containing the LiCl salt, and (c) TiO pellets after the salt separation.

4. Conclusion

It was reported that injecting the heated Ar gas with adequate momentum is effective to remove the residual LiCl in the surface of TiO pellets. It is considered that this simple technique can be used in pyroprocessing to separate the residual salts from the reaction products of the electrochemical processes. Scale-up study should be followed for the application in engineering-scale equipment for pyroprocessing.

REFERENCES

- [1] H. Lee, G.-I. Park, K.-H. Kang, J.-M. Hur, J.-G.

Kim, D.-H. Ahn, Y.-Z. Cho, and E.H. Kim, “Pyroprocessing Technology Development at KAERI”, Nuclear Engineering and Technology, 43(4), 317-328 (2011).

- [2] I.S. Kim, S.C. Oh, H.S. Im, J.M. Hur, and H.S. Lee, “Distillation of LiCl from the LiCl- Li_2O Molten Salt of the Electrolytic Reduction Process”, Journal of Radioanalytical and Nuclear Chemistry, 295(2), 1413-1417 (2013).

- [3] E.-Y. Choi, M.K. Jeon, and J.-M. Hur, “Reoxidation of Uranium in Electrolytically Reduced Simulated Oxide Fuel during Residual Salt Distillation”, Journal of Radioanalytical and Nuclear Chemistry, online published, DOI: 10.1007/s10967-017-5405-x.

Effect of Cl_2 on Electrodeposition Behavior in Electrowinning Process

Si Hyung Kim*, Taek-Jin Kim, Gha-Young Kim, Jun-Bo Shim, Seungwoo Paek, and Sung-Jai Lee

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

*exodus@kaeri.re.kr

Abstract

Pyroprocessing at KAERI (Korea Atomic Energy Research Institute) consists of pretreatment, electroreduction, electrorefining and electrowinning. SFR (Sodium Fast Reactor) fuel is prepared from the electrowinning process which is composed of LCC (Liquid Cadmium Process) and Cd distillation et al. LCC is an electrochemical process to obtain actinides from spent fuel. In order to recover actinides inert anodes such as carbon material are used, where chlorine gas (Cl_2) evolves on the surface of the carbon material. And, stainless steel (SUS) crucible should be installed in large-scale electrowinning system. Therefore, the effect of chlorine on the SUS material needs to be studied.

$\text{LiCl-KCl-UCl}_3\text{-NdCl}_3\text{-CeCl}_3\text{-LaCl}_3\text{-YCl}_3$ salt was contained in 2 kinds of electrolytic crucible having an inner diameter of 5cm, made of an insulated alumina and an SUS, respectively. And, three kinds of electrodes such as cathode, anode, reference were used for the electrochemical experiments. Both solid tungsten (W) and LCC were used as cathodes. Cd of 45 g as the cathode material was contained in alumina crucibles for the deposition experiments, where the crucible has an inner diameter of 3 cm. Glassy carbon rod with the diameter of 0.3 cm was employed as an anode, where shroud was not used for the anode. A pyrex tube containing $\text{LiCl-KCl-1mol\% AgCl}$ and silver (Ag) wire having a diameter of 0.1cm was used as a reference electrode. Electrodeposition experiments were conducted at 500°C

at the current densities of $50 \sim 100 \text{ mA/cm}^2$. In conclusion, Fe ions were produced in the salt during the electrodeposition by the reaction of chlorine evolved from the anode and Fe of the SUS crucible and thereby LCC system using SUS crucible showed very low current efficiencies compared with the system using the insulated alumina crucible. Anode shroud needs to be installed around the glassy carbon not to influence surrounding SUS material.

Measurement of Evaporation Rates for Lanthanum and Neodymium Chlorides

S.W. Kwon*, Y.S. Lee, J.H. Jung, J.H. Chang, S.H. Kim, and S.J. Lee

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

*swkwon@kaeri.re.kr

Abstract

Electrorefining is a key step in pyroprocessing. The electrorefining process is generally composed of two recovery steps – the deposit of uranium onto a solid cathode and the recovery of the remaining uranium and TRU elements simultaneously by a liquid cadmium cathode. Uranium deposit recovered from the solid cathode is a dendritic powder. It is necessary to separate the adhered salt from the deposits prior to the consolidation of uranium deposit. The adhered salt is composed of lithium, potassium, uranium, and rare earth chlorides. Distillation process was employed for the cathode processing. One of the operation methods is distillation of the salt at low temperature (900°C), and then melting of the deposit at high temperature to avoid a backward reaction.

For the development of the salt distiller, the distillation behavior of the low vapor pressure chlorides should be studied. Rare earth chlorides in the adhered salt of uranium deposits have relatively low vapor pressures compared to the process salt (LiCl-KCl). In this study, the evaporation rates of the lanthanum and neodymium chlorides were measured for the salt separation from electrorefiner uranium deposits in the temperature range of 825~910°C. The evaporation rates of both chlorides increased with an increasing temperature. The evaporation rate of lanthanum chloride varied from 0.12 to 1.68 g/cm²/h. Neodymium chloride was more volatile than lanthanum chloride. The evaporation rate of neodymium chloride varied from 0.20 to 4.55

g/cm²/h. The evaporation rate of both chlorides are more than 1 g/cm²/h at 900°C. Even though the evaporation rates of the both chlorides were less than that of the process salt, the contents of the lanthanide chlorides were small in the adhered salt. Therefore it can be concluded that 900°C is suitable for the operation temperature of the salt distiller.

Modification of Tablet Press for Compaction of U_3O_8 Powder in Hot Cell

Jae-Won Lee*, Do-Yoon Lee, Young-Soon Lee, Sang-Chae Jeon, Ju-Ho Lee, and Yung-Zun Cho
Korea Atomic Energy Research Institute, Daedeok-daero 989-111, Yuseong-gu, Daejeon, Republic of Korea

*njwlee@kaeri.re.kr

1. Introduction

The head-end process in pyroprocessing converts a used PWR (Pressurized Water Reactor) fuel into feed forms suitable to the electrolytic reduction process that converts oxides to metals [1]. In the head-end process of KAERI, fuel material is first recovered in the form of fragments by slitting decladding of fuel rod-cuts. After slitting decladding, fuel material adhered to cladding is recovered in the form of U_3O_8 powder by low temperature oxidation of cladding around 500 °C. For removing the fission products of cesium and iodine during sintering of green U_3O_8 pellets in the head-end process, fuel fragments could be oxidized to be pulverized into fine U_3O_8 powder at low temperature of 500°C. The fine U_3O_8 powder cannot be directly applied to mesh-type cathode baskets with an opening size of 0.045 mm in the electrolytic reduction process. Hence, U_3O_8 powder need to be pelletized and then sintered into porous UO_2 pellets. In this present work, the modification of a small-sized commercial tablet press was performed to compact the U_3O_8 powder into green pellet in hot cell of DFDF.

2. Press modification

2.1 Modification requirements

Press must be modified to meet the following requirements:

- 1) Press must be smaller than rear-door because press must be charged through rear-door (W 730 X H 860 mm) into hot cell of DFDF.

- 2) Die filling of powder into the die must be improved by vibrator or mechanical feed system because U_3O_8 powder has a poor flowability.
- 3) Die and punches must be easily replaced.

2.2 Press size

The press size (before: W 450 X D 680 X H 740 mm, after: W 555 X D 680 X H 740 mm) was increased by an improvement of a die filling method, but press can be passed through rear door.

2.3 Die filling

Feeder scraper/hopper module for die filling of powder was designed to be driven by rack gear module connected with pinion gear of motor module as shown in Fig. 1. For vibrating filling of powder into die hole, vibration module consists of T-type pin horizontally inserted into coil spring. Whenever one tooth of pinion gear turns, vibration is generated by compression and expansion of coil spring due to repeating contact and non-contact of pin with an obstacle acting as a limit. Die filling of powder was controlled by timer and speed controller of motor module. Powder filling depth in die was controlled by the height of changeable insert blocks.

2.4 Die and punches module

The die and punches are not easy to be individually replaced by MSM in hot cell. However,

die and punches can be modularly replaced by die and punches module as shown in Fig. 2. Die and punch module can be replaced using MSM and remote tools by following procedures (Fig. 1):

- 1) move the feeder scraper/hopper module to the left side,
- 2) move the limit switch module from normal operation to replacement position,
- 3) move the feeder scraper/hopper module to right side end,
- 4) remove lock the pin and draw the die and punches module front using a L-type tool.

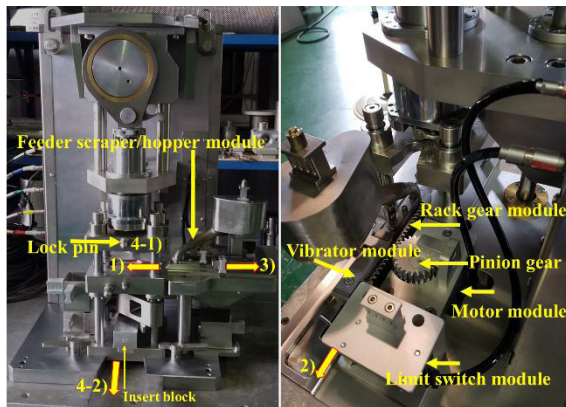


Fig. 1. Modified press and replacement procedures of the die and punches module.

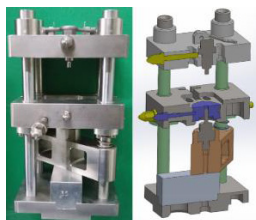


Fig. 2. Die and punches module.

3. Pre-operation test

Pre-operation test was performed using ceria powder of a very poor flowability. Filling density of ceria powder by effectively vibratory filling for three seconds (Table 1) was higher than pour density of that. Filling density of U_3O_8 powder for used nuclear fuel might be

estimated by pour density of U_3O_8 powder and ratio of filling to pour density.

Table 1. Compaction characteristics of ceria powder

Filling depth (mm)	Filling density (%TD)	Ratio of filling to pour density	Comp. pressure (MPa)	Green density (%TD)	Integrity of green pellet
8.1	1.35	1.12	119	53.2	Good
9.1	1.50	1.25	64	46.4	Poor
10.1	1.65	1.37	65	-	Poor

4. Conclusions

Die filling of powder into the die was improved by vibrator. It was confirmed by pre-operation test using ceria powder of a very poor flowability. Die and punches was easily replaced by modularization.

ACKNOWLEDGEMENTS

This work was supported by a grant from the Nuclear Research & Development Program of National Research Foundation (NRF) funded by the Ministry of Science, ICT & Future Planning (MSIP), Republic of Korea.

REFERENCES

- [1] H. Lee, et al., "Current Status of Pyroprocessing Development at KAERI", Science and Technology of Nuclear Installations, 2013, 1-11 (2013).

A Rough Estimation of the Repository Footprint of HLW for the Used Fuel Management Option

Do-Hee Ahn*, Heui-Joo Choi, HyoOn Nam, and Eun-Young Choi

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

*dhahn2@kaeri.re.kr

1. Introduction

The potential benefits of processing (reprocessing or recycling) the used nuclear fuel regarding the repository footprint, toxicity and nonproliferation have been discussed and estimated. The pyroprocessing technology linked to sodium fast reactor should be one of the perspective solutions of the used nuclear fuel management in Korea. The repository footprint of high level wastes for the used fuel management option are roughly estimated by only decay heat load and compared with the direct disposal option.

2. Estimation of the Repository Footprint for High Level Wastes

2.1 ANL in 2005 [1]

ANL quantified the potential for partitioning and transmutation to increase utilization of geologic repository space, described the heat transfer characteristics that control the loading of the repository, and identified the chemical elements that need to be removed from the spent fuel and recycled. As shown in Fig. 1, factors in increasing the loading of a repository depending on the separations efficiency were estimated with the following conditions;

- ♦ Built in volcanic rocks (tuffs) 344 m above the water table.
- ♦ Thermal criteria: 96 °C at midway between the drifts and 200 °C on the drift wall surface
- ♦ Heat removal: ventilation
- ♦ Elements separated: Pu, Am, Cm, Cs, Sr
- ♦ Reference fuel: 50 GWD/MTU
- ♦ Cooling time: 25 years
- ♦ Forced ventilation: for 75 years

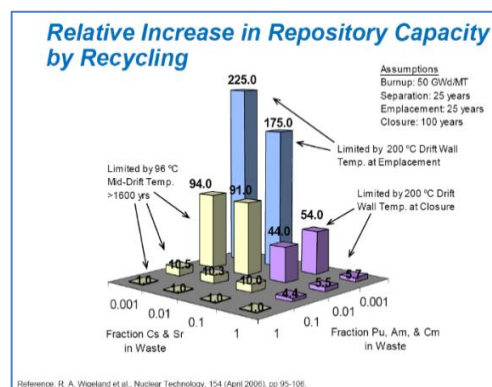


Fig. 1. Potential Repository Drift Loading Increase as a Function of Separation Efficiency for Pu, Am, Cm, Cs, Sr.

It was shown that removal of Pu, Am, Cm, Cs and Sr from used PWR fuel has the potential for reducing the size of a repository by up to a factor of 225 compared with direct disposal.

2.2 KAERI in 2017 (Preliminary Feasibility Study on Pyroprocessing) [2]

KAERI has performed the preliminary feasibility study on pyroprocessing in terms of economics, effectiveness, safety and safeguards. Based on the dynamic model of the Pyro-SFR option, the repository footprint of HLW were estimated with the following conditions;

- ♦ Built in granite 500 m below the water table.
- ♦ Thermal criteria: 100 °C in the bentonite
- ♦ Heat removal: conduction through bentonite buffer
- ♦ Elements separated: TRU, Cs, Sr
- ♦ Reference fuel: 55 GWD/MTU.
- ♦ Cooling time: 40 years

It was shown that removal of TRU, Cs and Sr by 99.9% from used PWR fuel has the potential for reducing the size of a repository by up to a factor of 270 compared

with direct disposal (13.54 km² for the 60,000 ton of PWR used fuel).

2.3 This Study

A repository footprint of HLW depends on various design features of a repository and HLW characteristics. Its detailed estimation requires much time and efforts due to many variables. In this study, repository footprints depending on separation efficiency of TRU, Cs and Sr were roughly estimated with assuming that it is directly proportional to only decay heat load of HLW and compared with those by ANL as shown in Fig. 2. Correlative relationship between two estimations is observed and the rough estimation could be used for relative comparison of a repository footprint.

Repository footprints as a function of separation efficiency of TRU, Cs and Sr from the used PWR fuel are roughly estimated as listed in Table 1. It is appeared that a repository footprint could be reduced by a factor of 10, 66 and 160 with a separation efficiency of 0.9, 0.99 and 0.999, respectively.

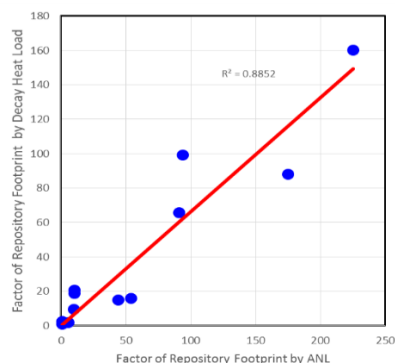


Fig. 2. Correlative Relationship between Repository Footprint Estimated by ANL and This Study.

Table 1. Repository Footprints Depending on Only Decay Heat Load of HLW with Used PWR Fuel Processing

Element	Separation Efficiency of TRU, Cs and Sr						
	0.000	0.900	0.950	0.990	0.995	0.999	1.000
TRU (watt/ton)	472.01	47.20	23.60	4.72	2.36	0.47	0.00
Cs/Sr (watt/ton)	632.09	63.21	31.60	6.32	3.16	0.63	0.00
U (watt/ton)	0.075	7.5E-05	7.5E-05	7.5E-05	7.5E-05	7.5E-05	0.00
Misc. (watt/ton)	5.83	5.83	5.83	5.83	5.83	5.83	5.83
Sum (watt/ton)	1110.00	116.24	61.03	16.87	11.35	6.93	5.83
Factor	1	10	18	66	98	160	190

3. Conclusion

A rough estimation of a repository footprint depending on decay heat load of HLW was proposed for the used fuel management option.

REFERENCES

- [1] R.A. Wigeland, et. al., "Separations and Transmutation Criteria to Improve Utilization of a Geological Repository", Nuclear Technology, 154, 95-106 (2006)
- [2] Won-Il Ko, et al., "Preliminary Feasibility Study on Pyroprocessing", KAERI/TR-6872/2017

Maintenance of Remote Handling Equipment in ACPF

SungHyun Kim* and Eun-Yong Choi

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Republic of Korea

*hyun@kaeri.re.kr

1. Introduction

The Korea Atomic Energy Research Institute (KAERI) has decided to demonstrate the electrolytic reduction process of the pyroprocess in the advanced spent fuel conditioning process facility (ACPF), a hot cell facility that has been constructed in the past. The hotcell is being improved for the composition. The existing cell was an air atmosphere hot cell, but an argon compartment was installed to improve some hot cells to argon atmosphere. In an argon atmosphere cell, an electrolytic reduction process device and a remote handling device are installed to perform process operation and maintenance.

Process equipment and cell equipment installed in the argon atmosphere cell should be operated and maintained remotely and remotely operated equipment handling it should be remotely maintained. In this study, we propose the maintenance method of the remote handling equipment remotely from the hot cell to verify the maintenance work so that the remote maintenance can be performed.

2. Maintenance of MSM

2.1 Configuration of ACPF

The existing ACPF facility is an air atmosphere hot cell. The hot cell has two cells, a process cell (M8a) and a maintenance cell (M8b), as shown in Figure 1. The ACPF facility requires an argon atmosphere cell to perform the electrolytic reduction process. Therefore, part of the process cell was modified as shown in Fig. 2, and a part of the cell was constructed with a cell of argon atmosphere. The argon compartment should be able to operate the process unit in an argon atmosphere and should not leak out. Due to the leakage of argon outside and the retrograde of existing remote handling devices (master-Slave Manipulator), existing these devices have been replaced by new ones.

All devices are operated remotely using crane and MSM. As the cell device, there are installed a temperature sensor for measuring the temperature in

the cell, a light for illuminating the cell, a camera for observing the operation in the cell, and a material transferring device for transferring the material between cells. A designed crane to ensure maximum operating area in the confined space of the compartment was installed. In addition, a crane operated from the maintenance cell to the process cell was installed.

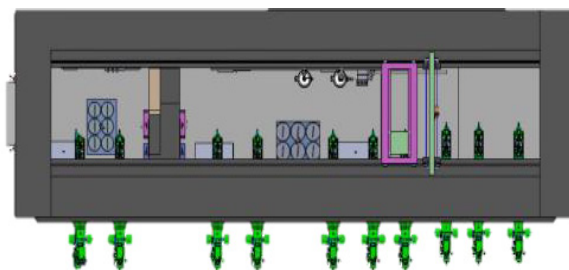


Fig. 1. Structure of ACPF.

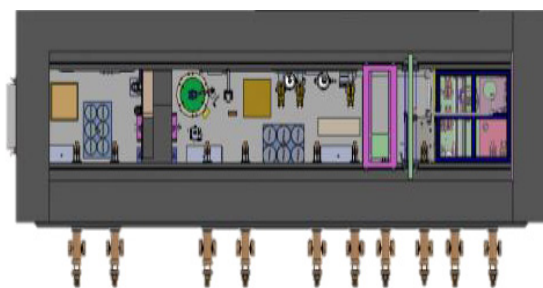


Fig. 2. Improved ACPF.

2.2 Mechanical Master-Slave Manipulator

The mechanical master-slave manipulator installed in the ACPF was a walischmiller A110 model. The MSM has a working area that can reach the rear wall of the hot cell. The MSM consists of three parts: a slave arm, a through tube, and a master arm. The MSM unit has a function of sealing the inside of the hot cell and the outside of the hot cell through a through tube installed on the wall of the hot cell, Argon gas does not leak out.

The MSM installed in the hot cell can be divided

into three parts. If a MSM has malfunctions while operating the hot cell, the corresponding arm can be removed from the hot cell wall and assembled to perform maintenance.

2.3 Maintenance of MSM

The MSM can be repaired by isolating the failed part. Since the master arm is installed outside the cell, it can be easily attached and detached. The master arm can repair the fault.

For maintenance of the slave arm, it must be taken out of the cell. To repair a failed device in a hot cell, there must be room for the decontamination of the device and for the operator to repair the device. At present, there is no room for maintenance in the ACPF. For the repair of the slave arm it was used as a way to replace it with a new slave arm. Corrosion can occur as the salt is used in the process. All materials are carried in and out through the material transport system between cells. The slave arm can't enter the argon cell using the transfer system. In the argon compartment, the spare slave arm is mounted on the wall of the hot cell so that the slave arm can be maintained without carrying it out. Two spare slave arms are installed in the argon compartment. In the process cell, three spare slave arms are installed on the wall.

Replacement of the slave arm uses a remote handling equipment crane. Separation of the through tube and the slave arm is carried out in the operating area. Maintenance work requires securing the operator's view because the slave arm is separated, fastened and transported on the hot cell window. The maintenance work was performed by using a camera to secure the operator's view. It is also important to ensure visibility using a mirror in case a camera malfunctions. In this paper, the procedure for the failure of the left and right slave arms of the work window is established and verified according to the installed position.

3. Conclusion

The maintainability of the MSM, a remote handling device installed in the hot cell, was examined and verified. In order to operate and maintain the process equipment installed in the hot cell, the remote handling device must be able to maintain its integrity. Therefore, a maintenance procedure simulating the faults of MSM was established and maintenance was performed

according to the scenario. By performing remote maintenance work on the argon compartment and the MSM in the cell, we have successfully achieved the objective.

ACKNOWLEDGEMENT

This work was supported by Nuclear Research & Development Program of National Research Foundation of Korea (NRF) funded by the Ministry of Science and ICT, Republic of Korea (MSIP)

REFERENCES

- [1] S.H. Kim, et al., "Evaluation Test of Remote Handling and Maintenance for the Ar Cell of ACPF", KAERI Report KAERI/TR-6241/2015.
- [2] S.H. Kim et al., "Evaluation Test of Remote Handling and Maintenance for the Air Cell of ACPF", KAERI Report KAERI/TR-6715/2016.

Development of Kr Adsorption Process in Pyroprocessing

Seok-Min Hong*, Ahreum Han, Jae Hwan Yang, and Yung-Zun Cho

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Republic of Korea

*seokminhong@kaeri.re.kr

1. Introduction

An accumulation of spent fuel has brought a considerable interest due to its energy and environmental issue [1]. Pyroprocessing has been investigated in Korea Atomic Energy Research Institute (KAERI) which reduces the accumulated spent fuel and increases the efficiency of fuel cycle. In head-end process of pyroprocessing, UO_2 pellets are produced from a fuel bundle and various radioactive gases from a spent fuel are released during thermal treatment. Within these gases, Kr is a volatile gas which has a long half-life of 11 years and high radioactivity, and therefore it is imperative to capture Kr [2]. Several methods such as cryogenic distillation, absorption using solvents, and adsorption on solids have been widely applied to Kr capture. Of these approaches, adsorption has many advantages such as its low energy-consumption processes and easy regeneration. In this study, Kr adsorption process was designed and tested using a packed bed column filled with adsorbents. The Kr adsorption performances were determined from breakthrough curves obtained from a gas chromatography, and the effects of adsorption conditions on capacities were investigated.

2. Experimental Section

2.1 Preparation of Adsorption Test

The adsorbents were pretreated at 130°C in a vacuum oven for overnight to remove moisture and impurities. The dried adsorbents were packed in a column of a diameter of 30 mm with flowing inert gas to prevent adsorbing water from atmosphere. The flow was $126\text{ cm}^3\text{ min}^{-1}$ and the pressure was 1 atm.

2.2 Gas Analysis

The gas chromatography (GC, DS Science, Republic of Korea) equipped with a pulsed discharge detector (PDD) was used to measure a gas concentration which

passed through the adsorption column. A 30 m of GC column (ValcoPLOT Molesieve 5A) was installed. He gas was used as a carrier gas in GC, and Kr was calibrated using standard gases. The temperature of GC was maintained at 60°C .

2.3 Breakthrough Curve

An adsorption capacity was calculated using a breakthrough time as follows:

$$q = \frac{F}{M} \int_0^t (C_0 - C) dt \quad (1)$$

where q represent adsorption capacity (mmol kg^{-1}), F is a flow rate ($\text{cm}^3\text{ min}^{-1}$), M is a mass of adsorbent (g), t is a adsorption time (min). C_0 and C are initial and measured concentration, respectively.

Breakthrough capacity (q_b) and equilibrium capacity (q_e) were calculated using a breakthrough time (t_b) and an equilibrium time (t_e), respectively. The breakthrough time is measured when Kr is detected at first time in GC, while equilibrium time is when the Kr concentration reaches to the initial feed gas concentration.

3. Adsorption Test

3.1 Adsorption Conditions

The commercial adsorbents, activated carbon and zeolite, and a novel adsorbent, metal organic framework (MOF), were selected as adsorbents.

In addition, Kr adsorption was tested with controlling parameters such as Kr concentration and adsorption temperature. From the results, it was found that the adsorption capacities were highly influenced by the Kr concentration and adsorption temperature. The both breakthrough and equilibrium capacity increase as increasing Kr concentration (Fig. 1), however, they decrease with increasing adsorption temperature.

Dynamic results were obtained through composition of gas using GC, and the adsorption performances such as working capacities, equilibrium capacity, and kinetic. Furthermore, the effects of adsorption condition on the adsorption capacities were investigated. These parameter effects will be further applied in optimizing adsorption process.

3.2 Adsorption performances

On the other hand, the feed gas in Pyroprocessing contains Hydrogen for reduction, and therefore Kr adsorption test was carried out in 3.2% H₂-balanced with Ar. It was confirmed that similar result was obtained in the presence of hydrogen which indicates a feasible adsorption process in Pyroprocessing (Fig. 2).

4. Conclusion

Adsorption process was designed for Kr capture using packed bed column with adsorbents. Kr adsorption performance was investigated by varying adsorbents and process parameters. It was found that Kr adsorption was highly influenced by Kr concentration and adsorption temperature.

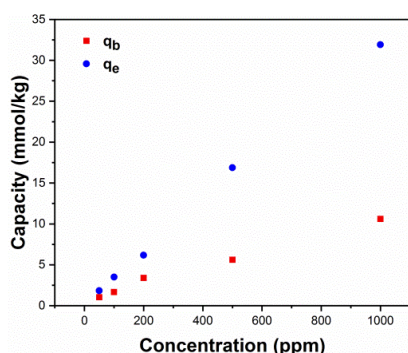


Fig. 1. Adsorption capacities at various Kr concentrations.

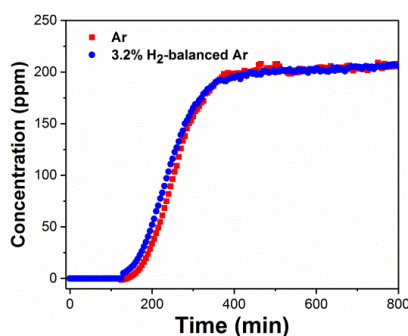


Fig. 2. Breakthrough curves measured at argon and reducing condition.

REFERENCES

- [1] L. Deliere, B. Coasne, S. Topin, C. Greau, C. Moulin, D. Farrusseng, "Beakthrough in Xenon capture and purification using adsorbent-supported silver nanoparticles", *Chemistry: A European Journal*, 2, 9660–9666 (2016).
- [2] M. Greenhalgh, T. G. Garn, J. D. Law, "Development of a hydrogen mordenite sorbent for the capture of krypton from used nuclear fuel reprocessing off-gas streams", *Journal of Nuclear Science and Technology*, 51, 476481 (2014).

Electrochemical Behavior of Tm ion and Bi ion in LiCl-KCl Molten Salt

Beom-Kyu Kim*, Hwa Jeong Han, Ji Hye Park, Won Ki Kim, and Byung Gi Park,

Department of Energy & Environmental Engineering, Soonchunhyang University,

Soonchunhyangro 22, Asan, Chungnam 336-745, Republic of Korea

*qu@sch.ac.kr

1. Introduction

The Pyro process is a process for recovering reusable uranium and TRU metals that are useful resources in spent nuclear fuel. The pyro process is actively carried out in many countries that have nuclear technology [1]. The waste molten salt generated in the pyro process contains residual actinides. Lanthanide and actinide elements coexist as various compounds in molten salt waste produced. Various process methods have been proposed for recovering and recycling them [2]. PyroRedSox process has been proposed to separate lanthanides from used salt include trans-uranium element using liquid bismuth and bismuth chloride. Pyroredsox processes consist of electrowinning and selective oxidation [3].

In this paper, the electrochemical behavior of Tm ion with bismuth ion was investigated by the electrochemical method. Also, investigated how Bi ions affect Tm ions.

2. Experimental

Electrochemical test were conducted in Glove Box. Its atmospheric environment is filled with argon gas and the concentration of oxygen and water is less than about 1 ppm.

An electrochemical cell was prepared with a one-end closed quartz tube (OD: 13 mm, ID: 10 mm), in which high purity LiCl-KCl eutectic salt (alfa-aesar, 99.99%) containing 1 wt% high purity TmCl_3 (Sigma-Aldrich, 99.99%) was placed.

A W wire (Alfa-Aesar, 99%, OD: 1mm) was used as a working & counter electrode. The reference electrode was consisted of an one-end closed Pyrex tube, in which LiCl-KCl eutectic salt containing 1wt% AgCl was placed and a silver wire (Alfa-Aesar, 99%, OD: 1 mm) was immersed in the salt.

All electrochemical measurements were performed using PAR Versastate potentiostat with Versastudio software. Temperature of the salt was measured with Chromel-Alumel thermocouple.

3. Results and Discussion

Results of Cyclic voltammograms of LiCl-KCl- TmCl_3 at 773K are shown in Fig. 1.

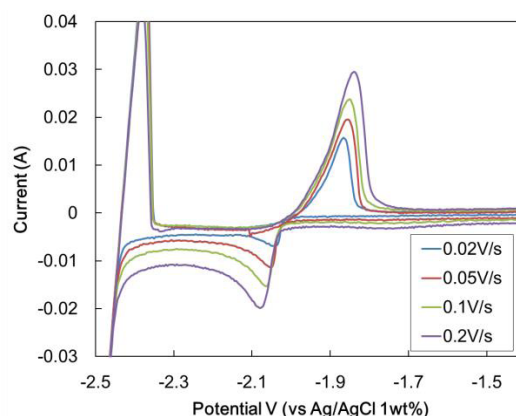


Fig. 1. Cyclic voltammogram obtained for Tm^{3+} in LiCl-KCl melt using the tungsten electrode ($A=0.664 \text{ cm}^2$) at 773K, 1wt%, scan rate: 20, 50, 100, 200 mV/s.

The values of the red-ox currents are shown in Table 1.

Table 1. The Value of the Current at 773K

Current(A)		
20mV/s	$\text{Tm} \rightarrow \text{Tm}^{3+} + 3\text{e}^-$	0.016
	$\text{Tm}^{3+} + 3\text{e}^- \rightarrow \text{Tm}$	-0.0066
50mV/s	$\text{Tm} \rightarrow \text{Tm}^{3+} + 3\text{e}^-$	0.019
	$\text{Tm}^{3+} + 3\text{e}^- \rightarrow \text{Tm}$	-0.0118
100mV/s	$\text{Tm} \rightarrow \text{Tm}^{3+} + 3\text{e}^-$	0.0234
	$\text{Tm}^{3+} + 3\text{e}^- \rightarrow \text{Tm}$	-0.0158
200mV/s	$\text{Tm} \rightarrow \text{Tm}^{3+} + 3\text{e}^-$	0.029
	$\text{Tm}^{3+} + 3\text{e}^- \rightarrow \text{Tm}$	-0.0193

We gradually added BiCl_3 in LiCl-KCl- TmCl_3 to investigate the effect of Bi ion on the Tm ion. The results of the measured CV are shown in Fig. 2.

Comparison of the cyclic voltammograms obtained in LiCl-KCl- TmCl_3 and after the addition BiCl_3 (0.2~2 wt%) is shown in Fig. 2. The results are consistence with those obtained Tm peak, correspond to the formation of a Tm^{3+}/Tm in the first time. And Cathodic/anodic peak around 0.1~0.6V should be

ascribed to the deposition/dissolution of Bi Metal, respectively. Many anodic and cathodic peaks Tm-Bi alloys observed, so expected to the formation of Tm-Bi intermetallic compounds.

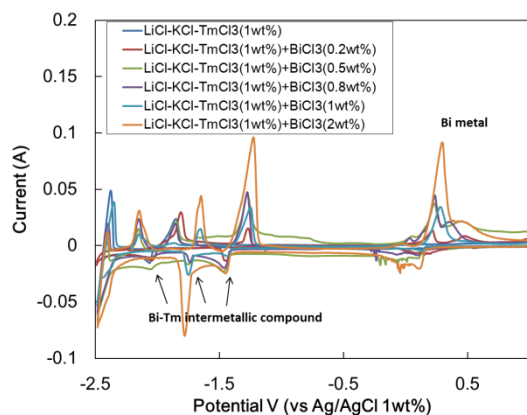


Fig. 2. Cyclic voltammograms of $\text{LiCl-KCl-TmCl}_3(2\text{wt}\%)$ at with $\text{BiCl}_3(0.2\sim 2\text{wt}\%)$ using tungsten electrode and scan rate 100mV/s .

In order to consider the reduction of intermetallic compound, potential was applied in the positive to negative direction. As a result of Fig. 3, various reduction peaks current appeared. It is estimated the reduction of Bi-Tm intermetallic compound. Also, as Bi ion more increased, the reduction peak of Tm, which appeared at $0.2\text{wt}\%$, disappeared.

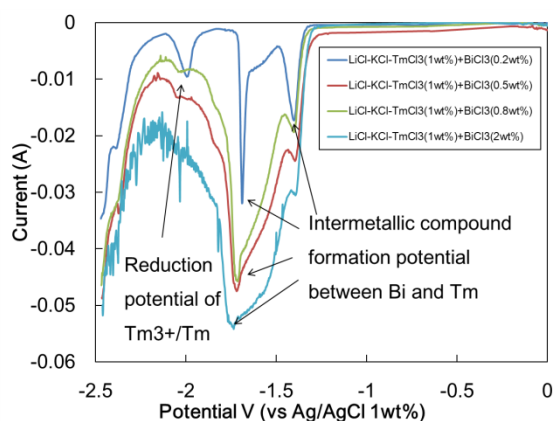


Fig. 3. Square Wave Voltammogram for the reduction (from 1 to -2.5V) of Tm ion with Bi ion (Tm-Bi alloy) in the LiCl-KCl on the tungsten electrode ($A=0.664\text{cm}^2$) at 773K .

4. Summary

This study, electrolytic behavior of Thulium and Tm-Bi ion system was studied. The electrochemical behavior of Tm was studied in LiCl-KCl-TmCl_3 molten salts using electrochemical techniques Cyclic Voltammetry on tungsten electrodes at 773K . During

the process of CV and SWV, intermetallic compound were observed Bi-Tm.

Further study, in order to determine clarity of diffusion coefficient in this experiment, we will compare result of electrochemistry method and we also need to quantitative research.

REFERENCES

- [1] H. Lee, G.-I. Park, K.-H. Kang, J.-M. Hur, J.-G. Kim, D.-H. Ahn, Y.-J. Cho, and E.-H. Kim, "Pyroprocessing Technology Development at KAERI", Nuclear Engineering and Technology, 43 (4), 317-328 (2011).
- [2] Kensuke Kinoshita et al., "Separation of uranium and transuranic elements from rare earth elements by means of multistage extraction in LiCl-KCl/Bi system," Journal of Nuclear Science and Technology, Vol. 36, pp. 189-197 (1999).
- [3] H. O. Nam, "All the Spent Nuclear Wastes to Low and Intermediate Level Wastes: PyroGreen", Proceedings of Global 2009, Paris, France, September 6-11 (2009).

Evaluate of the Spent Nuclear Fuel Rod Model Using the Monte Carlo Simulation

Hwa Jeong Han, Beom-Kyu Kim, Ji Hye Park, Won Ki Kim, and Byung Gi Park*

Department of Energy & Environmental Engineering, Soonchunhyang Univ.,

Soonchunhyang-ro 22, Asan, Korea

*Corresponding author: byunggi@sch.ac.kr

1. Introduction

The safeguards verification, material control and accountability requirements of a reprocessing facility are costly and very time consuming because they mainly rely on destructive analysis. With the growth occurring in the nuclear industry and the possibility of closing the fuel cycle by embracing fuel recycling, the number of reprocessing facilities is likely to grow. The IAEA will need supplemental systems that are less man-power intensive to provide verification assurance [1]. The CFOGRS is a next generation safeguarding detection system which is non-destructive, autonomous, and operates in near-real-time. The efficacy the gamma ray detection technique to the spent nuclear fuel rod has been evaluated. The goal of this work is to develop the spent nuclear fuel rod model and to evaluate the developed model to measure radiation of spent nuclear fuel using CFOGRS developed previously work. This model can be used as a radiation source to measure of the gamma ray emitted.

2. Modelling and Results

2.1 Modelling of the Spent Nuclear Fuel Rod

The Monte Carlo simulation package GEANT4 has been used to develop a model of the spent nuclear fuel rod. The decay of the fuel rod was constructed based on the Radioactive Decay libraries, which is provided with the GEANT4. And gamma ray emitted from the fuel rod were detected by Cerenkov-based FOGRS. This sensor was modeled based on the Optical libraries.

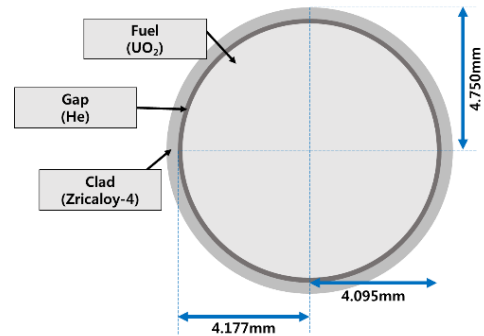


Fig.1. Plant view of a single fuel rod with the geometric properties.

A single spent fuel rod was designed to consist of a three-layer cylindrical tube which consists of Fuel, Gap, and Cladding. Where a single fuel rod consists of enriched uranium, in the form of UO₂ pellets, contained in Zircaloy-4 tubing. And rods are pressurized internally with helium during fabrication to reduce clad creepdown during operation and thereby prevent clad flattening. A schematic layout of modeled fuel rod is presented in Fig. 1.

2.2 Source of the Spent Nuclear Fuel Rod

Complex source lists of modeled spent nuclear fuel rod samples were generated using irradiation and decay calculations performed using ORIGEN-ARP and a Matlab script to convert the output files to a format recognized by GEANT4. In ORIGEN, the Express Form was used to define fuel parameters, burnup, irradiation pattern, cooling time, and enrichment. The sample rod was in the reactor for a total of five cycles. The initial enrichment of the rod was 2.94%. The burnups have been cooling for approximately 28 years [2].

2.3 Simulation results

The entire structure is the square-shaped FOCRS, which consists of four CFOGRS. When four CFOGRS were irradiated by gamma rays, the four separate Cerenkov radiations generated in the optical fiber were guided to the MPPC. The output file was written wavelength, which was calculated by convert equations.

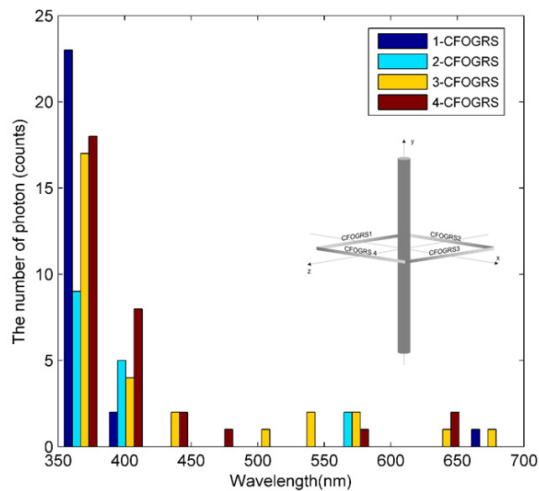


Fig.2. Wavelength spectra of detected Cerenkov photons.

3. Conclusion

In this work developed the spent nuclear fuel rod model and evaluated the developed model to measure radiation of spent nuclear fuel using CFOGRS. Spent nuclear fuel rod was modeled the general type of fuel rod. And the complex source of fuel rod was defined refer to the previous research. The gamma rays emitted from the modeled fuel rod was detected using CFOGRS. The simulation results showed that the CFOGRS can be detected gamma rays emitted from the source of the fuel rod. This model only can see the total gamma ray. Therefore, this work should be supplemented in the future. In order to be suitable for the purpose, it is considered that further improvement of this model.

REFERENCES

- [1] C. R. Orton, C. G. Fraga, R. N. Christensen, and J. M. Schwantes, "Proof of concept simulations

of the Multi-Isotope Process monitor: An online, nondestructive, near-real-time safeguards monitor for nuclear fuel reprocessing facilities," Nuclear Instruments and Methods in Physics Research Section A: Accelerators, Spectrometers, Detectors and Associated Equipment, 629(1), 209-219 (2011).

- [2] Sarah E. Bender, "Application of Monte Carlo modelling of Compton suppression spectroscopy to spent fuel material accountancy", M.S. thesis, Dept. Nuclear Eng., The Pennsylvania State Univ., 2011.

Design of Furnace Tray for Improved UO_2 Fragment Oxidation Rate

JuHo Lee*, Yung-Zun Cho, Sang-Chae, Jeon, and Young-Soon Lee

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989 beon-gil, Yuseong-gu, Daejeon, Republic of Korea

*jhlee835@kaeri.re.kr

1. Introduction

In KAERI, porous UO_2 pellet is considered as feed material for oxidation reduction of Pyroprocessing. Since the fabrication of porous UO_2 pellet begins with U_3O_8 powder produced by oxidation of UO_2 fragment recovered from fuel rod slitting, for the optimal oxidation of UO_2 fragment, a study is required to examine UO_2 fragment oxidation in macroscopic scope distinguished from the conventional study aimed at microscopic level [1].

2. Experiment

2.1 Equipment change

The present PRIDE furnace was slightly modified since the large heating zone compared to the amount of sample and gas circuit prevent the rapid response of oxygen concentration in output gas stream. An assembly of alumina crucibles was installed in the heating zone to reduce the control volume involved in oxidation process and part of discharged output stream was introduced to the furnace to maintain furnace pressure above the specified value.

2.2 Experimental Procedure

The sample with known amount of UO_2 fragment is introduced into quartz beaker of 250 ml and

located in the assembly of alumina crucibles. After fresh air is introduced into the furnace at a flowing rate of 2 liter/min, furnace is heated to 500°C at a heating rate of $5^\circ\text{C}/\text{min}$ and maintained until the cell voltage of output gas stream, an alternative expression of oxygen concentration, reaches to maximum. A moisture trap was installed into the sampling loop of oxygen analyzer of output gas stream to minimize the effect of moisture on oxygen measurement. After oxidation process is ended, the oxidized fragments are recovered and the density and particle size distribution to the depth of produced U_3O_8 powder are evaluated. Two samples are used; 230 g UO_2 fragment corresponding to packing height of 1 cm and 345 g of 1.5 cm.

3. Results

Fig. 1 demonstrates that oxidized U_3O_8 powder is expanded to about four times its UO_2 fragment volume. Reaction time was estimated approximately to be 6 hr 20 min for packing height of 1 cm and 8 hr of 1.5 cm and both samples were found to be completely oxidized to U_3O_8 powder based on the evaluated weight change. Table 1 shows the density of U_3O_8 powder of two samples to the powder depth. It implies that the powder in lower region are relatively more compacted than that in above region and thus density becomes increased as the powder depth increases.

Table 1. Density of U_3O_8 powder to the powder depth

Depth interval (cm)	Density (g/cm^3)	
	Packing height of 1 cm	Packing height of 1.5 cm
0 – 2	1.92	2.04
2 – 4	1.35	1.70
4 >	-	1.37

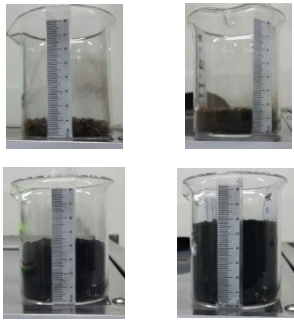


Fig. 1. Volume expansion of two samples after oxidation: packing height of 1 cm (left) and 1.5 cm (right).

4. Conceptual Design of furnace tray

As exhibited in Table 1, the density of U_3O_8 powder increases with depth since upper part of U_3O_8 powder exerts pressure against below part of powder. This implies that the improved oxidation rate of UO_2 fragment can be achieved by continually eliminating upper part of growing U_3O_8 powder preventing oxygen penetration to lower part. Fig. 2 shows the furnace tray satisfying such characteristic by inducing overflow of powder; as oxidation proceeds, part of growing U_3O_8 powder is discharged into the hole and thus the unreacted UO_2 fragment is likely to be more readily oxidized compared to the oxidation with simple tray

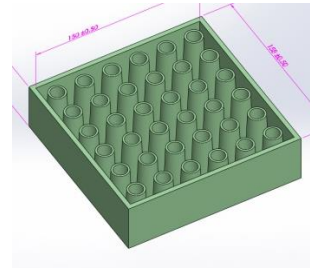


Fig. 2. Proposed furnace tray.

5. Conclusion

The effect of packing height on oxidation behavior is evaluated through oxidation using quartz beaker. The oxidized U_3O_8 powder was found to expand approximately four times the original volume of UO_2 fragment regardless of packing height. However, the U_3O_8 packing density of bottom was increased as the packing height increases. The proposed furnace design is expected to yield improved oxidation rate, however, the demonstration and the revision of the furnace design is currently in progress.

REFERENCES

- [1] P. Taylor, "Thermodynamic and kinetic aspects of UO_2 fuel oxidation in air at 400 – 2000K", *Journal of Nuclear Material*, 344, 206-212 (2005).

The Fabrication and Characteristics of Gd_2O_3 Particle Reinforced Aluminum Matrix Composite

Tae Gyu Lee and Ho Jin Ryu*

Department of Nuclear and Quantum Engineering, Korea Advanced Institute of Science and Technology (KAIST),
291 Daehak-ro, Yuseong-gu, Daejeon 305-701, Republic of Korea

*hojinryu@kaist.ac.kr

1. Introduction

Neutron absorbing materials are very important for spent nuclear fuel (SNF) management, such as SNF transport and SNF storage. Neutron absorbing materials not only maintain sub-criticality but also conduct roles of mechanical support and heat spreader.

Recently, B_4C reinforced composite is used for neutron absorbing materials because of its high neutron absorbing capacity [1] and structural integrity. However, B_4C reinforced composite has its limitation of brittleness. When contents of B_4C increased, the brittleness of B_4C reinforced composite also increases [2].

Gd_2O_3 is the most promising candidate for replacing B_4C due to its superior neutron absorbing ability. Gd has 60 times higher thermal neutron capture cross section so that it has 15 times better neutron capture ability than B_4C when Gd_2O_3 is used for reinforcement of composite.

By adopting Gd_2O_3 as a reinforcement of composite, 15 times smaller reinforcement contents are needed than B_4C composite for making neutron absorbing materials that have same neutron capture ability which means the composite can maintain its own ductility and thermal conductivity.

However, characteristics of Gd_2O_3 and Gd_2O_3 reinforced composites have to be investigated to check its feasibility for using actual neutron absorbing materials. The characteristics that should be checked are mechanical properties, thermal properties, and chemical reaction between matrix alloys.

In this study, bulk Gd_2O_3 and Gd_2O_3 reinforced composite were fabricated and characterized by Vickers hardness test, tensile test and scanning electron microscope (SEM).

2. Experimental procedure

2.1 Sample preparation

The fabrication procedure of bulk Gd_2O_3 consists of the following steps. First, Gd_2O_3 powder of 1 μm diameter was compacted by a uni-axial press with 400 MPa pressure and sintered at 1500 $^{\circ}C$ for 1, 2, 3 hours in air mood.

The fabrication procedure of bulk Gd_2O_3 reinforced composite consists of the following steps. 20 μm , 60 μm , and 100 μm sized Gd_2O_3 powder were mixed with aluminum alloy 7075 powder with 1.5 : 98.5 ratio by a 3D mixer, 40 RPM. The mixtures were pressurized at 50 MPa for sintering in a spark plasma sintering machine, heated to 450 $^{\circ}C$ for 5 minutes and held for 5 minutes.

2.2 Hardness test

The surface of bulk Gd_2O_3 and Gd_2O_3 reinforced composite for hardness test was polished to 2000grit using SiC polishing paper. The hardness was measured through Vickers hardness tester.

2.3 Tensile test

For the tensile test, the sintered samples were cut to a size of 2.7 * 2.0 * 11 cubic mm. Then, the tensile test was performed using a universal testing machine (UTM, Instron 5583, Instron Corporation, USA). At this time, the test was performed following ASTM E8.

2.4 Microstructure

The dispersion of the reinforcement and the microstructure of the specimen were observed with a

scanning electron microscope (SEM). To determine the mode of failure of the material, the fracture surface after the bending test was also observed with SEM.

3. Result

3.1 Hardness test

The bending test has not yet been carried out.

3.2 Tensile test

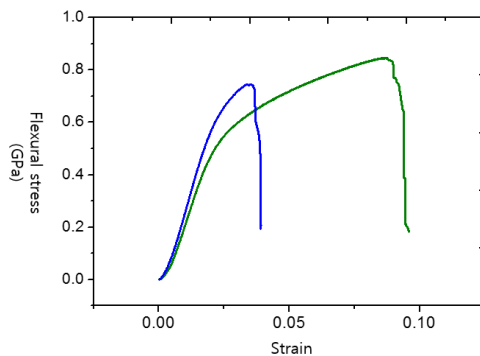


Fig. 1. Comparison of Gd_2O_3 1.5 vol% reinforced composite (green) and B_4C 20 vol% composite (blue) in bending test.

The tensile test has not yet been carried out, but according to bending test results (Fig. 1), the maximum strength is increased and maximum strain increased almost twice.

3.3 Microstructure

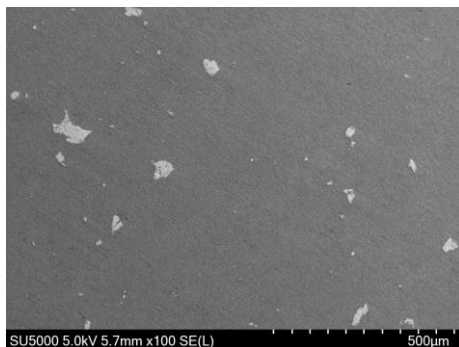


Fig. 2. Cross section image of 1.5 vol% Gd_2O_3 composite.

The Gd_2O_3 reinforced composite shows well - dispersed microstructure and no macro scale secondary phase (Fig. 2).

4. Conclusion

In this study, characteristics of bulk Gd_2O_3 and Gd_2O_3 reinforced composite were investigated. The bending test result shows that two times higher maximum strain than 20 vol% B_4C reinforced composite. The microstructure analysis shows the well dispersion of Gd_2O_3 and no macro secondary phase.

In conclusion, Gd_2O_3 reinforced composite has no disadvantages in terms of mechanical strength and interfacial chemical reaction with aluminum 70705 matrix.

With proper further studies, Gd_2O_3 reinforced composite can replace the B_4C reinforced composite as a next generation neutron absorbing material.

5. Acknowledgments

This work was supported by the National Research Foundation of Korea (NRF, No. 2014M3C1A-9060721) grant funded by the Ministry of Science, ICT and Future Planning.

REFERENCES

- [1] Zhang, P., Li, Y., Wang, W., Gao, Z., & Wang, B., "The design, fabrication and properties of B 4 C/Al neutron absorbers." *Journal of Nuclear materials*, 437(1), 350-358 (2013).
- [2] Viala, J. C., Bouix, J., Gonzalez, G., & Esnouf, C., "Chemical reactivity of aluminium with boron carbide.", *Journal of Materials Science*, 32(17), 4559-4573 (1997).

Technical Issues of Remote Assembler for TRU Fuel Assembly

Young-Ho Lee*, Sang-Gyu Park, Ki-ho Kim, Jeong-Yong Park, and Chan-Bock Lee

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Republic of Korea

*leeyh@kaeri.re.kr

1. Introduction

Recently, the amount of accumulated spent fuels will exceed the capacity of the interim storages because domestic nuclear power plants discharges approximately 760 tons of spent fuel per year. A prototype generation-IV sodium-cooled fast reactor (PGSFR) is being considered as a future nuclear system to reduce the amount of LWR spent fuels and radiotoxicities by recovering TRU materials from the spent LWR fuels through pyro-electrochemical processing (PEP) [1]. Thus, complete nuclear fuel cycle can be achieved by remotely fabrication of TRU element, rod and fuel assembly between PEP and PGSFR. However, the fabrication of TRU fuels should be undergone with completely remote operation in radiation-shielded hot cell. Especially, there are lack of experiences on the design of remote fabrication facility for TRU fuel assembly except FCF (Fuel Cycle Facility) at EBR-II (Experimental Breeder Reactor II, U.S.) in 1960s [2,3]. Fig. 1 shows the schematic view of fuel assembly for PGSFR. In this study, design criteria for remote assembler for TRU fuel assembly are reviewed and technical issues for remote fabrication of TRU fuel assembly are proposed.



Fig. 1. Schematic view of fuel assembly for PGSFR.

2. Design criteria and technical issues

2.1 Assumption

Fuel rods are remotely fabricated in pre-processes from fuel elements to wire wrapping and final products of fuel rods are vertically stored near the

assembling machine with adequate cooling system for removing decay heat of fuel elements.

- The structural parts of fuel assembly are manufactured and supplied with two preassemblies (i.e., upper and lower preassemblies) outside of the hot cells by normal shop procedures. Upper preassembly is consisted of handling socket, upper shield and duct, and lower preassembly is noise piece and lower shield.
- Fabrication procedure of fuel assembly can be divided into three main steps; (1) Rod bundle assembling, (2) Assembling upper and lower preassemblies and duct welding process, (3) Dimension inspection and non-destructive examination (NDE) of welds.

2.2 Design criteria

The objective of the remote assembler is to assemble the TRU fuel assemblies that can be loaded in SFR. The operations of assembling procedures should be done by totally remote methods. Especially, fuel rods should not be inclined from the vertical during the entire assembling process because of redistribution of bonded sodium by their decay heat.

- The machine should be designed to have modular parts to maintaining remotely in the hot cell with master-slave manipulation (MSM), overhead cell crane and viewing equipment. All parts of the machine inserted through a port within a limited diameter and handled with a specific weight limit, each part of the machine should be considered its size and weight.
- During the assembling operation, the application of complex mechanisms should be avoided and mechanical MSMs are to mainly be used. The use of overhead cell crane or other different-type MSM should be avoided or minimized because these in-cell facilities are not always ready to use.
- Radiation-shielded materials should be used for

all parts of the machine, which can be substituted simplified replacement procedure.

- Maximum temperature of the TRU fuel assembly should be below allowable temperature determined by the properties of the final PEF products.
- During the operation of assembling procedure, the working hours should be limited.

Table 1. Summary of technical issues of each remote manufacturing steps at the assembling machine

Steps	Technical issues
Rod bundle	• Fuel rod remote handling method with MSM (Length: 2,250 mm, OD: 7.4 mm) • Tolerance control of bottom end cap of fuel rod, mounting rail, lower shield that are assembled each other • Maintaining rod bundle shape during assembling procedure
Mating assemblies and welding	• Guiding methods and maintaining straightness for vertically mating of two preassemblies • Handling methods of heavy structural parts without additional in-cell equipment • Application of welding and post weld heat treatment (PWHT) methods in hot cell
Inspection and NDE	• Application of measuring sensors for fuel assembly dimensions with high irradiation resistance • Selection of remote NDE methods for welding parts and definition of defect threshold size for quality control • Application of tensile tests for welds and fixing deformed duct with a straightener
Common points	• Application of pneumatic system of each assembling steps and specification of cooling system (flow rate and positions) • Definition of allowable temperature in assembler, inspector, welding machine • Remote maintenance of the assembling machine

2.3 Technical issues of a remote assembler

Based on the above assumption of the TRU fuel assembly fabrication procedure, predictable technical issues of each assembling step are summarized in Table 1. For example, the length of the fuel rod in

PGSFR is 2,250 mm and more than two times longer when compared with design data at EBR-II. So, the fuel rod for PGSFR is too slender to handle remotely with single MSM without auxiliary tools. This means that an additional jig should be designed for maintaining the vertical of the fuel rod as well as resolving small tolerance between lower end cap of fuel rod and mounting rail.

3. Summary

In this study, assembling procedure of TRU fuel assembly was reviewed and divided into rod bundle assembling, mating preassemblies and welding, and inspection and non-destructive examination. Based on this assumption, the design criteria of a remote assembler for TRU fuel assembly of PGSFR is defined and predictable technical issues are proposed.

ACKNOWLEDGEMENTS

This work was supported by the National Research Foundation of Korea funded by the Ministry of Science and ICT(2017M2A8A5014888).

REFERENCES

- [1] C.B. Lee, J.S. Cheon, S.H. Kim, J.Y. Park, and H.K. Joo, "Metal Fuel Development and Verification for Prototype Generation IV Sodium-Cooled Fast Reactor", Nuclear Engineering and Technology, 48(5), 1096-1108 (2016).
- [2] C.E. Stevenson, "The EBR-II Fuel Cycle Story," American Nuclear Society, LaGrange Park, IL (1987).
- [3] "A State-of-Art Report on Remote Fabrication Technology Development for EBR-II Fuel," KAREI/AR-791/2008.

Thermal Conductivity Modeling of U-TRU-RE-Zr Metal Fuel for PGSFR

Byoung Oon Lee*, Chan Bock Lee, June Hyung Kim, Jin Sik Cheon, Jeong Yong Park, and Sung Ho Kim
Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Republic of Korea
*boleee@kaeri.re.kr

1. Introduction

The U-TRU-Zr metal fuel for PGSFR (Prototype Gen-IV Sodium-cooled Fast Reactor) is being developed in combination with the pyro-processing. Since RE (Rare Earth) and MA (Minor Actinide) will be included in the metal fuel, the behavior of RE and MA in the metal fuel is needed to be evaluated. In this paper, the thermal conductivity model of U-TRU-RE-Zr was developed based on the theory of two phase solid bodies using the existing thermal conductivity model of U-Pu-Zr alloy, the available data in the literature and the thermal conductivity theories of the solid solution.

2. Thermal conductivity model

2.1 Thermal conductivity model of the two phase solid bodies

RE and some MA (such as Am and Cm) are precipitated in the U-Pu-Zr matrix. The effect of the precipitates on the unirradiated thermal conductivity for U-TRU-RE-Zr was evaluated based on the Bruggeman model [1]. The thermal conductivity of U-PU-Zr alloy fuel is based on the model by Billone [2].

Fig. 1 compares the calculated thermal conductivities for U-19Pu-10Zr-(MA-RE) using the Billone and Bruggeman models with the data measured in METAPHIX [3].

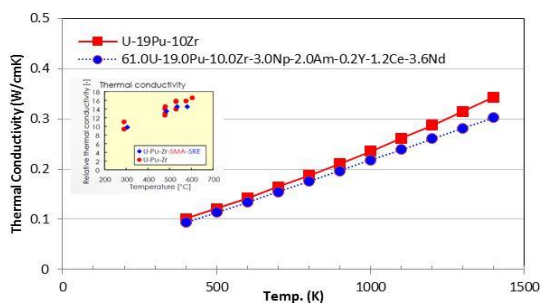


Fig. 1. Calculated and measured thermal conductivity for U-19Pu-(MA-RE)-Zr.

According to the METAPHIX results, the thermal conductivity decreases by about 10% when containing 5wt% MA and 5wt% RE. Bruggeman's model also predicts about 10% reduction of the thermal conductivity. So it is estimated that Bruggeman's model has a good capability for predicting the thermal conductivity of U-TRU-RE-Zr.

2.2 Adjusted thermal conductivity model of U-Pu-Zr alloy

As shown in Fig. 2, Billone's model shows a very low thermal conductivity value according to Pu or Zr contents. So, the U-Pu-Zr thermal conductivity model is needed to be adjusted for the case where the Zr or Pu content is greatly increased in the fuel.

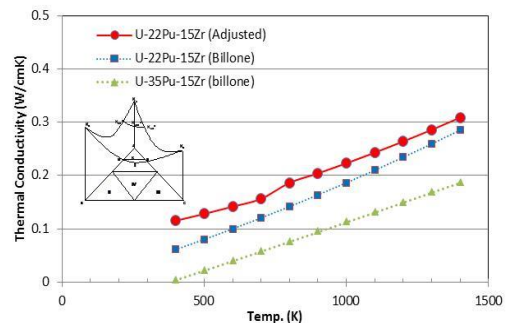


Fig. 2. U-Pu-Zr thermal conductivity comparison.

The ternary solid solution theory by Zarichnyak [4] was applied for the development of U-Pu-Zr alloy thermal conductivity.

Fig. 2 also shows the comparison of thermal conductivities. It was estimated that the value by Zarichnyak's model is higher than that of Billone's model for the higher Zr contents.

2.3 Thermal Conductivity of U-TRU-RE-Zr

U-TRU-RE-Zr thermal conductivities based on the Bruggeman's model were calculated using the adjusted U-Pu-Zr model as well as Billone's model. Fig. 3 shows the comparison the measured data [5] with the calculated thermal conductivities.

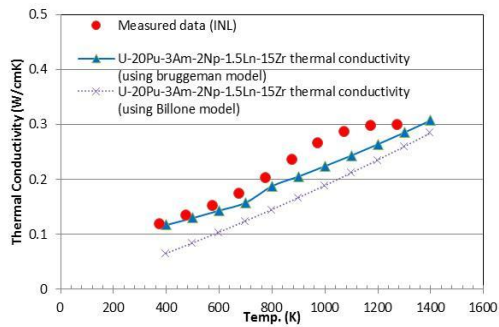


Fig. 3. Calculated and measured thermal conductivity for U-TRU-RE-Zr.

Even though the thermal conductivity model in this paper is likely to be conservative, it is estimated that the prediction by the adjusted model has a reasonable capability.

3. Conclusion

The thermal conductivity model of U-TRU-RE-Zr for PGSFR was developed and compared with the measured data. Even though more experimental data are needed, the developed models have a good capability to calculate the thermal conductivity. This work forms the basis for the establishment of key technology that will evaluate the performance of U-TRU-RE-Zr metal fuel.

ACKNOWLEDGEMENT

This work has been carried out under the National Nuclear R&D program supported by the Ministry of Science, ICT and Future Planning.

REFERENCES

- [1] J.P.Stora, Nuc. Tech., 17, March (1973).
- [2] M. C. Billone, et al., Proc. ANS Conf. Reliable Fuels for LMRs, Tucson, (1986).
- [3] T. Ogata et al., 11IEMPT, San Francisco, U.S.A., November 3, (2010).
- [4] Yu. P. Zarichnyak, et al., J. Engineering Physics, 435(1978).
- [5] J. Rory Kennedy et al., INL/EXT-07-13128, September (2007).

Re-fabrication of U-Zr Alloy System Fuel Slugs Recycling Fuel Scraps

Ki-Hwan Kim^{1,*}, Seong-Jun Ha^{1,2}, Seung-Uk Mun^{1,2}, Young-Mo Ko¹, Seok-Jin Oh¹, and Jeong-Yong Park¹

¹ Next Generation Fuel Development Division, KAERI, 111, Daedeok-daero 989beon-gil,
Yuseong-gu, Daejeon, Republic Korea

² Advanced Materials Engineering Department, Chungnam National University, 99, Daehak-ro,
Yuseong-gu, Daejeon, Republic Korea

*Corresponding author: khkim2@kaeri.re.kr

1. Introduction

Sodium cooled fast reactors (SFRs) are considered Gen-IV reactors. The spent fuel generated in nuclear power plants is converted into uranium and TRU, a raw material for SFRs [1]. The U-TRU-Zr-RE metal fuel slug has been developed as a candidate fuel slugs for SFR. It has many advantages such as simple fabrication procedures, good neutron economy, high thermal conductivity, excellent compatibility with a Na coolant, and inherent passive safety [2-3]. It is fabricated using an injection casting method. The efficient fabrication process of U-TRU-Zr-RE metal fuel slug is largely based on an injection casting method. Nevertheless, reducing radioactive waste has been an important challenge worldwide and thus recycling of the slug should be considered. A considerable amount of fuel scraps upto 50% of charge amount for injection casting has been made consisting of the heel of the melt residue and unsuitable fuel slugs.

In this study, to increase the yield rate of injection casting, metallic fuel slugs were re-fabricated using fuel scraps consisting heel residue and unsuitable fuel slugs. The metallic fuel scraps have been treated on the impurity layer of the surface using chemical and mechanical method. And then, metallic fuel slugs have been re-fabricated for the recycling of the fuel scraps. After the surface treatment, metallic fuel slugs were re-fabricated using a heel residue and fuel slug butts as the raw material through an injection casting apparatus to evaluate the feasibility of the recycling of the fuel slug scraps.

2. Methods and Results

2.1 Experimental Methods

The metallic fuel scraps such as heel residue and

slug butts of U-10wt.%Zr and U-10wt.%Zr-RE(rare-earth elements) fuels after injection casting were used as raw materials for re-fabrication of fuel slugs. RE is a rare-earth alloy consisting of 53wt%Nd, 25wt%Ce, 16wt%Pr, and 6wt%La. The metallic fuel scraps have been treated on the impurity layer of the surface by either chemical or mechanical method. Casting variables, e.g., casting temperature and pressure, pressurizing rate, mold coating method were adjusted with graphite crucibles coated with ceramic plasma-spray coating and quartz molds coated with slurry-coating. At a predetermined superheat, the mold was lowered with pressurization of atmospheric gas, immersing the open tip into the metal melt. The metallic fuel slugs were fabricated using recycled metallic fuel scraps by injection casting method.

To examine the soundness of the metallic fuel slugs, the density of the metallic fuel slugs was measured using an Archimedeian immersion method. The alloy compositions of the metallic fuel slugs were investigated using inductively coupled plasma atomic emission spectroscopy (ICP) and an elemental analysis (EA). The microstructure and the composition of the metallic fuel slugs were analyzed using scanning electron microscopy (SEM) and energy-dispersive spectroscopy (EDS).

2.2 Experimental Results

Metallic fuel slugs consisting of U-10wt%Zr and U-10wt%Zr-RE with a diameter of about 5.5 mm and a length of about 300 mm were fabricated per batch. They were generally sound without cracks or thin sections, as shown in Fig. 1. Average density of the U-10wt%Zr fuel slugs were measured as 15.7g/cm³. Average density of the U-10wt%Zr-RE fuel slugs was measured as 14.9 g/cm³. The density of the metallic fuel slugs slightly differed according to the composition of fuel alloys. The measured densities of

the metallic fuel slugs were considerably similar to the theoretical density of metallic fuel slug.

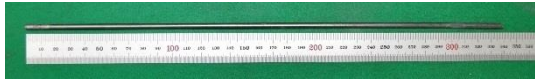


Fig. 1. Typical re-fabricated metallic fuel slug recycling fuel scraps.

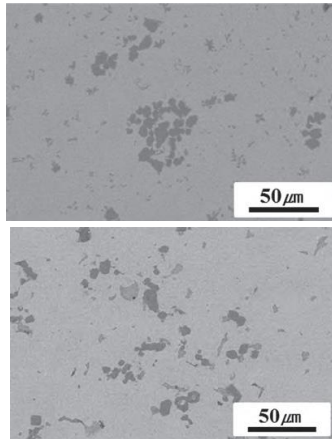


Fig. 2. Typical cross-section SEM images of re-fabricated metallic fuel slug; (a) U-10wt.%Zr, (b) U-10wt.%Zr-7wt.%RE.

Table 1. Chemical composition of recycled metallic fuel slugs

Alloy Composition	U-10wt.%Zr	U-10wt.%Zr-7wt.%RE
U (wt.%)	90.6	86.1
Zr (wt.%)	9.6	10.1
Nd (wt.%)	-	2.0
Ce (wt.%)	-	1.3
Pr (wt.%)	-	0.5
La (wt.%)	-	0.2
C (ppm)	87	237
N (ppm)	10	10
O (ppm)	653	430
Si (ppm)	239	379

Fig. 2 shows the microstructures of the respective metallic fuel slugs analyzed using SEM. All metallic fuel slugs were fabricated using heel scraps as the raw material. Metallic fuel slugs were fabricated using metallic fuel scraps added to a pure metal material. The precipitates were distributed uniformly at the top, middle, and bottom positions. A lot of Zr-rich precipitates or RE-rich inclusions were observed randomly in the metallic fuel slug. RE elements was separated as an inclusion with U-Zr alloy because of

immiscibility. From the specifications of the fuel slugs, the total impurities of carbon, nitrogen, oxygen, and silicon must be less than 2,000 ppm. The chemical analyses of the fuel slugs showed that the total impurities were satisfied with the specification requirements, as shown in Table 1.

3. Conclusion

The recycling of metallic fuel scraps is necessary to maximize the utilization of the uranium resources. Metallic fuel slugs were re-fabricated using metallic fuel scraps including heel residue and unsuitable fuel slugs. The fuel slugs were generally sound and fabricated to the mold length of 300 mm. The total impurities of oxygen, carbon, nitrogen, and silicon were less than 2,000 ppm for the recycled metallic fuel slugs. The feasibility of the recycling of the fuel slug scraps has been demonstrated by the re-fabrication of the metallic fuel slugs. The recycling of the metallic fuel scraps would be able to contribute to the maximization of yield rate for the fabrication process of the fuel slugs.

4. ACKNOWLEDGEMENTS

This work has been carried out under the Nuclear Research and Development Program supported by the Ministry of Science and Technology in the Republic of Korea.

5. REFERENCES

- [1] L. C. Walters, B. R. Seidel, J.H. Kittel, "Performance of Metallic Fuels and Blankets in Liquid-Metal Fast Breeder Reactors. Nuclear Technology", 65, 179-231, 1984.
- [2] G. L. Hofman, L. C. Walters, T. H. Bauer, "Metallic Fast Reactor fuels". Progress in Nuclear Energy, 31, 83-110, 1997.
- [3] Crawford DC, Porter DL, Hayes SL. "Fuels for Sodium-cooled Fast Reactors: US Perspective", Journal of Nuclear Materials, 371, 202-231, 2007.

Improvement of Spent Fuel Inspection System

Mu-gap Shin* and Young-hwan Kim

KEPCO E&C, 269 Hyeoksin-ro, Gimcheon-si, Gyeongsangbuk-do, 39660, Republic of Korea

*mgshin@kepco-enc.com

1. Introduction

The primary purpose of Spent Fuel Inspection System is to detect the mechanical defects of spent fuel assemblies drawn from a reactor by the visual inspection and ultrasonic inspection devices. Visual inspection is to observe the surface of the spent fuel assemblies by using radiation-resistant and water-proof camera. The purpose of ultrasonic inspection is to detect the fuel rod defect in spent fuel assemblies which mechanical defects were detected by visual inspection first. Since the visual inspection system on SKN 3&4 nuclear power plant is located at a relatively long distance from spent fuel handling path, it causes the delay of the fuel inspection time, and eventually leads to the extension of refueling operation period. For these reasons, the position of visual inspection system has been changed from Cask Loading Pit (CLP) to the refueling canal, and it contributes reducing fuel inspection time and operator exposure, and increasing efficiency of work.

2. Body

2.1 Composition of Fuel Inspection System

The Spent Fuel Inspection System primary consists of Fixed Fuel Inspection Stand, Movable Fuel Inspection Stand, Visual Inspection System and Ultrasonic Test (UT) System. The camera of Visual Inspection System which is originally placed on the Fixed Fuel Inspection Stand is designed to perform the surface of the whole spent fuel assemblies unloaded from reactor core by the visual inspection device on top of CLP. UT probe of Ultrasonic Test System placed on the Movable Fuel Inspection Stand has been designed to detect the defect of fuel rod in spent fuel assembly in Spent Fuel Pool (SFP). Ultrasonic Test System is designed to find the defected fuel rods using the UT probe when the radiation level is higher than the reference value. UT probe is installed on the manipulator and operated in

the water, and the manipulator is designed in the form of X-Y Table for UT probed to be conducted between each of the fuel rods. The original location of the Fuel Inspection System and fuel handling path in Aux. Building is illustrated in Figure 1.

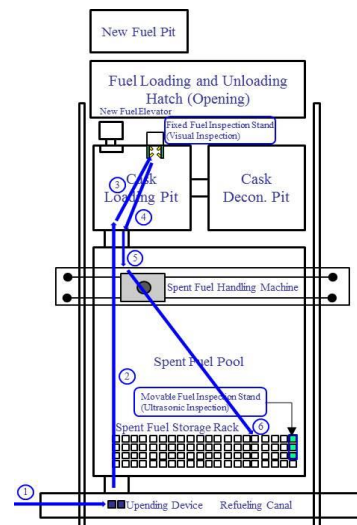


Fig. 1. Original Fuel Handling & Inspection Path.

2.2 Fuel Handling with Fuel Inspection System

Fixed Fuel Inspection Stand is located on top of the CLP and camera of Visual Inspection System has originally installed on the Fixed Fuel Inspection Stand. The surface of the spent fuel assembly has been scanned by camera attached to the inspection stand which moves up and down. Since the visual inspection has performed on the CLP, the spent fuel assembly erected as vertical position by upending device on refueling canal had to move to the CLP through SFP, and back to SFP to store it in the Spent Fuel Storage Racks after visual inspection. And then spent fuel assembly with the defect on the surface as a result of visual inspection is detected using Ultrasonic Test System. For this reason, fuel handling path without considering convenience cause the delay of the fuel inspection time and eventually leads to the extension of refueling operation period.

2.3 Improvement of Fuel Inspection System

Movable Visual Inspection Stand which is located above the upending device has been additionally installed as a purpose of fuel inspection which contributes reducing the fuel inspection time and operator exposure, and increasing efficiency of work. It is composed of under frame and upper frame, and four (4) set of the non-browning camera with the light. They are installed on the refueling canal. The Movable Visual Inspection Stand is illustrated in Figure 2.

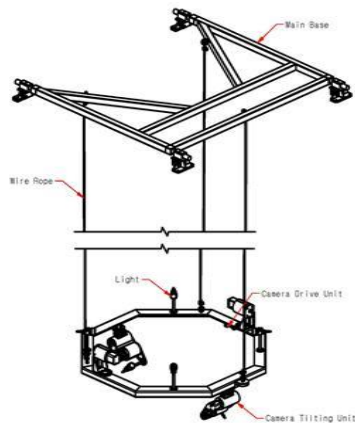


Fig. 2. Movable Visual Inspection Stand.

Visual inspection can be performed in the middle of unloading operation of spent fuel assemblies which is moved from upending device to hoist box of fuel handling machine at the same time. Thus, there is no need to have a time for only visual inspection of spent fuel assemblies. In addition, efficiency of work can be increased due to the reduced fuel inspection time. Furthermore, Fixed Fuel Inspection Stand used for visual inspection need not to be purchase on the following project because the role of the Fixed Fuel Inspection Stand and the Movable Visual Inspection Stand is overlapped, so it reduces purchase cost. The changed location of the Fuel Inspection System and fuel handling path in Aux. Building is illustrated in Figure 3.

Approximate time for fuel handling and inspection in Aux. building has been calculated as shown in Table 1 and 2. Actual operation time can definitely vary with field conditions. Occupying Movable Visual Inspection Stand instead of using Fixed Fuel Inspection Stand reduces fuel inspection time (about 7 minutes) and ultimately decreases the refueling operation period.

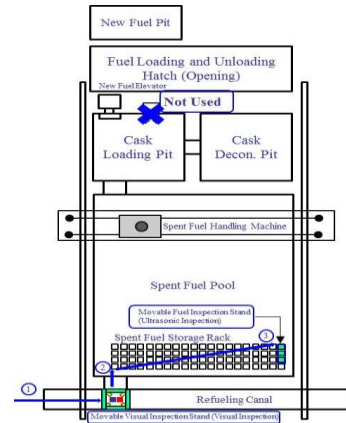


Fig. 3. Changed Fuel Handling & Inspection Path.

Table 1. Handling Time using Fixed Fuel Inspection Stand

Equipment		Speed (m/min)	Moving Distance (m)	Lead Time (min)
Fuel Handling Machine	Bridge	~ 15.2	25	3
	Hoist	~ 1.5	5	5
	Trolley	~9.1	15	2
Upending Device		2	N/A	2
Fuel Inspection		N/A	N/A	3
Total				15

Table 2. Handling Time using Movable Visual Inspection Stand

Equipment		Speed (m/min)	Moving Distance (m)	Lead Time (min)
Fuel Handling Machine	Bridge	~ 15.2	13	1
	Hoist	~ 1.5	5	5
	Trolley	~9.1	10	1
Upending Device		2	N/A	2
Fuel Inspection		N/A	N/A	N/A
Total				8

3. Conclusion

The location of visual inspection has been changed from on top of CLP to the refueling canal by using Movable Visual Inspection Stand. It contributes reducing fuel inspection time and operator exposure, and increasing efficiency of work and finally it decreases the refuel operation period.

REFERENCES

- [1] M.G.Shin, Specification for Fuel Inspection System, KEPSCO-ENC (2013).
- [2] C.B.Shim, Tech. Manual for Spent Fuel Handling Machine, Doosan Heavy Industry Construction (2012).

Estimation of On-Site Dry Storage Capacity Needed Based on Current Nuclear Policy in Korea

Hong Jang* and Hyo On Nam

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

*janghong@kaeri.re.kr

1. Introduction

Most countries where the national policy for the spent fuel (SF) is “wait and see” are faced with a big problem in wet storage capacity of the nuclear power plant (NPP). One solution for those countries is constructing an on-site or off-site interim storage facility, such as a dry storage facility. Based on an assumption that one dry storage facility would be arranged at each NPP site in Korea, this work estimated the required construction time and capacity of dry storage through the dynamic simulation.

2. Korean Phase-out Scenario

Based on the present government policy on the nuclear power generation, its portion in the energy mix will fall away to zero in the future. Total 27 NPPs (23 PWRs and 4 PHWRs) are expected to be operated for about 60 years without extension of their lifetime and addition of new NPP (Fig. 1a). According to this policy, the accumulated SFs generated from NPPs are expected to reach 35,000 tHM (Fig. 2b). The annual SF generation is calculated as follows,

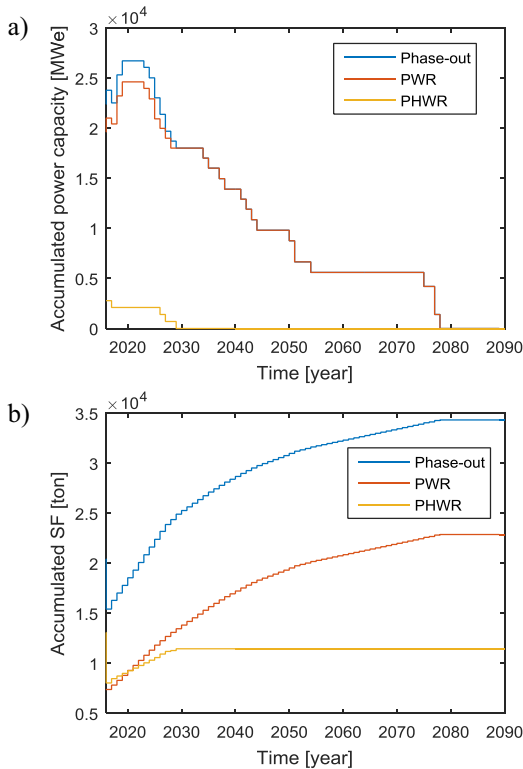


Fig. 1. Accumulated power capacity (a) and the amount of SFs generated (b) in the phase-out scenario.

$$m_{SF} = (P \times CF \times 365) / (\varepsilon \times BU) \quad (1)$$

where P is the accumulated power capacity, CF is the capacity factor, ε is the efficiency and BU is the burn up.

The government plan for decommissioning Kori#1 stopped 2017 [1] is to start work after cooling the PWR SFs in the wet storage for 7 years and releasing them from the wet storage. The released PWR SFs will be moved to the dry storage constructed until 2024 and stored temporarily for the final disposal. Like the plan for the Kori site, the required times of dry storage construction for other sites, Hanbit, Hanul and Wolsung, can be roughly estimated as 2032, 2034 and 2059, respectively.

Meanwhile, the committee for publicizing the SF management has estimated the saturation time of wet storage of all NPPs, which can be considered as the required construction time of dry storage [2]. Except for the Kori site, the estimated times based on the saturation of wet storage were earlier than the times based on the NPP decommissioning (Table 1). However, the required capacity of interim storage in the future has not been analyzed before.

Table 1. The estimated saturation times for wet storage of every NPP sites

Site	Capacity (bundle)	Current SFs (bundle)	Saturation rate (%)	Estimated saturation time (yr)
Kori	6494	5322	82.0	2028
Hanbit	9017	5413	60.0	2024
Hanul	7066	4652	65.8	2026
Wolsung	1046	64	6.1	2038

3. Dry Storage Capacity Estimation

Assumed one dry storage facility for one site, the required construction time and capacity of dry storage can be estimated based on the dynamic simulation of the PWR SF generation. Every NPPs are assumed to adopt the compact rack (at least twice) for maximizing the wet storage capacity. And also the transportation of SFs between NPPs in the same site is assumed to be allowed. The centralized interim storage and permanent disposal facilities are expected to be constructed 2035 and 2053, respectively, as announced from the government [3]. Their lifetimes are assumed to be 50 years.

As a result shown in Fig. 2, the required capacities are 2366.6 tHM for Kori, 2380.6 tHM for Hanbit and, 1204.1 tHM for Hanul. Wolsung doesn't require the dry storage. The required capacities for the interim storage facility and the disposal facility are 9258 tHM and 22875 tHM, respectively.

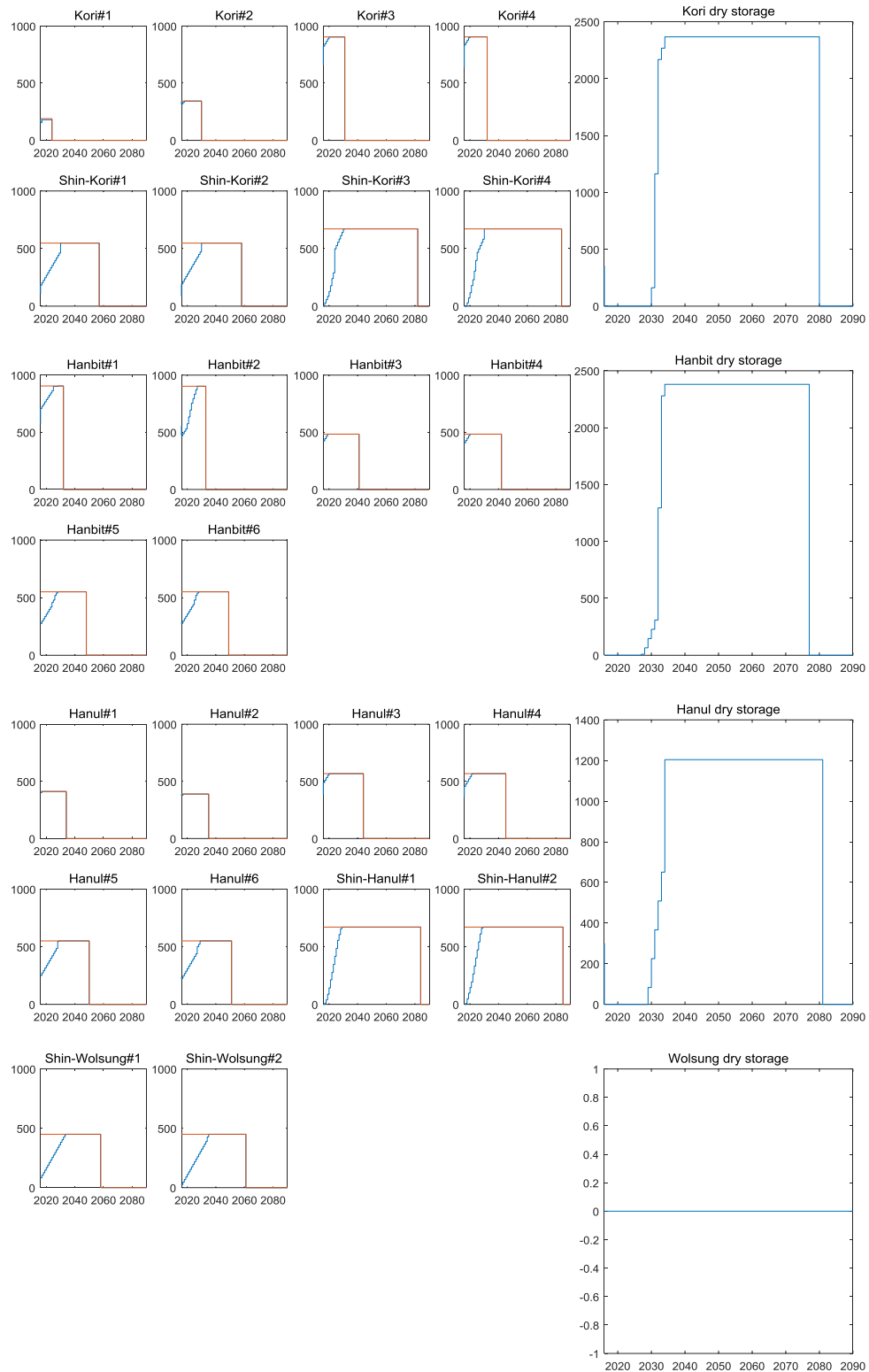


Fig. 2. The required time and capacity of on-site dry storage facility (y axis: accumulated SF [tHM], x axis: time [year]).

4. Conclusion

Based on the government policy for the nuclear power, we could estimate the required construction time and capacity of on-site dry storage for PWR SFs. As a future work, the constraint for the PWR SF transportation will be considered in the estimation.

REFERENCES

- [1] MOTIE, Permanent shutdown of Kori No.1, 2017.
- [2] PECOS, Proposal of recommendation for spent fuel management, 2015.
- [3] MOTIE, Proposal of basic plan for high-level radioactive waste management, 2016.

Line Balancing of an Integrated Pyroprocessing Using IOMP

Doyeon Kim*, Hyo jik Lee, Hunsuk Im, and Hong Jang

Korea Atomic Energy Research Institute, 989-111 Daedeok-Daero, Yuseong-Gu, Deajeon, Korea

*doyeon1301@kaeri.re.kr

1. Introduction

The Korea Atomic Energy Research Institute (KAERI) has been developing the multi-purpose integrated pyroprocessing operation model [1]. The model is based on the implementation of the dynamic material flow. The model of a so-called 'Integrated Operation Model for Pyroprocessing (IOMP)' is expected to provide an intuitive and various interfaces to facilitate user's access and comprehensive understanding of the complicated integrated pyroprocessing technology, and to enable users to analyze in-depth dynamic material flow [2,3]. IOMP v1 is available in KAERI and can be used to give insights into what optimized Integrated Pyroprocessing operation is by varying the various process parameters.

In this study, The IOMP will be introduced briefly, and case study will be presented by attempting line balancing analysis that can equalize capability values between unit processes.

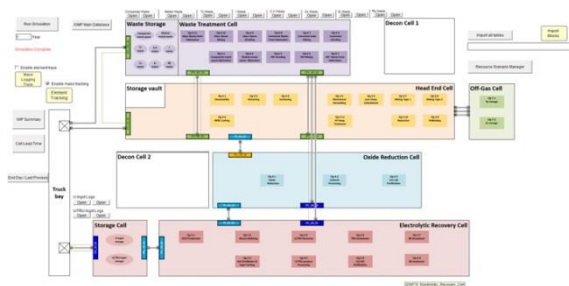


Fig. 1. 1st Level User Interface of IOMP.

2. Proposal for IOMP

2.1 Key Function

IOMP v1 can be used in various fields such as process control, safety measure and facility. The detailed functions of each field are as follows.

- Material transfer and operation tasks
- Element-wise material tracking
- Work-In-Progress(WIP) and hold-up analysis
- Periodic-and non-periodic operation handling
- Diversion scenario analysis
- Process operation scenario analysis

The IOMP shall include every unit process that KAERI's up-to-date pyroprocessing flow-sheet. The process parameters include the composition of Spent Nuclear Fuel (SNF) feed, process capacity, process time, the number of equipment, separation ratios, the number of human operators and material handling systems, and the size of WIP (Work In Process) storage.

2.2 Line Balancing of Integrated Process

One of the main function that the IOMP must have is operation analysis. The IOMP is capable of process line balancing analysis to design and efficient process line and give insight into the configuration of the process line. In order to design a well-balanced process line, a user might try to adjust the process time, process capacity, the number of process equipment and run a simulation. To check each cell lead time, data value will be investigated by clicking the [Cell Lead Time] or [IOMP Main Database-Report]. The lead time is defined as the time from the start of operation to completion. It is also possible to equalize the capabilities between unit processes.

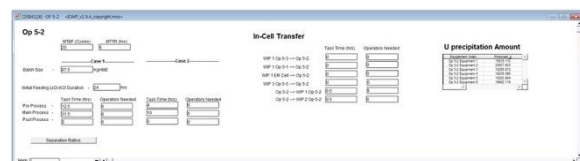


Fig. 2. Third level of window interface to adjust parameters and to display results.

2.3 Case study: Cell lead time analysis

In the process simulation operation, 72 assembly processing time is assumed to be 1 year and annual number of operation aims at 200 days. The simulation's initial value is recorded by engineering judgment. The initial setting data value about 1 year simulation run are as follows.

Table 1. Initial data of IOMP's default capacity value

Cell name	Lead time
Head End Cell	196.38
Oxide Reduction Cell	184.75
Electrolytic Recovery Cell	224.16
Storage cell	177.51
Waste Cell	242.95
Off-gas Cell	189.30

The data show that the Electrolytic Recovery (ER) cell is 244.16 days and the Waste cell is 242.95 days, which is more than 200 days. Therefore, the process line balancing is performed by adjusting the capacity of the corresponding cell. The batch size is doubled and the process task time is reduced by half for improving capacity. Capacity can be modified by importing or modifying the user-defined parameters in the 'IOMP import parameters' EXCEL file or directly from the Third Level window of the user interface. Table 2 Shows the simulation results after using the modified parameters.

Table 2. Result data of adjusted capacity value

Cell name	Lead time
Head End Cell	196.38
Oxide Reduction Cell	184.75
Electrolytic Recovery Cell	176.92
Storage cell	173.00
Waste Cell	187.65
Off-gas Cell	189.30

As a result of adjusting the capacity of ER cell and Waste cell, the target standard value was achieved for 200 days a year.

3. Conclusion

The IOMP v1 was developed at KAERI to present the engineering development direction of integrated pyroprocessing technology. The IOMP can propose insights of what is optimized integrated pyroprocessing operation while providing multiple interfaces and changing process parameters. It is possible to perform various analyzes such as equilibrium concentration analysis, logistics analysis, and scenario change analysis of operating conditions.

In this study, IOMP was used to analyze process line balancing. Integrated pyroprocessing aims at 200 days of operation per year, and the initial process capability of the model is set by engineering judgment. However, the lead time of each process cell is analyzed, and ER cell and Waste cell exceed 200 days. Therefore, in order to improve the capacity of each cell, the process task time is reduced by half and the batch size is doubled to change the parameters and attempt line balancing. As a result, efficient process line balancing, which achieved the annual 200 day operating target, was analyzed.

REFERENCES

- [1] H.J.Lee, H.S.Im, H.Jang, J.H.Ku, D.KrahlandJ. Luo, "Development of Integrated Operation Model for Pyroprocessing (IOMP)," Transactions of the Korean Nuclear Society Spring Meeting Jeju, Korea, May 18-19 (2017).
- [2] H. J. Lee, H. S. Im, and B. H. Won, Requirement of an Integrated Operation Model for Pyroprocessing (IOMP), KAERI/TR-6445/2016 (2016).
- [3] H.J.Lee, H.S.Im, H.Jang, J.H.Ku, Material Tracking of an Integrated Pyroprocessing Using IOMP, GLOBAL 2017, Korea, Sep 24-29 (2017).

Application of the Technical Safety Requirements for the Fuel Cycle Facility

Woo Jin Jo*, Seok-Jun Seo, Siwan Noh, and Seung Nam Yu

Korea Atomic Energy Research Institute, 111 Daedeok-daero, 989beon-gil, Daejeon, Korea

*wjjo@kaeri.re.kr

1. Introduction

Korea Atomic Energy Research Institute (KAERI) has been developing a pyroprocessing technology to reduce the waste volume and recycle some elements in a spent nuclear fuel. Because the operation with some fissionable materials such as pyroprocessing operations may cause very high risks of accidents resulting in high radiation doses for the public, facility worker, and collocated worker, the safety analysis is prerequisite to fuel cycle facility design. However, regulation guides in Korea are not prepared and are insufficient as compared with commercial nuclear power plants. With such situation in mind, KAERI performed Integrated Safety Analysis (ISA) for large-scale pyroprocessing facility, as suggested in NUREG-1520 rev.2, and radiological and chemical hazards are identified for accident analysis using ISA as required by 10 Code of Federal Regulation (CFR) 70.61 (Performance requirements)[1]. For the safe operation of the facility, the safety analysis report, which is the result of the ISA, provides the limiting conditions for the operation and administrative control throughout the facility operation. 10CFR 830.205 requires responsibility for nuclear facilities to develop Technical Safety Requirements (TSRs)[2]. In this summary, an example of derived TSRs applicable to nuclear fuel cycle facilities, consisting of safety limits, limiting condition, administrative controls, and other requirements is proposed. The guides and requirements for the TSR are also introduced.

2. Technical Safety Requirements

2.1 Regulations and Requirements

In Korea, there are no regulations and requirements for the nuclear facility including the fuel cycle facilities. However, 10CFR 50.36 of the

Technical Specifications, regulated by the US Nuclear Regulatory Commission (NRC), requires applicants for a license authorizing to include proposed technical specifications in their application. Technical Specifications will include items in the following categories: (1) Safety limits, limiting safety system settings, and limiting control settings (2) Limiting conditions for operation (3) Surveillance requirements (4) Design feature (5) Administrative controls (6) Decommissioning (7) Initial notification. 10CFR 830.205 of the Nuclear Safety Management Rule requires Department of Energy (DOE) contractors responsible for Hazard Category 1, 2, and 3 DOE nuclear facilities to develop TSRs. International Atomic Energy Agency (IAEA) also proposes the preparation of operational limits and conditions for the nuclear fuel cycle facilities in safety requirements No. NS-R-5.

2.2 Guides for TSRs Development

TSRs define the performance requirements of Structures systems and components (SSCs) and identify the safety management programs used by personnel to ensure safety. TSRs are aimed at confirming the ability of the SSCs and personnel to perform their intended safety functions under normal, abnormal, and accident conditions.

Through analysis of the encompassing bounding accidents, the necessary safety systems and accident mitigating system are identified and their characteristics are defined. Flowing from the analysis is information that provides the bases for controls, limits, and conditions for operation, known as TSRs.

2.3 TSR elements

As mentioned previously, Safety analysis identifies some parameters that must be controlled to ensure the safety requirements for the facility are met. There are several components for TSRs, but the important TSR components are as follows[2]: Safety Limits (SL) are limits on important process variables

needed for the facility function that, if exceeded, could directly cause the failure of one or more of the passive barriers that prevent the uncontrolled release of radioactive materials, with the potential of high consequences to the public. Limiting Control Settings (LCS) define the settings on safety systems that control process variables to prevent exceeding and SL. Limiting Conditions for Operation (LCO) defines the limits that represent the lowest functional capability or performance level of safety SSCs required to perform an activity safely. Action statements should describe the actions to be taken if an operating limit is not met and establish the steps and time limits to correct the conditions that are beyond the TSR limits. Surveillance Requirements (SRs) are used to ensure operability or availability of the safety SSCs identified in the operating limits. SRs are most often used with LCOs to periodically validate the operability and availability.

3. TSR examples

The small transfer lock system is one of the components of the hot cell boundary that is most important to prevent the leakage of radioactive materials to public in nuclear fuel cycle facilities. Thus, small transfer lock system shall be operated safely and ensured the reliability. Applicable LCO for small transfer lock system is shown below as an example.

[Example]

KRS LCO 1 – Small Transfer Lock System

The Small Transfer Lock System shall be operable.

Mode applicability – Operation mode

Actions

Condition	Required action	Completion time
One Small Transfer Lock System door is inoperable	Ensure the Small Transfer Lock System companion door is closed.	Immediately
Both Small Transfer Lock System doors are inoperable	Place main hot cell in the standby mode.	Immediately
Surveillance for compliance with KRS LCO 1		Frequency
Visually verify the Small Transfer Lock System doors are closed	Quarterly and following maintenance	
Verify that the Small Transfer Lock System leak rate is lower than 0.05 m ³ /hr	Quarterly and following maintenance	

4. Conclusion

Safety analysis is required for the safe operation of nuclear fuel cycle facilities. Through conducting the safety analysis for the facility and operation, continuous management or operational guidelines are naturally derived so that safety SSCs and administrative control perform their intended safety functions.

The TSRs consisting of several limitations must be prepared before the operation of the facility, and any changes to the process or facility will require the TSRs to be revised.

Regulatory requirements for establishing TSRs will be needed for nuclear fuel cycle facility, although the legal requirements are not currently available in Korea.

Acknowledgements

This study was supported by the Ministry of Science & ICT, and Future Planning through its National Nuclear Technology Program.

REFERENCES

- [1] U.S. NRC, "Standard Technical Specification", NUREG-2194, Vol. 1 (2016).
- [2] U.S. DOE, "Implementation guide for use in developing technical safety requirements", DOE G 423.1-1A (2010).

Pressure Drop of U-shape Bellows at Argon Circulation System

Ji-sun Lee*, Byungsuk Park and Jaehoon Lim

Korea Atomic Energy Research Institute, 111, Daedeok-dearo 989beon-gil, Yuseong-gu, Deajeon, Korea

*Hijisun84@kaeri.re.kr

1. Introduction

Advanced spent fuel conditioning process facility (ACPF) consists of an air cell and an argon cell. In an argon cell, an electrolytic reduction process is performed in which oxygen is separated from the spent fuel by an electrochemical method to reduce it to metal. Therefore, the inside of the cell is filled with argon gas to remove oxygen, and it is a space for handling the radioactive material, so it must be kept at a negative pressure so that the radioactive material can't move to the outside. At this time, the argon circulation system is used to maintain the moisture, oxygen concentration and pressure inside the cell at a constant level. This system consists of a process in which the fluid inside the argon cell is passed through the purification system through the HEPA filter in the air cell and then back to the argon cell. This process is repeated. This system also uses a blower to circulate the working fluid and maintain the design flow rate. In this case, the flow rate is lower than the design value because the pressure loss in the entire circulation system is larger than expected and the differential pressure of the blower is higher than the design pressure. To solve this problem, we investigated the characteristics of the fluid flow from the air cell to the bellows of the HEPA filter system, where pressure drops are large and can easily be improved. In this study, we used flow simulation, a computational fluid dynamics analysis program.

2. Model description

2.1 General Information of U-shape bellows

In this research facility, a circular bellows of the U-shape type is used, and it is always bent at an

angle of 120 degrees or more.

Generally, the bellows are known to have a significant impact on pressure loss due to the pitch value, inner diameter, outer diameter and the corrugate area [1]. Also, there is an experimental result on the occurrence of pressure loss according to the radius of curvature [2]. However, CFD analysis is rarely performed on the fluid flow characteristics of the bellows due to the change of the radius of curvature. Therefore, we analyzed the internal flow characteristics and the pressure drop at a specific flow rate using the bellows standard used in this research facility.

The bellows code is 50A and the detailed dimensions are shown in Table 1.

Table 1. Specifications of U-shape bellows

Material	Product Co.	I.D (mm)	O.D (mm)	Thickness (mm)	Pitch (mm)
SUS 304	50A	49.8	67.5	0.3	5

As shown in Fig. 1, the bellows length of all analytical models is 1000mm and a 50A general piping with a length of 100mm was connected to the inlet and the outlet so that the flow is fully developed.

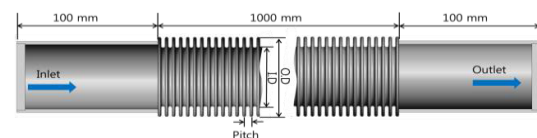


Fig. 1. Shape of bellows.

The analytical model is a total of nine models with 20 degrees interval from 0 to 160 degrees.

2.2 Condition of CFD analysis

In order to compare the experimental results with the analytical results, the pressure and temperature

conditions were set to -23.7 mmAq and 24 °C, respectively. The initial conditions were -23.7 mmAq of static pressure at the inlet and 98.5 m³/h of volume flow rate at the outlet.

The analysis used the k-ε model, which is a turbulence analysis model.

Table 2. Boundary condition and Initial condition

Pressure (mmAq)	Tem. (°C)	Inlet (mmAq)	Outlet (m ³ /h)
-23.7	24	Static pressure -23.7	Volume flow rate 98.5

3. Results and discussion

As shown in Table 3, the pressure drop increase with the bending angle. Also, as shown in Fig. 2, pressure drops are increased more than 25 mmAq at from 0 degree to 60 degree of bending angle, whereas the increase width decreases to 8 mmAq less when the bending angle is 80 degree or more.

Experimental results show a pressure drop of 170 mmAq in the bellows. The cause is two bends, 40 degree and 160 degree. Compared with the analytical results, although the absolute comparison is impossible, it can be seen that the tendency is somewhat consistent.

Table 3. Pressure drop versus bending angle for bellow

B.A (degree)	0	40	80	120	160
P.D (mmAq)	36.8	58	80.8	94	104.6

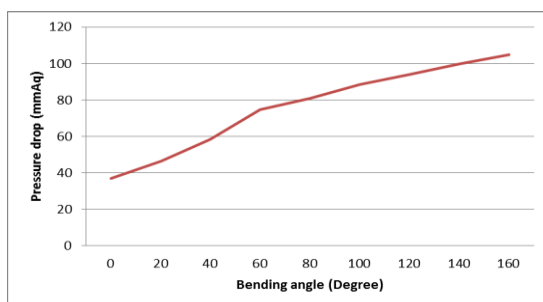


Fig. 2. Pressure drop versus bending angle for bellow.

4. Conclusion

It was found that the bending angle of the bellows significantly influenced the fluid resistance. Also, the pressure drop tendency according to the bending angle was able to be obtained. However, further analysis of the bellows shape and the same model of the experimental facility is needed to make a more accurate comparison.

REFERENCES

- [1] Youngsun yun, Junone Kang, Hyungjung Kim, Jaisuk Yoo "The experimental study for the pressure drop and friction factor of flexible metal tubes".
- [2] Kenneth Lloyd Riley, "Flow Losses in flexible hose", Louisiana State Univerdity and Agricultural Mechanical college, 1967.

Conceptualization of Dry Room for Verification of Pyroprocessing Equipment

Jaehoon Lim*, Byungsuk Park, Dongsuk Ryu, Hyo Jik Lee, and Seung Nam Yu

Korea Atomic Energy Research institute, 111, Daedeok-dearo 989beon-gil, Yuseong-gu, Deajeon, Korea

*jhl85@kaeri.re.kr

1. Introduction

Pyroprocessing equipment uses high-temperature molten salt (LiCl, KCl) which is highly corrosive. Thus, the pyroprocessing equipment should be verified for operability in high temperature and corrosive environment. However, the existing pyroprocess facilities in Korea Atomic Energy Research Institute (PRIDE, ACPF) are designated as the radiation controlled area. Thus, those existing facilities are not suitable for performing verification experiment of various prototypes.

Therefore, to perform operability verification in high temperature and corrosive environment, construction of a dry room which is not located in radiation controlled area is required.

By the way, the dry room used in the general industry is generally operated 24 hours a day, and continuously supplies and discharges a large amount of dry air. This operation method requires a lot of electricity consumption.

Therefore, in this paper, a brand new method to decrease the operating time of the dehumidifier is suggested by using the existing dry air supplied to the facility.

2. Conceptualization of Dry Room

2.1 Requirements of dry room

The established design requirements for the dry room are shown in Table 1. In order to accommodate two pyroprocessing equipment inside the dry room,

the dimension of the dry room is determined. Stainless Steel (STS) is selected to the material of inner panel, bottom plate, and ceiling to prevent corrosion during salt test.

Table 1. Design requirements for the dry room

Dry room due point	-40°C
Dry room temperature	25~30°C
Dimension (internal space)	5.2 m × 4.1 m × 4.5 m
Leak rate	< 0.01 × (dry room volume)
Material	STS (inner walls and bottom plate)
No. of people in the room	2 people

2.2 Layout of the dry room

Based on the design requirements for the dry room, layout is obtained as shown in Fig. 1.

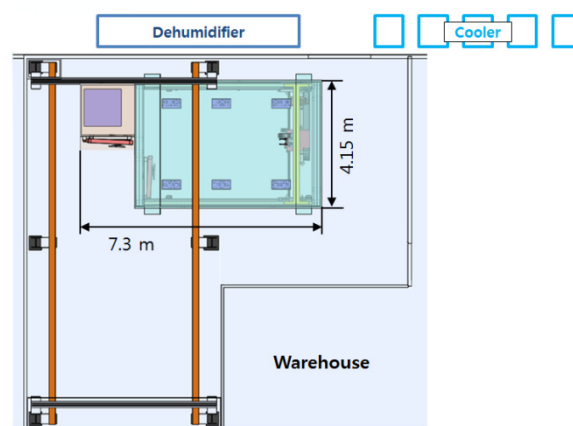


Fig. 1. Layout of the dry room.

It is planned that the dry room will be installed in

the remote device experiment building in Korea Atomic Energy Research Institute. In the building, a 5 ton crane is constructed as shown in Fig. 1, and a vehicle is accessible through the south shutter door.

The dry room and preparation room will be located in indoor space, and the dehumidifier systems will be located outdoors due to lack of indoor space.

A 1 ton overhead crane will be installed in the dry room for transportation and maintenance of demonstration process equipment, and a gantry robot for remote handling and automation research will be also installed. The gantry robots has a precision of more than 1 mm will be equipped with non-contact type considering the dust and corrosive environment.

2.3 Hybrid operation strategy

As mentioned previously, the dry room is needed to perform operability verification of pyroprocessing equipment. However, general industrial dry room is used for battery manufacturing plants. These plants operate 24 hours a day, 365 days a year. Therefore, there are always workers inside the dry room, and thus, the dried air should be continuously supplied to the dry room. On the contrary, the operability verification and automation research will not be conducted frequently. Therefore, it is not economic to operate the dehumidifier system for dry room for 24 hours a day, 365 days a year.

By the way, there is a dry air supply line in the remote device experiment building. The dew point of the dry air is approximately -40°C .

If the dry air is supplied to the dry room, the humidifier does not have to be operated when the process equipment does not operate and there is no person inside. The dehumidifier is operated when the humidity of the dry room is rapidly increased. The detailed operating procedure will be prepared in the future. By applying the brand new hybrid operating method, it is expected that the electric consumption can be significantly reduced.

3. Conclusion

The conceptualization of the dry room for mechanical demonstration of pyroprocessing equipment was performed. A few of design requirements were established and the layout of the dry room is determined. In addition, a hybrid operation method has been proposed for efficient operation. Further detailed design review and study will be conducted in the future.

Acknowledgement

This work was supported by National Research Foundation of Korea (NRF) (NRF-2017M2A8A5015083).

A Consideration on the Dehumidification Methods for Designing the Dry Room

Byungsuk Park*, Jaehoon Lim, Dongseok Ryu, Hyojik Lee, and Seungnam Yu

Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon, Korea

*nbspark@kaeri.re.kr

1. Introduction

Pyro-process researches are performed in a salt environment. Because salt causes corrosion, moisture in air must be removed. Therefore, in a small-scale research, a glove box is prepared. However, in medium- and large-scale research, dry room is required because worker access must be considered. In this study, the dehumidification methods for the dry room construction were reviewed and the contents are described below.

2. General Dry Room System

2.1 Clean Room

Clean rooms are typically used in manufacturing, including of pharmaceutical products or scientific research, as well as aerospace semiconductor engineering applications with a low level of environmental pollutants such as dust, airborne microbes, aerosol particles, and chemical vapors.

2.2 Constant Temperature and Humidity Room

These rooms are widely used in the temperature variation tests, cold resistance tests and storage with low-temperature in the fields of aerospace, aviation, electronics, instrument, electric products, materials, parts and components and equipment to analyze and evaluate the property and performance of the samples under the simulated conditions.

2.3 Dry Room

Dry rooms are typically controlled at much lower humidity than clean rooms and constant temperature/humidity rooms. To better control the humidity in the

lithium ion battery processing areas, dry rooms are hermetically sealed to maintain very low humidity levels.

3. Dehumidification Methods

Two procedures are primarily used for dehumidification: dehumidification through condensation, drying through sorption. The accompanying diagram shows the operating characteristics of both systems:

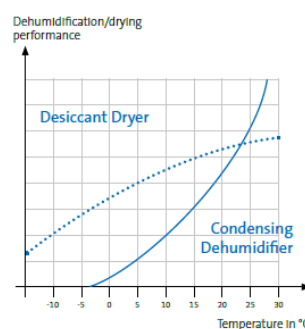


Fig. 1. Operating Characteristics of Desiccant Dryer and Condensing Dehumidifier.

3.1 Condensing Dehumidifier

As shown in Fig. 2, this process involves cooling the humid air below the dew point by conducting the air stream across the cold surface of a heat exchanger (evaporator of a cooling circuit).

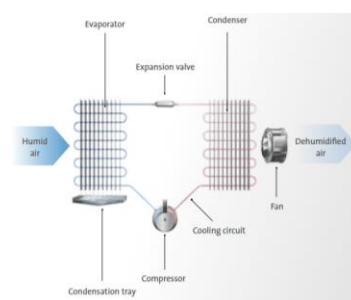


Fig. 2. Condensing Dehumidifier.

3.2 Desiccant Dehumidifier

Desiccant dryers are used wherever condensing dehumidifiers physically reach their limits and compliance with minimum absolute humidity or water vapor levels is required. Fig. 3 and 4 show discontinuous dehumidifiers and a continuous dehumidifier.

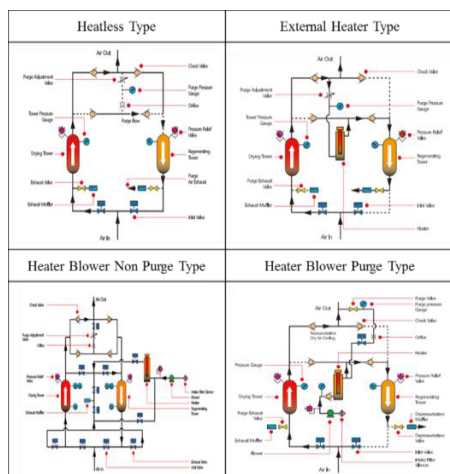


Fig. 3. Dehumidifiers of Discontinuous Type.

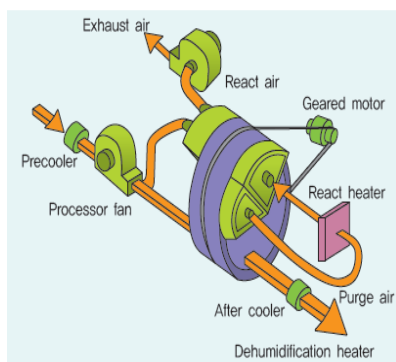


Fig. 4. Continuous Dehumidifier.

4. Adsorbent Type and Properties

Adsorption refers to the phenomenon that a gas powder adheres to a surface of solid. Thus, gases or liquids can be selectively removed using adsorbent. Table 1 shows the properties of adsorbents.

Table 1. Adsorbent Properties

	Alumina Gel	Silica Gel	Molecular Sieves
Size	2 - 6.4	2 - 5	1.5 - 5
Al ₂ O ₃	93%	0.3%	-
SiO ₂	0.02%	99%	-
RH 50%	18 wt%	26 wt%	20 wt%
RH 90%	40 wt%	43 wt%	21 wt%
Regen. Temp.	175 - 250°C	150 - 180°C	200 - 300°C

5. Conclusion

In this paper, various dehumidification methods used in the industry have been reviewed, and the results of this review will be used to construct the drying system in the future.

REFERENCES

- [1] Lewis G. Harriman III, "The Dehumidification Handbook", MUNTERTS CORPORATION, 2002.

Enforced Risk Criteria for SF Recycling Facilities in US and Its Availability in Korea

Gilsung You*, Seok-jun Seo, Siwan Noh, Hongrae Jeon, Hyo jik Lee, Woo jin Jo, Hyunsook Im, Hohee Lee, Wonmyung Choung, Seung nam Yu, Jaehoon Lim, and Jeonghoe Ku
KAERI, Daeduk-daero 989-111, Yuseong-gu, Daejeon, 305-353, Korea
*yougil@kaeri.re.kr

1. Introduction

KAERI is being developed pyroprocessing technology for spent fuel recycling. For this it is necessary to develop a facility with hot cells, which can handle the processes safely [1]. In this paper the enforced risk criteria suggested by the US NRC for spent fuel recycling facilities and its bases are introduced and also studied for its availability in Korea.

2. Enforced Risk Criteria of the US NRC and its Bases for SF Recycling Facilities

2.1 Enforced risk criteria of the US NRC for SF recycling facilities

The US NRC recommends and regulates fuel cycle facilities to follow the Integrated Safety Analysis (ISA) method through the code 10CFR70 [2, 3]. The main reason to use the ISA method is that the fuel cycle facilities have some chemical processes which are different from the nuclear reactors.

Recently the US NRC is studying and suggesting the risk criteria (the consequence thresholds, likelihoods, and qualitative risk bins for risk indexing method) for SF recycling facilities as following Table 1 and 2 [4-7].

Table 1. Consequence thresholds

	Workers	Offsite Public	Environment
Very High Consequence Event	RD>>1Sv CD>endanger life	RD>1Sv CD=endanger life	Radioactive release >500,000 x Table 2 of 10CFR20, Appendix B

High Consequence Event	RD>1Sv CD>endanger life	RD>0.25Sv sol U intake>30mg CD=long-lasting health effects	Radioactive release >50,000 x Table 2 of 10CFR20, Appendix B
Intermediate Consequence Event	RD>0.25Sv CD=long-lasting health effects	RD>0.05Sv CD=mild transient health effects	Radioactive release >5,000 x Table 2 of 10CFR20, Appendix B
Low Consequence Event	Accidents of lower radiological and chemical exposures than those above in this column	Accidents of lower radiological and chemical exposures than those above in this column	Radioactive releases producing lower effects than those referenced above in this column

Table 2. Qualitative risk bins

		Likelihood (Events Per Year)			
		Very Highly Unlikely (<1E-6)	Highly Unlikely (<1E-5)	Unlikely (<1E-4)	NOT Unlikely (>1E-4)
Consequence	VHCE	Acceptable	Not Acceptable	Not Acceptable	Not Acceptable
	HCE	Acceptable	Acceptable	Not Acceptable	Not Acceptable
	ICE	Acceptable	Acceptable	Acceptable	Not Acceptable
	LCE	Acceptable	Acceptable	Acceptable	Acceptable

2.2 Bases of the enforced risk criteria for the ISA [7]

The SECY-11-0163 of the US NRC [7] reports that spent nuclear fuel materials and Mixed Oxide (MOX) fuel prepared from reactor-grade plutonium have higher dose conversion factors: some 200,000 times greater than low enriched uranium (LEU) materials. Thus, a potential scenario at a facility involving LEU materials might have low consequences, but the same type of event at a reprocessing facility could potentially have consequences orders of magnitude larger because of this greater radiotoxicity of

materials (as shown in Table 3), thus requiring additional safety controls to achieve the same level of safety.

Table 3. Comparison of unit mass dose conversion factors (specific and relative inhalation doses)

Isotope/Mixture	Specific Inhalation Dose, rem/gram	Relative Dose, Ratio to "Ideal LEU"
Uranium-234 (U-234)	8.21E5	1.64E4
Uranium-235 (U-235)	2.58E2	5.16
Uranium-238 (U-238)	3.91E1	7.81E-1
Depleted Uranium (DU) U-235: 0.25%, U-234: 0.00194%, balance U-238	5.55E1	1.11
Natural Uranium U-235: 0.71%, U-234: 0.0055%, balance U-238	8.58E1	1.72
Low-Enriched Uranium (LEU) U-235: 5%, U-238: 95%	5.00E1	1 (reference)
LEU: U-235: 5%, U-234: 0.0055%, balance U-238 (similar to laser enrichment product)	9.72E1	1.9
LEU: U-235: 5%, U-234: 0.03873%, balance U-238 (similar to GC/gaseous diffusion plant enrichment product)	3.68E2	7.36
High-enriched uranium: U- 235: 80%, U-234: 0.88%, balance U-238	7.44E3	1.49E2
Mixed oxide (MOX): plutonium (Pu)-239: 5%, U-238: 95%	9.55E5	1.91E4
MOX: weapons Pu, 5% Puf, balance DU	1.27E6	2.54E4
MOX: reactor Pu, 5% Puf, balance DU	1.00E7	2.01E5
MOX: reactor Pu, 5% Puf, 0.25% Am- 241, balance DU	1.40E7	2.81E5
Spent nuclear fuel: 60,000 MWD/MTIHM Only fission products considered are Cs-135, Cs- 137, and Sr-90 isotopes	1.11E7	2.2E5
Cs-135, Cs-137, and Sr-90 isotopes from 60,000 MWD/MTIHM spent nuclear fuel	2.05E6	4.1E4
Inhalation doses are based upon 50-year committed effective dose equivalent (see ADAMS Accession No. ML102720167, slide 5). Specific data for isotopes are from EPA-520/1-88-020, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September 1988		

3. Availability of the US NRC's enforced risk criteria in Korea and Conclusions

In Korea SF recycling facilities are regulated as one of fuel cycle facilities under the nuclear safety law. But the safety analysis methods are nearly same and use the technical criteria with nuclear power plants. But in US the fuel cycle facilities are regulated by a different safety analysis method, such as the ISA, and use some different risk criteria. The ISA method suggested by the US NRC and the enforced 4x4 matrix risk criteria based on the Table 3 will be good references to help the improvement of the Korean nuclear safety law for licensing of SF recycling facilities. The consequences, likelihoods, and unit mass dose conversion factors as shown in Table 3 should be also reevaluated to comply with the Korean nuclear safety law.

Acknowledgements

This study was supported by the Ministry of Science & ICT, and Future Planning through its National Nuclear Technology Program.

REFERENCES

- [1] Gil-Sung You, et al., Concept and safety studies of an integrated pyroprocess facility, Nuclear Engineering Design, No. 241, pp. 415 ~424, (2011).
- [2] US NRC, "Integrated safety analysis guidance document," NUREG-1513 (2001).
- [3] US NRC, "Standard review plan for the review of a license application for a fuel cycle facility," NUREG-1520-rev.2 (2015).
- [4] US NRC, "Regulatory option for licensing facilities associated with the Global Nuclear Energy Partnership," SECY-07-0081 (2007).
- [5] US NRC, "Regulatory structure for spent fuel reprocessing," SECY-08-0134 (2008).
- [6] US NRC, "Update on reprocessing regulatory framework summary of gap analysis," SECY-09-0082 (2009).
- [7] US NRC, "Draft regulatory basis for licensing and regulating reprocessing facilities," SECY-11-0163 (2011).

Structural Evaluation Under Accident Condition for Concrete Storage Cask of Spent Fuels

Tae-Chul Moon* and Chun-Hyung Cho

Korea Radioactive Waste Agency, 168 Gajeong-ro, Yuseong-gu, Daejeon 34129, Republic of Korea

*tcmoon@korad.or.kr

1. Introduction

The concrete storage cask has been developed with containing canister loaded with spent fuel. The storage consists of reinforced concrete is filled between inner and outer shell of cask body and the carbon steel made cask lid filled with concrete is bolted into cask body to maintain structural integrity. The evaluation results were reviewed with respect to every design criterion, in terms of whether the results satisfy the criteria, provided by US 10CFR Part 72 and NUREG-1536 [1.2]. Description of the concrete storage cask and the maintainability and assumptions include in the analysis were confirmed through detailed explanations of the acceptable standards, analysis model and analysis method. ABAQUS 6.10, a widely used finite element analysis program, was used in the structural evaluation [3]. The evaluation results of this work show that the maximum stress was below the allowable stress under accident condition, and the concrete storage cask satisfied the design criteria.

2. Model description

2.1 General information of concrete storage cask

A concrete cask body is a cylindrical shell structure for the storage of canisters loaded with spent fuels. Concrete for shielding is filled between the internal and external carbon steel shells of the cask body, and air intake and exhaust vents are installed at the top and bottom of the cask body to eliminate the decay heat from the spent fuel through air circulation due to natural convection. The concrete cask body of the concrete cask protects the canister from the external environment to maintain the integrity of the spent fuels, and the canister maintains the confinement boundary while it is loaded with the spent fuels as like Fig. 1 [4].

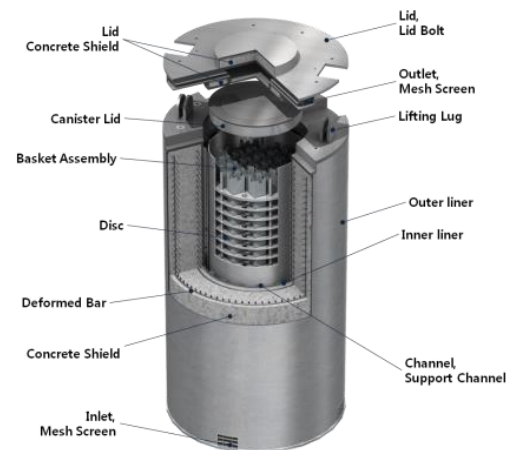


Fig. 1. The cross-section of concrete storage cask.

2.2 Load combinations of accident condition

The structural analysis based on the accident condition of the concrete storage cask was considered with pressure load, thermal load, earthquake load, cask drop and tip-over. ASME B&PV Code Section III, Division 1, Subsection NB (Level D) was applied to the stress evaluation of the canister, and ASME B&PV Code Section III, Division 1, Subsection NF (Level D) was applied to the cask body. The design criteria under accident condition are listed in Table 1.

Table 1. Design criteria under accident condition

Type	Criteria	Basis
Drop height	60 cm	-
Tip-over	Assumed	-
Fire (Duration/Temp)	8min/800 °C	10CFR72.122(c)
Blockage of air inlets	100%	10CFR72.128(a)
Flood	15m (4.6m/sec)	10CFR72.73
Seismic (Horizon, Vertical)	0.3g, 0.2g	10CFR72.102(f)
Extreme temp	40 °C	ANSI/ANS-57.9

2.3 Material properties

The concrete storage cask body was produced from carbon steel, and the elements other than the body were produced from stainless steel. The canister utilized SA-240 316L material to increase the salt resistance due to the external environment under storage conditions [4].

2.4 Structural analysis

For the finite element analysis model used in the structural analysis of the concrete storage cask, one structural model was prepared and the initial condition and boundary condition were varied to apply to every analysis. Considering the symmetry of the cask, the analysis model was composed of a 1/2 model. The analysis model was composed of 809,046 nodes and 556,905 solid elements [4].

3. Results and discussion

The internal pressure 0.335 MPa in the accident condition computed from the thermal analysis results was applied to the inside of the canister. The analysis results showed the highest stress of 70 MPa in the lower part of the canister body adjacent to the bottom plate of the canister.

The analysis results of vertical drop showed stress of 303 MPa at the bottom plate of the cask body, but it was evaluated to be below the allowable stress.

A tip-over analysis of the cask was conducted by applying the angular velocity 1.488 rad/sec at the moment the side of the cask body comes in contact with the bottom surface of the concrete pad. The analysis results showed that every part was evaluated to satisfy the allowable stress. The stress distribution regarding the tip-over analysis is shown in Fig. 2 (a).

The analysis results of flooding showed stress of 32 MPa at the bottom of the canister body and this was evaluated to be below the allowable stress.

In the analysis, the earthquake load was assumed to act simultaneously in the horizontal and vertical directions. The analysis results showed almost no stresses in the internal elements, and the highest stress observed in the canister was 32 MPa. However, stresses in every part were evaluated to be below the allowable stress. The stress distribution regarding the seismic is shown in Fig. 2 (b).

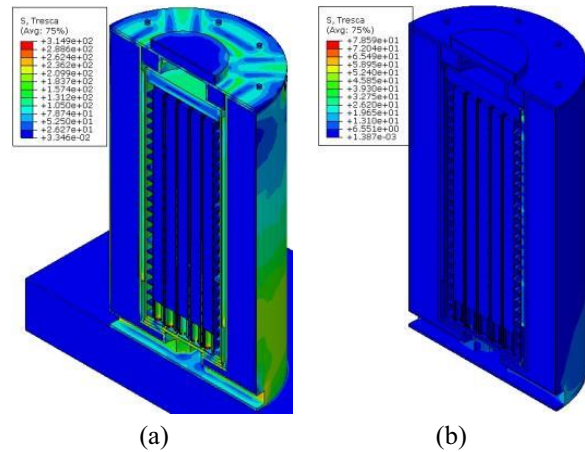


Fig. 2. The stress distribution for accident condition
(a) cask drop, (b) earthquake.

4. Conclusions

This paper verified the stabilities of the structural elements that influence the safety of the concrete storage cask for spent and summarizes the structural evaluation results of a concrete storage cask with respect to the design criteria under accident condition. The evaluation results show that the maximum stress was below the allowable stress under accident condition, and the concrete storage cask satisfied the design criteria. Therefore, based on the results of the stress evaluation, the design requirements were satisfied in the accident condition.

REFERENCES

- [1] 10CFR Part 72, Title 10 (2006), Chapter 1 of the Code of Federal Regulations, Licensing Requirement for the Independent Storage of Spent Nuclear Fuel, High-level Radioactive Waste, and Reactor-Related Greater than Class C Waste, 1-1-06 Edition.
- [2] NUREG-1536 (2010), Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility, Revision 1.
- [3] Dassault Simulia (2010), ABAQUS 6.10 Documentation.
- [4] TC Moon, CY Baeg, and CH Cho (2017), Structural Evaluation under Off-normal Condition for Concrete Storage cask of Spent fuels, Trans. of the Korean Radioactive Waste Society Spring Meeting, Pusan, Korea, May 24-26.

Requirements for the Transportation of Spent Nuclear Fuel (SNF) in Terms of Fuel Integrity and Data Needed According to

J. S. Noh¹, Y.K. Kim², and T.W. Kim³

¹ ACT Co., Ltd. , Techno 9 Ro-35, Yuseong-Gu, Daejeon, Korea

² KEPCO Nuclear Fuel Co., 242, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

³ Hanyang Univ., 222, Wangsimni-ro, Seongdong-gu, Seoul, Korea

jsnoh@actbest.com, ykkim@knfc.co.kr, idtaiwan57@gmail.com

1. Introduction

It would be desirable to delineate the requirements for the transportation of SNF in terms of fuel integrity, out of which the data needed to assure the satisfaction with the requirements could be enlisted and pursued according to, i.e. impact toughness, DBTT etc.(hydride behavior).

To meet the transportation regulations as well as to be able to predict confidently how SNF will behave during transportation and when handled after transportation to a repository or a processing facility also, we need to know the materials' properties and behavior.

Especially for the high burnup fuels, the higher the burnup level of the fuel increases beyond 45 GWd/MTU, the more vulnerable it should become. The thickness of the oxide layer on the exterior of Zircaloy cladding increases at a faster rate than at lower burnup levels possibly causing localized degradation of the cladding material. Additionally, as burnup levels increase, the cladding absorbs more hydrogen which affects the rate of material deformation and, depending on the environmental conditions of the cladding, can affect the mode of ultimate cladding failure. Hydride re-orientation and the increase of DBTT out of it could be a factor of its failure resulting from the cyclic vibration and impact encountered during transportation. Those factors should be evaluated in detail especially for the high burnup SNF. To have a look at the requirements in terms of fuel integrity and to draw a research plan for getting necessary data should be valuable to prepare SNF transportation near future.

2. Requirements in transportation of SNF in terms of fuel integrity

Regardless of the length of storage, used fuel eventually will have to be moved from the reactor sites either to off-site interim storage facilities or to used-fuel processing facilities for recycling or for waste management. Transportation regulations are largely focused on the integrity of the transportation casks, which contain the used fuel, and maintaining sub-criticality of the fuel. The primary goal is to ensure that the cask does not fail in the event of a transportation accident, with the potential for release of radioactive materials. The regulations require that under both normal and accident conditions, the transportation cask and its contents are capable of meeting stringent

performance specifications that include maintaining geometric configuration of the fuel to certain limits largely for criticality and to address concerns about external radiation levels [1].

The most important technical issue should be whether the cladding could maintain its integrity against the impact encountered during transportation, despite that there shall be some degradation factor resulted from the extended storage period. The regulations regarding to the fuel integrity in transportation are as follows:

Fuel-Specific Regulations:

A fuel-specific regulation means a characteristic or performance requirement of the fuel specifically named in the applicable Code of Federal Regulations (CFR). These are regulations that specify capabilities that the SNF must have. For transportations:

- 10 CFR 71.55(d) states, in part: "A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in 10 CFR 71.71 ('Normal conditions of transport')

(1) The contents would be subcritical.

(2) The geometric form of the package contents would not be substantially altered."

- Interim Staff Guidance-11, Revision 3, 'Cladding Considerations for the Transportation and Storage of Spent Fuel': "For hypothetical accident conditions, the licensee must assure that there is no significant cladding failure"[2].

System-Related Regulations:

A transportation and storage system must satisfy all applicable regulations in 10 CFR Parts 71 or 72. A system-related regulation is a performance requirement placed on the fuel for the system (i.e., transportation or storage cask) to meet a regulation that does not specifically require performance capabilities of the SNF. Examples include:

- 10 CFR 71.55(d): Requiring the system to remain subcritical and unchanged during normal transport, must be met. If gross breaches occurred on cladding, the SNF would be classified as damaged for transport because the requirements of 10 CFR 71.55(d) might not be met.
- 10 CFR 71.55(e) states in part: "A package used for the shipment of fissile material must be so designed and constructed and its contents so limited that under the tests specified in 10 CFR 71.73 ('Hypothetical accident conditions'), the package would be subcritical."

This regulation does not place a specific requirement on the SNF. However, if the package requires the SNF

to maintain its geometric configuration to ensure sub-criticality, then a function is imposed on the SNF [3].

•NUREG-1617: For the containment review, fractions of rods that develop cladding breaches were assumed 3 % for normal condition of transport, 100 % for hypothetical accident conditions, respectively. Although these bounding values are for the evaluation of transportation system itself and have been developed from reasoned argument and experimental data, it does not mean the fraction of fuel breach is allowed to that value. The SAR should justify release fractions and specific activities, as appropriate.

3. Fuel Integrity during Transportation

Verification of the predicted mechanical performance of fuel after extended dry storage during cask and container handling and normal transportation operations shall be considered in the process of licensing. As the regulations describe the requirements for SNF integrity above, the integrity of SNF shall be maintained for normal operation and there should be no significant cladding failure for hypothetical conditions during transportation as well.

This is to ensure that the transported fuel assemblies will be retrievable at every type of facility they may be shipped to and that the other regulations will be met [1].

However it is not possible to design a transportation system which is able to maintain the SNF intact in perfect condition throughout whole transportation process. Then here comes a question, 'What fraction of fuel rods could be allowed to fail during transportation?'

The answer to this question would be that those failure percentages assumed in evaluating transport system are only guidance and an applicant would be able to come in with fuel values that are greater or less than those percentages as long as they demonstrate compliances to the other regulations. [Private correspondence with NRC staff]

4. Needed Data for evaluating of the Integrity of Cladding

Then, what kinds of material properties of cladding should we look at to evaluate the integrity of fuel rod in terms of failure rate?

The most significant potential degradation mechanisms affecting the fuel cladding during extended storage and transportation are expected to be those related to hydriding, creep, stress corrosion cracking and delayed hydride cracking, affecting its ductility, strength, and fracture toughness. Cladding can become brittle at low temperatures than DBTT. As the burnup goes higher, the pellet-cladding gap decreases, which would increase the cladding stresses and might lead to creep and SCC of cladding in extended storage. Pellet also acts as a resistance of cladding to the pinch mode stress.

Upon reaching the repository site, or other processing facility, the SNF will have to be handled,

and that's why the integrity of it should be verified prior to transportation. If the fuel degrades during extended storage, it could be susceptible to damage from the vibration and shocks encountered during transportation. Here are two main players considered to be affecting cladding integrity mostly during transportation. 1) Hydride Re-orientation during dry storage is perceived to be of critical importance. 2) Ductile-Brittle transition Temperature (DBTT)

Radial hydrides can represent an additional embrittlement mechanism that could potentially result in cladding failure if the cladding temperature decreases below a DBTT and the rods are subjected to stresses higher than the fuel's mechanical limit. Recent research indicates that the hydride reorientation results in a shift of the DBTT of the cladding to a higher value.

To be able to evaluate cladding integrity during transportation, we need to model hydrogen content and hydride profile, hydride re-orientation, and 3-phase metal/hydride mixture in cladding, and also to implement a few of mechanical tests, i.e. bending and fatigue tests. Radial hydride continuity factor and critical strain energy density could be calculated out of those models and with which we can predict what fraction of fuel cladding would be intact or failed by analyzing of pinch loads to the cladding during transportation. This failure fraction should be incorporated in reviewing the transportation system for licensing. The most important thing as a foundation for the modeling of them would be that knowing the exact condition of microstructure of cladding and loaded condition with pellets in fuel rod. The difference between the condition of real SNF and the simulated experiment could misguide the conclusion to the wrong side, especially for the high burnup fuels. Currently, reviews of the transportation of high burnup SNF will be handled on a case-by-case basis due to lack of confirming data until further guidance is developed [4].

5. Summary

For the safe transportation of SNF and licensing, the integrity of SNF should be evaluated carefully. Researches to obtain the data for SNF cladding properties, i.e. impact toughness, DBTT (hydride behavior) when evaluating transportation of SNF, shall be precisely implemented by simulating the condition of real SNF to the hilt, accordingly.

REFERENCES

- [1] US NWTRB, 'Evaluation of the Technical Basis for Extended Dry Storage and Transportation of Used Nuclear Fuel' (2010).
- [2] NRC Interim Staff Guidance-11, Revision 3.
- [3] NRC Interim Staff Guidance-1, Revision 2.
- [4] NRC RIS 2015-XX, 'Considerations in licensing HBF in dry storage and transportation'.

Thermal Analysis of KORAD21 CASK Using COMSOL MULTIPHYSICS

Sunwoo Park and Kwangheon Park*

Nuclear Engr. Department, Kyung Hee Univ., 1732 Deogyong-daero, Giheung-gu, Yongin-si,
Gyeonggi-do 17104, Republic of Korea

*kpark@khu.ac.kr

1. Introduction

Spent fuel is the most important waste to be carefully managed after the release from the nuclear power plants. Due to the decay heat and radiation emitted from the spent fuel, it must be handled carefully for a long time. In Korea, spent fuel has been stored in a wet storage tank located in a nuclear power plant. However, as the wet storage tank becomes saturated, the necessity of using other methods is increasing. The dry storage method is preferred as a substitute for the wet storage method. Dry storage is relatively safe from degradation of the cladding could occur during the wet storage. Dry storage is known to be cheaper to operate than the wet storage facility. In general, Safety analysis of dry storage cask is done in four fields. Thermal analysis, Structural analysis, Radiation shielding analysis, and Criticality analysis are included in four fields. In this study, thermal analysis of KORAD21 CASK designed and fabricated by KORAD was carried out.

2. Methods and Results

2.1 COMSOL MULTIPHYSICS Program

COMSOL MULTIPHYSICS is a multiphysical phenomena analysis program that analyzes various complex phenomena using finite element method. In this study, thermal evaluation of KORAD21 CASK was performed using thermal analysis and flow analysis module of COMSOL MULTIPHYSICS program. In the COMSOL MULTIPHYSICS program, time-dependent analysis method and stationary analysis method are supported. In this study, stationary analysis method was used.

2.2 Modeling and material setup

The KORAD 21 CASK modeled in this study consists of a concrete cask and a metal canister. The

basket is supported by the 22 discs inserted inside of the canister. The air entering through the lower passageway is expanded by the heat released from the metal canister and form natural convection. As the expanded air release through the passageway at the top of the cask, cooling effect appears. Cooling by natural convection is also occurs on the outer surface of the cask. The homogenized fuel assemblies inserted inside the canister reflect the properties of Zircaloy cladding, UO_2 , and Helium. The nuclear fuel burns 50GWD/MTU and has a cooling period of 10 years, resulting in an average heat output of 758.3W and a maximum of 796.2W per assembly. Inside the canister, helium gas was injected with 1atm. Excluding air, the density property is a constant value. The heat transfer coefficient and the specific heat value of materials are described as a function of temperature. In the case of air, density, heat transfer coefficient, and specific heat values were described as functions of temperature.

2.3 Calculation results and analysis

A practical temperature evaluation of KORAD21 CASK using electric heaters was performed. The inside of the KORAD21 CASK is divided into six part in the z-axis direction. Temperature evaluation was performed by an inserted thermocouples. Each thermocouple measure the temperature of the canister shell, cask inner liner, concrete, and cask outer liner at each height. Based on the above experimental results, KORAD verified the thermal analysis result by FLUET code. In this study, the thermal analysis was performed using the COMSOL MULTIPHYSICS program and the results were verified in the same way as KORAD did. From the bottom, the heights of 1630.8 mm, 2630.8 mm, 3630.8 mm, 4130.8 mm, 4630.8 mm and 4970.8 mm were designated as Level 1 through Level 6 in order. In this article, only the results of Level 4, 5, and 6 are shown.

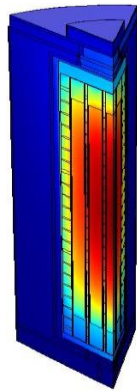


Fig. 1. Thermal analysis result of KORAD21 CASK using COMSOL MULTIPHYSICS.

In the graphs below, the results of the code analysis performed by KORAD, the actual measured results, and the results analyzed by COMSOL MULTIPHYSICS program are compared at the points of Level 4, 5, and 6.

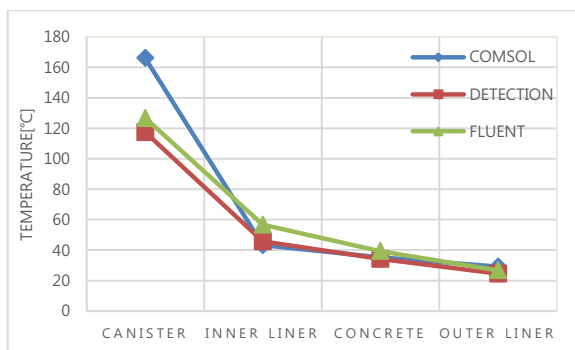


Fig. 2. Temperature profile at Level 4.

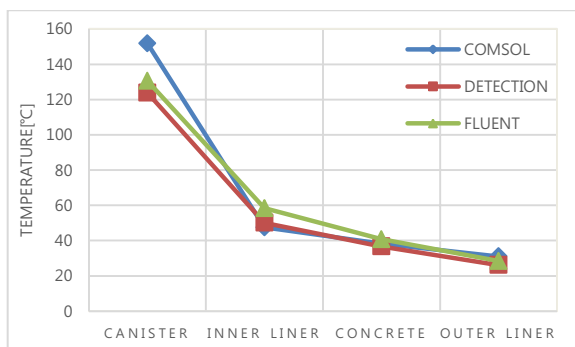


Fig. 3. Temperature profile at Level 5.

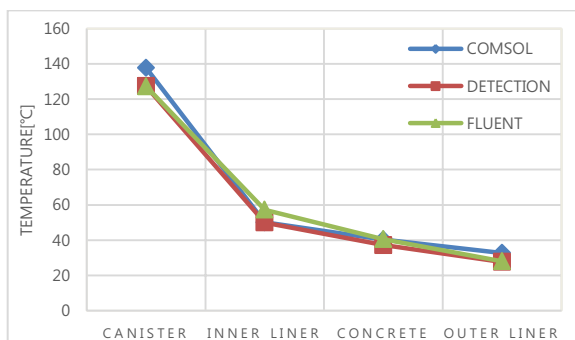


Fig. 4. Temperature profile at Level 6.

Except for the canister part, the analysis results using COMSOL show a tendency to be more consistent with the DETECTION results than the FLUENT code results. In addition, the analysis results of concrete, canister, neutron absorber, and cladding were evaluated to satisfying the temperature limit criteria.

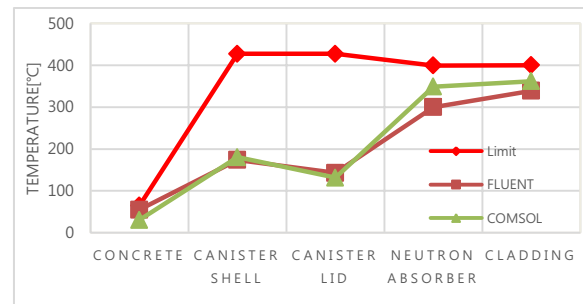


Fig. 5. Comparison of Code Analysis Results and Temperature Limit criteria.

3. Conclusion

The thermal analysis results of the KORAD21 CASK using COMSOL MULTIPHYSICS program were similar to those of code analysis results and the actual measurement results from KORAD. In addition, thermal analysis results of KORAD21 CASK have been found to meet the temperature limit criteria.

REFERENCES

- [1] Ju-Chan Lee, Kyung-Sik Bang, Ki-Seog Seo, Ho-Dong Kim, Byung-Il Choi, Heung-Young Lee, Myung-Jae Song, "Thermal Analysis of a Storage Cask for 24 Spent PWR Fuel Assemblies", 14th International Symposium on the Packaging and Transportation of Radioactive Materials, September 20-24, 2004.
- [2] KORAD, "Concrete Storage System Thermal Safety Analysis Report", 14220-P1-N-TR-030, 2013.

Considerations of Drying System Design for the Dry Storage of Spent Nuclear Fuel

Kyung-Wook Shin*, Byeong-Mok Park, Jae-Hyun Han, Geon-Hui Lee, Gyung-Sun Chae, and Jae-Seok Park

SAE-AN ENGINEERING CO., 184, Gasan digital 2-ro, Geumcheon-gu, Seoul, Republic of Korea

*and9211@sae-an.co.kr

1. Introduction

In the case of Spent Nuclear Fuel(SNF) of domestic the Pressurized Water Reactor (PWR), SNF storage pools reaches sequentially saturation with the beginning at Hanbit nuclear power plant in 2024. In the national basic plan for high-level radioactive waste management, the interim storage facility promotes construction in the site of the permanent disposal facility, adopts the dry storage method considering operation and expansion, economical efficiency. Drying technology of SNF is necessary to store the SNF by the dry storage method. The first trial research for SNF drying from the Pressurized Water Reactor (PWR) is progressing.

The short-term operation is a series of working process within an SNF storage building before the SNF transportation from the storage pool to a dry storage [1]. The processes of short-term operation are SNF loading in cask, cask transfer, cask decontamination, canister internal drainage, canister internal drying and inert gas filling in the canister. Canister internal drying process is the important process to secure the integrity of long-term dry storage of SNF. We conducted the review the design of drying system for the dry storage of SNF from PWR. It includes the survey of commercialized drying methods and system, related standards, any criteria to complete drying process.

2. Considerations of drying system design

The current drying process is divided into a commercial nuclear fuel vacuum drying system and a force gas dehydration system. Vacuum drying system is relatively simple, and forced gas hydration system has the advantages of short drying time and less heat cycle.

2.1 Drying process

In order to secure the safety of SNF transportation

and the long-term integrity of dry storage, the residual water in SNF transportation and storage container should be removed to under the relative standard [2]. The objective of drying process is to remove the residual water in canister to prevent geometric reconfiguration of the packaged SNF, and to prevent canister damage from over-pressurization or corrosion, and to prevent hydrogen induced damage or material corrosion. An evaluating adequate dryness must consider the starting system, the types of water, and water inventory and determine whether the appropriate technology has been applied through the drying process meet the regulatory requirements. Fig. 1 shows the flow diagram to preview the conceptual design of SNF drying process.

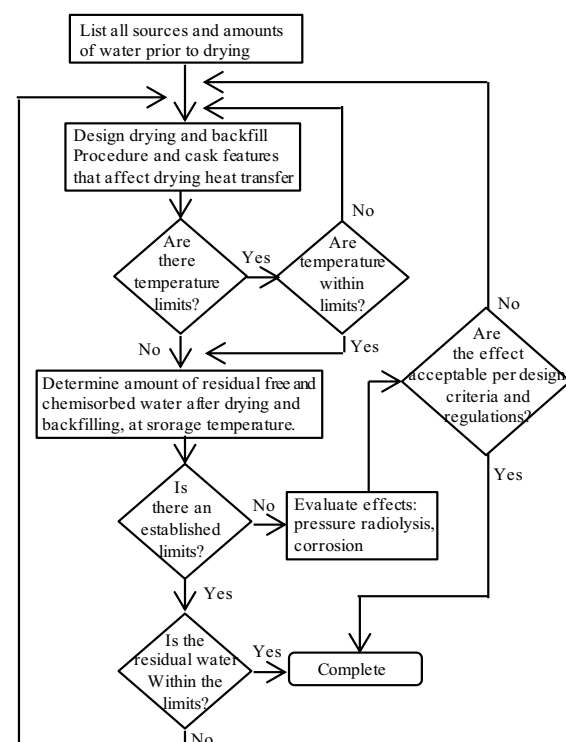


Fig. 1. Flow diagram of SNF drying process.

2.2 Vacuum drying system

Vacuum drying system consists of vacuum pumps, vacuum pipe lines, oil mist filter, gauges, valves,

condensers and chiller unit. In vacuum drying process, vacuum drying system connect to the container exhaust and drain connector on the canister lid part, and reduce the container internal pressure to $4.0\text{E-}04$ Mpa (3 torr) or less. Close the valve in the exhaust system and observe the pressure change for 30 minutes. If it is no change in pressure, the inside of canister complete drying process, and if the pressure increases, the drying operation must repeat. Vacuum drying system is relatively simple than gas circulation, but has some of drawback such as long drying time and ice formation during drying process.

2.3 Forced gas drying system

Forced gas drying system consists of a condensing module, a demohumidizer module, a gas circulator module and a preheater module, which are schematically shown in Fig. 2. The forced gas drying process is effectuated by circulating dry heated inert gas through the storage container loaded with SNF.

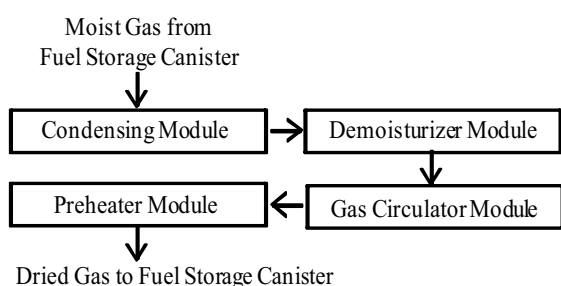


Fig. 2. Flow diagram of forced gas drying process.

The condensing module serves to cool the gas and vapor mixture exiting the canister to a temperature well below its dew point such that water may be extracted from the gas stream. The demohumidizer module receive partially cooled gas exiting the condensing module, progressively chills the recirculating gas to a temperature that is well below the temperature corresponding to the partial pressure of water vapor at $4.0\text{E-}04$ Mpa (3 torr). The gas circulator module provides the motive energy to circulate the gas, which is sized to provide the pressure rise necessary to circulate gas at the requisite rate. The preheater module serves to add supplemental heat energy to inert gas so as to facilitate rapid conversion of water into vapor form. It also acts to dry enough inert gas before entering the canister.

2.4 Dryness requirement

The U.S. Nuclear Regulatory Commission recommends acceptance criteria that the forced gas dehydration be maintained at a gas vapor pressure of less than or equal to $4.0\text{E-}04$ Mpa (3 torr).

In the case of the Forced Helium Dehydration system commercialized by Holtec International incorporation, the acceptance criteria is as follows [3]. The partial pressure of the water vapor in the canister will not exceed 3 torr. The limit will be met if the gas temperature at the demohumidizer outlet is verified by measurement to remain $\leq -6.1^\circ\text{C}$ for ≥ 30 minutes, or if the dew point of the gas exiting the canister is verified by measurement to remain $\leq -5.06^\circ\text{C}$ for ≥ 30 minutes.

3. Conclusion

Canister internal drying process can consider the important process to secure the integrity of long-term dry storage of spent nuclear fuel. We conducted the review the design of drying system for the dry storage of SNF from PWR. It includes the survey of commercialized drying methods and system, related standards, any criteria to complete drying process. This will be a major issue that can be reflected in the design of the forced gas dehydration under study.

Acknowledgement

This work was financially supported by Korea Institute of Energy Technology Evaluation and Planning(KETEP) grant funded by Ministry of Trade, Industry & Energy(20161720200930).

References

- [1] C.Y.Baeg and etc., "Development of the Vacuum Drying Process for the PWR Spent Nuclear Fuel Dry Storage" JNFCWT Vol.14 No4. 435-443, December 2016.
- [2] ASTM C1553-16, "Standard Guide for Drying Behavior of Spent Nuclear Fuel¹", ASTM, 2016.
- [3] "Holtec International Final Safety Analysis Report for the HI-STORM 100 Cask System*" 2016.

Thermal Expansion Analysis of Concrete Spent Nuclear Fuel Dry Storage Developed in Korea

Wooyong Choi and Kwangheon Park*

Nuclear Engr. Department, Kyung Hee Univ., 1732 Deogyong-daero, Giheung-gu, Yongin-si, Gyeonggi-do,
Republic of Korea

*Corresponding author: kpark@khu.ac.kr

1. Introduction

SNF dry storage facility is temporary storage stage before the SNF is reprocessed or moved to final disposal facility. For that reason, the SNF has to be maintained in safe and retrievable in anytime. Concrete SNF dry storage component can be expanded when SNF is loaded because of decay heat of the SNF. In this paper it is evaluated how much SNF dry storage is expanded at the different conditions.

2. Thermal expansion evaluation

2.1 COMSOL MULTIPHYSICS program

COMSOL MULTIPHYSICS program is a simulation tool for electrical, mechanical, fluid flow and chemical analysis in complex natural phenomenon by using finite elements method.

2.2 Structure modeling and materials

To model the concrete SNF storage simply and calculate fast, major components which affect heat transfer and solid mechanics analysis significantly, are selected and modeled in two dimensionally.

Several types of materials are adopted to consist of each components. Zircaloy-4 is used for Fuel assembly in the fuel basket. SA 240 type 304 stainless steel is used for fuel basket and disc. Canister shell consists of SA 240 type 316L. Lastly, concrete and carbon steel are used for concrete cask and liner. Because material properties change as

temperature changes, material properties are adopted as functions of temperature.

2.3 Evaluation method

SNF peak temperature, already evaluated in Safety Analysis Report (SAR), is used as heat source. Heat distribution evaluation of dry storage is conducted in normal and off-normal (50% and 100% air inlet is blocked) conditions in two dimensional model. Thermal expansion evaluation is conducted by using the temperature from heat distribution evaluation.

2.4 Result and discussion

Fig. 1 and Table 1 are the result of heat distribution evaluation in two dimensional model. Surface temperature of canister shell is almost similar to heat distribution evaluation in SAR but surface temperatures at concrete and liner are lower than SAR.

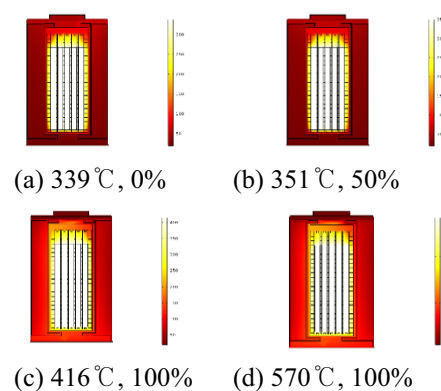


Fig. 1. Heat distribution depending on fuel temperatures and percentage of blockage.

Table 1. Max. Temperature of components depending on fuel temperatures and percentage of blockage

(Unit: °C)

	Blockage of air inlet	Adopted Fuel temp.	Canister shell	Liner	concrete
Normal condition	0 %	339	176.01	69.276	67.095
	0 %	400	205.14	78.081	75.609
Off- normal condition	50 %	351	184.44	55.817	52.893
	100 %	416	288.85	147.26	140.35
	100%	570	403.64	205.45	196.62

Fig. 2 and Table 2 are the result of thermal expansion evaluation. Displacements of most case are not severe in both radial and axial directions. However, when the air inlet is 100% blocked and SNF cladding temperature is 570°C, which is temperature limit in off-normal conditions, displacement of canister shell is 13.085 mm. Since the gap between canister shell and cask structure (T-channel) is 10 mm, the gap can be disappeared although fuel cladding temperature is less than limit.

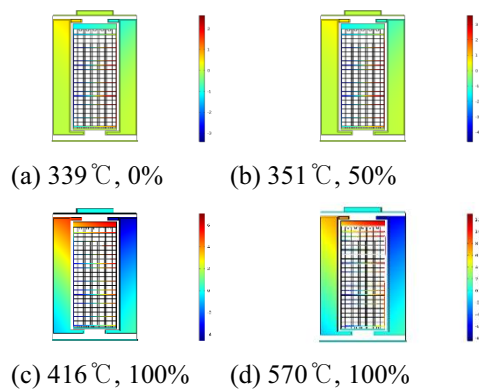


Fig. 2. Thermal expansion evaluation depending on fuel temperatures and percentage of blockage.

Table 2. Max. Displacement of canister shell based on heat distribution evaluation result

(Unit: mm)

	Blockage of air inlet	Adopted Fuel temp (°C)	Radial displace- ment	Axial displace- ment
Normal condition	0 %	339	2.643	5.2483
	0 %	400	3.6016	6.9406
Off- normal condition	50%	351	3.029	5.0813
	100%	416	6.1393	15.758
	100%	570	13.085	29.817

3. Conclusion

According to evaluation result, it seems that canister shell cannot maintain retrievable condition in this off-normal condition. Therefore, thermal expansion effect needs to be considered in safety analysis importantly.

REFERENCES

- [1] KORAD, "Concrete Storage System Thermal Safety Analysis Report", 14220-P1-N-TR030, 2013.

A Study on Drop Analysis of the Spent Nuclear Fuel Metal Cask (OASIS-32D)

Hye Jin Lim*, Jang Won Lee, Jung Gyu Kim, Ki Kwang Sung, and Hyun Min Kim

NSSS Division, KEPCO Engineering & Construction Company, INC., 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Republic of Korea

*jin@kepc-co-enc.com

1. Introduction

The purpose of this study is to evaluate the structural integrity of spent nuclear fuel cask, which OASIS-32D is transport and storage metal cask developed by KEPDO-E&C [1]. OASIS-32D should be designed to meet the requirements of 10CFR 71 [2]. The 9m free drop accident presented in 10CFR 71 is considered as the most critical scenario of hypothetical accident conditions during transport.

In this paper, the 9m free drop for OASIS-32D using the LS-DYNA program was evaluated. The results of the analyses confirm that the structural integrity of OASIS-32D is maintained.

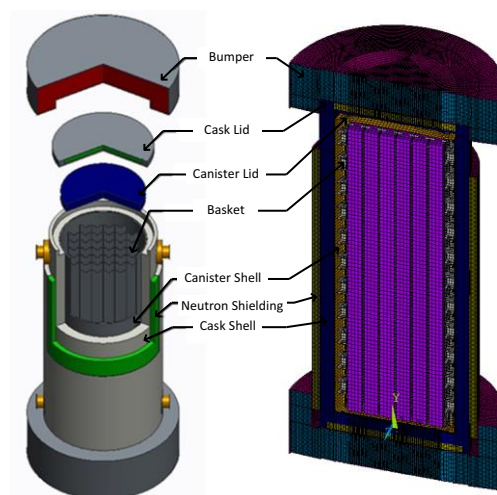


Fig. 1. Geometry and FE model of OASIS-32D.

2. Development of modeling

2.1 Geometry

OASIS-32D consists of the canister and cask as shown in Fig. 1. The canister is comprised of a canister shell, canister lid, fuel bundles, basket, and so on. The cask is composed of a cask lid, cask shell and bumpers used as energy-absorbing components to protect the cask.

2.2 Modeling

To improve the efficiency of analysis, the components such as trunnions, heat transfer fins, etc. cause little influence on the drop analysis were not considered for modeling. The fuel assembly is modeled as a simplified hexahedron with solid elements. However, the total weight of OASIS-32D is maintained using the equivalent density. The canister and its contents are modeled using the shell element because these components are very thin relative to its height, while cask is modeled as the solid element. Fig. 1 shows the finite element model.

3. Drop analysis

3.1 Method

The 9m free drop analyses are performed for two typical conditions, horizontal and vertical drop.

The target surface is conservatively considered as the rigid, and all DOF of the surface are restrained for boundary condition. The initial velocity of 13.3 m/s corresponding to the 9m free drop condition is applied on the cask and canister. The contact condition between bumpers and target surface is the node to surface contact used in impact analysis.

3.2 Results

Table 1 shows the ratio of the maximum stress intensities to allowable stress based on the Regulatory Guide 7.6 [3]. The stress intensities of the horizontal drop case are less than those of the vertical drop case because the bumpers in the horizontal drop case absorb more impact energy.

Fig. 2 and 3 show the energy histories and stress

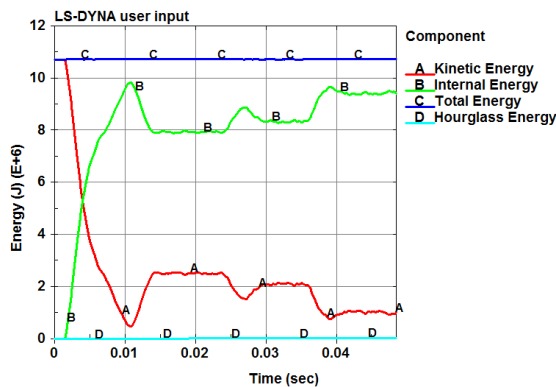
results of drop analyses. The appropriateness of the 9m drop analysis is confirmed from the constant total energy and small hourglass energy as shown in Fig. 2.

Although Table 1 shows that the maximum stress intensity of cask shell is close to the allowable, it is occurred only at the very small area in the inner surface of the cask bottom as shown in Fig. 3(a). Therefore, it is ensured that the structural integrity of OASIS-32D is maintained for the 9m free drop analyses at two conditions.

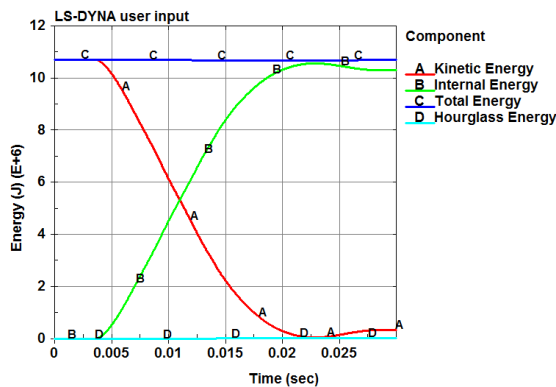
Table 1. Maximum stress intensities

Components		Maximum S.I.	
		Vertical	Horizontal
Cask	Lid	0.80	0.51
	Shell & Bottom	1.00	0.63
Canister	Lid	0.53	0.26
	Shell	0.79	0.60
	Bottom	0.75	0.59

Note: Normalized values to allowable stress [3].

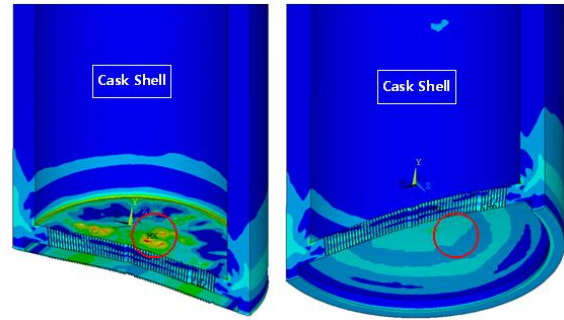


(a) Vertical drop



(b) Horizontal drop

Fig. 2. Energy histories for 9m free drop analyses.



(a) Vertical drop



(b) Horizontal drop

Fig. 3. Stress distribution.

4. Conclusion

The 9m free drop analyses of OASIS-32D were carried out for typical conditions. The results of the analyses show that the structural integrity of OASIS-32D is maintained for the 9 m free drop analyses of horizontal and vertical drop conditions.

In the near future, the additional drop analyses considering various drop conditions will be performed to evaluate the structural adequacy of OASIS-32D.

REFERENCES

- [1] K.J. Ko, S.H. Kim, M.G. Kim, Y.H. Cho, H.M. Kim, J.G. Ahn, "Design Features of an OASIS-32D Metal Cask for both Transport and Storage of SNF" Proc. of the KRS 2016 Autumn Conference, 14(2), Oct. 12-14, 2016, Jeju.
- [2] US NRC 10CFR 71, "Packaging and Transportation of Radioactive Material", Jan. 2017.
- [3] US NRC Regulatory Guide 7.6, "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels", March 1978.

Seismic Evaluation for a Fuel Assembly in Storage Rack of Spent Nuclear Fuel Pool

Sunghwan Chung* and Kiyoun Kim

Central Research Institute, Korea Hydro & Nuclear Power Co., Ltd., 70, Yuseong-daero 1312beon-gil,
Yuseong-gu, Daejeon, Republic of Korea

*sunghchung@khnp.co.kr

1. Introduction

Storage rack of spent nuclear fuel pool (SFP rack) of nuclear power plants (NPPs) is a structure to store safely spent nuclear fuel generated from NPPs. SFP rack must be designed and evaluated in accordance with seismic category I structure classified by US NRC's RG 1.29 [1], however, SFP rack evaluation was done on SFP rack itself rather than on the fuel. US NRC suggested that the fuel in SFP rack must be maintain the integrity without damage due to seismic loads. US NRC recommended in NUREG-0800 [2] Sec. 3.8.4 Appendix D to demonstrate that the impact load on fuel to seismic excitation does not lead to damage of the fuel and to evaluate fuel deformation resulting from seismic loads does not degrade the coolable configuration of fuel. This is also issued in US NRC's design review process for APR 1400 NPP. In this study, a fuel assembly in a cell of SFP rack was modeled 3-dimensionally in detail and the behavior of fuel rods against seismic loads was analyzed. Because fuel damage refers the fuel assembly elements stressed beyond the material allowable limit such that the fuel rods were no longer able to provide confinement for radioactive materials, maximum stresses generated on the fuel rods caused by seismic loads were assessed. Relative displacements and spacing between the fuel rods of fuel assembly were analyzed to ensure the coolable configuration was maintained even if fuel rods contacted with seismic loads.

2. Seismic evaluation

Seismic analysis was carried out on one fuel assembly in the storage cell by fixing the fuel bottom to SFP rack and allowing all loads to directly act on the fuel assembly under maximum displacement conditions of the upper and lower ends of SFP rack using seismic analysis results for SFP rack (Fig. 1).

Reinforced standard of 0.3g seismic acceleration was applied. Analyses were carried out using the explicit FEM program of LS-DYNA R7.1 [5]. Analysis models of one storage cell and one fuel assembly using shell, beam and solid elements are shown in Fig. 2.

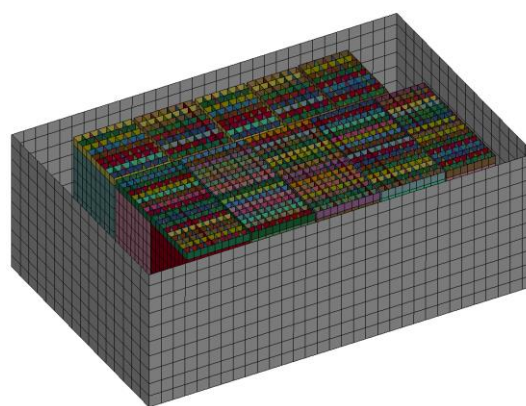


Fig. 1. Analysis model for a whole SFP rack.

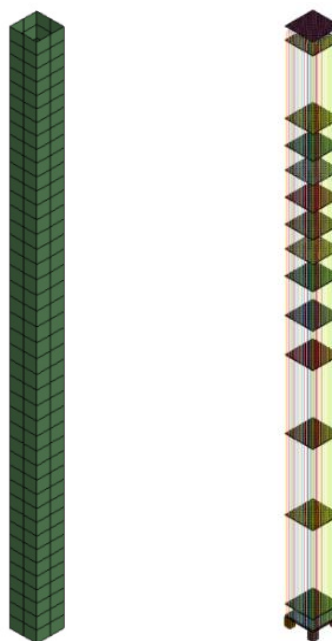


Fig. 2. Analysis models of a cell and a fuel assembly.

Maximum relative spacing and maximum stresses between the fuel rods of fuel assembly in the storage cell due to seismic loading are shown in Table 1 and 2, respectively. Spacing between the fuel rods and maximum stresses on the fuel rods were obtained by dividing the height of fuel assembly into 12 parts in East-West and North-South directions (Fig. 3). Negative values of the fuel rod spacing indicates that the fuel rods were in contact with each other and many fuel rods were contacted. There was no contact between the fuel rods at locations 8 and 9 where grid gap is narrow, but the fuel rods were in contact with each other at locations 1, 10 and 12 having a relatively large grid gap. As the fuel rod was modeled as a beam element, contact due to the thickness of fuel rod could not be considered. Stresses generated on the fuel rods were caused by the contact of fuel rods. Maximum stresses of the fuel rods were compared with the tensile strength at which at which the fuel rod is broken, but the fuel rods were not broken because all maximum stresses did not exceed the tensile strength of 651 Mpa.

Table 1. Maximum relative fuel rod spacing

Loc.	East-West (mm)		North-South (mm)	
	1-2	5-6	2-3	4-5
1	-0.46	-0.14	-0.57	-0.79
8	1.19	1.03	1.27	1.24
9	1.38	0.93	1.47	1.05
10	-0.37	-0.43	-0.54	-0.68
12	-0.74	-0.48	-0.64	-0.38

Table 2. Maximum stresses on the fuel rods

Location	Max. stress (MPa)	Tensile strength (MPa)
1	178	651
8	166	651
9	154	651
10	181	651
12	148	651

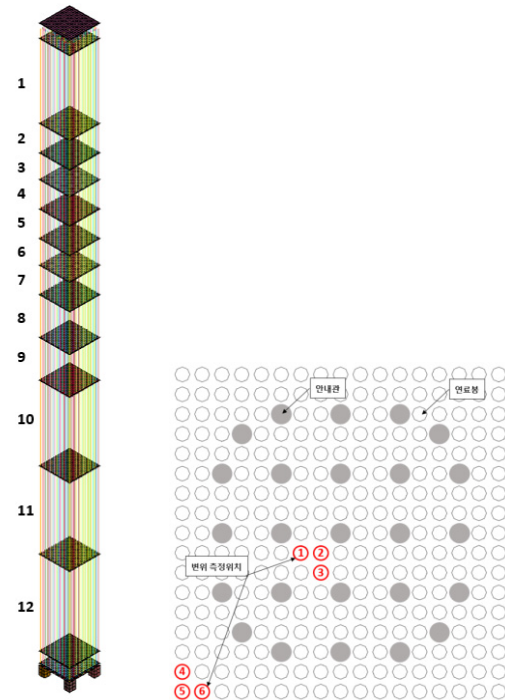


Fig. 3. Measurement locations.

3. Conclusion

It is considered that the coolable configuration of fuel assembly is difficult to be maintained because the fuel rods of fuel assembly contact each other due to seismic impact loads. However, as all maximum stresses generated on the fuel rods did not exceed the tensile strength, so that the fuel rods can be said to maintain the confinement of fuel rod. It is not easy to simulate and analyze all the fuel assemblies in the SFP rack in seismic evaluation of the fuel stored in SFP racks, so it is necessary to seek a more efficient method.

4. References

- [1] US NRC, RG 1.29, Seismic Design Classification.
- [2] US NRC, NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.
- [3] US NRC, RG 1.60, Design Response for Seismic Design of Nuclear Power Plants.
- [4] US NRC, OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, 1978.
- [5] US LSTC, LS-DYNA Program Version 7.1, 2014.

Electro-Deposition of Cr Layer to Prevent Fuel Cladding Chemical Interaction in Sodium Fast Reactor (SFR)

Sunghwan Yeo*, Koo Hye Yeon, and Junhwan Kim

SFR Fuel Development Division, KAERI, Daedeok-daero 989-111, Yuseong-gu, Daejeon, 305-353, Korea

*Corresponding author: syeo@kaeri.re.kr

1. INTRODUCTION

Disposal of high level nuclear waste is crucial problem in many countries that have nuclear plants. Small countries such as South Korea especially has limited disposal area and suffer from the continuously produced nuclear waste. A sodium cooled reactor (SFR) have attentions due to its capability to transmute nuclear waste.

In the core of a SFR, there are nuclear fuels and cladding. Unlike a conventional light water reactor, sodium reactors consider to use metallic fuels due to high thermal conductivity, great capability with sodium, and easier to fabricate [1]. Cladding is a important structural component that maintain the integrity of nuclear fuel and prevent fission product release, and transmit thermal energy to sodium coolant. Ferritic-martensitic steels are utilized for the cladding material. They have great swelling resistance up to 200 dpa, have high thermal conductivity, and low thermal expansion coefficient. HT9 has been adopted for the SFR research programs in many countries.

A crucial concern however is the reaction between actinide or rare earth elements in a metallic fuel and ferritic-martensitic steel cladding [2]. This phenomenon is named Fuel Cladding Chemical Interaction (FCCI). Lower melting point eutectic compounds are formed by this FCCI decreasing the integrity of cladding and nuclear fuel.

The coating of inner surface of FM steel cladding has been researched world widely to reduce FCCI. The coated layer acts as a diffusion barrier to actinide and rare earth elements. Various research teams use diverse coating methods and materials. Fauzia Khatkhatay et al., [3] performed pulsed laser deposition to coat TiN and ZrN on the surface of cladding. They performed Fe/Ce diffusion couple tests. TiN and ZrN coating layers with only 500nm thickness perfectly prevented chemical reaction between cerium and steel. Vahid Firouzidor et al., [4,

5] developed Yttrium Stabilized Zirconia (YSZ) and TiO_2 coating layers on steel substrates using electrophoretic deposition. Fe/Ce diffusion couple experiments showed that TiO_2 coating reduced up to 83% solid state inter diffusion. Moreover, no diffusion was observed in $8\mu\text{m}$ YSZ coated Fe and Ce diffusion couple tests.

In this study, Cr coating layer was deposited on HT9 substrate using electro-deposition method. When compared to other candidate materials such as TiN, ZrN, YSZ, and TiO_2 , Cr obtains higher level of comparability with cladding. For example, thermal expansion coefficients of Cr and Fe are similar. Therefore, Cr coating is expected to sustainably prevent chemical diffusion between metallic fuel and cladding. Our previous publications showed FCCI was successfully prevented by Cr coating layer. Electro-deposition is simple and efficient method for Cr coating. Hexavalent chromium (Cr^{6+}) in an aqueous solution is deposited on substrate by applied electric current.

Because the service environment of cladding is the core of SFR, a cladding is exposed to elevated temperature $\sim 650^\circ\text{C}$ for thousands hours. Therefore, it is essential to investigate if the microstructure and property of Cr coating layer is modified at the target temperature for prolonged time. However, to the best of the authors' knowledge, the information of Cr coating layer property at that temperature is limited. In this study, Cr coated HT9 disks were heated at 650°C for 50 and 250 hours. The microstructure of Cr coating layers was then observed to examine the influence of thermal cycle on the property of Cr layers.

2. EXPERIMENTAL

The detailed experimental procedure was described in [2, 6, 7]. Therefore, only a brief description is provided here. Cr electro-deposition was performed on polished HT9 disks. While HT9

disk was coated by Cr at cathode, Pb-Sn alloy was selected as anode. A sargent bath is water based solution composed of 250 g/l chromic acid (CrO_3) and 2.5 g/l sulfuric acid (H_2SO_4). The temperature of sargent bath was maintained at 50°C and 80°C during the electro-doposition. Both temperatures were selected because the property of Cr coating layer was dramatically changed at those temperatures. While the hardness of Cr coating layer is greater and cracks are observed at 50°C coating, 80°C coating result in crack-free and softer Cr layer. Moreover, the thickness of Cr coating layer at 50°C is almost twice the layer coated at 80°C [6]. HT9 disks Cr coated at 50 and 80°C for 70 mins were vaccum sealed in a glass to prevent oxidation at high temperature. The vaccum sealed glass was then inserted in a box furnace and heated at 650°C for 50 and 250 hours. Specimens were cut in half after heating tests to reveal the cross section. Microstructure of polished cross sections was observed using Scanning Electron Microscopy (SEM, Model Tescan, VEGA easy probe).

3. RESULTS AND DISCUSSION

While the thickness of Cr layer at 50°C was 40µm, only 25µm thickness was obtained by 80°C coating. This thickness difference implies that Cr deposition is more efficient at 50°C than 80°C. Another important feature observed in the Cr layer at 50°C is cracks. The direction of most cracks is proportional to the growth direction of Cr. Moreover, they seem to be indistinct and looked as slight defects. It is known that cracks are formed due to the volume shrinkage of chromium hydrides deposited in Cr layer. The stress increases with increasing thickness of coating layers.

The slight cracks observed in as-deposited Cr layer became more noticeable in heat treated Cr layer for 50 hours. The cracks were even more obvious in the heat treated Cr layer for 250 hours. Interfacial debonding was observed at the interface between HT9 and Cr layer. In the light of the fact that actinide and rare earth elements could easily diffuse through those deep cracks, it is beneficial to produce Cr layer without cracks. Meanwhile, unlike the deposited Cr layer at 50°C, the 80°C Cr layer maintained crack-free microstructure even after heat treatments.

4. CONCLUSIONS

Electro-deposition was performed to coat Cr layers on HT9 disks to hinder FCCI at both 50°C and 80°C. Cracks are observed only in the Cr layer coated at 50°C due to the thickness. The observed cracks became severer after heat treatment at 650°C for 50 and 250 hours. Meanwhile, the Cr layer coated at 80°C maintained its crack-free microstructures even after the heat treatments. Because cracks are diffusion paths for actinides and rare earth materials, 80°C bath temperature seems to produce better Cr coating layer to prevent FCCI.

ACKNOWLEDGEMENT

This work was conducted under the ‘SFR Cladding Development Project’ supported by the National Nuclear R&D program through the Sodium-cooled Fast Reactor Development Agency funded by the Ministry of Science, ICT and Future Planning.

REFERENCES

- [1] I. Kim, F. Khatkhatay, L. Jiao, G. Swadener, J.I. Cole, J. Gan, H. Wang, *Journal Of Nuclear Materials*, 429 (2012) 143-148.
- [2] J.H. Kim, H.J. Ryu, J.H. Baek, S.J. Oh, B.O. Lee, C.B. Lee, Y.S. Yoon, *Journal Of Nuclear Materials*, 394 (2009) 144-150.
- [3] F. Khatkhatay, J. Jian, L. Jiao, Q. Su, J. Gan, J.I. Cole, H. Wang, *Journal Of Alloys And Compounds*, 580 (2013) 442-448.
- [4] V. Firouzdor, J. Brechtel, L. Wilson, B. Semerau, K. Sridharan, T.R. Allen, *Journal Of Nuclear Materials*, 438 (2013) 268-277.
- [5] V. Firouzdor, J. Brechtel, B. Hauch, K. Sridharan, T.R. Allen, *Applied Surface Science*, 282 (2013) 798-808.
- [6] S.W. Yang, H.J. Ryu, J.H. Kim, B.O. Lee, C.B. Lee, *Journal Of Nuclear Materials*, 401 (2010) 98-103.
- [7] H.J. Ryu, B.O. Lee, S.J. Oh, J.H. Kim, C.B. Lee, *Journal Of Nuclear Materials*, 392 (2009) 206-212.

3분과

고준위폐기물처분 (Oral)



Application of Thermo-Hydro-Mechanical Damage Model on Triaxial Test of Granite

Jin-Seop Kim^{1*}, Jun-Seo Jeon², Jung-Woo Kim¹, and Geon-Young Kim¹

¹Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

²Korea Advanced Institute of Science and Technology, 291, Daehak-ro, Yuseong-gu, Daejeon, Korea

*kjs@kaeri.re.kr

1. Introduction

In a repository for radioactive waste disposal, the coupled thermo-hydro-mechanical (THM) behavior in porous media is one of the important issues from the perspective of a long-term safety concerns. Formation of microcracks, which is primarily caused from damage evolution in a repository are essential consideration in the evaluation of the long-term performance of waste disposal system. The accumulation of damage induces changes in the mechanical, hydraulic, and thermal properties of the rock.

The objective of this study is to propose the coupled thermo-hydro-mechanical damage (THMD) model to investigate the time-dependent deformation and failure process of rock.

2. Approach

In the model development, Young's modulus, permeability and thermal conductivity of rock are changed in accordance to damage evolution. The THMD model development is primarily performed by COMSOL Multiphysics ver. 5.3. Two-phase Darcy's law, solid mechanics and heat transfer in porous media modules are used in this study. The model includes the accumulation of damage applied to individual elements, which modifies the modulus, permeability and thermal conductivity. With regard to the damage model application to THMD

development, Mazars damage model is used in the structural mechanics module and it is incorporated into COMSOL as an external material in this study (www.comsol.com/blogs).

3. Model setup

The mode geometry is presented in Fig. 1 below. A cylindrical apparatus with a 10 cm diameter and a 20 cm height, presses the rock sample from the top by a prescribed pressure (50 MPa). The sample is radially constrained by a pressure of 10 MPa.

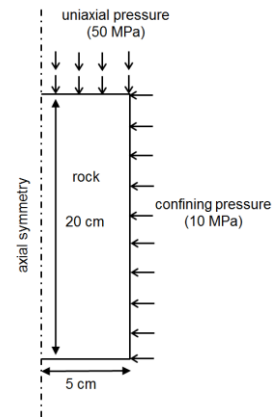


Fig. 1. Dimensions and loading condition for triaxial test.

Governing equations for water and gas phases used in the model development are as follows:

$$\rho_w S \frac{\partial p^w}{\partial t} + \nabla \rho_w \left\{ -\frac{kk^rw}{\mu^w} \cdot (\nabla p^w) \right\} = Q_m \quad (1)$$

$$\rho_g S \frac{\partial p^g}{\partial t} + \nabla \rho_g \left\{ -\frac{kk^rg}{\mu^g} \cdot (\nabla p^g) \right\} = Q_m \quad (2)$$

where S is a storage coefficient, which can be interpreted as weighted compressibility of the bulk aquifer material and the fluid in the pores. Q_m is a mass source term.

Damage is described by two different evolution laws under tension and compression in Mazars' damage model.

$$D = \alpha_t * D_t + \alpha_c * D_c \quad (3)$$

$$D_t(k) = 1 - \frac{k_0(1-A_t)}{k} - A_t e^{B_t(k-k_0)} \quad (4)$$

$$D_c(k) = 1 - \frac{k_0(1-A_c)}{k} - A_c e^{B_c(k-k_0)} \quad (5)$$

where D is total damage, D_t and D_c indicate tensile and compressive damage. A_t , B_t , κ_0 , A_c , and B_c are material parameters which can be determined by experiments.

4. Main Results

4.1 Variation of thermal and hydraulic properties

Fig. 2 shows the changes of porewater pressure and corresponding gas pressure with time.

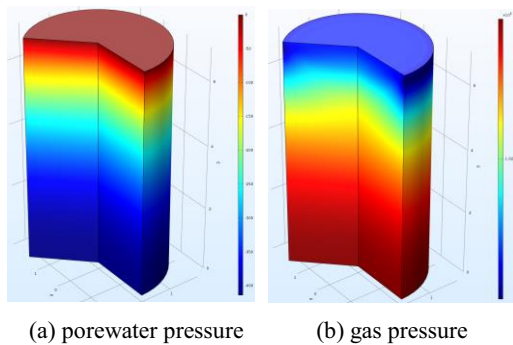


Fig. 2. Pressure variation at 100d under THM triaxial test.

4.2 Damage evolution under THM triaxial test

Damage evolution of rock is presented in Fig. 3. Deviatoric stress of 40 MPa causes the maximum degree of damage of 0.12 in the THM triaxial test. The additional displacement of a specimen has been

produced due to the temperature rise. This is attributed to the fact that thermal expansion term is additionally taken into account in contrary to HM test condition.

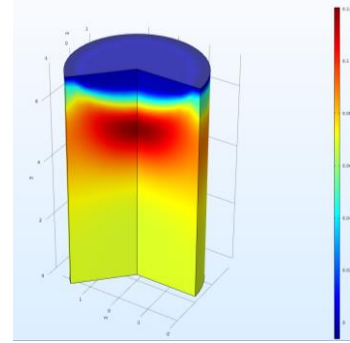


Fig. 3. Damage evolution at 1000d under THM triaxial test.

5. Conclusion

The THM model development in which damage evolution is taken into account can contribute to the better prediction of THM behaviors in a nuclear waste repository.

REFERENCES

- [1] C. Wei, W. Zhu, S. Chen and P.G. Ranjith, "A coupled Thermal-Hydrological-Mechanical damage model and its numerical simulations of damage evolution in APSE", *Materials*, 9(841), 1-19 (2016).
- [2] J. Mazars, "A description of micro-and macroscale damage of concrete structures", *Engineering Fracture Mechanics*, 25(5/6), 729-737 (1986).
- [3] R.W. Lewis and B.A. Schrefler, "The finite element method in the static and dynamic deformation and consolidation of porous media", John Wiley & Sons, England, 1998.
- [4] COMSOL AB. Comsol Multiphysics Ver. 5.3, Reference Manual and Application Libraries.

Proposal on Regulatory Guidelines for the Assessment of Human Intrusion Into a Radwaste Repository

Dong-Keun Cho ^{a*}, Hyosub Kim ^b, Jung-Woo Kim ^a, Jongtae Jeong ^a, Heui-Joo Choi ^a, Jong-Youl Lee ^a,
Jae-Won Lee ^a, and Min-Hoon Baik ^a

^aKorea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

^bDepartment of Nuclear Engineering, Hanyang University, 222 Wangsimni-ro, Seongdong-gu, Seoul, Korea

*dkcho@kaeri.re.kr

1. Introduction

Recently, IAEA has carried out the HIDRA (Human Intrusion in the Context of Disposal of Radioactive Waste) project [1] in order to derive an assessment methodology of human intrusion into radwaste disposal facility through extensive discussions among participants from member states. Since the methodology derived from the discussions could not reflect all legal and technical circumstances of each country, the application of this methodology to disposal facilities in our country might not work. Therefore, regulatory guidelines needed to make the methodology suggested in the HIDRA project work adequately to the disposal facility in Korea were proposed in this paper.

2. HIDRA Approach

Since HIDRA methodology for an evaluation of human intrusion is presented in detail in References 1 and 2, we describe only overall things related with this study.

The method for assessing human intrusion into a disposal facility is described in detail in Ref. 2, published by the author. To summarize, it is necessary to identify the stylized scenarios taking into account the regulatory requirements and the disposal facility, and to establish a protective measure against human intrusion. The ultimate goal is to reduce the likelihood of a human intrusion and to reduce the consequences when the intrusion occurs. Protective measures are continuously updated and optimized to increase robustness of the disposal facility [1, 2].

Assessment of a human intrusion begins with the description of the safety framework, which identifies the considerations on human intrusion for a given disposal facility, followed by description of site- and facility-specific human intrusion scenarios and potential measures. After site- and facility-specific scenarios and potential measures are identified, protective measures is derived by taking into account safety function, human and environmental impacts, economics, and cost. What is important is that no matter how effective it is in terms of prevention of human intrusion, if it jeopardizes the safety function,

it cannot be accepted as a protective measure against human intrusion [1, 2].

The procedure on how to derive a protective measure is described in References 2 in detail.

3. Disposal Facility

The disposal silos for low- and intermediate-level waste at Gyeongju repository [3] were considered in this study. This facility was completed in August 2015, and is now in operation.

Since the silo has already been constructed and is in operation, it is not possible to consider the potential measures to be applicable in the stage of site selection, facility design, and construction. Only the potential measures at the operational stage can be considered.

4. Discussions for Application of HIDRA Approach

The clauses for regarding human intrusion at LILW repository facility are only described in Notice No. 2016-25 of NSSC (Nuclear Safety and Security Commission) [4], describing “Human intrusion refers to all activities affecting the performance of isolation of radioactive waste, which results in radiation exposure to human being” and “After the institutional control period, the radiological effects by the human intrusion into a disposal facility should be maintained below the dose limit for the public as low as possible”.

The premises of the HIDRA methodology are as follows: First, the purpose of the human intrusion analysis is to reduce the likelihood of the intrusion and to optimize for increase in robustness of the disposal facility, not to determine whether dose limit is satisfied or not. Second, only those human actions that result in direct disturbance of the disposal facility (i.e. the waste itself, the contaminated near field or the engineered barrier materials) are considered as human intrusion. Third, a human intrusion is evaluated based on stylized scenarios to exclude uncertainty from detailed scenarios. Fourth, a deliberate human intrusion does not need to be considered, and even if an inadvertent human

intrusion occurs, human actions after recognizing the fact that there is hazardous material need not to be taken into account. Fifth, the time at which a human intrusion occurs is reasonable to assume to occur after promptly after the institutional control period. Sixth, it is reasonable to assume that the level of technology, eating and living habits of the future society are identical to those of the current society in order to exclude uncertainty.

According to the Notice of NSSC in Korea, the disposal facility for LILW should be designed to satisfy the exposure dose limit for all human activities near the disposal facility less than 1 mSv/yr based on the two clauses regarding human intrusion by NSSC. This could be interpreted that human intrusion scenario where the well is installed penetrating the disposal silo at Gyeongju repository should satisfy dose limit of 1 mSv/yr. However, according to the HIDRA methodology, installing a well through silo is considered to be the human intrusion to optimize the disposal system by adoption of a protective measure, rather than satisfying the dose constraint. There are obvious differences in the interpretation of human intrusion between domestic regulation and HIDRA methodology. Namely, there is too much conservatism in the domestic regulations on interpreting human intrusion issue. In addition, there is no description on a human intrusion scenario, the time of intrusion, the technology and habit of the future generations. Therefore, supplementary guidelines are needed to exclude ambiguities. Table 1 describes further guidelines concerning regulatory aspects of human intrusion assessment.

Table 1. Proposed Guidelines for HIDRA Approach

Premises for Human Intrusion Assessment

- Human intrusion into a disposal facility could be regarded as following three classes:
 - (i) intrusion with full knowledge of the existence, location, nature and contents of the facility, (ii) Intrusion without prior knowledge of the disposal facility, (iii) Intrusion with knowledge on the underground facility but without understanding what it contain.
- Developer/operator should consider the second and third of these classes.
- Examples of classes that developers/operators should consider as human intrusion are as follows:
 - (a) human intrusion directly into the disposal facility,
 - (b) other human activities that damage the facility barriers or degrade their functions, such as closed tunnel excavation.
- Human intrusion in area of the above events should be optimized to increase robustness rather than to satisfy dose constraints.
- Human actions except above activities should satisfy the risk limit of 10^{-6} /yr.

Human Intrusion Scenario

- Human intrusion scenario should be separately considered from both scenarios: normal scenario considering an evolution of the disposal facility and abnormal scenario considering unforeseen events.

- The human intrusion should be assessed based on the stylized scenarios that represent the typical activities that the future generations can carry out on the disposal facility. The custom and habits of the future generation considered in the scenario should be assumed to similar to those of the past and current generations.

Ways to Reduce Likelihood of Human Intrusion

- Developers/operators should consider all practical ways to reduce the likelihood of human intrusion and implement all these to the disposal facility. These kinds of measures should not compromise the safety performance of the disposal system.

Human Intrusion Impact Analysis on the Public

- Developers/operators should evaluate the radiation dose from human intrusion for the person living near repository.

5. Summary and Conclusion

In this paper, regulatory guidelines were proposed to make the methodology suggested in the HIDRA project work adequately to the disposal facility in Korea. As a result, further guidelines regarding definition of the human intrusion, purpose of assessment of human intrusion, level of human intrusion scenario, and other premises were proposed.

It is expected that new guidelines proposed in this study could be useful when the guidelines regarding assessment of a human intrusion are established in Korea.

ACKNOWLEDGEMENT

This work is based on the outcomes from HIDRA (Human Intrusion in the context of Disposal of RadioActive Waste) Project by International Atomic Energy Agency [1] and was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government (the Ministry of Science and ICT (MSIT)) (No. 2017M2A8A5014856)

REFERENCES

- [1] HIDRA-The International Project on Human Intrusion in the context of Disposal of RadioActive Waste, Version 1.1, International Atomic Energy Agency Report, 2015.
- [2] D.K. Cho, J.W. Kim, J.T. Jeong, M.H. Baik, "Review on Assessment Methodology for Human Intrusion into a Repository for Radioactive Waste", J. of Nuclear Fuel Cycle and Waste Technology, 14(3) 297-305, 2016.
- [3] J. C. Ha et al., Gas Migration in Low- and Intermediate-Level Waste (LILW) Disposal Facility in Korea', Journal of Nuclear Fuel Cycle and Waste Technology, Vol. 12(4), pp. 267-274, 2014.
- [4] Nuclear Safety and Security Commission, Criteria for Prevention of Radiological Hazard from Disposal Facilities for Low- and Intermediate-Level Radioactive Waste, Notice No. 2016-25, 2016.

Integrated Risk-Weighted Radioactivity Flux Ratios as Performance Indicators of Defense-in-Depth for HLW Disposal System

Chan Woo Jeong*, Jinyong Park, Eun Jin Seo, and Sung Il Kim

Korea Institute of Nuclear Safety, 62 Gwahak-ro, Yuseong-gu, Daejeon, Korea

*jcw@kins.re.kr

1. Introduction

The heart of safety cases for developing a high-level radioactive waste (HLW) disposal system is to construct defense-in-depth arguments supporting the disposal safety, the basic considerations of which are presented in Article 7 of the related NSSC Notice to deep geological repository [1]. In addition, the relevant safety case may need more concrete indicators of system performance in order to optimize the subsystems, to improve the system by applying Best Available Techniques, or to compare the system alternatives for protective capability. The performance may be expressed in many ways linking the related factors to the system characteristics and behaviors and according with the purpose of system development in different phases [2]. This study proposes some performance indicators supporting compliance with the disposal safety objectives in harmony with the integrative meaning of the safety case which the Article emphasizes.

2. Cumulative radiological fluxes

2.1 Cumulative expected dose

While the peak curve with time appearing as a result of disposal safety assessment has a visual forte for compliance with the safety objective, it will have large uncertainty also with, in particular, respect to time. The cumulative expected dose through time span T , $E^T(D)$ may show an overall system performance as a complement. In relation to the safety objective of HLW disposal [1],

$$E^T(D) = \int_0^T E[D(t)] dt \approx \int_0^T [\sum P_k(t) D_k(t)] dt, \quad (1)$$

where the subscript k denotes an exposure scenario or a corresponding exposure pathway through relevant scenarios [3]. The scenarios in (1) may include a proper range of potential exposure situations, as well as normal ones, to the purpose of the assessment. From (1) we can construct $E^T(D)$

with time, in which the values corresponding to a thousand, ten thousands and the overall performance period may have representative meanings in relation to the safety functions of main barriers. Furthermore, we may compare this trend with similar expectation of cumulative dose due to natural radionuclides, including uranium, from the disposal site.

2.2 Weighted releasing fraction of the waste

In connection with the concept of cumulative expected dose mentioned above, a disposal safety function which the waste form constructs by confining radionuclides within itself may be estimated as the following ratio of cumulative release to the original radionuclide inventory disposed of:

$$CRW(T) = \sum r_i^T w_i^T / \sum r_i^T q_i^0, \quad (2)$$

where w_i^T is the activity (Bq) of radionuclide i released from the waste for a period of $0 \sim T$ ($= \int_0^T w_i(t) dt$); q_i^0 is the radionuclide inventory at $t=0$; and r_i^T is the fractional contribution of radionuclide i to the expected dose from influent to exposure pathways for a period of $0 \sim T$, defined as

$$r_i^T \equiv [E_i^T / b_i^T] / \sum [E_j^T / b_j^T], \quad (3)$$

where b_i^T is the influent activity (Bq) of radionuclide i to exposure pathways for a period of $0 \sim T$ in relation with Expression (6) below. In this manner, the characteristics of biosphere may influence the CRW in terms of r_i .

2.3 Weighted net effluent fraction of the EBS

We may represent the capability of engineered barriers for containment of radionuclides released from the waste by the relevant fluxes ratio:

$$CRE(T) = \sum r_i^T e_i^T / \sum r_i^T w_i^T, \quad (4)$$

where e_i^T is the effluent activity of radionuclide i to the geosphere, or the natural barrier, for a period $0 \sim T$.

2.4 Weighted net effluent fraction of the geosphere

Similarly to the CRE defined above, we may represent the capability of natural barriers for isolation of radionuclides released from the engineered barriers by the relevant fluxes ratio:

$$CRG(T) = \sum r_i^T g_i^T / \sum r_i^T e_i^T, \quad (5)$$

where g_i^T is the effluent activity of radionuclide i from the geosphere to the biosphere for a period $0 \sim T$.

2.5 Weighted influent fraction to exposure pathways

The fractional amount of radionuclides influent to the exposure pathways in effluent from the geosphere may be an index on radiological receptivity of the neighboring habitat: i.e.,

$$CRB(T) = \sum r_i^T b_i^T / \sum r_i^T g_i^T \quad (6)$$

where b_i^T is the influent activity of radionuclide i to the relevant exposure pathways for a period of $0 \sim T$.

3. Application of the flux ratio indicators

The above-defined performance indicators of disposal system components are applicable in many ways to estimating the degree of defense-in-depth through multiple barriers or to comparing system candidates overall.

From a viewpoint of radiological protection, the overall performance of barriers in a disposal system may be expressed by combining the individual performance indicators of subsequent barrier components as:

$$CR(T) = CRW \cdot CRE \cdot CRG \cdot CRB \quad (7)$$

For the purpose of defense in depth, we may expect the following set of conditions for a relevant period of time T ($\gg 1$):

$$CRW < 1 \quad (8)$$

$$CRE \ll 1 \quad (9)$$

$$CRG \ll 1 \quad (10)$$

$$CRB < 1 \quad (11)$$

$$CR \ll \ll 1 \quad (12)$$

Furthermore, even if there should be some wide uncertainties in barrier performance, considering a probable range of performance and the uncertainties,

$$CRE \sim CRG. \quad (13)$$

Fig. 1 illustrates the above-defined indicators for an imaginary disposal system. In real systems, the indicator curves may give us some useful intuition, showing various forms and trends. We can apply this information to carry out systems analysis, compare the alternatives, and optimize the system. For these purposes, we need to assess the system rather practically than to simply demonstrate the safety conservatively.

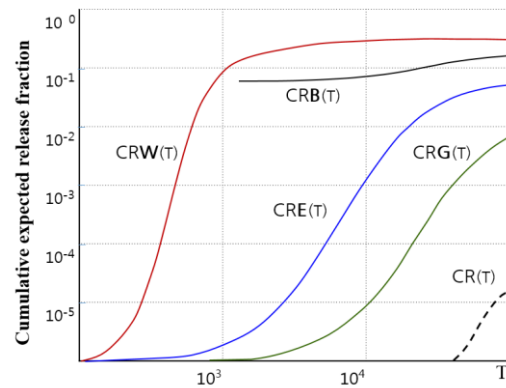


Fig. 1. An example of the cumulative flux ratios.

4. Conclusion

Safety cases of the deep geological repository for HLW, which inevitably involve broad uncertainties over time and space, shall include complementary arguments of defense in depth supporting compliance with the safety objective [1, 3]. The performance indicators proposed here which are based on cumulative radionuclide fluxes weighted in terms of exposure risk can answer this requirement, being used to estimate the defense-in-depth performances of the overall system and its barrier components.

REFERENCES

- [1] NSSC Notice 2016-27, General criteria on deep disposal facility for high-level radioactive waste.
- [2] IAEA TECDOC-1372, Safety Indicators for the Safety Assessment of Radioactive Waste Disposal, IAEA, 2003.
- [3] KINS/RR-1221 (v.2), A Draft Guide to Constructing the Safety Case, KINS, 2015.

Spectroscopic Studies on Pu(III) Hydrolysis Under Reducing and Elevated Temperatures Conditions

Hye-Ryun Cho*, Hee Kyung Kim, Euo Chang Jung, and Wansik Cha

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

*hrcho@kaeri.re.kr

1. Introduction

Most countries operating nuclear power plants have planned development of deep geological facilities for the high-level radioactive waste disposal. Understanding the aquatic chemistry and determining thermodynamic data of actinides are important for the assurance of long-term safety for such permanent isolation of the high-level radioactive materials. The near-field environments of the repositories in an early stage of the operation are expected to be high temperatures due to the decay heat of the high-level radioactive waste. In addition, reducing conditions in groundwater surrounding the planned underground repositories are expected because there are many kinds of reducing materials such as Fe(II) compounds, sulfides, humic substances, and microorganisms living in anaerobic environments.

Plutonium exists significant amounts in the spent nuclear fuel and contributes to the long-term risk due to the long half-life of ^{239}Pu ($t_{1/2} = 2.4 \cdot 10^4$ year). The plutonium chemistry in aqueous solutions (solubility, hydrolysis, complexation, redox-reaction, sorption, etc.) is related to the transport of plutonium in geological environments and thus has been widely investigated [1]. However, reliable thermodynamic data on Pu(III) hydrolysis and solubility of Pu(III) hydroxide are few [1-3] because it is difficult to control the oxidation state of Pu(III) in the laboratory. In the previous report [4], Pu(III) hydrolysis in a neutral pH region was investigated under well controlled reducing conditions for maintaining Pu(III) by electrolysis. In this work, the previous investigation is expanded to the elevated temperature condition in order to determine thermodynamic data, such as ΔH and ΔS .

2. Experimental

2.1 Sample preparation

Chemically purified plutonium ($> 99.9\%$ ^{242}Pu) in

a solution of 0.1 M HClO_4 was prepared. A Pu(III) stock solution in 0.1 M HClO_4 was prepared through electrolysis by applying -0.2 V to the platinum working electrode versus Ag/AgCl reference electrode. The electrolysis setup and process were described in detail elsewhere [4]. Standard samples for absorption measurements were prepared by dilution of the Pu(III) stock solution with an 0.1 M HClO_4 solution. The concentrations of plutonium were determined using liquid scintillation counting (LSC, TriCarb 2500 TR/AB, Packard). For the investigation of Pu(III) hydrolysis, aliquots of the Pu(III) stock solution were diluted using an 0.1 M NaClO_4 solution and then the concentration of hydrogen ions of the solution is controlled to slowly decrease by electrolytic reduction described in the previous report [4].

2.2 Spectrophotometry

The absorption spectra of the plutonium samples are measured using a spectrophotometer (Cary5000, Agilent Technologies) with a Peltier-thormostatted multicell (6×6 cells) holder, which can control the temperature of the samples from 10 to 100 °C. The actual temperature of the sample solutions was monitored by dipping a built-in temperature probe in a blank solution during the measurements.

3. Results and discussions

Fig. 1 shows representative absorption spectra of Pu(III) measured at different temperatures. Absorption feature of trivalent plutonium in 0.1 M HClO_4 is well known, which shows absorption peaks centered at 556, 601 and 665 nm. Molar absorption coefficient at 601 nm is $37.5 \text{ M}^{-1}\text{cm}^{-1}$ at room temperature [4]. As increasing the temperature, decrease of molar absorption coefficient at 601 nm and increase of the full width at a half maximum (FWHM) were observed. The absorbance at 601 nm of the Pu(III) solution measured at 80°C was only 85 %

compared to that measured at 25°C, while peak areas, which are the summation of absorbance from 520 to 640 nm, remained constant.

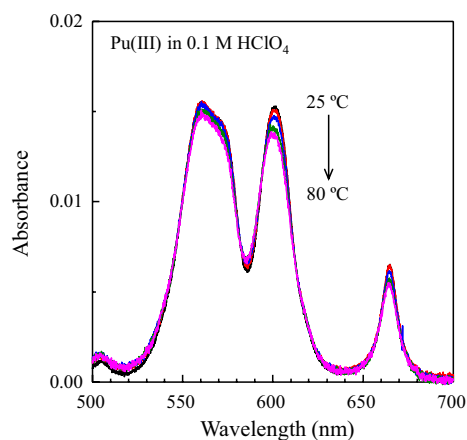


Fig. 1. Absorption spectra of Pu(III) in 0.1 M HClO₄ measured at different temperatures. The concentration of Pu(III) is 0.4 mM determined by LSC.

Fig. 2 shows absorption spectra of Pu(III) samples measured at 60°C. The total concentration of plutonium is constant throughout the coulometric titration ([Pu] = 0.5 mM). Absorbance of Pu(III) decreases with increasing pH. Based on the spectroscopic measurement, the formation of Pu(III)-OH species is determined and the results will be discussed.

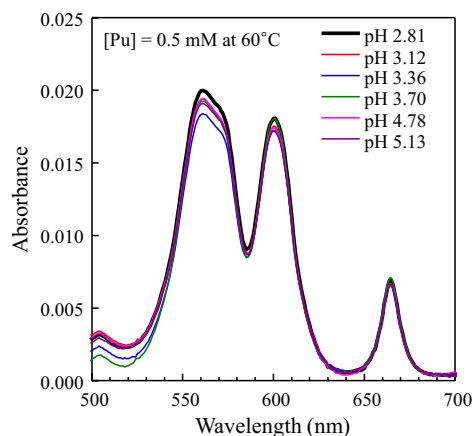


Fig. 2. Absorption spectra of Pu(III) at 60°C were measured at various pHs.

4. Summary

The spectroscopic reference data for plutonium at different temperatures are necessary information for the chemical speciation and evaluation of thermodynamic data at elevated temperature. This work is the initial step

to extend research activities for understanding the plutonium chemistry in aquatic solutions at high temperature. The hydrolysis of Pu(III) and the solubility of Pu(III) hydroxide at the elevated temperature will be discussed.

ACKNOWLEDGMENTS

The work was supported by a grant from the Nuclear Research and Development Program of the National Research Foundation of Korea funded by the Korean Ministry of Science and ICT (Grant code: 2017M2A8A5014719).

REFERENCES

- [1] A. B. Kersting, "Plutonium Transport in the Environment", *Inorganic Chemistry*, 52, 3533-3546 (2013).
- [2] R. J. Lemire, J. Fuger, H. Nitsche, P. Potter, M. H. Rand, J. Rydberg, K. Spahiu, J. C. Sullivan, W. J. Ullman, P. Vitorge, H. Wanner, "Chemical thermodynamics of neptunium and plutonium", pp. 295-450, Elsevier, North-Holland, Amsterdam (2001).
- [3] R. Guillaumont, T. Fanghänel, V. Neck, J. Fuger, D. A. Palmer, I. Grenthe, M. H. Rand, "Chemical thermodynamics of uranium, neptunium, plutonium, americium and technetium", pp. 313-388, Elsevier, North-Holland, Amsterdam (2003).
- [4] H.-R. Cho, Y.-S. Youn, E. C. Jung and W. Cha, "Hydrolysis of trivalent plutonium and solubility of Pu(OH)₃ (am) under electrolytic reducing conditions," *Dalton Transactions*, 45, 19449-19456 (2016).

Investigation of Present-Day Stress States at KURT Site

Yeonguk Jo*, Kyung-Woo Park, Sung-Hoon Ji, and Changsoo Lee

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Republic of Korea

*joy@kaeri.re.kr

1. Introduction

In situ stress states of bedrock are a fundamental parameter, needed to analyze a wide range of potential geomechanical problems, including reservoir characterization, faulting and earthquake mechanism, and geotechnical problems such as underground excavation [1].

The states of stress (orientations and magnitudes) can be estimated through various stress indicators obtained from various sources, such as hydraulic fracturing test, borehole observations, rock core data, and earthquake focal mechanism solutions [2, 3].

In this study, we investigated in situ stress states in KURT site, using hydraulic fracturing tests and borehole wall failure observations.

2. Tested boreholes

To investigate the stress conditions at KURT site, we conducted hydraulic fracturing (HF) tests and borehole image logging, using three wells located in that site (Fig. 1).



Fig. 1. Locations of boreholes tested in this study. Altitudes of individual boreholes KP-2, YS-4, DB-2 are 120, 95, 115 m, respectively.

A series of HF stress measurements were carried out in two boreholes of KP-2 and YS-4. Based on results from HF tests, we determined stress states in shallow depths. The borehole DB-2, a 1 km deep test hole, was image-logged using an acoustic televiewer (BHTV) to identify fractures and faults crossing the borehole, and also probable borehole stress indicators such as breakouts. The stress indicators allowed us to understand stress states in deep depths in the study site.

3. In situ stress estimation

3.1 Stress orientations

Based on the image logs and rock core analysis, we identified several intact rock intervals in the boreholes that were suitable for hydraulic fracturing stress measurements. A total of six depths in KP-2 and six depths in YS-4 (117.5, 160.5, 234, 315, 328, 343.5 m) were tested where the intervals are devoid of pre-existing natural fractures. After the tests, we used impression packers to identify the orientations of HF induced fractures, which were used to determine stress orientations.

Borehole ultrasonic image logs show borehole breakouts (BO) and drilling induced tensile fractures (DITF) along the whole logged section to ~ 738 m in DB-2. The observed BO and DITF, although scattered in their azimuth somewhat (within $\pm 13^\circ$), generally indicating a maximum horizontal principal stress (S_{Hmax}) direction of E-W.

All stress indicators, derived from boreholes, consistently argue the S_{Hmax} direction of E-W compression. The determined average orientation of S_{Hmax} at KURT site is $N98 \pm 13^\circ E$ (Fig. 2) [4].

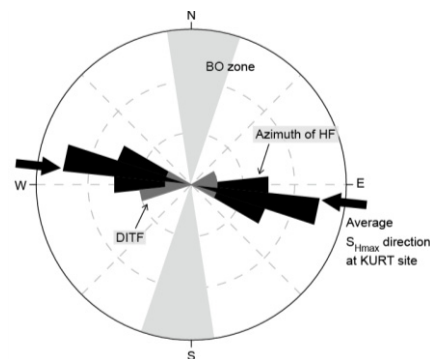


Fig. 2. Result of S_{Hmax} orientation determinations merging stress indicators, taken from [4].

3.2 Magnitudes of stress

The magnitudes of the minimum horizontal principal stress (S_{hmin}) were determined using the

pressure-time curves recorded during HF tests and those of S_{Hmax} using the relationship between logged borehole wall failure and rock strength. The vertical stress (S_v), calculated from overburden weight of rocks, was either the minimum or the intermediate principal stress depending on depths (Fig. 3) [4].

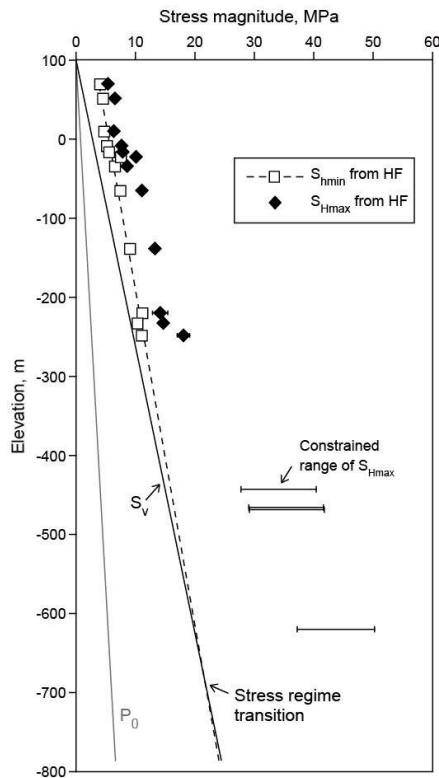


Fig. 3. Investigated in situ stress state profile as a function of elevation, taken from [4].

Results of our stress estimations indicate that stress regime in KURT site is in favor of thrust faulting ($S_v < S_{hmin} < S_{Hmax}$). Another noticeable aspect of the stress condition at this site is that S_v is intermediate between two horizontal stresses ~ 700 m below. It signifies a transition from near surface thrust faulting regime to one favoring strike-slip ($S_{hmin} < S_v < S_{Hmax}$). These results are in agreement with the regional scale crustal stress pattern in Korean Peninsula reported from the previous studies [5].

4. Conclusions

In this paper, we investigated in situ stress conditions at KURT research site, using the results of hydraulic fracturing and borehole observations.

The estimated stress regime in study site is in favor of thrust faulting (< 700 m) or strike-slip

faulting, which agrees with earthquake focal mechanisms in Korea and previous stress results.

We expect that our stress estimation results can be provided as an initial parameter for various studies to exploit the study site, such as construction of underground facility in the future.

REFERENCES

- [1] C.D. Martin and G.W. Lanyon, "Measurement of in-situ stress in weak rocks at Mont Terri Rock Laboratory, Switzerland", *International Journal of Rock Mechanics & Mining Sciences*, 40, 1077-1088 (2003).
- [2] M.D. Zoback et al., "Determination of stress orientation and magnitude in deep wells", *International Journal of Rock Mechanics & Mining Sciences*, 40, 1049-1076 (2003).
- [3] O. Stephansson and A. Zang, "ISRM suggested methods for rock stress estimation-part 5: Establishing a model for the in situ stress at a given site", *Rock Mechanics and Rock Engineering*, 45, 955-969 (2012).
- [4] Y. Jo, et al., "In situ stress estimation in KURT site", *Journal of the geological society of Korea* (in press). (Korean with English abstract)
- [5] J.H. Synn, C. Park, and B.J. Lee, "Regional distribution pattern and geo-historical transition of in-situ stress fields in the Korean Peninsula", *Tunnel and Underground Space*, 23, 457-469 (2013). (Korean with English abstract)

Long-Term Corrosion Behavior of Canister Materials in KURT Environment

Minsoo Lee*, Heui-Joo Choi, and Sung-Hoon Ji

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Republic of Korea

*minm@kaeri.re.kr

1. Introduction

Main purpose of these corrosion tests was to know the corrosion rate of the candidate materials for an engineered barrier system in underground environment. The long term corrosion test cells have been launched since Oct. 2010 at KURT site in KAERI. The corrosion cell was designed simulating the EBS suggested in A-KRS as depicted in Fig. 1 [1].

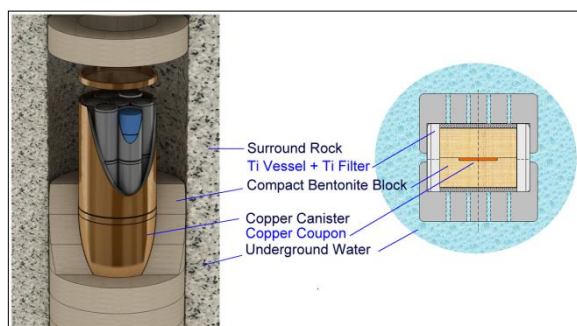


Fig. 1. Schematic diagram of corrosion cell.

In this study, cold spray coating copper, compressed copper, STS 304, cast nodular iron, and Ti(Gr.2) have been kept in KURT environment. Recently lots of 3 and 6 year old test cells were dismantled and analyzed, and the test results were disclosed in this article.

2. Experiment

2.1 Environmental corrosion test

A test specimen (D 15.0 × t 1.0 mm) was surrounded by two compact bentonite blocks (D 30.0 mm × t 10.0 mm) similar to the actual disposal system, and then the blocks were stored in a small titanium vessel. The titanium vessel was composed of a hollow body, two titanium filters (D30.0 mm × t 1.5 mm) of 10 μm pore size and two caps blocking the both ends of the body by screwing them.

In the KURT field test, the lots of corrosion cells were kept in several environmental chambers sustained at 70

°C in KURT. Fresh underground water was flowing through the chamber continuously at 10~20 cc/min.

After the test, corrosion rate was determined by weight loss measurement. And the corrosion product was analyzed by XRD and SEM-EDX. The used bentonite was also analyzed by XRD and ICP-OES.

3. Result

3.1 Dismantling of corrosion cells

The corrosion cells of 3.2 and 6.5 year old were dismantled as in Fig. 2. The corrosion products in compressed copper, coating copper, and cast nodular iron were combined with bentonite strongly. But STS304 and Ti showed clear surface without any kind of corrosion product.



Fig. 2. The 3.2 year old test specimens with compact bentonite right after dismantling.

3.2 Analysis

There were white precipitations on copper specimens and cast iron, which were analyzed by XRD. The white precipitation on copper specimen was presumed to be anorthite (Ca type feldspar), which can be formed by the reaction of dissolved feldspar with Ca^{2+} [2]. And the white on cast iron was presumed to be Aragonite, which was known to be formed by the reaction of carbonate (CaCO_3) with Ca^{2+} .

But the Albite was also detected by XRD in 6.5 year old Copper specimen.

3.3 Corrosion depth

The corrosion product on copper specimen was removed with weak HCl solution, but the removal process was not smooth since some black CuO spots on the surface were hard to remove with weak HCl, which caused severe dissolution of pure copper and produced a thicker corrosion depth. Ti and STS304 did not showed any increase in corrosion depth though they experienced 6.5 year corrosion test. The cast iron showed continuous increase of corrosion depth with test time

4. Conclusion

There were considerable increases of corrosion depth in cast iron and copper specimens. And the compact bentonite exhibited some change in its mineral composition after 3.2 and 6.5 years.

Acknowledgement

This work was supported by the Ministry of Science, ICT and Future Planning within the framework of the national long-term nuclear R&D program (NRF-2017M2A8A5014836).

REFERENCES

- [1] KAERI Research Report, "High-Level Waste Long-term management Technology Development", KAERI/RR-3100 (2009).
- [2] C.W. Parmelee, A.J. Monack, Journal of the American Ceramic Society Volume 13, Issue 6, Version of Record online: 2 JUN 2006.

3분과

고준위폐기물처분 (Poster)



Recent Techniques for Measuring the SWRC of Unsaturated Compacted Bentonite Buffer

Jaw Owan Lee* and Geon Young Kim

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

*jolee@kaeri.re.kr

1. Introduction

This paper introduces the principal concept of soil-water retention curve (SWRC) and reviews the SWRC measurement technique reported in the literature. In addition, a measurement technique suitable for the SWRC measurement of unsaturated compacted bentonite buffer is suggested.

2. Soil-Water Retention Curve

The soil-water retention curve represents the relationship between suction versus volumetric water content in unsaturated media. This is an important input data for analyzing the water movement and mechanical behaviour of unsaturated media. The suction (S) is determined by measuring the water potential (ψ), and it is the negative value of water potential. That is, $S = -\psi$.

The water potential is conceptually the water movement capacity in the unsaturated medium. The total water potential (ψ_t) is expressed as the sum of the matric potential (ψ_m), osmotic potential (ψ_o), gravitational potential (ψ_g), and mechanical pressure potential (ψ_p) as follows:

$$\psi_t = \psi_g + \psi_m + \psi_o + \psi_p \quad (1)$$

Once the SWRC is determined by measuring the water potential, the next step is to determine the retention model that best fits the experimentally obtained SWRC. Many retention models are presented in literature. Of these models, van Genuchten equation has been widely used for analyzing the hydro-mechanical behaviour of the unsaturated bentonite buffer.

3. Axis translation technique(ATT)

The ATT was first proposed by Hilf (1956). This technique, which has been widely used in the field of civil engineering, assumes liquid transfer (i.e., the

liquid phase in a porous medium undergoes an increase in pressure equal to that of the gaseous phase) and does direct measurement of matric water potential. The main reason for the use of this technique is to avoid cavitation when measuring matric water potential greater than 10 kPa. In the ATT, the pore water pressure in the porous medium is increased to a positive reference pressure (u_w). Matric water potential in the medium is applied by increasing the pore air pressure (u_a). A high-air entry ceramic disk is normally used to provide separation between water and air pressure. The ceramic disk is not an impermeable membrane to dissolve salts in the medium. Therefore, water flow due to osmotic effects is bypassed since the water flow is an advective process in this case. The matric water potential (ψ_m) is determined by the following equation:

$$\psi_m = u_a - u_w$$

4. Vapor equilibrium technique(VET)

This technique overcomes the limitation of the ATT for applying high suction. The VET, which indirectly determined the total water potential from the measured relative humidity (RH), assumes vapor transfer and equilibrium with pore water in the porous medium. The thermodynamic relationship between total water potential and relative humidity is based on Kelvin's law:

$$\psi_t = 10^{-6} (RT/V_w) \ln(RH/100) \quad (2)$$

where ψ_t is total water potential[MPa], R is universal gas constant[8.3143 J/K mol], T is absolute temperature, V_w is molar volume of water[1.8×10^{-5} m³/mol]. The equilibrium requirement is that the potential of water in its liquid and vapor phases must be the same. Therefore, the water exchange between the pore water and the vapor includes the effect of capillary action, osmotic force, and sorptive forces. This technique is applied especially in the case of high water potential values. Two types of tests are mainly used in this VET technique: cell method and block-sensor method.

Cell method The water potential has been determined in cells kept inside desiccators in which the water potential can be controlled. Imposing a relative humidity is based on the fact that this conditions the pressure of the water and gas in the pores. This humidity may be imposed by means of any salt solutions that know its water activity. The measurement sample exchanges water with the atmosphere until thermodynamic equilibrium is reached with vapor pressure of the solution, as a result of which total water potential is changed. The water potential in the pores of the sample is determined by means of the Kelvin's law of eqn. (2). However, this technique has a limitation to take a long time to reach the thermodynamic equilibrium when applying it to highly compacted bentonite buffer.

Block-sensor method The water potential can be determined at constant volume and as a function of the salinity of the interstitial water as well as temperature. The most important feature of this method is that the time required for measurement can be greatly shortened comparing with that of the cell method. This method consists of the measurement, through a capacitive sensor, of the relative humidity of blocks of bentonite highly compacted to different dry densities and water contents. Fig. 1 shows an experimental apparatus for measuring the total suction of compacted bentonite with hygroscopic water content [1]. The compacted bentonite is prepared by adjusting the water content of the powdered bentonite and then compacting it to a desired dry density. The adjustment of the water content is done by adding de-mineralized water to the dried bentonite by pre-weighed amounts using an ultrasonic humidifier. After the water content adjustment, the bentonite is compacted in a cylindrical cell (0.05 m in inner diameter and 0.05 m in length) to a desired dry density using a hydraulic press; a hole is drilled in it to allow a capacitive hygrometer sensor to be installed inside; then it is left in a desiccator for more than 3 days for its water content equilibrium. The compacted bentonite and connecting devices are assembled, which is followed by measuring the relative humidity of the compacted bentonite. The total suction of the unsaturated compacted bentonite is determined from the measured relative humidity using Kelvin's law of eqn. (2).

5. Summary and Conclusions

The SWRC represents the relationship between suction and volumetric water content. The suction

can be determined by measuring the water potential. The total water potential of the unsaturated media includes matric potential, osmotic potential, gravitational potential and mechanical pressure potential. The soils of the civil engineering field usually determine the suction by measuring the matric potential, while the suction of the bentonite buffer should be determined by measuring the total water potential because the contribution of the osmotic potential to it is also important. To determine the suction of the unsaturated compacted bentonite buffer, a block-sensor method based on VET was appropriate and effective.

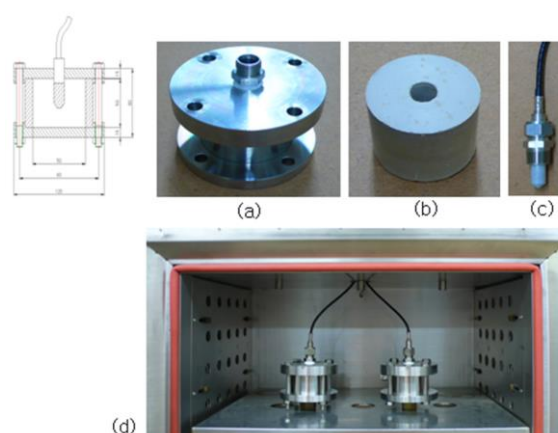


Fig. 1. Experimental apparatus for the suction measurement: (a) cylindrical cell and its housing (b) compacted bentonite (c) relative humidity sensor (d) dry oven with installed cylindrical cells [1].

Acknowledgement

This work was supported by Nuclear Research & Development Program of the National Research Foundation of Korea (NRF) Grant funded by the Ministry of Science, ICT and Future Planning (MSIP).

REFERENCES

- [1] Jae Owan Lee, Won Jin Cho, Sangki Kwon, "Suction and water uptake in unsaturated compacted bentonite", *Annals of Nuclear Energy* 38, 520-526 (2011).

Preliminary Study on Paleostress Reconstruction Around KURT Area

Soolim Jung* and Sung-Hoon Ji

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

*sljung@kaeri.re.kr

1. Introduction

Disposal facilities for radioactive waste shall be sited to provide isolation from the accessible biosphere. The features shall aim to provide this isolation for several hundreds of thousands of years after closure. For the safety assessments of a repository, the long-term natural evolution and possible events of the site, that can cause disturbances to the facility over the period of interest, should be considered. Understanding the history of paleo-stress evolution is one of the key factors to verify long-term safety of the given area.

The objective of this research is to introduce several techniques to reconstruct the paleo-stress fields and compare them, to develop the effective methodology of constructing a geological evolution model.

2. Literature investigation

2.1 Historical models of deformation in Finland

The Finnish organization for final disposal of nuclear waste, POSIVA, has conducted geological research around the site-specific Underground Research Laboratory (URL) located in Olkiluoto. To assess the long-term safety of Olkiluoto area, POSIVA researchers built regional-scale lithological models and regional-scale conceptual models of each ductile and brittle evolution history [1]. The brittle

deformation model shows changes in paleo-stress field and ranges of intrusion age for each recorded magmatism from 1.7 billion years ago.

2.2 Long-term evolution models in Japan

Japan Atomic Energy Agency (JAEA) has studied the reconstruction and estimation of long-term geological evolution around URL site [2]. JAEA focused on the period of interest, ten to a hundred thousand years after closure, and constructed evolution models with major tectonic movements during the period.

3. Methods

3.1 Methodology to paleostress reconstruction

To interpret the geodynamic evolution history, it is necessary to have a complete paleostress analysis of fracture systems. In a typical case study for understanding the paleostress, the orientations are measured by stress tensor analysis, and the ages of the events are estimated by relative age analysis [3].

Microcracks can be another indicator to find out the orientation and age of paleostress [4]. A set of oriented microcracks indicates the orientation of principal stress and fluid inclusion formed within the microcracks can classify the formation age of the microcracks.

4. Paleostress reconstruction of KURT area

The research was performed around KURT (Korea Atomic Energy Research Institute Underground Research Tunnel), Daejeon, Korea. To understand the geological information around the research area, various geological surveys have been conducted, including a surface geophysical survey, geological lineaments analysis, field observations, borehole investigations, chemical analyses, age dating of the rock samples, and a structural analysis of the exposed wall and flat-cut surface inside KURT.

Compared to preceding research, it is necessary to get more data from field geometry for the paleostress reconstruction around KURT site. The deep borehole investigations and KURT observations will contribute to reconstruct more precise paleostress evolution model around the KURT site.

granites from the Sogrisan and the Wolagsan area”, Journal of Geological Society of Korea, 32(3), 291-301 (1996).

REFERENCES

- [1] POSIVA Oy, “Geology of Olkiluoto”, POSIVA 2016-16 (2016).
- [2] H. Kosaka, H. Saegusa, K. Yasue, Y. Kusano, and H. Onoe, “Development of a Method-ology for the Characterisztion of the Long-term Geosphere Evolution (2) Estimation of the long-term evolution of goundwater flow conditions in a Tono area case study”, Proceeding of ICON19, Japan, ICON19-43567 (2011).
- [3] J. Hwang, and J. Angelier, “Paleostress Reconstructions and the Tectonic Evolution of the Kyungsang Basin : a case example near Goryeong-eub”, Journal of Geological Society of Korea, 28(5), 471-482 (1992).
- [4] B. Jang, and J. Gim, “Cretaceous paleostress from healed microcracks and fluid inclusions in

Biogeochemical Alteration of KJ-II Bentonite Under Alkaline-Reducing Conditions

Hyo-Jin Seo*, Seung Yeop Lee, Jae-Kwang Lee, and Wan Hyoung Cho

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Republic of Korea

*hjseo@kaeri.re.kr

1. Introduction

Bentonite has been used as buffer or backfill materials for deep geological repositories of high-level radioactive wastes. The bentonite can react with the cement used as construction materials in the repositories [1]. They may cause alkaline conditions. When bentonite and cement interacts, OH^- and Ca^{2+} can move into bentonite and pH rises to 10~13 in bentonite pore water. In this process, a dissolution of bentonite can occur, and secondary minerals may precipitate [2]. Therefore the study on the alteration of bentonite under alkaline condition is needed. Furthermore, the alteration of bentonite may be related with the behavior of radionuclides.

In the study, we investigated the alteration of KJ-II (Gyeongju) bentonite based on microbial activity under alkaline-reducing conditions. This work was focused on the effect of indigenous microbe in the bentonite, NaCl, and corrosion products of disposal materials on the alteration of KJ-II bentonite.

2. Experimental

2.1 Material

The bentonite used in this study was taken from Gyeongju. We named it as “KJ-II bentonite”. It is a Ca-type bentonite that has montmorillonite content of about 65% with the ratio of $\text{CaO} : \text{Na}_2\text{O}$ as 5.5:1 [3]. To examine the effect of corrosion products for the bentonite alteration, Cu_2S , FeS , and Fe_2O_3 powders were mixed with KJ-II bentonite (10% w/w).

2.2 Methods

2.2.1 Bentonite. KJ-II bentonite was mixed with corrosion products, and then distilled water that was sterilized was added to make a solid-liquid ratio (S/L) of 1g/30ml in 50ml centrifuge tubes. To remove some oxygen remaining in the bentonite, it was previously shaken for 3 days (120 rpm, 30°C). Finally, the supernatant was removed after a centrifugation (10,000 rpm, 5 minutes), and the centrifuge tubes containing the materials were placed in a glove box filled with N_2 gas.

2.2.2 Solutions. Solutions were prepared by adding some reagents to water using a clean bench. Before addition of the reagents, they were sterilized by 0.20 μm filtration (ADVANTEC). To investigate the effect of indigenous microbe in KJ-II bentonite, distilled water, distilled water+10mM Na-lactate, and distilled water+10mM Na-lactate+2mM sulfate were prepared respectively. To investigate the effect of salt water, 0.5M NaCl and 2.0M NaCl solutions were used. All solutions were purged with N_2 gas for 40 minutes to remove oxygen in solution and then immediately placed in a glove box.

2.2.3 Experiments. All experiments were carried out in the glove box filled with N_2 gas (anaerobic condition). The prepared solutions were poured into centrifuge tubes containing bentonite mixed with corrosion products (FeS , Fe_2O_3 and Cu_2S). NaOH solution was used to adjust the pH of the media as 9, 10, and 11, respectively. The experimental conditions are summarized as follows (Table 1).

Table 1. Experimental conditions in our study

Case	Type of bentonite	Solutions	pHs
1	KJ-II bentonite	distilled water	9, 10, 11
2	KJ-II bentonite	distilled water+10mM Na-lactate	9, 10, 11
3	KJ-II bentonite	distilled water +10mM Na-lactate +2mM sulfate	9, 10, 11
4	KJ-II bentonite	0.5 M NaCl	9, 10, 11
5	KJ-II bentonite	2.0 M NaCl	9, 10, 11
6	KJ-II bentonite +FeS(10% w/w)	distilled water	9, 10, 11
7	KJ-II bentonite +Fe ₂ O ₃ (10% w/w)	distilled water	9, 10, 11
8	KJ-II bentonite +Cu ₂ S(10% w/w)	distilled water	9, 10, 11

3. Results and discussion

The bentonites that were incubated in anaerobic condition showed visual changes (Fig. 1). In particular, case 3 showed that the color of bentonite was changed for 2 months regardless of its pH conditions. It is assumed that this phenomenon is caused by the indigenous bacteria that are active in the bentonite, if some electron donors (e.g., lactate) are supplied. The sulfate that was initially injected into the tube may be reduced to sulfide that is black as combined with metals (e.g., Fe).

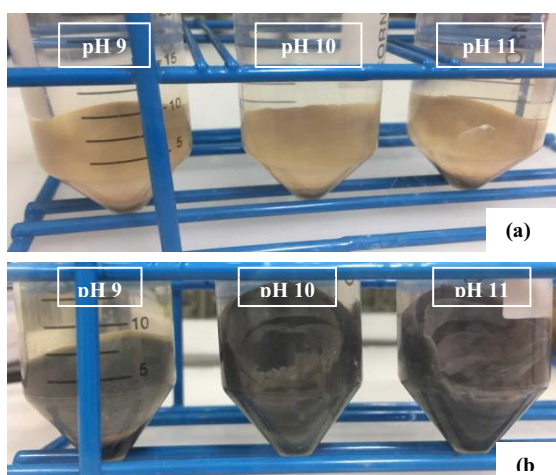


Fig. 1. Some KJ-II bentonites with different pHs; (a) after 1 week, (b) after 2 months.

4. Conclusion

The effect of microbial activity under alkaline-reducing conditions was investigated experimentally. In the study, the color of bentonite was changed by the indigenous bacteria that are active in the

bentonite under the alkaline condition. When KJ-II bentonite is used as buffer material, bentonite alteration may be caused by the microbial activity in deep geological repositories of high-level radioactive wastes.

REFERENCES

- [1] EC. Gaucher and P. Blanc, "Cement/Clay interactions - A review: Experiments, natural analogues, and modeling", *Waste Management*, 26, 776-788, (2006).
- [2] D. Savage, et al., "Alteration of bentonite by hyperalkaline fluid: A review of the role of secondary minerals", *Physics and Chemistry of the Earth*, 32, 287-297, (2007).
- [3] Y. Malgobalgaebitnala, et al., "Measurement of Properties of Domestic Bentonite for a Buffer of an HLW Repository", *JNFCWT*, 14(2), 135-147 (2016).

Statistical Approach of the Geochemical Mobility and Behavior of Natural Uranium in KURT Groundwater

JiHye Jeong*, Jang-Soon Kwon, Yong-Kwon Koh, and Jae-Kwang Lee

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Republic of Korea

*yz0201@kaeri.re.kr

1. Introduction

For the high level radioactive waste disposal facilities it is important to evaluate the mobility of radionuclides in groundwater and the geochemical factors that control its behavior in case of unplanned leakage of radionuclides to groundwater. So far, a number of studies have been conducted to investigate hydrochemical and biogeochemical properties of the groundwater in the KAERI Underground Research Tunnel (KURT) [1]–[4], but statistical approach of mobility and transport of natural nuclides in the groundwater have not been carried out. The purpose of this study is to statistically identify factors that affect uranium mobility in the KURT groundwater and to understand how the transport of natural uranium changes under the geochemical controls.

2. Results and discussions

2.1 Geochemical mobility of uranium using statistical methods

The main factors affecting uranium mobility in groundwater are pH, DO (ORP) and partial pressure of carbon dioxide (PCO_2). Uranium mobility shows the increases in the groundwater with higher oxicalkaline condition and PCO_2 . The uranium concentration in the KURT groundwater is the highest in the i3 section (mean value: 720.34 $\mu\text{g/L}$) and lowest in the i5 section (mean value: 0.85 $\mu\text{g/L}$). The uranium concentration in each section showed a statistical difference through the analyses of Variance (ANOVA) test was performed ($p < 0.05$). Then the influences of the aforementioned factors on the uranium concentration in each section (depth) were statistically expressed (Fig. 1). The result confirmed that uranium mobility is higher under the pH 8.0 - 9.5, $DO > 0.5$ and high PCO_2 conditions. Furthermore, the Eh-pH diagram for uranium species in the

groundwater showed that uranium existed as $UO_2(CO_3)_3^{4-}$, which has the highest mobility.

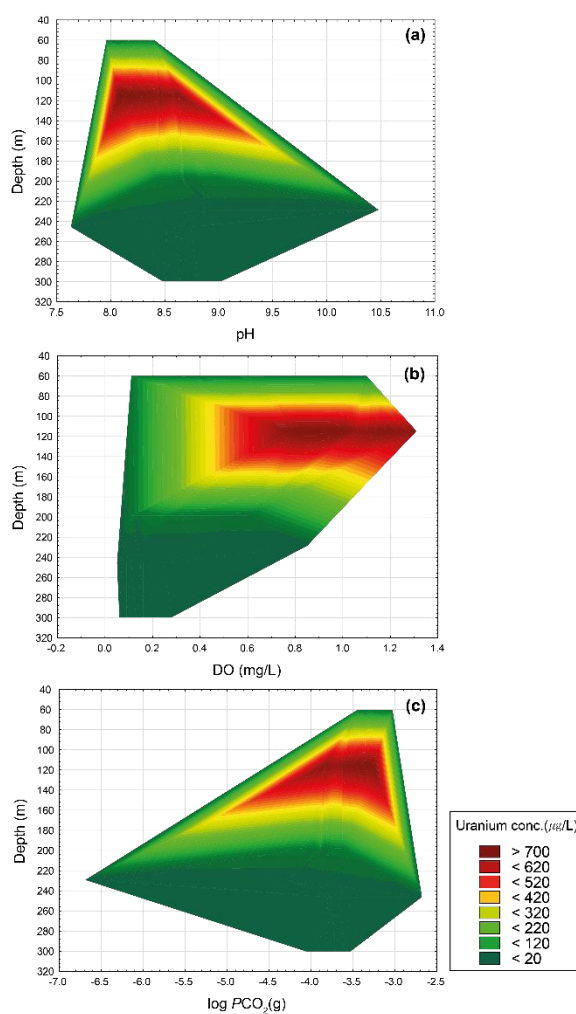


Fig. 1. Wafer plots of U concentration within pH (a), DO (b) and PCO_2 (c) in KURT groundwaters using Statistica program.

2.2 Geochemical behavior of Uranium in groundwater

According to a report published by KAERI in 2012, there is information on the mineral adsorption of uranium in KURT groundwater [5]. However, there is a limit to the accuracy of the results because

the analysis is performed on the cores of other wells, not the cores of the corresponding groundwater. Therefore, this study analyzed the saturation index (SI) of the mineral by depth in the DB-1 groundwater through PHREEQC program. Assuming there is a possibility that the mobility was lowered by the adsorption of uranium, this is probably due to the influence of chlorite in KURT granite minerals. The SI value of chlorite is supersaturated only in the i5 section where uranium concentration is lowest (Fig. 2).

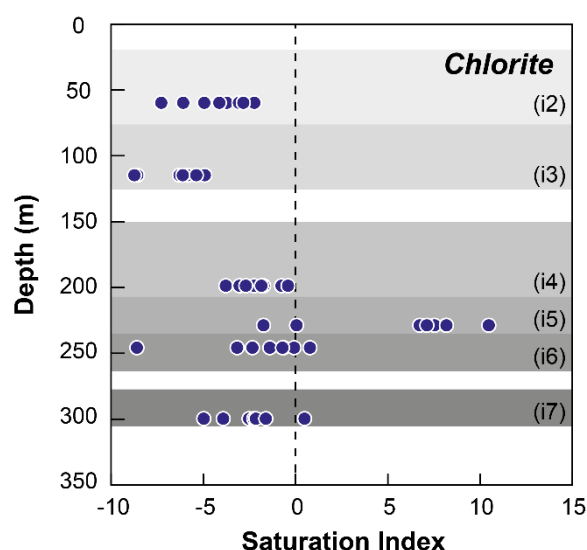


Fig. 2. Saturation index for Chlorite in KURT groundwater using PHREEQC program.

3. Conclusion

Natural Uranium in the KURT groundwater within granite bedrock had different mobility in each section (depth). Uranium mobility in groundwater is controlled by pH, DO, and PCO_2 its mobility in the KURT groundwater is the highest when the groundwater is under oxic-alkaline and high PCO_2 conditions. In addition, it is reported that uranium is well adsorbed to iron and manganese oxides and on chlorite; Assuming that the decrease in uranium concentration affects the adsorption of minerals, uranium in KURT groundwater might be predicted to adsorbed on chlorite.

REFERENCES

- [1] J.-H. Ryu, J.-S. Kwon, G.-Y. Kim, and Y.-K. Koh, "Geochemical Characterization of Rock-

Water Interaction in Groundwater at the KURT Site," *J. Nucl. Fuel Cycle Waste Technol.*, vol. 10, no. 3, pp. 189–197, 2012.

- [2] N.-Y. Ko, K. W. Park, K. S. Kim, and J. W. Choi, "Groundwater Flow Modeling in the KURT site for a Case Study about a Hypothetical Geological Disposal Facility of Radioactive Wastes," *J. Nucl. Fuel Cycle Waste Technol.*, vol. 10, no. 3, pp. 143–149, 2012.
- [3] J.-H. Lee, H. Jung, E. Lee, and S. Kim, "Estimation of groundwater level fluctuation of the crystalline site using time series analyses in south Korea," *J. Nucl. Fuel Cycle Waste Technol.*, vol. 11, no. 3, pp. 179–192, 2013.
- [4] S. Y. Lee, Y. Roh, and J. T. Jeong, "Changes of the Oxidation/Reduction Potential of Groundwater by the Biogeochemical Activity of Indigenous Bacteria," *Econ. Environ. Geol.*, vol. 47, no. 1, pp. 61–69, 2014.
- [5] M. Baik, C. Park, and S. Kim, "Investigation on nuclide migration behaviors," Korea Atomic Energy Research Institute, 2012.

A Preliminary Deep Borehole Disposal Concept for Cs-Wastes From Pyro-Processing

Jong-Youl Lee*, Heui-Joo Choi, In-Young Kim, and Dong-Keun Cho
Korea Atomic Energy Research Institute, 1045 Daeduk-daero, Yuseong, Daejeon
*njylee@kaeri.re.kr

1. Introduction

The purpose of disposal for radioactive wastes is not only to isolate them from humans, but also to inhibit leakage of any radioactive materials into the accessible environment.

Among the eight kinds of radioactive wastes generated from the pyro-processing that KAERI is developing to reuse the fissile nuclides and reduce the footprint of final repository, there are two highly-heat generating wastes, and they are called Cs-spent filter waste and Sr-spent filter waste. Generally, these two wastes are considered to be stored for some time to get rid of the decay heat at a storage facility of the above ground or the underground. And then they are to be disposed of as low and intermediate level radioactive waste.

In this study, the characteristics of the Cs-waste from pyro-processing such as waste form, decay heat and weight etc., were reviewed. Based on the review results, the concepts of Cs-waste disposal container and disposal at the depth of more than 2 - 3 km were designed preliminarily. And with this concept, thermal analyses were carried out and disposal area for the total Cs-waste amount based on the recent national energy basic plan was estimated.

2. Characteristics of Cs-waste

2.1 Decay heat of Cs-waste

Cs-spent fuel filters will be generated from the preprocessing of spent nuclear fuel. The decay heat from one Cs-spent filter waste form after discharge from a reactor was calculated and listed in Table 1.

Table 1. Decay heat of Cs waste with time

time		decay heat	
after discharge [yr]	after disposal [yr]	basis [W/10tonSF]	1 Cs-Waste [W/1 block]
40	0	3,293	548
60	20	2,068	344
80	40	1,299	216
100	60	816	136
110	70	647	107
120	80	513	85
140	100	323	53
160	120	204	34
180	140	129	21
200	160	82	13
250	210	28	4
300	260	11	1

Table 1 shows that most of decay heat disappears about 200 years after discharge.

2.2 Amount of Cs-waste

According to the current version of material balance, the amount of the waste from 30 tHM of PWR spent fuels is given in Table 2. With the consideration of 23,618 tHM of PWR spent fuels from 24 reactors, around 14,171 waste forms are expected.

Table 2. Amount of Cs waste from pyro-processing of 30 tHM of PWR SNF

Total weight of waste, kg	2,750
Size of waste form, mm	300 (D) ×900 (H)
Weight of waste form, kg	160
Number of waste forms	18

3. DBD Concept for Cs-waste

3.1 Disposal container concept

A double layered disposal container concept was proposed for the deep borehole disposal of Cs-waste. Inner part material of the disposal container was SiC to endure the pressure of disposal depth with high density brine water. And outer part of that was stainless steel to protect the inner SiC vessel and to make it easy to handle the containers strings in the deep boreholes. A container can hold 4 waste storage cans and 10 containers make one string to be emplaced in the borehole. This disposal container concept is shown in Fig. 1.

3.2 Disposal concept

A recent deep borehole disposal concept consists of drilling a borehole (or array of boreholes) into crystalline basement rock to a depth of about 5 km, emplacing disposal containers containing radioactive wastes in the lower 2~3 km of the borehole, and sealing the upper part of the borehole. The waste disposal containers could be emplaced individually or as a string of 10~20 containers. In the waste emplacement procedure of deep borehole disposal system, bridge plugs will be constructed to support the weight of container strings. A single borehole could emplace up to 400~600 containers (1,600~2,400 waste blocks), each approximately 5 m in length. The sealing material for the borehole can be compacted bentonite, asphalt and concrete (Fig. 2).

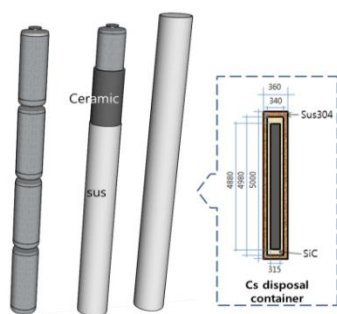


Fig. 1. A Concept of DBD container.

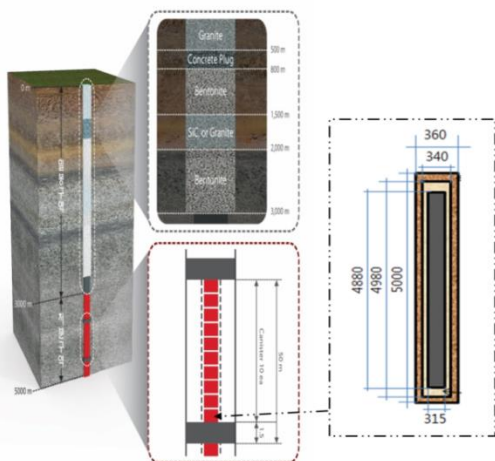


Fig. 2. A Concept of DBD for Cs waste.

4. Some Analyses for Cs-waste DBD System

4.1 Thermal Analyses

With the preliminary Cs-waste deep borehole disposal concept, some thermal analyses carried out and Fig. 3 shows the results. As shown in Fig. 3, the maximum temperature of canister surface and hole wall will be 270°C, 250°C respectively for the Cs-waste at the 40 years after discharge from reactors. For the Cs-waste at the 80 years after discharge, those will be decreased to 206°C, 196°C respectively.

For the integrity of the borehole or the bridge plugs in the disposal area, thermal condition at the disposal depth should be considered and a requirement related to this condition need to be established.

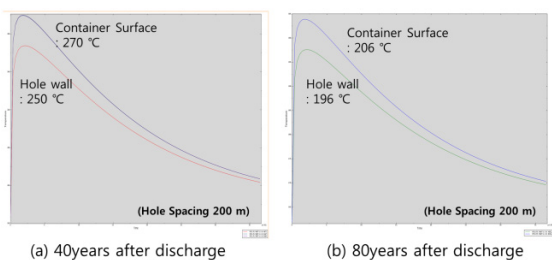


Fig. 3. Conceptual drawing of buffer.

4.2 Disposal Area for Cs-waste

With the total amount of Cs-waste from pyro-processing, DBD area for Cs-waste was estimated. Two cases of disposal depth, depth from 3 km to 5 km and that from 2 km to 5 km, were considered and the disposal area were 0.36 km², 0.24 km² respectively, as shown in Table 3.

Table 3. Disposal area for Cs-waste

No. of Deep Borehole. (No. of total waste form : 14,170 ea)	Disposal area	
	Array	Area km ²
Disposal depth 3 ~ 5 km	9	3 × 3 = 0.36
Disposal depth 2 ~ 5 km	6	3 × 2 = 0.24

* Borehole spacing : 200 m

5. Conclusion

In this study, a preliminary deep borehole disposal concept for Cs spent filter waste which was one of highly decay heat generating waste from pyro-processing of PWR spent nuclear fuels was proposed. To do this, characteristics of the waste were reviewed and a concept of disposal container was drawn. Then, based on the DBD concept, some analyses such as thermal analyses and disposal area was carried out.

These kind of analyses need to be carried out in detail with the real site data later. And the result of this study can be used as an input data for the establishment of the disposal policy.

Acknowledgement

This work was supported by the Ministry of Science and ICT within the framework of the national long-term nuclear R&D program (NRF-2017M2A8A-5014856).

REFERENCES

- [1] NIREX, A Review of the Deep Borehole Disposal Concept for Radioactive Waste, Nirex report no. N/108, Oxfordshire, UK, 2004.
- [2] Patrick V. Brady, Bill W. Arnold, Geoff A. Freeze, Peter N. Swift, Stephen J. Bauer, Joseph L. Kanney, Robert P. Rechard, Joshua S. Stein., Deep Borehole Disposal of High-Level Radioactive Waste, SAND2009-4401, SNL, Albuquerque, NM, 2009.
- [3] B. W. Arnold, P. V. Brady, S. J. Bauer, Courtney Herrick, Stephen Pye, and John Finger, Reference Design and Operations for Deep Borehole Disposal of High-Level Radioactive Waste, SAND2011-6749, Sandia National Laboratories, Albuquerque, NM, 2011.

Investigation on Regulator's Role for Confidence Building of Stakeholders in Radioactive Waste Disposal

Kyongwon Han^{1*}, Ji Hyun Kim¹, Sang Hoon Park¹, Jinyong Park², and Yohan Kim²

¹RadWorks Corp., 530 Daeduckdae-ro, Yuseong-gu Daejeon, Korea

²KINS, 62 Gwahak-ro, Yuseong-gu, Daejeon, Korea

*kwhan0101@gmail.com

1. Introduction

The disposal of radioactive waste is an important issue in Korea, where experience has been gained from the implementation of low and intermediate level radioactive waste disposal, while preparation is under way for high level radioactive waste disposal. Implementation of the disposal has to take account of not only the assurance of the safety in technical aspects but also the achievement of confidence from stakeholders [1, 2]. For the latter, the regulator in Korea has focused on regulation on the safety, while acted a limited role in the process of stakeholder involvement. However, there are views in the Korean nuclear community that an active regulator's role in the process of stakeholder involvement is necessary for the achievement of confidence from stakeholders, not to mention the operator's role of regulation [1, 2].

This paper presents the investigated information of regulator's role for the stakeholder involvement from international organizations and some countries advancing their disposal programs, which is followed by an identification of implications for the consideration of regulator's role in Korea.

2. Method and Results

2.1 Investigation on oversea status

An investigation has been conducted regarding regulator's role in the stakeholder involvement for radioactive waste disposal, in terms of recommendations from IAEA and NEA, and regulatory framework and involvement practices in Finland, Sweden and USA.

2.1.1 IAEA and NEA. Stakeholder is defined as those who have a specific interest in a given issue or decision, and can be classified into two types, i.e.

internal and external [1]. Participation by stakeholders in decision making processes are recommended to allow a stakeholder to be informed and to be involved in decisions that affect his or her well-being, and to have stakeholders' reasonable issues and concerns are factored into decisions. Stakeholder involvement makes regulatory organizations and other authorities acutely aware that their actions are under public scrutiny. Regulator's role has been changed from a limited involvement until the beginning of licensing process to an active involvement already in pre-licensing stage. Accordingly an early involvement of regulators in the decision making process from the stage of establishing a safety case concept and a siting plan, and in early consultations with local communities as "safety communicators" and "peoples' experts" are recommended [1, 2]. NEA Forum on Stakeholders Confidence (FSC) found that there was a leap in a decade (2003-2010) from tokenism to real participation in the approach of public involvement [3].

2.1.2 Finland. The Finnish Nuclear Energy Act requires regulator to establish regulations and involve in the radioactive waste management from the pre-licensing stage. Actually, the Finnish regulator, STUK, reviewed drafts of safety case and R&D programs of the repository operator, Posiva, through close bilateral communications. In parallel, STUK carried out a program for co-operation and direct communication with the public media, local public and their representatives through oral and written materials, seminars, discussion meetings, and etc. STUK made clear to the public its role as an entity independent of Posiva, with the duty to look after the health and safety of citizens. Thus, STUK made a success story of being "peoples' experts" [4].

2.1.3 Sweden. The Swedish decision-making process

must be open, transparent, fair and participatory. The programs have also become more communicative by requirements of Environmental Impact Assessment at project level and Strategic Environmental Assessment at the planning and program implementation levels. The Swedish regulatory authority (SSM) communicates with the public including the involved municipalities through dialogue, participation in public hearing, etc. The former regulator (Swedish Nuclear Power Inspectorate (SKI) and the Swedish Radiation Protection Authority (SSI)) applied RISCUM model to the design and evaluation of a public hearing on the repository operator's (SKB) choice of municipalities for the program to build a spent nuclear fuel repository. The RISCUM model for transparency includes three basic elements: technical/scientific issues, normative issues and authenticity [5].

2.1.4 USA. Regulation in USA requires the regulator, NRC, to involve in the pre-licensing process and communicate with any affected Indian Tribe for the disposal of high level radioactive waste. In practice, NRC communicated closely with the repository operator, DOE, regarding site characterization and preparation of Environmental Impact Statement report on Yucca Mountain repository through reviews and dialogues on key technical issues. NRC also developed licensing criteria, 10CFR63, by having consensus with the public comments from public meetings. Meanwhile, NRC established strategy for the improvement of its system of communication with stakeholders and the use of communication tools [6].

2.2 Implications

Implications from the investigation can be drawn as follows:

- Internationally, gaining public confidence through stakeholder involvement in radioactive waste disposal is becoming more important than in the past when decisions were made by public information and consultation approach. At the same time, an active role of regulators in the stakeholder involvement is necessitated to build confidence of stakeholders.
- Regulator's role in the stakeholder involvement can be summarized as, a) establishment of required regulations and enforcement, b) stepwise interaction with the repository operator, and c) communication with other stakeholders (see Fig. 1).

- Regulators are establishing risk communication process and tools including models for their roles.

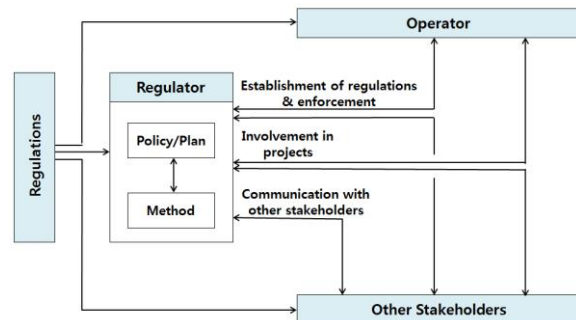


Fig. 1. The framework of regulator's role for stakeholder involvement.

3. Conclusions

The investigation has shown that an active role of regulator in stakeholder involvement activities is becoming important internationally for building confidence in the implementation of radioactive waste disposal. Accordingly, it is considered desirable for the Korean policy and/or regulations to mandate regulator's role in the stakeholder involvement in particular for the forthcoming disposal of high level radioactive waste. For this, the regulator may develop its own risk communication system including approaches, models and tools, making use of the information from this study.

REFERENCES

- [1] IAEA, Stakeholder Involvement in Nuclear Issues, INSAG-20, Vienna, 2006.
- [2] NEA, The Evolving Role and Image of the Regulator in Radioactive Waste Management: Trends over Two Decades, 2012.
- [3] NEA, From Information and Consultation to Citizen Influence and Power, FSC Flyer, 2010.
- [4] T. Varjoranta, The Regulator's Relationship with the Public and the Media: Satisfying Expectations; OECD, Stepwise Decision Making in Finland for the Disposal of Spent Nuclear Fuel Workshop Proceedings, Turku, Finland, 2001.
- [5] SKI, Design and Evaluation of Public Hearings for Swedish Site Selection, SKI Report 2003:32 August 2003.
- [6] US NRC, Report of the Public Communications Task Force, 2003.

Concept Development of Drift Storage-and-Disposal of Cs-Wastes

Heui-Joo Choi*, Jong-Youl Lee, In-Young Kim, and Dong-Keun Cho
Korea Atomic Energy Research Institute, 1045 Daeduk-daero, Yuseong, Daejeon
*HJCHOI@kaeri.re.kr

1. Introduction

KAERI is developing a pyroprocessing technology for the effective management of spent nuclear fuels generated in Korea. The main idea of pyroprocessing is the group separation of radionuclides to reuse the fissile nuclides and reduce the footprint of final repository. According to the current material balance, eight kinds of wastes will be generated from the pyroprocessing. Among them, the management of two highly-heat generating wastes, Cs-spent filter waste and Sr-spent filter waste, is not yet fixed.

Considering the relatively short half-life of Cs-137 and Sr-90, it is strongly recommended to store the wastes for several hundreds years at the surface facility and then dispose of them like low-level radioactive wastes. However, it is very difficult to operate the surface facility for several hundreds years. Thus, in this paper we propose a new concept to store and dispose of the Cs-spent filter wastes using drifts located around 300 meters below the ground surface.

The previous concepts, such as Japanese NUMO's CARE [1] and US DOE's Yucca Mountain Repository [2], were reviewed. It was found that the direct application of such concepts was not feasible. Thus, we propose that the concept of the Yucca Mountain Repository was applied to the storage for more than 100 years and then the Canadian NWMO's disposal concept is applied to the final disposal of the waste. In this study, we roughly designed the storage-and-disposal facility for the thermal analysis.

2. Concept Development

2.1 Amounts of Cs-waste

Cs-spent fuel filters will be generated from the preprocessing of spent nuclear fuel. According to the current version of material balance, the amount of the waste from 30 tHM of PWR spent fuels is given in Table 1. With the consideration of 23,618 tHM of

PWR spent fuels from 24 reactors, around 14,171 waste forms are expected.

Table 1. Amount of Cs waste from pyroprocessing of 30 tHM of PWR SNF

Total weight of waste, kg	2,750
Size of waste form, mm	300 (D) ×900 (H)
Weight of waste form, kg	160
Number of waste forms	18

The decay heat from one Cs-spent filter waste form was calculated and shown in Fig. 1. As shown in Fig. 1, most of decay heat disappears 200 years after discharge.

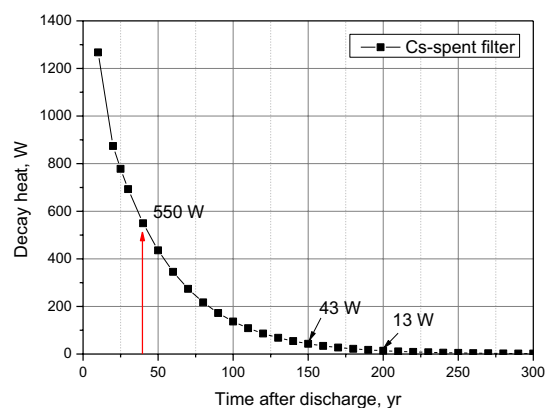


Fig. 1. Decay heat from one Cs waste form.

2.2 Disposal canister

A disposal canister was designed from the viewpoint of thermal management. According to the Yucca Mountain Repository (YRM) design, the thermal loading is around 1.5 kW/m. At the time of disposal, the decay heat from a Cs waste form showed around 0.55 kW as shown in Fig. 1. Thus, we designed a disposal canister which could hold eight storage canisters (Fig. 2), which was equivalent to around 1.83 kW/m. The inner structure was made of steel. However, outer copper coating was not decided

at this study, and it will be decided through a corrosion study.

During the storage period, the drift will be open and cooled by forced convection like YMR. The size of the drift was decided roughly 3.5 m by 3.5 m as shown in Fig. 3. The storage period will be determined through a thermal analysis, which is in progress. Thermal design criteria during the storage period are taken from those of the YMR and modified to the Korean situation:

- Temperature on the drift wall: below 200°C
- Temperature halfway between emplacement drifts: below 100°C

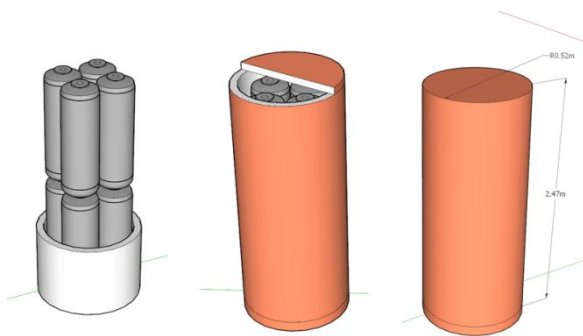


Fig. 2. Schematic of a disposal canister.

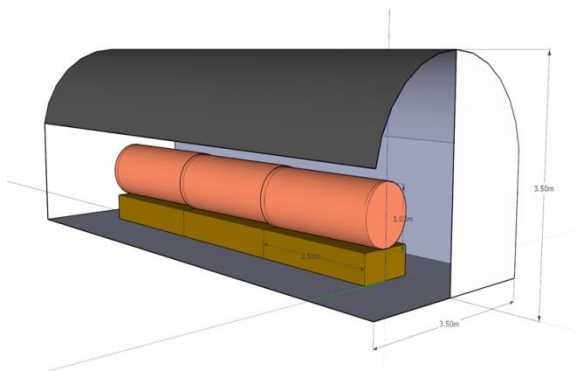


Fig. 3. Schematic of a disposal canister during storage period.

2.3 Buffer and final disposal

The Cs-spent filter waste contains very long-lived Cs-135 as well as Cs-137. Due to the long-lived Cs-135 we designed a disposal facility with bentonite buffer to mitigate the release of the radionuclides. Considering the small size of a disposal canister, we introduced the Canadian NWMO's buffer concept to the disposal of Cs waste [3]. Soon after the storage period when decay heat removed, the disposal canisters will be moved to a surface facility,

packaged with buffer blocks as shown in Fig. 4 and disposed of in the drifts for the final disposal. The size of blocks and drift spacing will be determined through the thermal analysis.

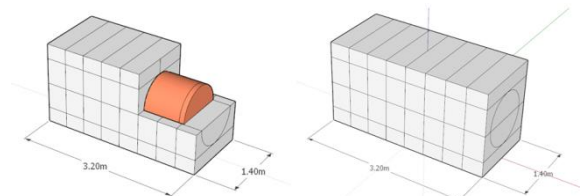


Fig. 4. Conceptual drawing of buffer.

3. Conclusion

Several kinds of radioactive wastes are expected from the pyro-processing of PWR spent nuclear fuels. Among them, the wastes containing cesium and strontium are planned to be stored for removing decay heats for several hundred years. In this paper, a concept for the storage-and-disposal of high-heat generating Cs waste was proposed. Based on the US DOE Yucca Mountain Repository and Canadian NWMO's geological repository, the storage-and-disposal concept was developed for the thermal analysis. Depending on the final calculation results, the specific design features of the system will be determined.

Acknowledgement

This work was supported by the Ministry of Science and ICT within the framework of the national long-term nuclear R&D program (NRF-2017M2A8A5014856).

REFERENCES

- [1] NUMO, Development of Repository Concepts for Volunteer Siting Environment, NUMO-TR-04-03, 2004.
- [2] US DOE, Yucca Mountain Science and Engineering Report, Revision 1, DOE/RW-0539-1, 2002.
- [3] Ruiping Guo, Thermal Response of a Mark II Conceptual Deep Geological Repository in Crystalline Rock, NWMO-TR-2016-03, 2016.

Effect of Initial Nuclide Concentration on the Purification of LiCl Waste Salt by Layer-Melt Crystallization Process

Tae-Kyo Lee*, Jung-Hoon Choi, Ki Rak Lee, Seung-Youb Han, and Hwan-Seo Park

Korea Atomic Energy Research Institute, Daedeok-daero 989-111, Yuseong-gu, Daejeon, Korea

*tklee@kaeri.re.kr

1. Introduction

The wastes salt generated from pyrochemical process after sufficient use of chloride salts as molten state electrolytes are LiCl waste salt from the electrolytic reduction process, and LiCl-KCl eutectic waste salt from the electrorefining and electrowinning processes. In particular, the waste salts have highly heat generative group I & II fission products such as Cs, Sr and Ba nuclides and highly radioactive generative rare earth fission products. Therefore, the fission products within the waste salt should be separated and concentrated in small volume to reduce the volume of final waste [1].

In this study, the effect of initial concentration on separation performance of group I & II nuclides from LiCl waste was investigated using layer-melt crystallization at various initial concentrations of nuclides.

2. Experimental

Layer-melt crystallization process was carried out in a 3 Kg/batch lab-scale apparatus as shown in Fig. 1(a). The apparatus consisted of crystallization furnace, melting furnace (recover purified LiCl from the crystallizer plate), crystallizer module and Ar circulation system (as coolant), data acquisition system. All the parts are located in a glove box maintained Ar atmosphere to prevent moisture and oxygen. LiCl crystals were grown by layer-melt crystallization, where crystallization plate is rectangular shape with thickness of 15 mm, 60 mm wide, and 200 mm height. Module of crystallizer consisted of three thermocouples, three crystallization plates [2]. The temperature monitoring system is applied to recover high purity LiCl crystal by stopping the crystallization process when the crystal is grown properly without over-growing. The stop point of process was determined by temperature monitoring of around crystallizer module: surface of

crystallization plate (Crystallizer TC), center of between crystallization plate (Center TC) and molten salt (Salt TC), as shown in Fig. 1(b).

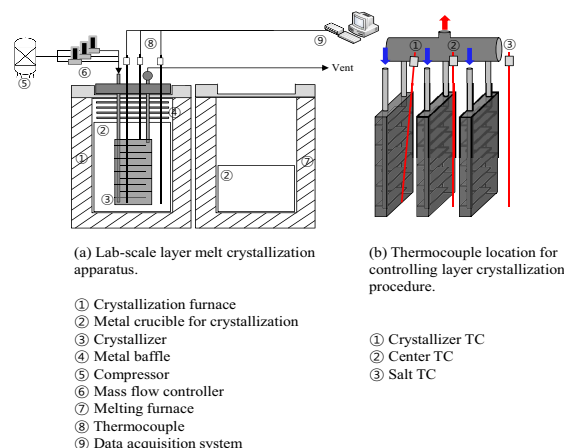


Fig. 1. Scheme of lab scale layer-melt crystallization apparatus.

The initial concentration ratio of cesium and strontium/barium nuclides are set based on the properties of LiCl waste salt generated after oxide reduction process. These tests were performed under various initial concentrations. After crystallization process, concentration of impurities in purified LiCl salt was analyzed using an Inductively Coupled Plasma Spectrometer.

3. Results and discussion

Solid-liquid phase diagram was identified separation possibility and temperature condition of crystallization process. Fig. 2 shows the solid-liquid phase diagram for the LiCl-CsCl-SrCl₂-BaCl₂ system drawn using the FactSage software. As shown in the Fig. 2, there is temperature range that exists only in a solid state of LiCl at 500~600°C. The solidus temperature of LiCl is decreased as the initial concentration of cesium and strontium/barium nuclides in the LiCl increased. The effect of the

initial concentration of impurity nuclides on the separation efficiency can be seen in Fig. 3. The results show that the separation performance decreased when the initial concentration of cesium and strontium/barium nuclides in the LiCl increased. This is because at a high initial concentration, many impurities of mother liquid are entrapped in the crystal layer under the same growth conditions.

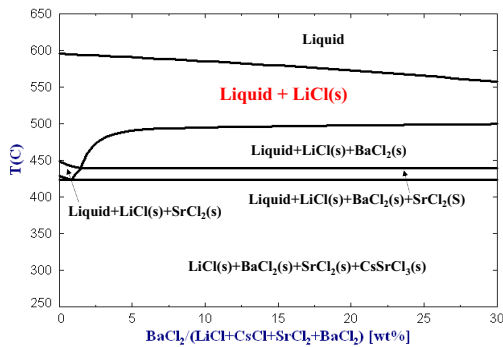


Fig. 2. Phase diagram of LiCl-CsCl-SrCl₂-BaCl₂ system.

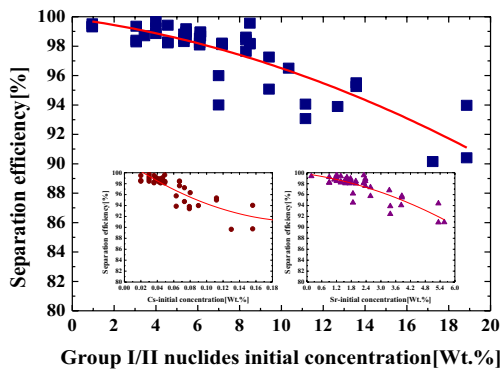


Fig. 3. Effect of the initial nuclides concentration on the separation efficiency.

REFERENCES

- [1] J. H. Choi, H. C. Eun, H. S. Park, D. H. Ahn, "Study on a Separation and Solidification of Sr and Ba Nuclides from LiCl Waste Salt Generated from Pyrochemical process", JNFCWT, 13(S), 77 (2015).
- [2] Y. Z. Cho, T. K. Lee, J. H. Choi, H. C. Eun, H. S. Park, I. T. Kim, G. I. Park, "Study on LiCl Waste Salt Treatment Process by Layer Melt Crystallization", Proc. of the Global 2013, Sept 29-Oct 3, 2013, Salt Lake City.

Melt-Crystallization Monitoring System for the Purification of LiCl Salt Waste

Jung-Hoon Choi*, Tae-Kyo Lee, Ki Rak Lee, Seung-Youb Han, Na-Young Kim, Seon-Ah Jang,
and Hwan-Seo Park

Korea Atomic Energy Research Institute, Daedeok-daero 989-111, Yuseong-gu, Daejeon, Korea

*mrchoijh@kaeri.re.kr

1. Introduction

Korea Atomic Energy Research Institute (KAERI) is developing pyrochemical process for the recycling of PWR spent nuclear fuels using electrochemical method to achieve the volume reduction of radioactive waste [1-3]. During the process, LiCl salt wastes are planned to be recycled using melt-crystallization process. In this study, modified crystallization monitoring system has been developed to prevent over-growing of LiCl crystal on the crystallization plate during melt-crystallization process.

2. Experimental

Melt-crystallization tests were conducted using engineering-scale melt-crystallization system capable of 20kg/batch treatment. In this study, the tests were conducted using 10 kg of surrogate LiCl salt waste. The surrogate LiCl salt waste having 0.03wt% CsCl, 1.11wt% SrCl₂ and 2.31wt% BaCl₂ were charged in the crystallization vessel and melted at 700 °C in the crystallization furnace. Then, crystallizer having 6 crystallization plates were immersed in the molten LiCl salt waste and the LiCl crystals were recovered on the 6 crystallization plates by cooling the plates using Ar coolant gas. Melt-crystallization process was terminated using modified crystallization monitoring system by measuring temperatures at each characteristic points. Nuclide separation efficiencies (%) were calculated by the relationship

of $(C_{before} - C_{after})/C_{before} \times 100$, where C_{before} and C_{after} are nuclides concentration in the LiCl salt waste before and after crystallization process. Concentration of cesium nuclide was measured using an Inductively Coupled Plasma Mass Spectrometer (ICP-mass) with a Thermo Elemental (Xseries X5) and concentrations of strontium/barium nuclides were measured by Inductively Coupled Plasma Atomic Emission Spectrometer (ICP-AES) with a Perkin Elmer (Optima 4300DV).

3. Results and Discussion

Fig. 1 shows the recovered LiCl crystal on the crystallization plates after melt-crystallization test. As shown in the Fig. 1 (a), when used previous monitoring system, over-grown LiCl crystal was found at the center part of crystallization plate owing to the temperature gradients. Meanwhile, modified crystallization monitoring system has been developed in consideration of temperature gradients within the reaction vessel, where the crystallization process of center part of crystallization plate is first terminated owing to the relatively high temperature. Therefore, as shown in the Fig. 1 (a), when used the modified crystallization monitoring system, we could obtain pure LiCl crystals on the crystallization plates with uniform distribution in terms of LiCl quantity. Also, the nuclide separation efficiencies were found to be enhanced as over 90% relative to that of purified LiCl crystal recovered using previous monitoring system.

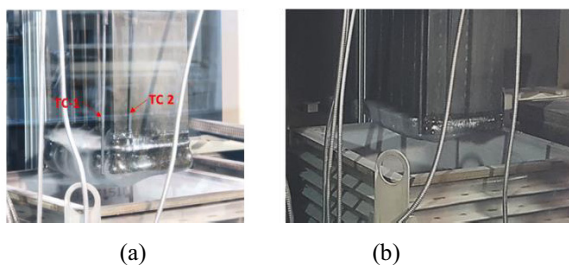


Fig. 1. (a) Over-grown crystal grown by previous monitoring system, (b) Recovered LiCl crystal without over-growing using modified crystallization monitoring system.

4. Conclusion

In the present study, melt-crystallization monitoring system has been modified to prevent over-growing of LiCl salt on the crystallization plates by considering temperature gradient within the reaction vessel. Using the modified crystallization monitoring system, high nuclide separation efficiency was obtained with uniform distribution of LiCl salt on the crystallization plates.

REFERENCES

- [1] H.S. Lee, J.M. Hur, J.G. Kim, D.H. Ahn, Y.Z. Cho, and S.W. Paek, "Korean Pyrochemical Process R&D activities", Energy Procedia, 7, 391-395 (2011).
- [2] H.S. Lee, G.I. Park, K.H. Kang, J.M. Hur, J.G. Kim, D.H. Ahn, Y.Z. Cho, and E.H. Kim, "Pyroprocessing technology development at KAERI", Nucl. Eng. Technol., 43, 317-328 (2011).
- [3] H.S. Lee, G.I. Park, J.W. Lee, K.H. Kang, J.M. Hur, J.G. Kim, S.W. Paek, I.T. Kim, and I.J. Cho, "Current Status of Pyroprocessing Development at KAERI", Sci. Tech. Nucl. Install., 2013, 1-11 (2013).

4분과

방사성폐기물관리 (Oral)



Removal of Iodide and Iodate Using Layered Double Hydroxides (LDHs) Under Aqueous Condition

Jaehyuk Kang* and Wooyong Um

Division of Advanced Nuclear Engineering, POSTECH, 77, Cheongam-ro, Nam-gu, Pohang, Korea

*jaehyukkang@postech.ac.kr

1. Introduction

Continuous operation of nuclear power plants (NPPs) produces an amount of electric power. However, because radioactive wastes including radioiodine are also generated, the wastes should be carefully managed during NPPs operation. In particular, radioiodine species are required to be more carefully managed because of their long half-lives ($t_{1/2}$ of $^{125}\text{I} = 59.40$ d and $t_{1/2}$ of $^{129}\text{I} = 1.57 \times 10^7$ y) [1], and toxicity which causes skin burns and thyroid cancer.

In this study, we used synthesized layered double hydroxides (LDHs) to remove iodide and iodate in aqueous condition. LDHs show high anion exchange performance and memory effect to reform their original structure by adsorbing anion after calcination. Our specific objective is to study the removal of radioiodine species in aqueous condition using synthesized LDHs and understand the removal mechanism.

2. Experimental section

2.1 Synthesis of LDHs

LDHs are synthesized using co-precipitation method. Divalent metal(II) chloride and trivalent metal(III) chloride reagents were mixed as concentration ratio of 2:1 and 4:1 in deionized water (DIW). Metal ions are Mg^{2+} , Co^{2+} , Fe^{3+} , and Cr^{3+} . The pH of mixed metal solution was kept about ~12 at room temperature. Then, it was aged at 65 °C under stirring vigorously for 12 h.

The resultant suspension was separated by centrifugation. Subsequently, supernatant was decanted into empty vial to obtain only slurry, then it was washed using DIW. Finally, LDHs were dried at 110°C in oven for 12 h. To check memory effect for iodine removal, dried LDHs were calcined at 300, 500, and 700°C for 2 h before use [2].

2.2 Iodine removal test

Kinetic test to conclude removal equilibration time was conducted for 168 h. Stable iodine, ^{127}I (surrogate for radioiodine) solutions were prepared in crimp-top glass amber vials. Iodide and iodate solutions (4 mg/L and pH 4.3-5.8) were prepared and reacted with LDHs at solid to solution ratio of 1 g/L. The samples were covered by the lids and placed in a basic platform shaker (os-3000) for complete mixing at 130rpm. After kinetic test, the aqueous samples were collected at different reaction times (0, 1, 3, 6, 12, 24, 72, and 168 h) and analyzed using inductively coupled plasma mass spectroscopy (ICP-MS, NexION 300D) to measure iodine concentrations.

3. Results

The results of iodide and iodate removal test are shown in Fig. 1. The iodide removal efficiency by MgFe_Cl_C_21 with calcination became 89.2% within 3 d, whereas MgFe_Cl_NC_21 without calcination showed 6.20% after 3 d. The iodate removal efficiency by CoCr_Cl_NC_41 without

calcination became 97.1% within 3 d, whereas CoCr_Cl_C_41 with calcination showed 17.0% within 3 d.

Korean government (MSIP : Ministry of Science, ICT and Future Planning) (No. NRF-2016R1D1A1B-02013310)

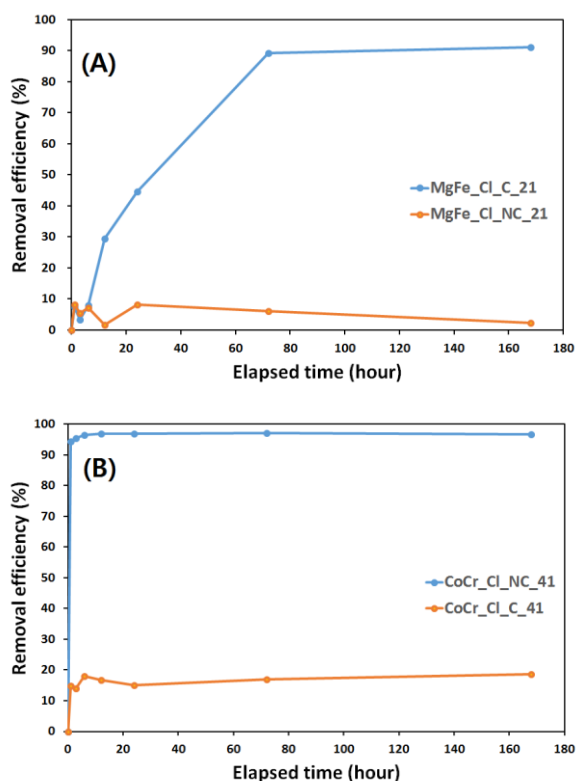


Fig. 1. Results of iodide (A) and iodate (B) removal kinetic test using synthesized LDHs.

4. Conclusions

Iodide was removed by synthesized LDHs with calcination better than synthesized LDHs without calcination. However, iodate was removed by synthesized LDHs without calcination better than synthesized LDHs with calcination. These results suggest that memory effect is important for LDHs to remove iodide, whereas anion exchange is dominant to remove iodate.

ACKNOWLEDGEMENT

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the

REFERENCES

- [1] Yujiro Watanabe, et al., "Novel Long-Term Immobilization Method for Radioactive Iodine-129 Using a Zeolite/Apatite Composite Sintered Body", ACS applied materials & interfaces, 1 (7), 1579-1584 (2009).
- [2] F.L. Theiss, et al., "A review of the removal of anions and oxyanions of the halogen elements from aqueous solution by layered double hydroxides", Journal of Colloid and Interface Science, 417, 356-368 (2014).

Concentration Averaging Option of NPP Decommissioning Low Level Radioactive Waste

Sun-Kee Lee*, Heon Kim, and Chang-Lak Kim

KEPCO International Nuclear Graduate School, 658-91 Haemaji-ro, Seosaeng-myeon, Uljgun, Ulsan, South Korea

*penda1226@naver.com

1. Introduction

Disposal choices according to domestic radioactive waste classification are as illustrated in Fig. 1. The intermediate and low level wastes occurred from immediately decommissioning Kori NPP Unit 1 are estimated mostly to be the low and very low levels with VLLW 9.2% higher than LLW. The net volume generation of intermediate and low level radioactive wastes expected to occur upon immediate decommissioning of Kori NPP Unit 1 is estimated to be 0.2% for the intermediate level, 45.3% for the low level, and 54.5% for the very low level. Scabbling concrete is estimated to have a very low level almost three times more generated than the low level [1]. Scabbling concrete is defined as dust-form concrete of the structure removed by scabbler and is expected to be potential target for concentration averaging. US NRC's position is that if concentration averaging or blending does not diminish the overall safety of LLW disposal, CA or blending can be allowed.

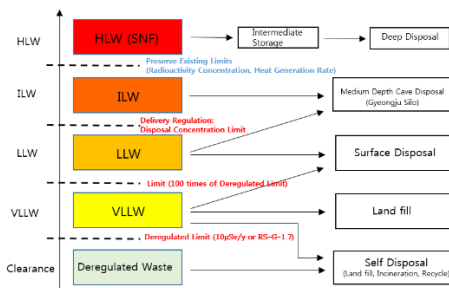


Fig. 1. Radioactive Waste Classification vs Disposal Option.

It is suggested that the activated bio-shield concrete could be concentration averaged to VLLW for disposal after sectioning the activated part [2].

2. Concentration Averaging of Decommissioning Waste

2.1 Blending concept

The NRC defines blending as a relatively uniform mixing of low level wastes with different radioactivity concentrations, so being appropriate for disposal. The types of wastes include those that have physical and chemical similarities, such as ion exchange resin from each system of a nuclear power plant, but may include other types of wastes as long as the final mixture can be homogeneous, such as soil or ash. To demonstrate that waste is adequately blended, there shall be no hot spots of waste $\geq 0.2 \text{ m}^3$

that have a sum of fractions >10 times the average concentration of the blended product for the specific radionuclides [3].

2.2 Concentration Averaging Application

In the United States, the concept of concentration averaging was newly established in 2015. This includes the mixing of wastes with different concentrations or the determination of the concentration of wastes with different concentrations. When decommissioning nuclear power plants in the future, many wastes are generated, so that the handling and disposal of wastes can be efficiently performed by using the concept of concentration averaging.

Intermediate and low Level radioactive wastes from decommissioning Kori NPP Unit 1 are possibly treated to lower radioactivity concentration or radioactivity by blending, concentration averaging or encapsulation. Scabbling concrete is likely to be appropriate for concentration averaging rather than blending a waste stream. The amount of scabbling concrete classified as low level is estimated to be tens of 200L drums.

In the case of decommissioned wastes, there is a few waste with hotspots, so concentration averaging is possibly applied to mix different waste streams with different concentrations or mass of concrete or metal with different concentrations. It is stipulated that the radioactive material should not be diluted or mixed intentionally according to the Notification Notice of Korean Nuclear Safety and Security Commission, but this is to prevent the release of the radioactive material into the general environment. That is, some of the high-level wastes are classified as low level waste by mixing them with low level wastes, and there is no problem because they are disposed of as radioactive waste.

2.3 Domestic Availability

In Korea, the acceptance criteria for intermediate and low level wastes are expected to be the same except for the concentration of wastes. However, the acceptance criteria for very low level wastes will be eased considerably compared to intermediate and low level wastes. Therefore, there is no blending effect of intermediate level wastes, and only a small part of low level wastes can be blended to be effective.

In the United States, blending is presented in a variety of ways including ion exchange resins, cut cartridge filters, soil, and charcoal. However, in the case of decommissioning wastes from Kori NPP Unit 1, the types of wastes applicable to blending are

limited. For example, the ion exchange resins generated in the decontamination of the primary system are at the intermediate level waste, and the ion exchange resins generated after the treatment of other liquid wastes are at the low level waste.

Very low level ion exchange resins, which may be blended with intermediate and low level ion exchange resins, are not generated. Likewise, waste filters do not have very low level waste. Contaminated soil can be blended to reduce low level waste. It is estimated that contaminated soil wastes are not generated by decontamination at present. In the case of metal, the slag after melting is disposed after being crushed to put it in a large amount in the container, but the radioactivity level is expected to be high, and there will be no blending effect. Thus, the effect of blending can be up to tens of drums in a maximum of almost one hundred drums.

3. Concentration Limit for Blending

3.1 Conditions to potentially facilitate blending

In order to smoothly perform blending, the radioactivity levels of scabbling concrete and contaminated soil, which are likely to be blended from the time of radioactive waste generation, must be determined and managed separately to facilitate blending. Lower level part of scabbling concrete classified as low level waste can be blended with very low level scabbling concrete.

3.2 Concentration Allowance for Mixture

It is recommendable that the radionuclide concentration limit of low level waste to be blended be set at 1.5 times the VLLW class limit of mixture as illustrated in Fig. 2. The factor of 1.5 is reasonable conservatively considering CA BTP 1995 and 2015 versions. However, if the LLW quantity is too high, the factor should be adjusted to 1.3 times, considering the amount of very low level waste in that blending is determined by the concentrations and amounts of very low level and low level wastes. It is estimated that the about 60 drums of scabbling concrete is capable of being blended since 189 drums of LLW and 551 drums of VLLW are generated based on scabbling concrete amounts.

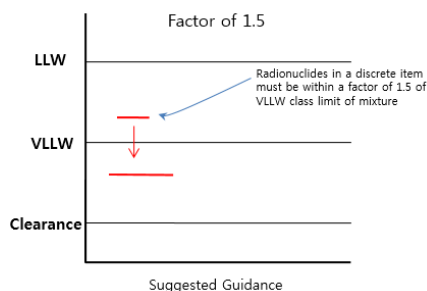


Fig. 2. Radionuclide Concentration Constraint for LLW.

4. Conclusion

Concentration averaging or blending has appropriately been performed if the waste mixture has homogeneous radioactivity concentrations and is radiologically safe without hot spots. Concentration averaging on scabbling concrete requires control of the size and shape of concrete particles as well as the radionuclide concentration or radiation level to ensure homogeneity of the mixture. Radionuclide concentration constraint provides the guidance for each item in LLW mixture to be concentration averaged to VLLW.

Acknowledgements

This research was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KOFONS), granted financial resources from the Nuclear Safety and Security Commission (NSSC), Republic of Korea (No. 1305009).

REFERENCES

- [1] Ministry of Commerce, Industry and Energy, "Final Technical Development Report on Evaluating Nuclear Power Plant Decommissioning Source Terms", pp.37-46, July 2016.
- [2] David S. Kessel and Chang-Lak Kim, "U.S. Policy and Current Practices for Blending Low-Level Radioactive Waste for Disposal", JNFCWT (Journal of Nuclear Fuel Cycle and Waste Technology) Vol.14 No.3 pp.235-243, September 2016.
- [3] US NRC, "Concentration Averaging and Encapsulation Branch Technical Position, Revision 1", Volume 1, February 2015.
- [4] US NRC, "Concentration-Averaging and Encapsulation Branch Technical Position Response to Stakeholder Comments and Technical Basis, Revision 1", Volume 2, 2015 February.
- [5] EPRI, "Implementation Guidance for the Nuclear Regulatory Commission Branch Technical Position on Concentration Averaging and Encapsulation, Revision 1", September 2016.

Decontamination Technology for Low Activity Spent Resin From PWR's

Young-Soo Jo^{*}, Hyun-Ki Kim, Jun Heo, Se-Ban Lee, and Jin-Hyeon Sung
HuvisWater Co., 16, Gongdan 1-daero 322beon-gil, Siheung-si, Gyeonggi-do, Korea
^{*}jys@huviswater.com

1. Introduction

The low activity spent resin from a pressurized water reactor(PWR) is stored in the radwaste building. The storage space of the radwaste building has been saturated due to the prolonged operation time of the nuclear power reactor[1].

Therefore, it is necessary to develop a proper method to handle and transfer the spent resin. However, because the spent resin contaminates a trace amount of ^{60}Co and ^{137}Cs , the spent resin is currently under long-term storage.

Particularly, ^{60}Co and ^{137}Cs , which are mainly generated in the nuclear reactor, are adsorbed physicochemically with ferrite as a form of the radioactive crud. This crud moves in the ion exchange resin tower and accumulate at the top layer. These nuclides are also physicochemically bonded to the ion exchange resin. This spent resin contaminates low concentration of radioactive, but it is necessary for decontamination because it is more concentrations than the IAEA Safety standards[2]. If the spent resin is sent to the radioactive waste disposal facility without decontamination, it will be a lot of expense, so the decontamination technique is necessary to treat the spent resin.

This study focuses on introducing of the method of the radioactive crud separation without the secondary waste.

2. Experiment method

2.1 Preparation of spent resin

Before of the separation and decontamination experiment, in order to evaluate the radioactivity of the spent resin from PWR, we transferred the spent resin from Hanbit Nuclear Power Plant to the isotope laboratory in Huviswater. The major nuclides of spent resin were ^{60}Co and ^{137}Cs . Table 1 shows the specific activity of the spent resin.



Fig. 1. The spent resin from Hanbit nuclear power plant.

Table 1. The specific activity of the spent resin

Sample	A		B		C	
Nuclide	^{60}Co	^{137}Cs	^{60}Co	^{137}Cs	^{60}Co	^{137}Cs
Specific activity (Bq/g)	0.413	MDA	0.462	MDA	0.075	0.049

MDA: Minimum Detectable Activity

The activity of all sample are more than IAEA safety standards, it is necessary to reduce below 0.1 Bq/g of ^{60}Co and ^{137}Cs .

2.2 Decontamination experiment method of spent resin

In order to separate the radioactive crud, that has been physicochemically bonded to the spent resin, the ultrasonic wave was applied. The radioactive crud is separated by the cavitation of the shock wave. The spent resin is separated radioactive crud and spent resin by the mesh filter, and the radioactive crud moves to the next step with process water. Because this process water gets mixed with the radioactive crud, it must treat water to remove the radioactive crud. As mentioned above, the nuclides are combined with ferrite as the form of radioactive crud. Therefore, the magnets are very useful to collect and remove the radioactive crud. We tested 2 kind of experiment(Lab. scale and Semi-pilot scale). Semi-Pilot devices are shown in Fig. 2.



Fig. 2. Semi-Pilot decontamination system(4L/hr).

3. Results and discussion

3.1 Results of Lab. scale experiment

Table 2 show that the total activity of the spent resin

decontaminated from 0.413Bq/g, 0.462Bq/g, 0.075Bq/g of ^{60}Co to MDA. As a result, it means that almost ^{60}Co and ^{137}Cs were removed completely.

Table 2. The specific activity of the spent resin before and after decontamination

Sample		A		B		C	
Nuclide		^{60}Co	^{137}Cs	^{60}Co	^{137}Cs	^{60}Co	^{137}Cs
Specific activity (Bq/g)	Before	0.413	MDA	0.462	MDA	0.075	0.049
	After	MDA	MDA	MDA	MDA	MDA	MDA

3.2 Results of Semi-pilot scale experiment

Table 3 show that the total activity of spent resin decontaminated from 0.11Bq/g, 0.01Bq/g, to 0.002Bq/g, 0.006Bq/g. As a result, it means that almost ^{60}Co and ^{137}Cs were removed as well. And also, it is much lower than the IAEA safety standards.

The total activity of treated process water was MDA value, it means ^{60}Co and ^{137}Cs were removed by magnet.

Table 3. The specific activity of the spent resin and process water before and after the decontamination

Sample		Spent resin		Process water	
Nuclide		^{60}Co	^{137}Cs	^{60}Co	^{137}Cs
Specific activity (Bq/g)	Before	0.11	0.01	MDA	MDA
	After	0.002	0.006	MDA	MDA

3.3 Optical microscope image of spent resin

Fig. 3 shows the optical microscope image of spent resin before and after decontamination. Fig. 3(a) is the image of the spent resin before the decontamination test, and Fig. 3(b) is the image of the spent resin after decontamination test. The optical microscope scales 75 times. Fig. 3(b) means that almost the radioactive crud have been removed.

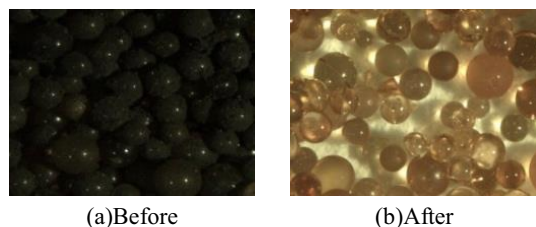


Fig. 3. The optical microscope image of spent resin before and after decontamination(x 75).

4. Conclusion

Through the decontamination experiment, it was assured that almost ^{60}Co and ^{137}Cs were removed. And, process water can be also reused because the nuclides were not detected in the process water.

Through the magnet separation of the spent resin, it was affirmed that the radioactive crud can be collected by magnet, and it can be expected that there will be considerably less amount of the secondary waste than radioactive crud collection by using the bag filter and the cartridge filter.

Therefore, if the spent resin is decontaminated by applying these technology in PWR, It must be an effective technology for dispose of spent resin with minimum amount of secondary waste.

REFERENCES

- [1] Hyun-ki Kim, besides 4, "The test of Co-60 decontamination from the spent waste resin using ultrasonic wave technology", Abstracts of Proceedings of the Korean Radioactive Waste Society Spring 2014, 12(1), 313-314 (2014).
- [2] Application of the Concepts of Exclusion, Exemption and Clearance (RS-G-1.7).

Development of Treatment Technology of Non-Disposal Radwastes for the Increase of Their Volume at Disposal Site

MinYoung Jung*, SangHyun Lim, and Jongsoon Song

Chosun University, 309, Pillmoon-daero, Dong-gu, Gwangju, Republic of Korea

*anstjdrh12@nate.com

1. Introduction

In general, the radioactive wastes (liquid, dried powder, and sludge) were solidified by mixing with a proper binding matrix (cement, polymer, asphalt, etc.), but their solidified volume inevitably increases due to the limitation of incorporation of radwastes to a binding matrix. To solve this problem, some of Korean NPPs operate both the CTS (Concentrate Treatment System, evaporation, and granulation) and the polymer solidification system. The size distribution of granulated wastes produced by CTS is so broad (diameter; several μm ~ several cm) that a polymer cannot impregnate into the void between granulates due to their fine particulate.

On the contrary, if the powdered radwastes can be compacted as the high-density pellets, and the pellets can be loaded in a waste drum as much as possible, and then the adhesive matrix can be seized while filling the void between pellets.

Through this study, we will contribute to the reduction of disposal cost of radioactive waste and the improvement of disposal technology.

2. Roll Compaction of Powdered Radioactive Wastes

2.1 Necessity of Technology Development

Non-Disposal radwaste is a term referring to particulate radioactive waste generated during the operation of a nuclear power plant or dismantling a nuclear power plant. Because it is not currently securing processing technology, it is targeted for concentrated waste liquid powder, slurry and sludge which are not processed. In addition, any powder waste (concrete crushed material, decontaminated sludge, etc.) generated in the process of dismantling nuclear power plants in the future, and contaminated soil of fine size not used for decontamination are also target waste.

These radioactive wastes are concentrated and

managed in specially designed and constructed areas in the country for comprehensive and safe management. This is referred to as "disposal of radioactive waste at disposal sites." These radioactive wastes must satisfy the acquisition criteria of the repository and the regulations of the country of India and safety must be secured [1][2].

Table 1. Acceptance criteria test method for disposal safety

classification	Test Items	Related standards
Structural stability	Compressive strength test	Hard case : KS F 2405 Soft case : KS F 2351
	Immersion test	NRC/Technical Position on Waste Form, Rev.1
	Heat cycle test	ASTM B553
	Free water measuring	Apply method similar to ANS 55.1
	Irradiation test	NRC/Technical Position on Waste Form, Rev.1
Leachability	Leaching Test	ANS 16.1

Table 2. Waste solidification criteria for disposal safety

Number	Test Items	Criteria	
		Hard solidification	Soft solidification
1	Compressive strength test	3.44 MPa (500 psig) or higher	0.41 MPa (60 psig) or more when the specimen's vertical strain is 3%
2	Compressive strength test after immersion test		
3	Compressive strength test after heat cycling test		
4	Compressive strength test after irradiation test		
5	Leaching Test	The leaching index of the sample is 6 or more for Cs, Sr, Co nuclides	
6	Free water measuring	1% of high-integrity containers, and others shall not exceed 0.5%.	

2.2 Differentiation of technology development

The feature of this technology is that the dried powder is pelletized and compressed to a high degree, and the pellets are packed in the drum to the maximum to solidify the pores between them into a high-adhesive and high-strength polymer.

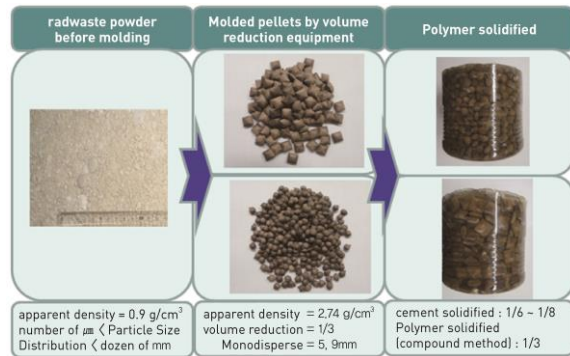


Fig. 1. Roll Compaction of Powdered Radioactive Wastes.

The feature of this technology is that the dried powder is pelletized and compressed to a high degree, and the pellets are packed in the waste drum to the maximum to solidify the pores between them into a high-adhesive and high-strength polymer.

If the bulk density of Non-Disposal waste is maximized and pellets are manufactured in a form that maximizes the filling rate in the waste drum and the pellet filling space is consolidated into a highly adhesive special polymer then it is more than 3 times as much as that of powder ($0.7 \sim 0.9 \rightarrow 3.0 \text{ g/cm}^3$), 1.2 times higher on pellet and drum basis and 6 times higher than cement solid dependent, and the polymer solid is 3 times or more.

Conventional solidification technology adopts a simple mixing method between waste and solidification medium, so the amount of waste is increased rather than being scraped. However, the present technology is differentiated to attain high sensitivity by multiplying the filling rate in the drum first.

3. Conclusion

If the radioactive waste is delivered to the disposal site without being highly reduced, the amount of waste disposed at the disposal site will increase

significantly, necessitating another disposal site. For this reason, it is considered that the development of waste reduction technology is essential and important, and it can contribute to the reduction of related burdens such as excessive waste disposal cost and solving the problem of saturation of temporary waste storage in NPP through technology development.

REFERENCES

- [1] Radwaste Acceptance Criteria, LILW-Operation-Radiation-028, Korea Radioactive Waste Agency.
- [2] Solidification Radwaste structural soundness verification of examination, LILW-Operate-Radiation-048, Korea Radioactive Waste Agency.

Carbon-14 Inventory Optimization Process for LILW Silo Repositories in a Normal Scenario by Adopting Triangular Distribution Function

Sol-Chan Han and Jong-Il Yun*

Dept. of Nuclear and Quantum Eng., KAIST, 291 Daehak-ro, Yuseong-gu, Daejeon, Korea

*jiyun@kaist.ac.kr

1. Introduction

For the disposal of low- and intermediate-level radioactive waste (LILW), a silo type repository is currently being operated in Gyeongju.

In our previous work, we developed the safety assessment model for LILW disposal system by using Monte Carlo based program, Goldsim, and evaluated the radiological effect in terms of annual dose rate.

Based on the safety assessments, carbon-14 (C-14) appeared as a radionuclide which dominantly contributes to the maximum annual dose rate [1] as reported in other study [2]. Therefore, the inherent uncertainties of C-14 related input data (i.e. solubility, distribution coefficient, etc.) may highly affect the reliability of safety assessment. Moreover, C-14 inventory, calculated from the scaling factor, necessarily involves uncertainty as well.

In this framework, the triangular distribution function is adopted to examine the effect of variation in C-14 inventory on the safety assessment results. In addition, the estimation of the safety margin and optimization process for C-14 inventory has also been achieved.

2. Model Description

2.1 Performance assessment module of silo repository

The figure below indicates a schematic diagram of the modelled silo and a part of performance assessment module of silo repository. Radioactive wastes are stored in 200-L waste drum and repacked in the concrete waste box.

From a conservative point of view, we presume that the silo is fully saturated with groundwater and waste drum loses its credibility immediately after its closure.

Once the silo is saturated with groundwater,

radionuclides can be released from the damaged waste drum and diffused to engineered barrier system (EBS). The released radionuclides are transported through the fractures in the geosphere and migrate to the biosphere. Finally, they affect human and environment radiologically.

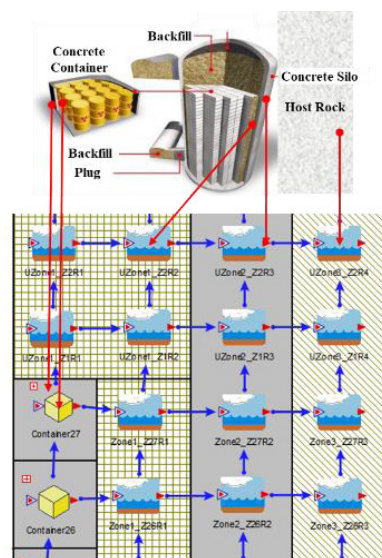


Fig. 1. Schematic diagram of modelled silo and performance assessment module.

2.2 Application of triangular distribution function to C-14 inventory in silo repository

In general, the normal distribution function and the triangular distribution function are widely used to represent the variation of data. In the case of the normal distribution function, both ends of which are unbounded, often samples outliers. On the other hand, the triangular distribution function can fix the exact boundary conditions. Therefore, the reasonable modeling results are achievable.

In this study, the triangular distribution function for the C-14 inventory is adopted to implement the variation of the inventory. The lower limit, upper limit, and the distribution center are set as 10^3 Bq, 10^{23} Bq, and 10^{13} Bq, respectively.

3. Modeling Result

For six silos, each C-14 inventory is randomly generated from the triangular distribution function and simulated 500 iterations. Fig. 2 illustrates the annual dose rate of all realizations. Within a few tens of years after closure, there are no significant differences in the annual dose rate among the iterations. This is because C-14 normally releases into biosphere in the next decades after closure. On the other hand, a hundred years after closure, C-14 inventory shows high sensitivity versus the annual dose rate.

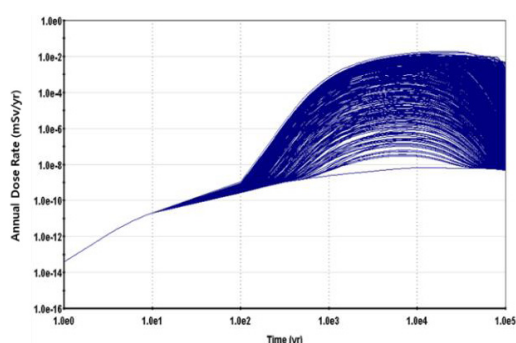


Fig. 2. Annual dose rate obtained from 500 simulations.

The maximum annual dose rates are also calculated from each simulation, as described in Fig. 3. The black dots indicate the maximum annual dose rates obtained from each simulation. The red dot represents the maximum annual dose rate which is calculated with actual C-14 inventory (1.7×10^{14} Bq) that is expected to be disposed of in the silo.

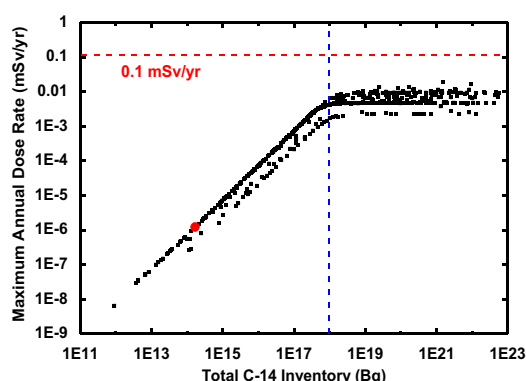


Fig. 3. Maximum annual dose rate according to C-14.

As can be seen in Fig. 3, the maximum annual dose rate becomes increasing until the total C-14 inventory is reached to nearly 5×10^{18} Bq and after that, the slope of the graph is dulled and saturated.

The solubility of carbon is fixed, there is no increase in the discharge rate of C-14, even though its inventory increases at these inventories. Accordingly, the maximum annual dose rate is not increased either. However, the maximum annual dose rate is still lower than the safety limit, 0.1 mSv/yr, although the maximum annual dose rate reaches the highest value.

In addition, the inventory margin for C-14 is calculated to be 5×10^{18} Bq, almost 30,000 times higher than the actual total C-14 inventory.

4. Conclusions

In our modeling environment, it was observed that the maximum annual dose rate reaches the highest value and does not exceed the safety limit when the inventory of C-14 becomes greater than 5.11×10^{18} Bq.

The margin of the C-14 inventory of silo is estimated to be nearly 30,000 times higher than that of originally planned to be disposed. Thus, the silo repository would be maintaining its safety even if the repository additionally disposes C-14 contaminated radioactive wastes at least several thousands of times higher than the original inventory. Moreover, based on the great inventory margin of the silo repository, it is expected to ensure its safety despite of uncertainties of C-14 related input data.

5. Acknowledgement

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. 1305032).

REFERENCES

- [1] S. C. Han and J.-I. Yun, "Sensitivity Analysis for Major Input Parameters in LILW Repository Safety Assessment Model", Proc. of the KRS 2016 Autumn Conference, 14(2), October 12-14, 2016, Jeju.
- [2] S. Lee and J. Kim, "Post-closure safety assessment of near surface disposal facilities for disused sealed radioactive sources", Nuclear Engineering and Design, 313, 425-436 (2017).

Development of C-14 Stripping Method From Waste Resin Generated From HWR via Microwave Treatment: Part 2, Demonstration Equipment for C-14 Stripping

Ki Rak Lee^{a*}, Hwan-Seo Park^a, Hyeon-Oh Park^b, Jung-Hoon Choi^a, and Seung-Youb Han^a

^a Pyroprocess Technology Division, Korea Atomic Energy Research Institute (KAERI), 111 Daedeok-daero 989, Yuseong-gu, Daejeon, Republic of Korea

^b Sunkwang T&S Co.,Ltd, Sunkyung Officetel 20F, #3, Gongwon-ro, Guro-gu, Seoul, Republic of Korea

*kirakki@kaeri.re.kr

Inorganic and organic ion exchange materials were generally applied to liquid processes in nuclear reactor. In the case of heavy-water reactor (HWR), zeolite, active carbon, anion resin, and cation resin were used to treat liquid processes such as reactor primary coolant cleanup and liquid radioactive waste management system. Then, used ion exchangers were stored at storage tanks. Various kinds of nuclides were adsorbed in ion exchange materials. Especially, C-14, long half-life nuclide, was highly concentrated in anion resin, and waste resin was treated as intermediated level radioactive waste (ILW).

In KOREA, 4 units of heavy-water reactor are operated and a lot of spent resins are generated. Until now, 580 m³ of waste resins were generated from Wolsung HWR unit 1 and 2 and stored at storage tank (storage tank capacity of Wolsung HWR unit 1 and 2: 986 m³). Enormous cost will be expected to dispose this radioactive waste if there is no process treating this waste.

Conventional process to treat radioactive wastes is not suitable for treating spent resin. To solve this problem, various kinds of processes were developed such as acid stripping, PLO process, activity stripping, thermal treatment, and etc. In this study, new process using microwave, is suggested. Basic characterization of anion and cation surrogate waste resin was conducted. Demonstration equipment was

manufactured using characterization data.

In our previous study, C-14 nuclide was easily desorbed from waste resin via microwave treatment. Then, demonstration equipment using microwave is manufactured in this study. The equipment was composed of a microwave reactor, a condenser, two adsorption tower, and circulation part. Closed loop system was applied to the equipment. Test using surrogate waste resin (mixture of anion and cation) is scheduled and real waste resin will be treated for demonstration.

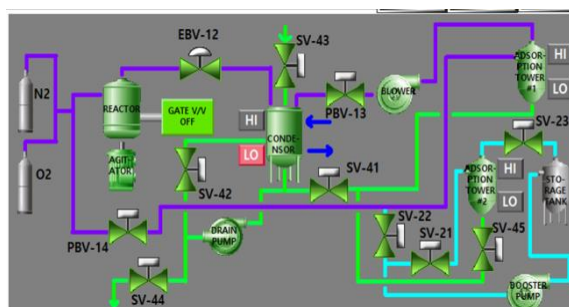


Fig. 1. P&ID of demonstration equipment.



Fig. 2. Image of demonstration equipment.

REFERENCES

- [1] Technical Reports Series no. 408, Application of ion exchange processes for the treatment of radioactive waste and management of spent ion exchangers, IAEA, Vienna, 2002.
- [2] M. Matsuda, K. Funabashi, T. Nishi, H. Yusa, M. Kikuchi, "Decomposition of ion exchange resins by pyrolysis", Nuclear Technology, 75 (1986), 187-192.

Chemical Pretreatment of Dry Active Waste for Determination of Radionuclides

Jaesik Hwang*, Simon Oh, Kwang-soon Choi, and Hong Joo Ahn
Nuclear Chemistry Research Division, Korea Atomic Energy Research Institute,
111 Daedeok-daero-989, Yuseong-gu, Daejeon, KOREA,
*jaesikhwang@kaeri.re.kr

1. Introduction

Dry active waste (DAW) is the most common type of radioactive waste in nuclear power plants. To permanently dispose of these wastes in a repository, the threshold for the activity concentration of ^{241}Am , ^{242}Cm , $^{58, 60}\text{Co}$, ^{137}Cs , ^{55}Fe , ^{94}Nb , $^{59, 63}\text{Ni}$, $^{238, 239, 240}\text{Pu}$, ^{90}Sr , and ^{99}Tc as well as the total alpha, based on the radioactive waste acceptance criteria specified by nuclear safety and security commission (NSSC) notice No. 2015-4 should be met. A chemical analysis of DAW for the determination of radionuclides has technical difficulties owing to its high volume, representative samples, and various decomposition characteristics [1]. Fig. 1 shows composition of low level radioactive wastes from nuclear power plant in 2014 years and picture of dry active wastes.

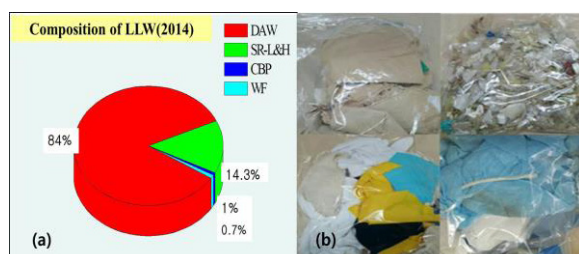


Fig. 1. Composition of low level radioactive waste from nuclear power plants in 2014 (a), and picture of dry active wastes (b).

The volume is therefore minimized through ashing, and the waste is then dissolved using various acids. However, because of the various chemical compositions, the ashing characteristics of DAW are different in terms of the ashing time and temperature. The dissolution of the produced ash is difficult due to the formation of certain oxides while dry ashing at high temperature. Using closed microwave digestion for sample dissolution has advantages, such as a high heating efficiency, a rapid reaction time, avoiding the loss of volatile elements, a low blank level, the prevention of sample contamination, and low reagent consumption, compared to hot plate heating [2].

2. Experimental

We investigated the dissolution properties of dry active wastes for analytical samples arising from nuclear power plants. The analytical samples are prepared using three types of decontamination papers, vinyl gloves, cotton gloves, plastics and linoleums. The weight of each sample is 50 g, and the samples were cut into dimensions of 10 mm x 10 mm. An electric muffle furnace was used to ignite the dry active waste samples. To prevent the formation of soot or a flame during ashing, the temperature was increased step by step from room temperature to 100, 200, 250, 300 °C, and maintained for 4 or 5 hrs at every step. The Table 1 shows the analytical samples for ashing.

Table 1. Sample by each type for ashing

Group		Content
1	Paper	Decontamination paper
2	Vinyl	Doggie bag, Latex gloves
3	Cotten	Polyurethane coated gloves, Cotten gloves, Shoes cover
4	Plastic	High-density polyethylene, Low-density polyethylene, Polypropylene
5	Pad	Linoleum

For evaluating the chemical pretreatment, chemicals including HCl, HF, and HNO₃ (Merck, Germany) were used of analytical grade. After dry ashing, the produced ash was dissolved with a mixed acid of HNO₃-HCl-HF by a closed vessel microwave digestion system. The operation conditions of the closed vessel microwave digestion system are shown in Table 2.

Table 2. Operation condition of closed vessel microwave digestion system

Step	Condition	
	Power (W)	Time (min)
1	290	4
2	400	10
3	300	1
4	250	10

3. Result

After completely ashing, the ashing ratio of the analytical samples were 1.04%, 0.61%, 0.45%, 0.54%, 11.83%, respectively. Most samples were completely dissolved with 10 mL of HNO₃, 4 mL of HCl, and 0.25 mL of HF using closed vessel microwave digestion system. However, linoleum were not dissolved with the same condition because of containing titanium and silicon as major compositions.

4. Conclusion

For the ashing results, the ashing ratio of linoleum were comparably much higher than the other samples. In addition, the dissolution of ash of linoleum were not dissolved. The dissolution of ash was affected by the sample weight of ash. For this reason, ashing using linoleum in the radiochemical analysis should be considered deeply because of the representatives in the samples.

REFERENCES

- [1] J. T. Marumo, "Progress in the characterization of radioactive waste in brazil", Annual meeting of the international network of laboratories for nuclear waste characterization, December 10-12, 2013, Vienna, Austria.
- [2] K-S. Choi, C. H. Lee, H-J Im, H-J Ahn, K. Song, "Sample pretreatment for the determination of gamma emitting nuclides in dry radioactive waste using a dry ashing and high performance microwave digestion system", Journal of Radioanalytical and Nuclear Chemistry, 301, 567-571 (2014).

Conceptual Design for Kori Unit 1 Reactor Vessel and Internal Segmentation Process

Young Hwan Hwang*, Seokju Hwang, Joo-Young Yoon, and Cheon-Woo Kim

KHNP-CRI, 70, 1312-gil, Yuseong-daero, Yuseong-gu, Daejeon 34101, Republic of Korea

*yhhwang7@khnp.co.kr

1. Introduction

The Kori unit 1 is a historic nuclear power plant (NPP) in Korean nuclear history. The experiences, including construction, criticality, continued operation permit, etc., enable the significant advancement in nuclear industry and R&D. As the permanent shut down of the Kori unit 1, the importance of decommissioning technology is increasing. The decommissioning strategy of the Kori unit 1 is a prompt decommissioning. The decommissioning activity will be implemented around 2025, after transient period and removal of spent fuels. One of the most important process in decommissioning activity is the segmentation and dismantling of activated large component. The reactor vessel (RV) and internals (RVI) are relatively highly activated among various systems. Since the direct disposal of RV and RVI is not available, the development of efficient RV and RVI segmentation and waste management plan are essential.

The conceptual design for the Kori unit1 RV and RVI segmentation process will be discussed in this paper. The accurate activation analysis is prerequisite for the successful development of the segmentation process. Also, the project duration and availability of remoted and underwater segmentation technology are considered to minimize radiation exposure during decommissioning.

2. Conceptual Design of RV/RVI Segmentation

The RV and RVI segmentation process consists of activation analysis, cutting plan development and experiment, and package plan development. It is important to understand the activation level of RV and RVI for the development of cutting plan, waste classification, and package plan. Using the cutting and packaging plan data, the mockup test follows. Through the mockup test, the verification of segmentation process and packaging plan will perform.

2.1 Activation analysis

The activation analysis of representative elements in PWR reactor vessel wall is shown in Table 1 [1]. It is generally accepted that the operation history, including nuclear designs and fuel distribution plans, should be considered for the accurate calculation of the neutron flux and activation. The previous result indicated that RVI near the core are classified as intermediate level waste (ILW). Upper and lower parts of RVI and RV are classified as low level waste (LLW) or very low level waste (VLLW) [2].

Table 1. Activation of reactor vessel wall [1]

Nuclide	PWR	
	Half Life (y)	Specific Activity (Bq/g)
Fe-55	2.70	2.18E+09
Co-60	5.27	1.18E+09
Ni-63	100.00	2.11E+08
Mn-54	0.85	3.07E+07
Zn-65	0.67	4.07E+06
Ni-59	8.00E+04	1.59E+06
T-3	12.30	4.81E+05
C-14	5730.00	2.70E+05
Cs-134	2.06	8.88E+04
Eu-154	8.6	3.37E+04
Ba-133	10.4	2.74E+04
Eu-155	4.8	1.33E+04
Cl-36	3.00E+05	5.92E+03
Sr-90	29.00	5.55E+03
Cs-137	30.1	5.55E+03
Mo-93	3500.00	2.85E+03
Nb-94	2.00E+04	2.33E+03
Kr-85	10.50	2.15E+03
Hf-178m	30.0	1.37E+03
Ho-166m	0.0	1.04E+03
Tb-158	150.0	7.70E+02
Ag-108m	130.00	7.40E+02
Ar-39	269.00	7.03E+02
Tc-99	2.10E+05	5.55E+02
Sm-151	93.0	1.41E+02
Pu-239	2.4E+04	9.99E+01
Ca-41	1.03E+06	5.18E+01
Mn-53	3.70E+06	1.70E+01
Pm-145	17.7	1.15E+01
U-233	1.6E+05	5.92E+00
Se-79	6.50E+04	3.70E+00
Kr-81	2.10E+05	5.92E-01
Zr-93	9.50E+05	3.70E-01

2.2 RV segmentation process

The RV is a massive and large component with diameter of 3.8 m, height of 9.7 m, and weight of 187 ton, without head. The RV and RVI are shown in Fig. 2. Since the space near the RV is not sufficient, the thermal cutting method is favored, which does not require heavy component to overcome the mechanical reaction force. In addition, the thermal cutting technology allows the short process period and minimal radiation exposure. The oxy-propane torch and auxiliary ventilation system are suggested in this study to prevent the generation of secondary wastes during segmentation process.

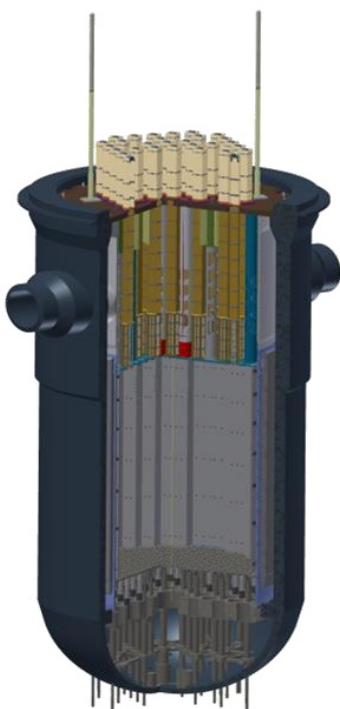


Fig. 1. Image of RV and RVI of Kori unit 1.

2.3 RVI Segmentation process

The RVI is a heavy and complex component, consists of various subsidiary items. Since they are assembled with attachments and have various shapes, it is important that the development of efficient disassembling equipment and various cutting method. In addition, the underwater cutting process is highly recommended to manage the radiation exposure efficiently. The example of RVI underwater cutting RVI is shown in Fig. 2 [3]. It is generally accepted that the segments of core part should be managed carefully due the high activation level. In this study, the

intermediate storage plan of them is also considered for the development of waste management plan.

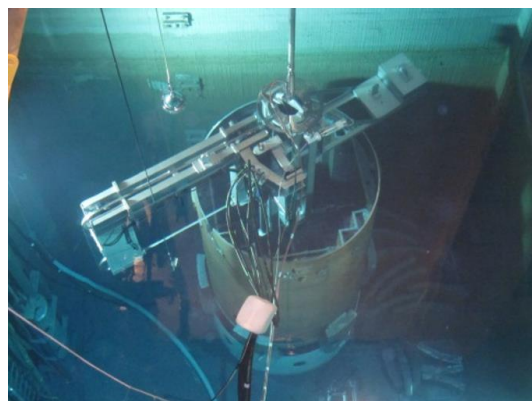


Fig. 2. Example of underwater cutting of RVI [3].

3. Conclusions

The conceptual design for the Kori unit 1 RV and RVI segmentation process is suggested. The segmentation process, waste classification, waste management plan, and packaging plan are systematically studied using activation analysis data. The thermal cutting method is mainly applied to RV segmentation to minimize the process period and radiation exposure to workers. In the case of RVI segmentation, disassembling technology and various underwater cutting processes are studied. In addition, remote segmentation technology and process are strongly favored for the efficient radiation protection.

REFERENCES

- [1] J.C. Evans, E.L. Lepel, R.W. Sanders, C.L. Wilkerson, W. Silker, C.W. Thomas, K.H. Abel, D.R. Robertson, "Long-Lived Activation Products in Reactor Materials", NUREG/CR-3474 (1984).
- [2] H.-K. Kim, et. al, "Characterization of Source Term in Nuclear Power Plant Decommissioning", Final report (2016).
- [3] R. Reid, R. McGrath, "Segmentation and Packaging of the Reactor Vessel Internals at the Jose Cabrera Nuclear Power Plant", EPRI Technical Report (2015).

4분과

방사성폐기물관리 (Poster)



Analysis of Data Field of Radioactive Waste Management for Development of Decommissioning Waste Management Program

Jin Ho Park, Ji Young Song, Hyun Woo Kim, Kang Hyun Song, and Kwang Pyo Kim *

Kyung Hee University, 1732 Deokyoung-daero, Giheung-gu, Yongin-si, Gyeonggi-do, Republic of Korea

*kpkim@khu.ac.kr

1. Introduction

The Korean government determined permanent shutdown of Kori 1 unit in 2017. As a result of decommissioning, large amounts of decommissioning waste will be generated. The proportions of the wastes by waste categorization were reported to be 4.2% for intermediate level waste (ILW), 28.7% for low level waste (LLW), and 67.1% for very low level waste (VLLW) [1]. The half of VLLW is estimated to be clearance waste after the delay and decay process.

Article 9 of Radioactive Waste Control Act in Korea requires collection, investigation, analysis, and management for radioactive waste control. Therefore, applicant or licensee should establish radioactive waste control system.

International Atomic Energy Agency (IAEA) proposed requirement for tracking radioactive waste management [2]. The countries with nuclear power plant (NPP) decommissioning experience, including the USA, Germany etc. constructed management system for decommissioning radioactive wastes and implemented. However, there is no such waste management system for NPP decommissioning in Korea.

Final objective of this study is to develop a waste management system for NPP decommissioning. As a beginning step for the objective, we analyzed data field of radioactive waste management program.

2. Materials and Methods

We reviewed radioactive waste management system to derive requirements for development of a decommissioning waste management program. Table 1 shows some examples of radioactive waste management programs. Each country and IAEA possesses independent management system to manage radioactive waste effectively.

Table 1. Examples of radioactive waste management programs

Country or Organization	Institute	Management program
IAEA	IAEA	WIRKS NEWMDB DIRATA
USA	EPRI	Waste Management Tracking System
	DOE	NMMSS, WWIS, SWITS, MIMS, IWTS, CID
Germany	EWN	ReVK
	GNS	AVK
England	Nirex	BRIMS
Japan	JNC	DES
	JAERI	COMSARD
Korea	KAERI	RAWMIS
	KHNP	DREAMS
	KNF	MES

Among various radioactive waste management systems, we selected WIRKS (Waste Inventory Record Keeping Systems), ReVK (Waste Flow Tracking and Control System), and RAWMIS (Radioactive Waste Management Integration System) for the review. WIRKS in IAEA was used as standard for control in member countries [2]. ReVK was developed by EWN (Energiewerke Nord GmbH), which has NPP decommissioning experience [3]. RAWMIS was developed by KAERI (Korea Atomic Energy Research Institute) and used to manage radioactive wastes [4].

3. Results and Discussions

Data field of radioactive waste management system and a common feature are given Fig. 1. WIRKS provides basic requirement of data field for radioactive waste management. ReVK was developed for the decommissioning wastes, thus data

of dismantling was included under ReVK. RAWMIS included data of clearance waste, such as clearance plan.

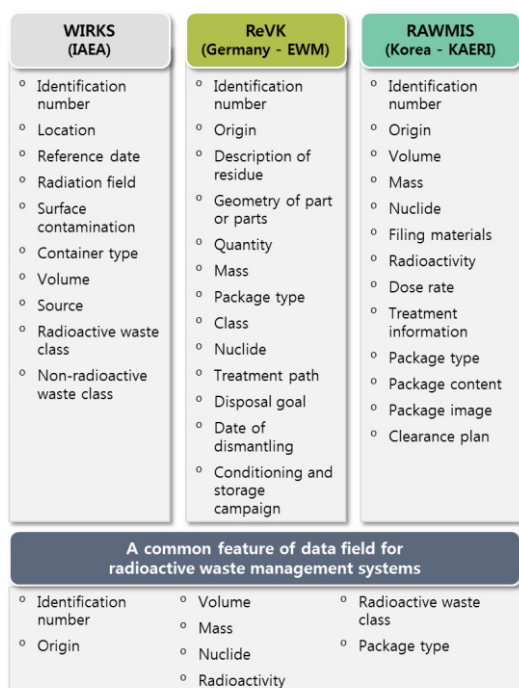


Fig. 1. Data field of radioactive waste management system and a common feature.

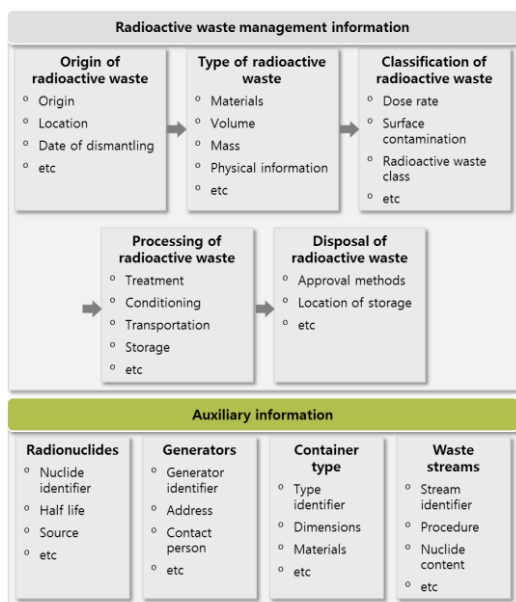


Fig. 2. Basic structure of radioactive waste management system.

The common features of data field for the radioactive waste management systems included identification number, origin, volume, mass, nuclide, radioactivity, radioactive waste class, and package type. Basic structure of radioactive waste

management system based on a common feature is given in Fig. 2. Basic structure of data field consisted of radioactive waste management information and auxiliary information. Radioactive waste management information included origin, type, classification, processing, and disposal of radioactive waste. Auxiliary information included radionuclides, generators, container type, and waste streams.

4. Conclusion

We analyzed data field of radioactive waste management program. Based on this data field requirement, we suggested basic structure of data field for radioactive waste management. The results of this study will be used to develop a waste management system for NPP decommissioning. In addition, it will contribute to the establishment of classification and management system in accordance with the characteristics of decommissioning 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. 20171510300580).

REFERENCES

- [1] KEITI, "Trend of NPP Decommissioning waste management", Korea Environmental Industry Technology Institute, Konetic report 2016-138, 2016.
- [2] IAEA, "Waste Inventory Record Keeping Systems for the Management and Disposal of Radioactive Waste", International Atomic Energy Agency, IAEA-TECDOC-1222, 2001.
- [3] B. Hartmann, M. Hager, "ReVK – a Tool for the Fulfilment of Requirements from National Rules for Tracking and Documentation of Radioactive Residual Material and Radioactive Waste", WM'06 conference, February 26-March 2, 2006, Tucson.
- [4] Y. Lee, H. Cho, J. Shon, K. Kim, K. P. Hong, H. S. Park, "Design of System and Prototype for the Radioactive Waste management", WM'04 conference, February 29-March 4, 2004, Tucson.

Radioactive Waste Treatment for Release of Korea Research Reactor Site

Seung-Geon An^{*}, Sang-bok Lee, Tae-ryong Kim, Jung-soo Park, Yeo-jib Choi, and Seong-jin Cho

Sunkwang T&S, 20F Sunkyung Officetel, Gongwon-ro 3, Guro-gu, Seoul, Korea

^{*}seunggeonan@hotmail.com

1. Introduction

Release project for KRR site was restarted in earnest from 2016 to 2021. Preparation works for release of KRR (Korea Research Reactor) were conducted by Sunkwang T&S from July, 2016 to May, 2017. The KRR site will be used for non-nuclear purpose (green field) after site release.

Although radioactive waste was satisfied the clearance criteria when it generated but now the regulation guide for clearance was revised in 2014[1], and as a result the radioactive waste in KRR didn't meet the clearance criteria any more.

So extra works, such as, taking representative sample, activity analysis, repacking, crushing and cutting for clearance were performed.

2. Preparation Works

2.1 Collecting of Representative Sample

Because the regulator generally want to(requires to) submit the latest analysis data, within two years, the validity of activity analysis data for radioactive waste in KRR site has already been expired. Therefore extra sampling was conducted to secure the representative sample.

2.1.1 Concrete and Soil. To take sample and classify the radioactive waste, Semi-automatic classification working table was manufactured.



Fig. 1. Classification Working Table.

Table 1. Specification of Classification Working Table

Item	Spec.
Max. Intake weight	500 kg/each
Classification table	No. 2
	Size 200 × 100 × 20
	[cm] [L × W × H]
Operation unit	Electric Hoist

After unloading the concrete and soil waste on classification table by forklift, then waste was divided by 10 zones for scan survey. Based on the scan survey result, representative sample was collected intensively at the zone that contaminated more than other zones slightly or highly.

2.1.2 Metal. Punching and drilling were applied to collecting representative sample of metal waste. An Object metal was divided four zones that were surveyed in the same manner as the concrete and soil.

Chips generated from the zone, gathered as a sample of metal. If the thickness of metal is thinner than 10 mm the punching was applied, if not the drilling was applied.



Fig. 2. Punching and Drilling of Metal.

2.2 Gamma analysis

Gamma analysis of representative sample was performed with gamma spectroscopy.[2] The conditions of analysis are as in the following.

Table 2. Conditions of gamma analysis

Conditions	Value
Analysis time	3,600 sec
Sample Geometry	1,000 ml M.B
MDA	Less than 10% of Permissible activity of nuclide*

* Notification of Nuclear Safety and Security Commission 2014-003 table 1

2.3 Repacking

Base on the result of gamma analysis, waste was classified two groups. If the ratio of gamma activity summation of nuclides are less than 0.5, applicable waste was classified as object for clearance and more than 0.5 it was classified as permanent disposal waste.

After classifying the waste, permanent disposal object was repacked to disposal container.



Fig. 3. Repacking to Disposal Container.

Every package was checked its radiation dose, radioactivity, weight, volumetric packing fraction, surface existence of free water, chelate agent, explosive material and so on.

3. Conclusion

Radioactive waste in KRR site was treated and classified from July, 2016 to May, 2017. Collecting representative sample, gamma analysis and repacking were performed and as a result 271,298 kg waste was classified totally. Classified waste has been stored in KRR 2 finally.

REFERENCES

- [1] Jong Soon Song, Dong Min Kim, Sang Heon Lee, "A Study on the Application of Standards for Clearance of Metal Waste Generated During the Decommissioning of NPP by Using the RESRAD-RECYCLE", Journal of Nuclear Fuel Cycle and Waste Technology, Vol.14, No.4, 305-320(2016).
- [2] Il Sik Kang, "Treatment of Radioactive Contaminated Soil and Concrete Wastes Using the Regulatory Clearance", KAERI/TR-3653/2008, 39.

Rainfall Infiltration Analysis for the Near-Surface Radioactive Waste Disposal Facility

Jin-Kyu Choi* and Han-Jeong Na

KEPCO E&C, 269, Hyeoksin-ro, Gimcheon-si, Gyeongsangbuk-do, Republic of Korea

*cjk@kepc-co-enc.com

1. Introduction

The near-surface type of repository is selected to the second phase of LILW disposal facility in Korea. Compared to the silo-type cave disposal facility, which was applied to the first phase of LILW disposal facility, the disposal cover for near-surface disposal facility should be designed to prevent rainfall from infiltrating into the waste package in disposal vaults.

The rainfall infiltration analysis for near-surface disposal facility in Korea is performed using Visual HELP Code (Hydrologic Evaluation of Landfill Performance) which is used to evaluate hydrologic behavior of percolated rainfall through disposal cover.

2. Concept of Disposal Cover Design

The design of disposal cover for near-surface disposal facility is developed based on the internationally proven technology which is applied for various near-surface repositories around the world.

Since the disposal cover consists of several types of layer such as unsaturated soil, water proof barrier and membrane, it prevents rainfall from infiltrating into waste package in disposal vaults. The conceptual section diagram is shown in Fig. 1.

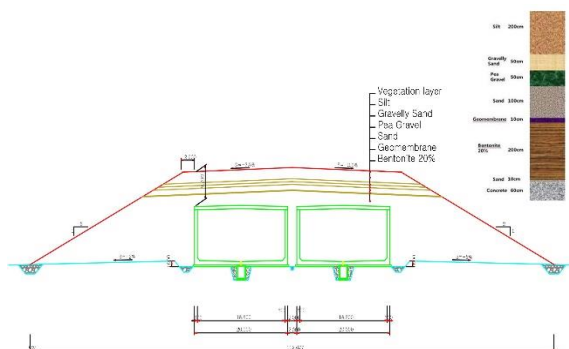


Fig. 1. The conceptual section drawing for near-surface radioactive waste disposal facility.

The performance of disposal cover will be decreased upon the time elapsed, and infiltration rate to waste package will be consequently increased. Since the disposal cover is monitored and repaired, the performance of disposal cover will be completely maintained for 300 years after the closure (institutional period). After institutional period, it is assumed that the disposal cover will be completely degraded and will not be able to prevent rainfall infiltration.

For concrete slab, the performance will be maintained until institutional period. The degradation of concrete slab will start after institutional period.

Table 1. The status of disposal cover and concrete slab

Phase	Period	Disposal cover	Concrete slab
1	Closure ~300 years	Maintain structural integrity	Maintain structural integrity
2	300 years ~ Degradation of concrete	Degraded	On degradation
3	Degradation of concrete ~	Degraded	Degraded

The conceptual model for disposal cover system is classified into Types I and II as follows:

- 1) Type I (Facility closure ~ 300 years after closure) : 3.5% slope

Table 2. The material and function of disposal cover Type I

Layer	Material	Function
1	Top soil (Silt)	Feeding water for plant or grass, maximizing runoff and evapotranspiration and surface erosion control
2	Soil and Gravel	Moving down infiltration water. Surface erosion control. Prevent penetration of small particles to sand layer
3	Gravel	Control the hydraulic head and prevent penetration of roots or underground animals

4	Soil	Drain infiltration water to planned drainage channel. Prevent the increase of hydraulic head.
5	Geomembrane	Prevent infiltration water
6	Bentonite(20% of sand)	Prevent infiltration water as a Hydraulic barrier
7	Soil	Same as layer 4
8	Concrete Slab	Prevent infiltration water

2) Type II (1,185 years after closure ~) : No slope

Type II has only three layers (layer 1, 7, 8 of Type I) and no slope. It is assumed that the layers are degraded after institutional period. The concrete slab is also degraded, and related input data are applied after degradation.

3. Rainfall Infiltration Analysis Results

The monthly average climate data during 1981~2010 in Wolsung site where the facility are located is used for this analysis. The Visual HELP creates the weather data for 100 years (maximum) using weather generator in the code.

The analysis results for disposal Types I and II for specific plan are described in Table 3.

Table 3. The annual average rainfall infiltration analysis results

Model Type	Percolation or Leakage (mm/yr)
I (%)	0.01977 (0.0016)
II (%)	435.98 (34.4)

4. Conclusion

Most of rainfall is blocked with evaporation rate and infiltration rate to waste package in disposal vaults is very low for Type I. By contrast, about 35% of rainfall passes through the disposal cover and concrete slab due to change in material characteristics by degradation. As the rainfall infiltration analysis results, the disposal cover which has the function of hydraulic barrier is strongly required to prevent rainfall from infiltrating into waste package for near-surface radioactive waste

disposal facility. The analysis results can be utilized to the analysis for potential radiological release from the waste package to the environment or input data for the associated drain system in the facility.

REFERENCES

- [1] NCRP/CR-4370, Vol.1, Update of Part 61 Impacts Analysis Methodology, U.S.NRC, 1988.
- [2] U.S.DOE/LLW-73T, Prototype License Application Safety Analysis Report Earth-Mounded Concrete Bunker, Ebasco Service.
- [3] U.S. New York State LLRW siting Commission, Disposal Method Evaluation Procedures, Rogers & Associates Engineering Corporation and Golder Associates, Inc., 1994.
- [4] IAEA, ISAM, the International Programme for Improving Long Term Safety Assessment Methodologies for Near-surface Radioactive Waste Disposal Facilities, 2001.

Cutting Force Test of Cutting Blade Modules for Slitter Design

Young-Hwan Kim*, Yung-Zun Cho, and Young-Soon Lee

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

*Corresponding author: yhkim3@kaeri.re.kr

1. Introduction

Mechanical head-end processing of SF disassembly, extraction of the rods, and the shearing of the extracted rods shall be performed in advance as the head-end process of the pyro electro-reduction process. Also, for oxidation processing of the spent fuel in the head-end process, slitting decladding device for decladding of the cut rods is necessary. For the concept design of the device, after checking the decladding condition of the pellet and the hull and considering the cutting force requirements of the cutting blade, a tool to test simulated rods and cutting force was manufactured. Also, we used the zry-4 tube with AL6061 pellets instead of spent fuel tube with high toxicity, and as the role of the simulated pellet, to prevent from running down, it was punched at both ends and fixed. As the major requirements, 50 kg HM/day throughput and 250 working days (full capacity) per 1 year were assumed, and based on KSFA type (16x16), it was conditioned to process 10 Ton HM/year fuel with 85 % availability. Cutting force test results of cutting blade can be utilized for the head-end decladding process of SF dry process.

2. Methods and Results

2.1 Summary of slitter

The slant slitter is cut as follows. When a cutting rod of the spent nuclear fuel is placed in the module entrance, it is inserted into the center of module by the extrusion pin. A cutting rod of spent nuclear fuel is passed through the module and is simultaneously separated into several pieces by the blades inside the module, and the pieces of the pellets in the fuel rod are recovered.

As in Fig. 1, screw press-in part transfers the force to the screw by the servo motor, and the load cell and punch fixing part are operated at the bottom. When the cutting blade module installed at the bottom has the structure of measuring the cutting force when the rods are penetrated and cut, and standard TM screws and nuts are used. The length of the part of the simulated rods and punch where they go through the cutting module and reach the bottom was set to be about 420 mm or more. Ball bush and guide axis were installed on both ends for the balance and for soft press-in without external interference. The bottom base plate was composed of the hole to fix the support axis, and to settle down the cutting blade module 2-CUT module and 3-CUT module (Fig. 2). Capacity of 100 kg- 1 Ton was selected for load cell, and RS232C is embedded in the controller.

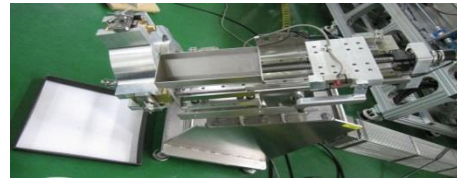


Fig. 1. Spent Fuel slant slitter for slitting test.



Fig. 2. Blade types and simulated fuel rods. (2-CUT module and 3-CUT module)

2.2 Experiment method

As in Fig. 3, the roller gradient of the cutting module was changed to visually observe the slitting degree of Zry-4 tube, and the data measured in the load cell was obtained using RS232 communication. During the slitting using Zry-4, the gap in the cutting blade was measured, and $\varnothing 9.5$ mm was used for Zry-4 rods. Also, the roller gradient used the adjustment plat as in Fig. 4. Also, aluminum pellet were inserted into the blank tube with length 250 mm for the testing to derive the optimal gradient value.

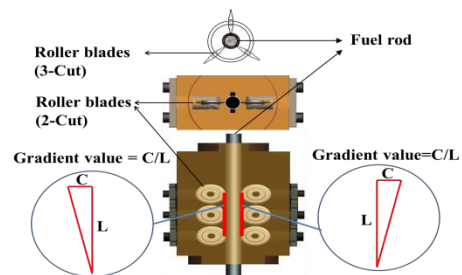


Fig. 3. Rollers gradient of 2-CUT and 3-CUT module.

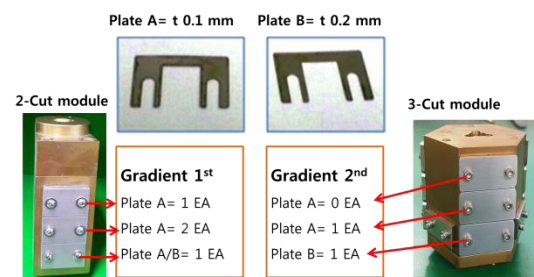


Fig. 4. Gradient variable of modules using the plate.

2.3 Experimental result and consideration

In the slitting decladding device (Fig. 1), cutting modules (Fig. 2) were used to give changes to the stepwise cutting speed, and using RS232 communication, the data measured in the load cell (BS-7220) were shown. Fig. 5 is the shape of the slitting after the experiment using 2-CUT module and 3-CUT module.



Fig. 5. Slitting shapes of 2-CUT and 3-CUT modules.

In Fig. 6, when 2-CUT module is used, speed changes of 5, 7.5, 10, 12.5 mm/s were given to measure the change of the force. As a result of the measurement, as the speed increases, you can see that the force also has the tendency of increasing. The maximum force at 5 mm/s was 181.28 kg_f, the maximum force at 7.5 mm/s was 191.72 kg_f, the maximum force at 10 mm/s was 185.95 kg_f, and the maximum force at 12.5 mm/s was 192.33 kg_f. Therefore, if you search for the force at adequate speed, you can observe that the maximum force at 5 mm/s is 181.28 kg_f, which is lower than other speeds.

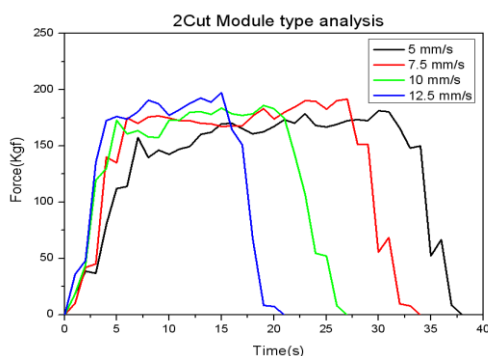


Fig. 6. Forces (kg_f) variations on the 2-CUT module.

In Fig. 7, when 3-CUT module is used, speed changes of 5, 7.5, 10, 12.5 mm/s were given to measure the change of the force. As a result of the measurement, as the speed increases, you can see that the force also has the tendency of increasing. The maximum force at 5 mm/s is 244.03 kg_f, the maximum force at 7.5 mm/s is 293.93 kg_f, the maximum force at 10 mm/s is 281.97 kg_f, and the maximum force at 12.5 mm/s is 393.4 kg_f. Herefore, if you search for the force at adequate speed, you can observe that the maximum force at 10 mm/s is 281.97 kg_f, which is lower than other speeds.

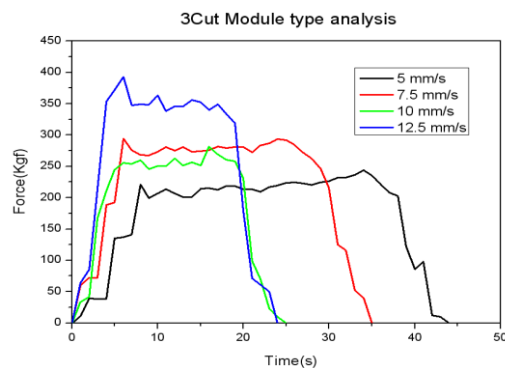


Fig. 7. Forces (kg) variations on the 3-CUT module.

3. Summary

For the concept design of the device, a tool was made to test the simulated fuel rods and cutting force and the cutting force was measured. When 2-CUT and 3-CUT modules were used, the maximum force in 2-CUT at 12.5 mm/s speed change was 197.5 kg_f and the maximum force at 3-CUT was 363.2 kg_f. The change of force in 2-CUT rapidly increases from about 1 second, and you can see that there are increase and decrease of the force change from about 5 seconds to 18 seconds, and it was rapidly decreased and the cut was made. The force change in 3-CUT has higher force at about 5 seconds later than 2-CUT at the speed of 12.5 mm/s, and you can see that it has the same tendency afterwards. If you search for the force at adequate speed from this cutting force test, 2-CUT module requires less slitting force than 3-CUT module, and the cutting time for 250 mm at 12.5 mm/s was 21 seconds, which can cut 4 m fuel rod in 5 minutes. But, there are cases of not completely slitting with 2-CUT module, so it is necessary to supplement this in the future through experiments.

Acknowledgements

This paper is the study carried out with the 2013 funding of the Ministry of Education, Science and Technology and the support from National Research Foundation of Korea (Nuclear Energy Technology Development Project, No. 2012M2A8A5025 696).

REFERENCES

- [1] J. Alagy, "Designing a cyclohexane oxidation reactor", Ind. Eng. Chem., Process Des., 13(4) 1974.
- [2] C.T. Kring and S.L. Schrock "Remote maintenance lessons learned on prototypical reprocessing equipment," Proceedings of 38th Conference on Remote Systems Technology, pp.23-27, 1990.
- [3] B. D. Cul, R. H. Hunt and B. Spencer, "Advanced head-end processing of spent fuel," 2004 American Nuclear Society Winter Meeting, Washington DC, Nov., 16, 2004.

Immobilization of Radioactive Waste Generated From Primary System Decontamination Using Geopolymer Waste Form

Jinmo Ahn^{1*}, Won-Seok Kim¹, Seon-Byeong Kim², Wangkyu Choi², and Wooyong Um¹

¹Division of Advanced Nuclear Engineering, POSTECH, 77, Cheongam-ro, Nam-gu, Pohang, Korea

²Decontamination and Decommissioning Research Division, Korea Atomic Energy Research Institute (KAERI), 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea
*jinmore@postech.ac.kr

1. Introduction

The decontamination of nuclear power plant (NPP) is a great concern as old NPPs are decommissioned. Among several decontamination processes for NPP's primary system, hydrazine based reductive metal ion decontamination (HYBRID) process with sulfuric acid is newly developed as an attractive method for achieving acceptable decontamination factor and lowering the damage of base metals. Despite of these advantages, high sulfate contents of HYBRID process lead to serious deterioration of general cement waste form due to the potential formation of ettringite [1].

Therefore, the new solidification method is needed to develop and evaluate the geopolymer waste form with high mechanical performance and even high sulfate resistance.

2. Geopolymer synthesis and results

This study used industrial byproducts (red mud and fly ash) as geopolymer precursors. All paste samples were cast in polypropylene vials (Φ23 x 46mm) with sealing material. The sealed samples were placed in the oven at 20°C for 7 days.

2.1 Geopolymer precursors

Industrial wastes (red mud and fly ash) were obtained from Daejoo KC and Sungshin industry, respectively. The chemical analysis of red mud and fly ash using X-ray fluorescence (XRF) is shown in Table 1. As water content of red mud is about 20%, red mud sample was dried in the 110 °C oven for 24 hr before use.

Table 1. Chemical compositions of red mud and fly ash (wt%)

	SiO ₂	Al ₂ O ₃	CaO	Fe ₂ O ₃	Na ₂ O	LOI*
Red mud	20.0	25.0	2.35	29.3	14.5	3.01
Fly ash	42.1	18.6	23.6	4.20	1.66	4.60

*LOI: loss of ignition

2.2 Identification of simulated HYBRID sludge waste

The properties of simulated HYBRID sludge waste were identified by XRF and XRD (Table 2 and Fig. 1). Sludge waste contains high amount of SO₃ (13.8%) which corresponds to XRD patterns of barite peak (PDF#24-1035).

Table 2. Chemical properties (%) of simulated HYBRID sludge waste

SiO ₂	BaO	SO ₃	Na ₂ O	Al ₂ O ₃	MnO	LOI*
45.8	28.9	13.8	2.58	2.04	2.04	4.49

*LOI: loss of ignition

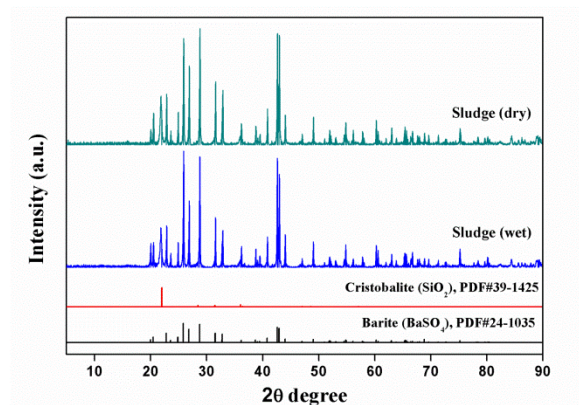


Fig. 1. XRD patterns of simulated HYBRID sludge waste.

2.3 Geopolymer synthesis

Geopolymer waste forms were successively formulated with a desired ratio of Si/Al (0.8-1.3) and H₂O/Al (13-17) using calculated matrix [2]. Geopolymer waste forms were successfully synthesized using red mud and fly ash to meet the waste form acceptance criteria (>3.45 MPa).

3. Conclusions

The immobilization of simulated HYBRID sludge waste using geopolymer waste form prepared with red mud and fly ash was tested and investigated. The results revealed that the mechanical stability of the geopolymer waste forms was better than that of cement waste forms. These results suggest the potential use of geopolymer based on industrial byproducts for radioactive waste solidification.

ACKNOWLEDGEMENT

This work was supported by the Korea Atomic Energy Research Institute (KAERI) granted financial resource (No. 1711049170).

REFERENCES

- [1] M.S.H. Khan, O. Kayali, and U. Troitzsch, "Effect of NaOH activation on sulphate resistance of GGBFS and binary blend pastes", *Cement and Concrete Composites*, 81, 49-58 (2017).
- [2] RP Williams and A Van Riessen, "Determination of the reactive component of fly ashes for geopolymer production using XRF and XRD", *Fuel*, 89(12), 3683-3692, (2010).

The Improvement for Operational Convenience of Solid Waste Management System

Kim Young-hwan*, Oh Chang-kyun, and Shin Mu-gap

KEPCO E&C, 269, Hyeoksin-ro, Gimcheon-si, Gyeongsangbuk-do, Republic of Korea

*dudhwan@kepc0-enc.com

1. Introduction

The solid waste management system in the nuclear power plant provides the means to collect, segregate, store, process, sample and monitor the radioactive solid waste. Low radioactive level of spent resin and spent carbon in types of solid radioactive waste are processed by the polymer solidification system after short term storage in Low Activity Spent Resin Tank.

Unlike spent resin, which is easily solidified with polymer, the spent carbon has difficulty to be solidified due to the small bid. Then, there is no choice but to pack it in the container such as High Integrity Container (HIC) and disposed. When the spent resin and the spent carbon are mixed, the polymer solidification has inconveniency to solidify. So HIC usage will be increased. In order to resolve this problem, bypass discharge piping for preventing mixing is installed.

2. Body

2.1 Design Concept of Solid Waste Management System

Wet solid radioactive waste, such as R/O concentrate and spent resin, is solidified by polymer and packaged in a waste drum in a form suitable for disposal after drying and dewatering. And all solid waste is packaged in accordance with Nuclear Safety and Security Commission Notice 2014-50.

2.2 Operation of Low Activity Spent Resin Tank

Low radioactive level of spent resin and spent carbon are processed after short term storage in Low Activity Spent Resin Tank. Spent resin discharged from the Low Activity Spent Resin Tank is disposed of after the polymer solidification process, and the spent carbon is packaged in HIC and disposed. When spent resin and spent carbon are mixed, the polymer solidification treatment is difficult and inconveniency. For example, when the spent carbon is introduced into the tank, the spent resin stored in the tank is discharged for treatment and then the tank and piping are washed using demineralized water.

2.3 Installation of bypass discharge piping

Low radioactive level of Spent resin and spent carbon in type of solid radioactive waste is designed to be discharged via a Low Activity Spent Resin Tank. In order to reduce operating cost, it is necessary to install and use bypass discharge piping so that the spent carbon can be bypassed without passing through Low Activity Spent Resin Tank when it is discharged. The spent carbon has a very low radioactive level compared to spent resin because it plays a role of filtering simply. So, it does not require short-term storage in a Low Activity Spent Resin Tank. Therefore, it is discharged through bypass pipe to improve operating convenience. In addition, if the radioactive level is high after measuring the surface dose at the time of discharging

spent carbon, the piping path is provided so that spent carbon flows through the Low Activity Spent Resin Tank by the valve operation to protect the operator.

2.4 Piping Stress Analysis

Stress analysis was carried out for various loads acting on the new piping system from the existing spent carbon transfer piping to the connector, and the integrity of the piping system was evaluated by applying KEPIC MND to the Non-Safety Related, Seismic Category II piping. Stress analysis was performed using the following equation of KEPIC MND. The results of the stress analysis are shown in Table 1.

2.4.1 Sustained Loads.

$$S_{SL} = B_1 \frac{P D_o}{2 t_n} + B_2 \frac{M_A}{Z} \leq 1.5 \times S_h \quad (1)$$

2.4.2 Occasional Load.

$$S_{OL} = B_1 \frac{P_{max} \times D_o}{2 t_n} + B_2 \frac{(M_A + M_B)}{Z} \leq 3.0 \times S_h \quad (2)$$

2.4.3 Sum of Sustained Loads and Thermal Expansion Loads.

$$S_{TE} = \frac{P \times D_o}{4 t_n} + 0.75 i \frac{M_A}{Z} + i \frac{M_C}{Z} \leq (S_h + S_A) \quad (3)$$

Table 1. Result of Stress Analysis

Code Equation	Node Point	Maximum Stress (psi)	Code Allowable Stress (psi)	Maximum Stress ratio to Allowable stress (%)
1	890	3,620	19,380	18.7
2	780	27,300	52,680	51.8
3	140	39,800	45,450	87.6

As shown above, the maximum stress value per code equations was compared with the allowable stress of code, all the analysis results have been satisfied the code allowable stress.

3. Conclusion

Bypass discharge pipe has been newly installed to enable bypass discharge without passing through the Low Activity Spent Resin tank when spent carbon is discharged. Therefore, this modification improves operating convenience.

REFERENCES

- [1] Oh Chang-kyun, "Stress analysis of bypass discharge piping in Low Activity Spent Resin Tank, KEPIC E&C (2017).
- [2] Regulation on Packaging and Transportation of Radioactive Material, ROK NSSC Notice, 2014-50.
- [3] Shin-Wolsong Nuclear Power Plant Units 1&2 Final Safety Analysis Report.

Review on Regulations and Practices of Periodic Safety Review for LILW Repositories

Jinyong Park^{1*}, Jaehak Cheong², Eunjin Seo¹, Hojin Lee¹, and Sangmyeon Ahn¹

¹KINS, 62 Gwahak-ro, Yuseong-gu, Daejeon, Korea

²Kyung Hee Univ., 1732 Deogyong-daero, Giheung-gu, Yongin-si, Gyeonggi-do, Korea

*jinyong@kins.re.kr

1. Introduction

Technical review for safety of disposal facilities have to be undertaken prior to construction and prior to operation as done for the LILW disposal facility in Korea. In addition periodic reviews have to be undertaken during the operation of the facility and following closure, up to termination of the facility license. The purpose of this periodic safety review(PSR) is to periodically evaluate and enhance their safety based on the operational experience acquired, recent regulatory requirement and new technological developments in the field of radioactive waste management[1].

According to Article 5(Renewal of Safety Assessment) of NSSC Notice No. 2016-26, on the basis of the experience gained through the process of managing and controlling disposal facilities and the results of safety assessment, the operator shall frequently reexamine the conditions regarding the safety of disposal facilities and supplement them, if necessary[2]. However, unlike the regulations for NPPs, scope and contents of the PSR for LILW disposal facilities are not clearly stipulated in the Nuclear Safety Act(NSA) and its subordinate statute.

This paper presents the investigated information of the PSR in terms of regulations, experiences and practices from some countries with advanced disposal programs in order to suggest scope and contents of the PSR for LILW disposal facilities in Korea based on the international experience and practices.

2. Regulations and Practices

2.1 Investigation on regulations of PSA

An investigation has been conducted regarding regulatory requirements of the PSR for disposal facilities in terms of its purposes, period, and scope and contents in Finland, Sweden and France as well as WENRA(Western European Nuclear Regulator's Association).

WENRA: Working Group on Waste and Decommissioning (WGWD) has worked on safety reference levels for disposal facilities for radioactive waste, and published the report which includes an issue related to the PSR[3]. It concluded that the licensee shall carry out at regular intervals a review of the operational and post-closure safety of the facility to confirm compliance with licensing requirements. The frequency of the review shall be established by the national legal and regulatory framework.

Finland: The licensee shall carry out a periodic safety review for the disposal facility at least once in every 15 years, and the PSR shall include assessments of the disposal facility's safety status and the long-term safety of the disposal as well as potential development targets in order to maintain and enhance safety[4]. The PSR document should include the equivalent information needed to be reviewed for a license application or a license amendment, and updated as appropriate.

Sweden: At least once every ten years, a new and integrated analysis and assessment of the safety of a facility shall be performed. Since 2010, the requirement regarding the PSR is stated in the Act on Nuclear Activities. The regulations for the PSA stipulated in the Regulatory Code, SSMFS 2008:1, are applicable both nuclear power plants and disposal facilities.

France: According to TSN Act(Concerning Transparency and Security in the Nuclear Field) established in 2006, the licensee of a basic nuclear installation including disposal facilities has to carry out PSRs of its installation by taking account of the best international practices (i.e. every 10 years).

2.2 Investigation on practices of PSA

An investigation has been conducted regarding practices of the PSR for disposal facilities in Finland, Sweden, France and Spain. Table 1 shows the summary of regulations and practices of the PSR for LILW disposal facilities from different countries.

Finland: The Olkiluoto LILW disposal facility was taken into operation in 1992 and consequently its safety assessment was submitted for review in 2007. The first stage of the Loviisa LILW disposal facility, the LLW disposal tunnel, was taken into operation in 1998. The construction of the second stage of the facility, the ILW disposal cavern, was completed in 2007. The operator of the Loviisa LILW disposal facility submitted its PSR of the facility to the regulator in 2013.

Sweden: The SFR disposal facility has been in operation since 1988. An updated long term safety assessment of the SFR repository was submitted in 2008. As a result of the review by the regulator, a more detailed assessment of SFR was submitted in 2014, especially addressing the long term performance of the rock vault for intermediate level waste[5].

France: The PSR is carried out every ten years. The latest safety assessment of the La Manche repository was carried out by ASN in December 2009. ASN notified its conclusions concerning the files in a letter sent out in 2010.

Table 1. Practices of PSR for LILW repositories

	Operat-ion	Regulation	Practi-ce
SFR (Sweden)	1988	● Act on Nuclear Activities(1984:3) ● SSMFS 2008:1 - every 10 years	2009 (Add. info in '10 & '14)
Loviisa (Finland)	1998	● YVL A.1 (Reg. control of the use of nuclear energy) - every 15 years	2013
Olkiluoto (Finland)	1992		2007
La Manche (France)	1969 ~1994 (closed)	● TSN Act - every 10 years	2009
El Cabril (Spain)	1992	● Not identified in law - every 10 years	2003

3. Contents for PSR

After reviewing the regulations and experience from some countries with advanced disposal programs, and current regulation of the PSR for NPPs in Korea, several suggestions are possible to propose regulations of the PSR for LILW disposal facilities.

The PSR in principle can include review of facility design; SSCs important to safety of facility and their

aging; radiological and non-radiological impact due to facility operation and their assessments including normal and abnormal conditions due to accident; use of experience during operation and research findings; operational procedures established; organization, management framework and safety culture; emergency planning; waste acceptance criteria; inventory of radioactive materials disposed; and monitoring results etc.

4. Conclusions

The PSR for LILW disposal facilities has been carried out at various periods in countries with advanced disposal programs, and is usually performed every 10 to 15 years. The PSRs are international practices for periodically evaluating and enhancing safety of facilities based on the operational experience gained and new technological developments, but the scope and contents of the PSR are variously defined and implemented in each country by law. In Korea, the scope and contents of the PSR for LILW disposal facilities should be established by taking into account the inherent properties of disposal facilities, e.g. passive safety function, unlike the properties of common NPPs.

ACKNOWLEDGMENTS

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. 1703005)

REFERENCES

- [1] IAEA Safety Standards Series No. SSR-5, Disposal of Radioactive Waste, IAEA, 2011.
- [2] NSSC Notice No. 2016-26, Technical Standards for the Operation, etc. of LILW Disposal Facilities, NSSC, 2016.
- [3] WENRA WGWD Report, Radioactive Waste Disposal Facilities Safety Reference Levels, WENRA, 2014.
- [4] STUK, 5th Finnish National Report under IAEA Joint Convention, STUK, 2014.
- [5] Ministry of the Environment, 5th Swedish National Report under IAEA Joint Convention, 2014.

Study on Mechanical Strength of External Wall Coating Material for Cold Crucible Induction Melter

Young Hwan Hwang*, Seokju Hwang, Mi-Hyun Lee, and Cheon-Woo Kim

KHNP-CRI, 70, 1312-gil, Yuseong-daero, Yuseong-gu, Daejeon 34101, Republic of Korea

*yhhwang7@khnp.co.kr

1. Introduction

The cold crucible induction melter (CCIM) of HanUl Vitrification Facility (UVF) consists of 48 sectors with mica sheet spacers. The internal water based cooling system maintains the temperature of external wall to the designed temperature, while the temperature of melted glass is over 1,000 °C. Since the induced current is the primary energy for the glass melting, the each sector is electrical separated using mica spacer for the efficient operation. The mica spacer also allows the sufficient mechanical flexibility of the melter. The external wall coating material of melter offers the solid maintenance of the physical structure of the melter and obstructs the emission of internal heat energy transfer, as shown in Fig. 1. The hybrid structure of glass fabric and heat resistance type chemical is favored for the external coating materials to achieve both good mechanical strength and high temperature resistance.



Fig. 1. Photo of external wall coating material of melter.

2. Experiment

2.1 General information

The glass fabric, mainly consists of E-glass, and high thermal resistance chemical, epoxy and silicon, are used for the fabrication of composite structure. The fabrication process and basic thermal property of the fabricated composite films are found in previous study [1]. Since the primary object of the external coating materials is a solid maintenance of physical structure of sectors, the construction of composite structure is important.

In this study, the mechanical property of the external coating materials is studied.

2.2 Tensile strength analysis

The tensile strength of the glass fabric and chemical composite is analyzed using INSTRON 5969. The measurements, using ASTM D 3039-14, were carried out at the room temperature, 95, 110, 150, 200, and 250 °C to understand the effect of temperature with 5 mm/min. The Fig. 2 shows the measurement of tensile strength using INSTRON 5969. The tensile strain analysis was repeated 3 times and recorded at the measurement temperature.

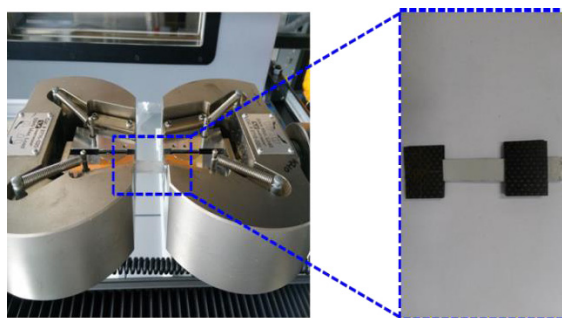


Fig. 2. Measurement of tensile stress using INSTRON 5969.

3. Discussion

3.1 Tensile strength of glass fabric

The tensile strength of glass fabric with respect to temperature is shown in Fig. 3. As the temperature increases, it is observed that the tensile strain slightly decreases. The average tensile strain at the 110 °C, melter operation temperature, is 0.63 kN.

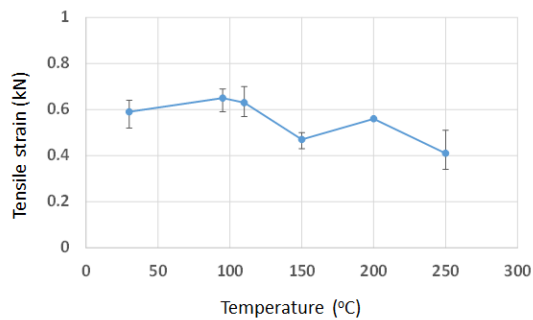


Fig. 3. Tensile strain of glass fabric.

3.2 Tensile strength of glass fabric/epoxy composite

The tensile strength of glass fabric and epoxy composite with respect to temperature is shown in Fig. 4. As the temperature increases, it is observed that the tensile strain slightly decreases, similar to the glass fabric. The average tensile strain at the 110 °C, melter operation temperature, is 0.5 kN.

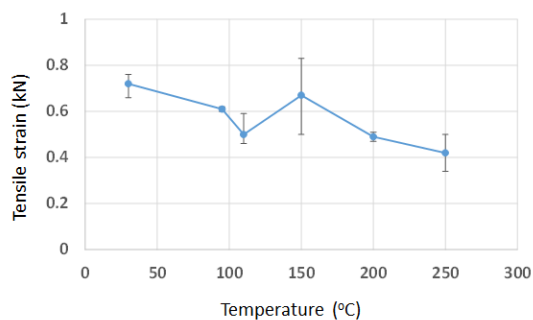


Fig. 4. Tensile strain of glass fabric and epoxy composite.

3.3 Tensile strength of glass fabric/silicon composite

The tensile strength of glass fabric with respect to temperature is shown in Fig. 5. As the temperature increases, it is observed that the tensile strain slightly decreases, similar to the glass fabric. The average tensile strain at the 110 °C, melter operation temperature, is 0.62 kN. According to the

measurement data, it is concluded that the fabric and epoxy or silicon composites exhibit sufficient tensile strain to maintain melter structure. In addition, the glass fabric and silicon composite, which is superior to epoxy based composite in terms of thermal resistance, shows the highest tensile strain at the high temperature.

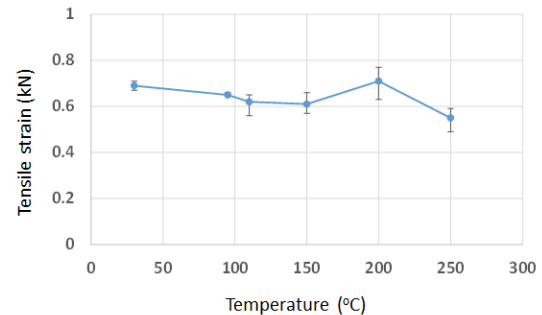


Fig. 5. Tensile strain of glass fabric and silicon composite.

4. Conclusions

The composite coating materials, hybrid materials using glass fabric and chemical, were studied for the employment to external coating materials for CCIM. The mechanical property, in terms of tensile strength, analysis indicated that the composite exhibits sufficient mechanical strength and enables the solid maintenance of the sectors. As a result, it seems that the glass fabric and silicon composite has a high-potential for the application of external coating material of CCIM, in terms of thermal and mechanical properties.

REFERENCES

- [1] Young Hwan Hwang, Seok-Ju Hwang, Mi-Hyun Lee, Cheon-Woo Kim, "Thermogravimetric Analysis on External Wall Coating Material of Cold Crucible Induction Melter", Proc. of the KRS 2017 Spring Conference, 15(1), May 24-26, 2017, Busan (2017).

Trapping Characteristics of Cesium Iodide by Aluminosilicate Filters

Jae Hwan Yang*, Joo Young Yoon, Ju Ho Lee, Seok-Min Hong, and Yung-Zun Cho

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

*yjh98@kaeri.re.kr

1. Introduction

Radioactive cesium (Cs-137), generated by nuclear fission reaction, is present as several chemical forms in spent nuclear fuels. During the pyro head-end process, spent fuels are heat-treated and cesium is released as off-gas [1]. Most species of Cs evaporated from spent fuel are known as metallic Cs (Cs(g)), cesium oxide (Cs₂O(g)), cesium hydroxide (CsOH(g)), and cesium iodide (CsI(g)) [2]. Because of negative health effects of Cs-137, gaseous Cs vaporized from spent fuel must be captured using proper sorbents. At Korea Atomic Energy Research Institute (KAERI), trapping of Cs has been studied by aluminosilicate-based filters, and it was validated that CsOH(g) could be effectively captured by Silica-Alumina (SA) filter [3]. In this contribution, trapping behavior of CsI(g) is investigated.

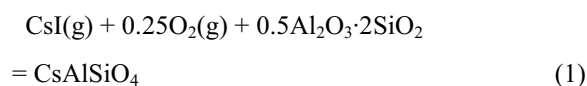
2. Experimental Section

Trapping experiments were performed using an electric tube furnace with multi heating zones. The SA filters were stacked at trapping zone. As a source of CsI(g), cesium iodide (CsI, 99.999%, Alfa Aesar) was purchased and put in an alumina boat crucible. The crucible was then placed at volatilization zone. The SA filters were thermally elevated at 1,000 °C prior to vaporization of CsI. Then, CsI was heat-treated at 1,000 °C to generate CsI vapors. As carrier gases, air, argon, or 4%H₂/Ar were provided with a rate of 0.8 liter/min. After the trapping experiments,

trapping efficiency was obtained by measuring increased mass of the SA filters. It should be mentioned that the increased mass was assumed to that of Cs and O when air was used as a carrier gas, for oxygen in the air was involved in the trapping reaction. Crystalline structure and morphology were examined by X-ray diffraction (XRD) method and scanning electron microscopy (SEM), respectively.

3. Results and discussion

Fig. 1 shows the increased mass in the SA filter after trapping experiments at different carrier gases. As indicated in the graph, CsI(g) is not effectively captured in the SA filter when 4%H₂/Ar was used as a carrier gas. Trapping efficiency of CsI calculated from the increased mass was only 19.5%. On the contrary, it was demonstrated that CsI(g) was effectively trapped in the SA filters when carrier gas was air instead of 4%H₂/Ar. The calculation of trapping efficiency showed that 86.8% of Cs in the volatilized CsI(g) was removed by the SA filters. The significant difference of trapping efficiency is related to the oxygen contained in the carrier gas. As presented in Fig. 2, CsI(g) reacts with the SA filter to form cesium aluminum silicate (CsAlSiO₄). A plausible equation of trapping reaction is estimated as follows:



Therefore, oxygen is required to form the reaction product. When air was used as a carrier gas, oxygen could be readily provided from the carrier gas, resulting in the high trapping efficiency. However, oxygen was not present in 4% H_2 /Ar gas, which was a reason of the low trapping efficiency.

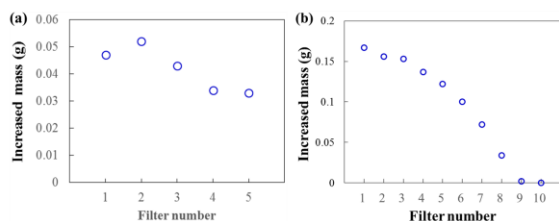


Fig. 1. Trapping of CsI(g) by the SA filters in (a) 4% H_2 /Ar and (b) air gas.

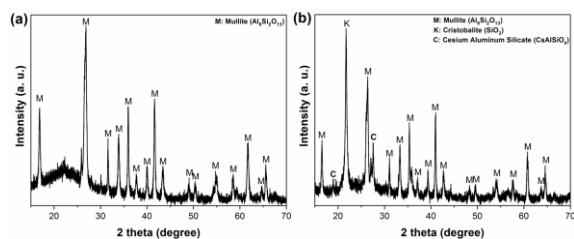


Fig. 2. XRD patterns of the SA filter (a) before CsI trapping and (b) after CsI trapping.

4. Conclusion

Trapping experiments of CsI(g) were performed using the SA filter in different carrier gases. In contrast to CsOH(g), trapping efficiency of CsI was very poor in 4% H_2 /Ar gas. However, CsI could be effectively captured when air was used as a carrier gas, which was attributed to the presence of oxygen that is engaged in the trapping reaction. It is expected that the CsI trapping efficiency in 4% H_2 /Ar could be improved if oxygen sources are added in the SA filter.

REFERENCES

- [1] H.S. Lee, et al., “Pyroprocessing technology development at KAERI”, Nucl. Eng. Technol., 43, 317-328 (2011).

- [2] D.K. Bose, et al., “Some thermochemical studies cesium urinate, molybdate and chromate”, J. Nucl. Mater., 130, 122-125 (1985).
- [3] J.H. Yang, et al., “A kaolinite-based filter to capture gaseous cesium compounds in off-gas released during the pyroprocessing head-end process”, Ann. Nucl. Energy, 103, 29-35 (2017).

Semi-Quantitative Analysis of Electron Probe Micro-Analyzer for Zr-2.5Nb Pressure Tube Material

Y. S. Choo*, Y. J. Kim, B. O. Yoo, S. J. Baik, Y. G. Jin, K.S. Heo, H. M. Kim, and S. B. Ahn
Irradiated Materials Examination Facility, Korea Atomic Energy Research Institute, 989-111 Daedeok-daero,
Yuseong-gu, Daejeon, Republic of Korea

*Corresponding author: yschoo@kaeri.re.kr

1. Introduction

The electron probe micro-analyzer (EPMA) is widely applied to analyze the chemical compositions of unknown materials, especially for irradiated nuclear fuels [1].

To know materials of unknown specimens for PTs, qualitative analyses, line analyses, mappings, and quantitative analyses are carried out without the standard specimen with the EPMA machine through calculating the K_{raw}(%).

When quantitative analyses is done without the standard specimen, semi-quantitative analyses are alternatively performed by adapting the mathematical method of probability and statistics designed and supplied by the JEOL.

In this paper, the adapted probability and statistics supplied by the JEOL is not only investigated, but also unknown specimens for PTs Zr-2.5Nb materials are analyzed.

2. Experimental & Results

2.1 Specimen

Four (4) specimens as shown in Fig. 1 are used to investigate the adapted probability and statistics for the semi-quantitative analysis.

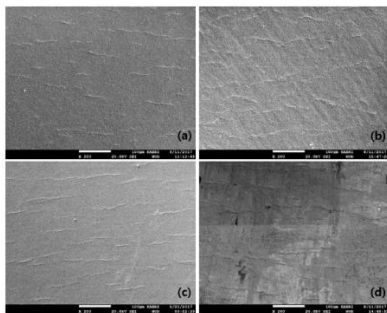


Fig. 1. The specimens of unknown for PTs material, (a) H82, (b) H122, (c) D82, and (d) D122.

2.2 Electron probe micro-analyzer (EPMA)[2]

EPMA was carried out using a state of the art shielded JEOL JXA-8230 model specially shielded and modified to permit the analysis of irradiated nuclear fuels shown in Fig. 2. This equipment has 4 WDXs (Wave Dispersive Spectrometers) to analyze elements from Boron to Uranium, and an additional function of SEM (Scanning Electron Microscope).



Fig. 2. The appearance of the EPMA at hot lab. in IMEF.

2.3 K_{raw}(%)

The K_{raw}(%) is calculated by using following equation (1).

$$K - raw(\%) = \frac{X-ray\ Intensity_{UNK_{net}}}{X-ray\ Intensity_{STD_{net}}} \times 100 \quad (1)$$

,where

X-ray Intensity : counts per beam current in cps/uA,

net : net count in cps,

UNK : unknown specimen,

STD : standard specimen.

But, the standard specimen is not used so the X-ray intensity of standard specimen is also not known. Then the semi-quantitative analysis is done by the machine, i.e., EPMA.

2.4 Semi-qualitative analysis

From the result of qualitative analysis, the semi-qualitative analyses were performed to know the contents (wt.%) of an unknown specimen under the following conditions, and the test results are shown in Table 1.

- HT (kV) & beam current (A) : 20.0/1.0E-08
- CH-3(Zr, PETH/Cal_STD)
- CH-3(Nb, PETH/Cal_STD)

2.5 Coefficients Semi-qualitative analysis for crystals

The coefficients of CH-3(PETH) crystal for Zr (z=40) and Nb (z=41) elements are suggested by the JEOL, and those are calculated by an equation (2) and summarized in Table 2.

$$\ln(I) = a + b \times Z + c \times Z^2 + d \times Z^3 \quad (2)$$

, where I = cps/100pA, Z= ln(z), z = atomic number

Table 1. The test results from the The EPMA of H122 (unknown) specimen by semi-quantitative analysis for Zr and Nb elements

Element	Zr			Nb		
Frequency	1st	2nd	3rd	1st	2nd	3rd
Peak(mm)	194.586	194.559	194.564	183.458	183.509	183.508
Net(cps)	10895.1	10675.5	10767.9	175.6	199.4	183.5
Bg-	56.8	65.6	63.2	26.6	30.2	31
Bg+	37.2	41.8	30.8	46.6	44.4	47.6
S.D.(%)	0.3	0.31	0.31	2.84	2.63	2.79
D.L.(ppm)	294	314	294	406	409	420
K-raw(%)	104.612	102.299	103.133	2.637	2.987	2.751
Curr.(A) :	2E-08	2E-08	2E-08	1.04E-08	1.04E-08	1.04E-08
Intensity (cps/uA)	5.45E+05	5.33E+05	5.37E+05	1.69E+04	1.92E+04	1.76E+04
Remarks	0104QNT			0105QNT		

Table 2. The coefficients of crystals CH-3(PETH) & CH-3(PETH) for Zr and Nb elements estimated/suggested by the JEOL and calculated results

Element	Zr	Nb	Remarks
Channel	CH-3	CH-3	
Crystal Name/Orbital	PETH/La	PETH/La	
z	40	41	Atomic No.
a	-1.97409	-1.97409	
b	-53.0312	-53.0312	
c	27.82146	27.82146	
d	-3.52669	-3.52669	
ln(Z)	3.688879	3.713572	
ln(I)	3.958137	4.155133	
cps/100pA	52.35967	63.76046	
CA(cps/uA)	523596.7	637604.6	Calculated
ME(cps/uA)	497069.8	605278.0	Measured
Difference	26528.6	32326.6	= CA-ME

2.6 Probability and statistics

If the population is known to normal as shown in Fig. 3, the sampling distribution of $\bar{X}$ will follow a normal distribution exactly, no matter how small the size of the samples [3]. So the confidence interval for μ is calculated by an equation (3).

$$\bar{x} - z_{\alpha/2} \frac{\sigma}{\sqrt{n}} < \mu < \bar{x} + z_{\alpha/2} \frac{\sigma}{\sqrt{n}} \quad (3)$$

, where

$$Z = \frac{\bar{X} - \mu}{\sigma/\sqrt{n}}, \quad P(-z_{\alpha/2} < Z < z_{\alpha/2}) = 1 - \alpha$$

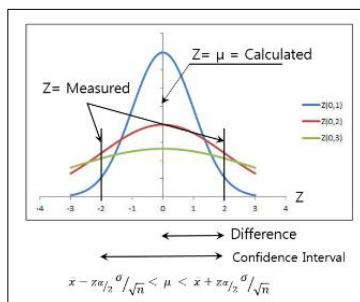


Fig. 3. The meaning of confidence Interval in normal distribution.

2.7 Calculation by adapting the probability and statistics

According to clause 2.6, confidence interval and probability of it were calculated by equation (4) and the whole results are shown in Table 3.

$$CI(\text{Confidence Interval}) = \left(z_{\text{value}} - \frac{\sigma_{UNK}}{100} \right),$$

$$PCI(\text{Prob. of it}) = P(Z < CI) - P(Z > -CI), \quad (4)$$

$$K - \text{raw}(\%) = K - \text{raw}(\%) - 1 \times PCI$$

2.8 Summary and discussion

The K-raw(%) values are almost same as shown in Table 1 and Table 3. As explained in clause 2.3, the K-raw(%) in Table 1 is calculated by program developed by JEOL considering the confidence interval with difference between calculated and measured X-ray intensity(cps/uA) for each elements such as Zr and Nb.

In case of testing the quantitative analysis for unknown specimen without standard specimens, therefore, it must be considered that the analysis results are carried out by the semi-quantitative analysis with probability 95.45%(2σ), and 97.33%(3σ).

Table 3. Final results for the elements of Zr and Nb

Element	Zr			Nb		
Frequency	1st	2nd	3rd	1st	2nd	3rd
Intensity_unk (cps/uA)	5.45E+05	5.33E+05	5.37E+05	1.69E+04	1.92E+04	1.76E+04
Intensity_Measured (cps/uA)	497073.8	497073.8	497073.8	605278.0	605278.0	605278.0
K-raw(%) -1	1.0965	1.0722	1.0810	0.0279	0.0317	0.0292
Confidence Interval	1.9970	1.9969	1.9969	1.9716	1.9737	1.9721
Probability of Confidence Interval	0.9541	0.9541	0.9541	0.9513	0.9515	0.9514
Probability of Difference	0.9493	0.9493	0.9493	0.9493	0.9493	0.9493
Z-value	2	2	2	2	2	2
K-raw(%)	104.62%	102.30%	103.14%	2.65%	3.01%	2.77%

3. Conclusions

Throughout the results of the semi-quantitative analyses between measured from the machine and calculated from calculations, the following items were investigated and concluded.

(1) The results, i.e. population, produced by the machine are seemed to have the normal distribution, $N(\mu, \sigma^2)$.

(2) Without standard specimen, the equation (2) is applied theoretically to obtain the calculated X-ray intensity for CAL_STD spectrometer.

(3) The theoretical calculated value includes the errors, i.e., the approximate 2.5 ~ 5.0% uncertainty basically.

REFERENCES

- [1] S. Bremier et al., "Electron probe microanalysis of a METHPHIX UPuZr metallic alloy fuel irradiated to 7.0 at.% burn-up", Journal of Nuclear Materials 480 p.109 (2016).
- [2] Y.S Choo et al., "The Development of Dismantling Machine for Capsule and HANARO fuel bundle Irradiated at HANARO", KAERI/TR-1078/98, KAERI (1998).
- [3] R.Walpole et al., "Probability and Statistics for Engineers and Scientists", third edition, ISBN 0-02424170-9 (1985).

Study on Bubbler Position Optimization of Cold Crucible Induction Melter

Seokju Hwang*, Young Hwan Hwang, and Cheon-Woo Kim

KHNP CRI, 70, 1312-gil, Yuseong-daero, Yuseong-gu, Daejeon 34101, Republic of Korea

*hsjhheh@khn.co.kr

1. Introduction

In the cold crucible induction melter(CCIM), bubblers are installed to efficiently mix wastes such as dry active waste and resin with molten glass to obtain high-quality stable glass solid. In this study, the thermal flow field is visualized in the CCIM according to the position of the bubbler, and the influence of the position of the bubbler on the flow of the molten glass in the CCIM is examined. Previously, study was the thermal- flow analysis inside the CCIM.[1]

2. Thermal-Flow Analysis

2.1 Analysis strategy

Thermal-Flow Analysis in the FLUENT 15.0 is used. Under the presumption of abnormal state and incompressible flow, the volume of fluid(VOF) model, and standard k-ε turbulence model were applied to simulate the bubble behavior, molten glass, and interface.

2.2 Initial Conditions

The physical properties and boundary conditions used in this study are shown in Table 1 and Table 2. [2] The cooling conditions outside the CCIM are evaluated by calculating the thermal transfer coefficient according to the given temperature, which influences the thermal flow simulation.

Table 1. Physical properties of the air and glass

	Density (kg/m ³)	Cp (J/kg·K)	Thermal Conductivity (W/m·K)	Viscosity (kg/m·s)	Molecular Weight (kg/kgmol)
Air	1.225	1006.23	0.0242	1.7894e-5	28.966
Glass	2350	1290	3	4.278	69.62

Table 2. Boundary conditions for interfaces

	Interface	Bottom Wall	Side Wall Air	Side Wall Glass
Heat Transfer Coefficient	80 W/m ² ·K	300 W/m ² ·K	80 W/m ² ·K	30 W/m ² ·K
Free Steam Temperature	30 °C	24 °C	30 °C	30 °C

In order to investigate the thermal flow phenomenon of the molten glass with respect to the position change of the bubbler, the bubbler position was changed to 4 positions based on the center. The bubbler position for each analysis case is shown in Fig. 1.

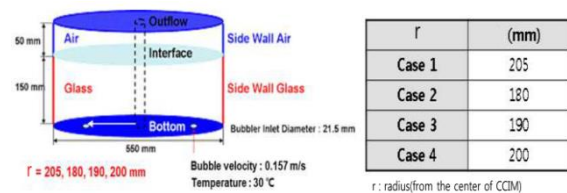


Fig. 1. Dimensions and boundary conditions of CCIM.

2.3 Analysis results

Fig. 2 shows the temperature distribution of the central section with respect to time in the case of Case 1.

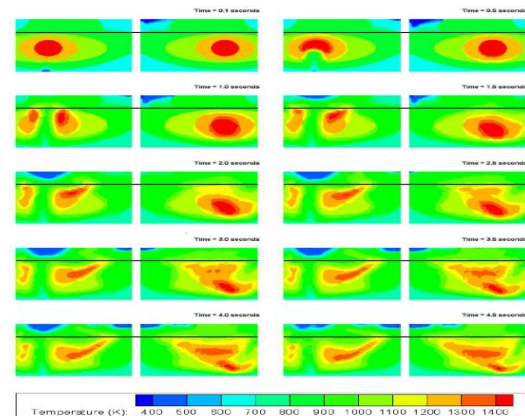


Fig. 2. Temperature distribution. (Case1)

Fig. 3 shows the temperature distribution with respect to time when the bubbler's position is 0.18m from the center.

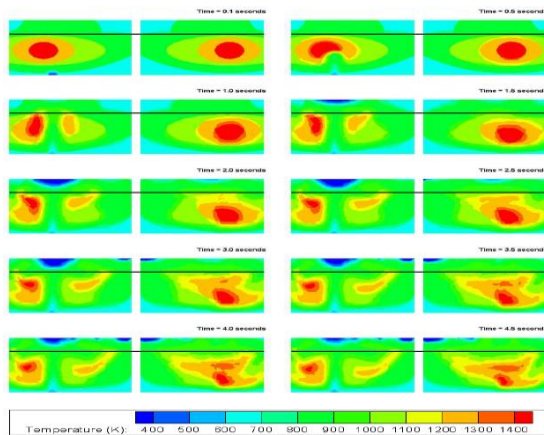


Fig. 3. Temperature distribution. (Case2)

Fig. 4 and 5 show the temperature distribution of the central section with respect to time when the position of the bubbler is 0.19m and 0.2m from the center.

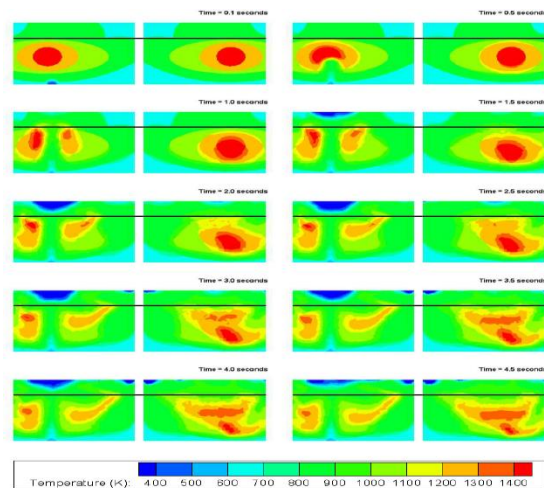


Fig. 4. Temperature distribution. (Case3)

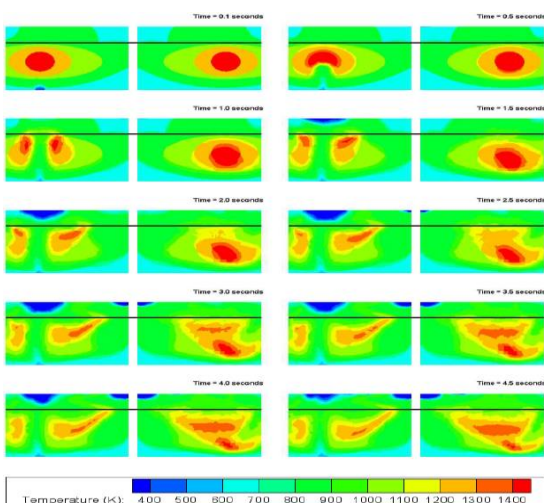


Fig. 5. Temperature distribution. (Case4)

It can be seen that the temperature of the wall of case 2~4 is lower than the wall temperature of case 1. This is because of the flow of fluid generated by bubbles, less active around the wall than in case 1.

In order to analyze the flow inside the CCIM, the streamline was identified in the middle section. Fig. 6 shows the streamline of the central section with respect to time when the position of the bubbler is 0.205m from the center (case 1). In the early stage of the bubble generation, it can be seen that the internal melt flow path rises due to the momentum generated by the bubble generation. As the bubble rises, it is observed that the molten glass circulates thanks to the vortex generated near the wall surface and the central position inside the CCIM. According to the analysis, the size of vortex only differed from the central part and wall surface according to the position of the bubbler.

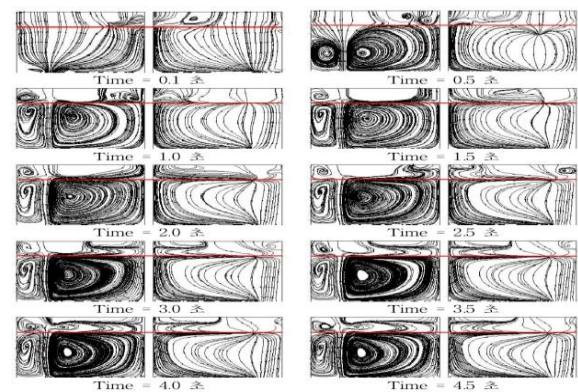


Fig. 6. Streamline inside the CCIM. (Case1)

3. Conclusion

The effect of the position of the bubbler on the flow of the molten glass inside the CCIM was analyzed by visualizing the thermal-flow field according to the position of the bubbler. It was confirmed that the flow the fluid did not flow toward the wall as the position of the bubbler became closer to the center. As a result, it is found that case1 is the optimal position.

REFERENCES

- [1] KHNP CRI, "Thermal fatigue life evaluation of the CCIM chamber.", Technical Report (2013).
- [2] Seokju Hwang, Young Hwan Hwang, Cheon-Woo Kim, "Study on Thermal-Flow Analysis of Cold Crucible Induction Melter", Proc. Of the KRS 2017 Spring Conference, 15(1) May 24-25, 2017, Busan.

Spent Nuclear Fuel Storage Rack Sampling and Process Development for Evaluation of the Radionuclide Inventory

Kyung-Ho Kim^{a*}, Bong-Ki Ko^a, Dae-Sung Kim^b, and Young-Ku Choi^a

^a Sunkwang T&S Co., Ltd Sunkyung Officetel 20F, #3, Gongwon-ro, Guro-gu, Seoul, Korea

^b KOREA HYDRO&NUCLEAR POWERCO., LTD, 696-13 Donghaean-ro, Yangnam-myeon, Gyeongju-si, Gyeongsangbuk-do 38218, Korea

*kyungho0121@naver.com

1. Introduction

Currently spent nuclear fuel is generated in 23 electricity generating reactors, i.e. 19 light-water reactors (Kori, Hanbit and Hanul) and 4 heavy-water reactors (Wolsong), and 1 research reactor (Hanaro in Korea Atomic Energy Research Institute) in Korea. The spent nuclear fuel from light-water reactors is first cooled in the storage water pool inside the reactor building. As the wet storage capacity of the spent nuclear fuel of nuclear power plants is saturated, the storage water pools were replaced by high-density storage racks, and the existing spent nuclear fuel storage racks are cut, compressed and stored in drums. Nuclear Safety and Security Commission Notification No. 2015-4 requires the measurement of the density of the radionuclides, held by Korea Hydro & Nuclear Power Co., Ltd., which must be identified during disposition of low- and intermediate-level radioactive waste, especially the radionuclides discharging alpha rays and beta rays in drums, which cannot be directly measured by radiation detectors. Accordingly, this study developed the specimen extraction and process for verifying the scale factors of existing spent nuclear fuel storage racks.

2. Main Subject

2.1 Current status of spent nuclear fuel storage racks

Hanbit Unit 1 was selected for extracting spent nuclear fuel storage rack specimens, and 262 drums were put on the spent nuclear fuel storage racks Overhaul period and Ordinary period in 2016. As illustrated in Fig. 1, the spent nuclear fuel storage racks are rectangular, made of 4M-long and 2T STS304. They are cut to a length of about 30cm,

compressed and put in drums. A device is used to extract 1 approximately 2 X 2 specimen.



Fig. 1. Spent nuclear fuel storage rack.

2.2 Developing the sampling extraction process

2.2.1 Setting up the Radiation Controlled Area.

The specimens were extracted in this study from the spent nuclear fuel storage racks in Hanbit Unit 1. Some drums are stored in drums in the 77ft temporary warehouse, and some spent nuclear fuel storage racks will be extracted as specimens from the 126ft fuel building before they are put in drums. To prevent the spread of dust, a tent will be installed, and it will be designated as a radiation controlled area during specimen extraction, and specimens in drums will be transported to the work space using the crane for drums. During the specimen extraction period, no one other than workers will be allowed in the radiation controlled area. To minimize the radiation exposure of workers, protective measures fit for the characteristics of the work must be taken, radiation shields must be installed, and work space must be secured. Radiation protection plans will be established based on surface dose.

2.2.2 Preparing the sampling device. As for devices for specimen extraction, the ready-made hydraulic punching machine and hydraulic pump will be used to extract 2 X 2cm specimen. Fig. 2 illustrates this process.



Fig. 2. Hydraulic punching machine and hydraulic pump.

2.3 Sampling

- 1) Prepare specimens before they are put in drums.
- 2) Use a crane to move specimens in drums.
- 3) Connect the hydraulic punching machine and hydraulic pump.
- 4) Install an immobilization frame to prevent the specimen from escaping.
- 5) Install the tent for preventing the spread of dust during specimen extraction.
- 6) Number each specimen.
- 7) Insert the prepared spent nuclear fuel storage racks into the hydraulic punching machine, and punch them.
- 8) Use the sealed vinyl to pack, and put it in the radioactive material in the ip-2 carrying container for transporting radioactive materials before transporting it.



Fig. 3. Sampling.

3. Conclusion

The spent nuclear fuel storage rack specimen extraction process for evaluating the radionuclide inventory was developed, and specimens were extracted from the waste from the spent nuclear fuel storage racks, which were replaced in Hanbit Unit 1 in 2016. The total weight was 91,700 kg, and a total of 10 sets of 20 specimens were extracted per 1,750 kg, which is equivalent to 5 drums. Two 2 X 2 specimens were extracted from each set.

REFERENCES

- [1] Yoon In Sik, A Study on the Evaluation of Source term for Spent Fuel of Nuclear Power Plants in Korea (2004).
- [2] Yong Soo Hwang, Sun Young Chang, Jae Jun Han, Spent Nuclear Fuel Management in South Korea Current Status and the Way Forward (2015).

Development of C-14 Stripping Method From Waste Resin Generated From HWR via Microwave Treatment: Part 1, Characterization of Anion and Cation Surrogate Waste Resin With Microwave Treatment

Ki Rak Lee^{a*}, Hwan-Seo Park^a, Hyeon-Oh Park^b, Jung-Hoon Choi^a, and Seung-Youb Han^a

^a Pyroprocess Technology Division, Korea Atomic Energy Research Institute (KAERI),

111 Daedeok-daero 989, Yuseong-gu, Daejeon, Republic of Korea

^b Sunkwang T&S Co.,Ltd, Sunkyung Officetel 20F, #3, Gongwon-ro, Guro-gu, Seoul, Korea

*kirakki@kaeri.re.kr

Inorganic and organic ion exchange materials were generally applied to liquid processes in nuclear reactor. In the case of heavy-water reactor (HWR), zeolite, active carbon, anion resin, and cation resin were used to treat liquid processes such as reactor primary coolant cleanup and liquid radioactive waste management system. Then, used ion exchangers were stored at storage tanks. Various kinds of nuclides were adsorbed in ion exchange materials. Especially, C-14, long half-life nuclide, was highly concentrated in anion resin, and waste resin was treated as intermediated level radioactive waste (ILW).

In KOREA, 4 units of heavy-water reactor are operated and a lot of spent resins are generated. Until now, 580 m³ of waste resins were generated from Wolsung HWR unit 1 and 2 and stored at storage tank (storage tank capacity of Wolsung HWR unit 1 and 2: 986 m³). Enormous cost will be expected to dispose this radioactive waste if there is no process treating this waste.

Conventional process to treat radioactive wastes is not suitable for treating spent resin. To solve this problem, various kinds of processes were developed such as acid stripping, PLO process, activity stripping, thermal treatment, and etc. In this study, new process using microwave, is suggested. Basic characterization of anion and cation surrogate waste resin was conducted. Demonstration equipment was manufactured using characterization data.

Surrogate waste resin was synthesized by using anion resin absorbed HCO₃⁻ ion. Surrogate waste resin was treated by microwave under different time condition from 1 min to 30 min. Treated surrogate waste resin was characterized by C NMR to confirm a stripping of adsorbed HCO₃⁻ without damage of main chain in resin. After 20 min microwave treatment, ammonium functional group was perfectly changed to amine group. It means that of adsorbed HCO₃⁻ was removed. Additionally, main chain and benzene group were detected at even 30 min treated sample.

Surrogate C-14 nuclide was perfectly removed from waste anion resin without damage of resin.

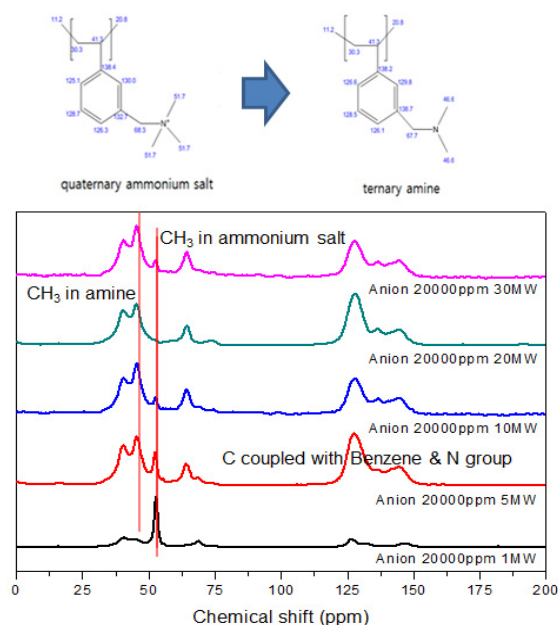


Fig. 1. C NMR results of surrogated anion resin treated by microwave.

REFERENCES

- [1] Technical Reports Series no. 408, Application of ion exchange processes for the treatment of radioactive waste and management of spent ion exchangers, IAEA, Vienna, 2002.
- [2] M. Matsuda, K. Funabashi, T. Nishi, H. Yusa, M. Kikuchi, "Decomposition of ion exchange resins by pyrolysis", Nuclear Technology, 75 (1986), 187-192.

Ultra-Low Temperature, Non-Conventional Sintering of Iodate Substituted Hydroxyapatite (IO-HAP) for the Immobilization of Radioiodine

Muhmood ul Hassan and Ho Jin Ryu*

Department of Nuclear and Quantum Engineering, KAIST, 291, Daehak-ro, Yuseong-gu, Daejeon, Korea

*hojinryu@kaist.ac.kr

1. Introduction

Immobilization of radioiodine (I-129) cannot be carried out by using conventional vitrification route by using borosilicate glass because of its low volatilization temperature. A number of different low temperature methods are being investigated in order to resolve the loading as well as volatilization issues during the loading process. Among all those on going and suggested waste matrices the use of hydroxyapatite has an advantage of not only effectively hold the loaded radioiodine as well as its decay product i.e. Xe-129. It also has good corrosion resistance properties in ground environment and resistance against radiation damages of the loaded radioactive material [1].

Iodate substituted hydroxyapatite (IO-HAP) $\{Ca_{10}(PO_4)_6(OH)_{2-x}(IO)_x\}$ has been synthesized by using the wet precipitation process with 10 wt% of iodine loading. Since the synthesized material is a porous material with high surface area which may accelerate the leaching of loaded iodine due to large surface area in contact with the ground water. In order to overcome this issue sintering of the IO-HAP has been carried out by using spark plasma sintering at temperatures higher than 200°C [2].

In our work we have demonstrated and shown that IO-HAP dried powder can be sintered at temperatures $\leq 200^\circ\text{C}$ by using a steel mold, a uniaxial press and heating band. We have also studied the sintering behavior under the presence of added as well as adsorbed water in the dried IO-HAP. We have optimized the sintering conditions to achieve the sintered relative density of $\geq 90\%$ without loss of loaded iodine.

2. Methods and Results

2.1 Iodate substituted Apatite Synthesis

Initially the crystalline apatite has been synthesized by using the wet precipitation method as described elsewhere [3]. Two solutions, anionic and cationic, were prepared by dissolving each di-ammonium hydrogen phosphate $\{(NH_4)_2HPO_4\}$ + ammonium iodate (NH_4IO_3) and calcium nitrate $\{Ca(NO_3)_2 \cdot 4H_2O\}$ in 100ml of double deionized water respectively. The pH of the solutions was adjusted at 10.5 by using concentrated ammonia $\{NH_4(OH)_2\}$. All used materials were of reagent grade. Then anionic solution was mixed in cationic solution dropwise during one hour under continuous stirring at the rate of 200RPM and temperature was maintained at 35°C. The pH of the solution was continuously monitored by using digital pH meter and was maintained at 10.5 throughout the synthesis period by the dropwise addition of concentrated ammonia. At the end of synthesis, precipitates were left inside the solution for 12hrs for ageing. Later on precipitate was filtered and thoroughly washed with double deionized water. Finally the filtrate was dried for 12hrs in a vacuum oven at 50°C.

2.2 Powder Characterization

XRD pattern of the IO-HAP powder was obtained by using SmartLab, RIGAKU, high resolution powder X-ray diffractometer and the 2θ range was between 10 and 70° (Fig. 1).

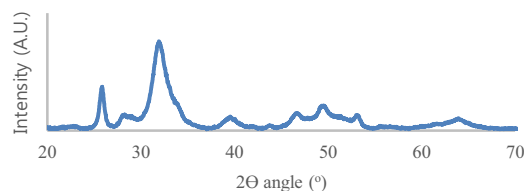


Fig. 1. XRD patterns of IO-HAP synthesized at 35 °C.

FTIR spectra of the sintered IO-HAP sample was also obtained to verify the formation of HAP and presence of main apatite groups (Fig. 2).

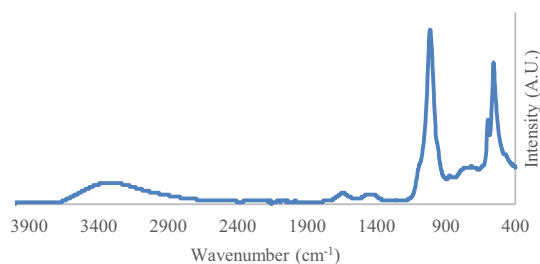


Fig. 2. FTIR spectra of IO-HAP synthesized at 35 °C.

Raman spectra of the sintered IO-HAP sample was also obtained to verify the stability and presence of iodine in the form of iodate in the sintered samples (Fig. 3).

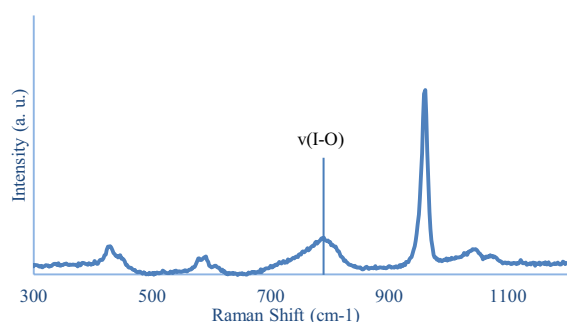


Fig. 3. Raman Spectra of the IO-HAP sintered at 200 °C.

2.3 Sintering of IO-HAP

The ultra-low temperature sintering of IO-HAP dried powder has been investigated first time. For sintering purpose 0.5g of IO-HAP dried powder was poured in a steel mold with internal diameter of 17mm. In order to provide controlled heating during the pressing, heating jacket of 0.5 kW connected with a temperature controller was tighten around the steel mold. An optimized uniaxial pressure of 500 MPa was applied for an optimized holding time of 10 minutes only.



Fig. 4. IO-HAP pellet sintered at 200°C under 500 MPa uniaxial pressure at holding time of 10 minutes.

After 10 minutes of holding at 200°C, pressure was released and a sintered/dense pellet of the IO-HAP was obtained (Fig. 4).

3. Conclusion

In this work, ultra-low temperature, non-conventional sintering of iodate substituted hydroxyapatite was demonstrated for the first time. We have studied the role of adsorbed water present in the dried powder even after drying at 50°C under vacuum conditions and found that adsorbed water has vital role in the realization of this ultra-low temperature sintering. We have optimized the sintering temperature as 170°C in order to get sintered relative density upto 88% whereas for relative densities $\geq 90\%$ the sintering temperature has been optimized as 200°C. The uniaxial pressure and holding time has been optimized as 500 MPa and 10 minutes. This leads to our final goal of development of an ultralow temperature sintered waste matrices for the immobilization of I-129 for long term geological disposal.

REFERENCES

- [1] A. Coulon, D. Laurencin, A. Grandjean, C. Cau Dit Coumes, S. Rossignol, and L. Campayo, "Immobilization of iodine into a hydroxyapatite structure prepared by cementation †," *J. Mater. Chem. A*, vol. 2, pp. 20923–20932, 2014.
- [2] L. Campayo, A. Grandjean, A. Coulon, R. Delorme, D. Vantelon, and D. Laurencin, "Incorporation of iodates into hydroxyapatites: a new approach for the confinement of radioactive iodine," *J. Mater. Chem.*, vol. 21, no. 44, p. 17609, 2011.
- [3] A. Coulon *et al.*, "Key parameters for spark plasma sintering of wet-precipitated iodate-substituted hydroxyapatite," *J. Eur. Ceram. Soc.*, vol. 36, no. 8, pp. 2009–2016, 2016.

Heavy Water Reactor(HWR) Spent Resin Mixture Processor Development

Hyeon-Oh Park^{a*}, Kyu-Tae Park^a, Jung-Min Yoo^a, Young-Ku Choi^a, and Yeo-jin Kim^b

^aSunkwang T&S Co.,Ltd., Sunkyung Officetel 20F, #3, Gongwon-ro, Guro-gu, Seoul, Korea

^bKHNP, 696-13, Donghaean-ro, Yangnam-myeon, Gyeongju-si, Gyeongbuk-do, Korea

*gusdhtpgml@nate.com

1. Introduction

The liquid radioactive waste, generated in the process of operating NPPs (Nuclear Power Plants), is purified using the ion exchange resin. PHWR Wolsong Units 1 and 2, which are Pressurized Heavy Water Reactors, temporarily store spent resin in the spent resin storage tank, and they are planning to process it according to the NPP decommissioning plan.

The spent resin, generated by heavy water reactors, contains various radioactive nuclides, and especially because it has a high concentration of C-14, a long-lived nuclide, it is classified as ILW (Intermediate Level Waste) radioactive waste. The spent resin mixture in the storage tank contains about 20% of active carbon and zeolite, and about 80% of mixed resin.

This study developed a system for processing the spent resin mixture, stored in the spent resin storage tank of a heavy water reactor, and derived the optimal performance of the system by conducting an empirical experiment using actual spent resin. The authors analyzed the factors affecting the performance of the system, and reviewed the relationships between these factors and the performance of the system. The spent resin used for the empirical experiment was sampled from storage tank No. 2 of Wolsong Unit 1.

2. Methods and Results

2.1 Making the heavy water reactor spent resin mixture processor

Based on the results of a laboratory-size simulation, the authors designed a 1-liter batch-type closed circuit circulation process, and the spent resin mixture processor (Fig. 1) consisted of the spent resin separation, C-14 desorption and CO₂ adsorption part.

The spent resin mixture separator is a separation system using the particle size in the liquid state based on the results of the analysis of the characteristics of the spent resin mixture. To desorb C-14 from the separated resin, the direct heating technology using the microwave was applied, and magnetrons with 3kw power (1kw, 3ea) were used through the simulation. For CO₂ adsorption, a circulating adsorption system, including an adsorption tower with adsorbents, was made.

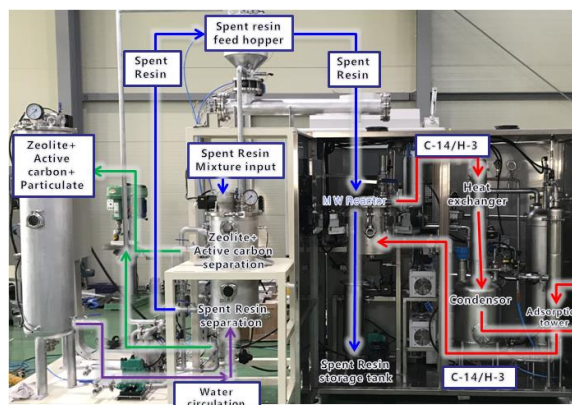


Fig. 1. Heavy water reactor spent resin mixture processor.

2.2 Separating the resin

The heavy water reactor spent resin mixture processor was used to separate resin from the spent resin mixture, and the optimal separation process was established. The spent resin separation procedure is shown in the process flow chart illustrated in Fig. 2 below.

- 1) Supply circulating water to the circulating water tank in ①.
- 2) Put the spent resin mixture in the separator in ②.
- 3) Use the line in ③ to supply circulating water to the separator.
- 4) Separate the spent resin while operating the impeller.
- 5) Retrieve the circulating water through the line in ④.
- 6) Re-supply the circulating water with the fine powder removed by the filter at the top of the tank in ① to the separator.
- 7) Discharge the separated active carbon and zeolite.
- 8) Re-supply the retrieved circulating water to the separator.
- 9) The separated spent resin will be stored in the buffer hopper along the line in ⑥.
- 10) The separated spent resin will be supplied to the MW reactor using the screw-type feeder in ⑧.

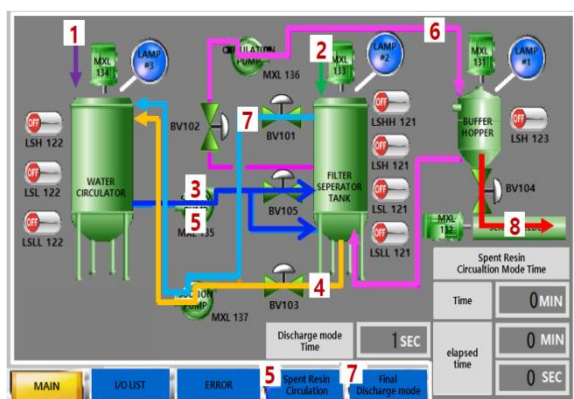


Fig. 2. Spent resin mixture separator process flow.

2.3 C-14 desorption

The separated spent resin will be supplied from the spent resin mixture separator to the microwave reactor, and desorbed according to the process in Fig. 3.

- 1) For effective reaction between the supplied resin and the microwave, the impeller (⑤) will be operated.
- 2) For circulation of the desorbed gas, the blower (⑥) will be operated.
- 3) To collect the moisture, the condenser (②) will be operated.
- 4) Desorbed C-14 will be transported to the adsorption tower along the line in ③.
- 5) To enhance the efficiency of C-14 adsorption, a closed circulation process was designed.

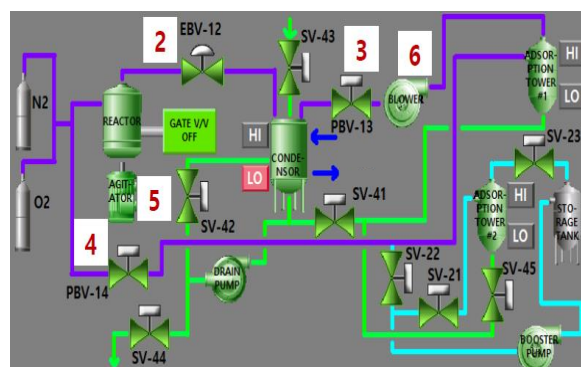


Fig. 3. Microwave C-14 desorption process flow.

2.4 CO₂ adsorption

C-14 in the C-14 desorption process will be adsorbed by the adsorption tower in 1. However, some unadsorbed C-14 will be collected in the storage tank, and the residual C-14 will be adsorbed by the adsorption tower through the circulation process. Inside the adsorption tower, there are six stages to increase the reaction surface area of the adsorbent and C-14, and Fig. 4 illustrates the interior of the adsorption tower.



Fig. 4. Image of the interior of the adsorption tower.

3. Conclusion

The authors developed a system for processing the spent resin mixture stored in the spent resin storage tank of the heavy water reactor, and derived the optimal performance of the system by using actual spent resin to conduct an empirical experiment.

To efficiently separate the spent resin from the spent resin mixture, appropriate meshes in consideration of the particle size of each component were selected, and to effectively transport the separated spent resin, appropriate flow and flux were determined. To improve the separation efficiency of each mixture with a different mass, the RPM of the impeller was adjusted, and to increase the resin feed rate, the optimal operation conditions of the diaphragm pump were derived.

To effectively desorb from the spent resin, the power supplied to the magnetron and the optimal reaction time were derived.

4. Acknowledgement

This study was conducted as part of the Ministry of Commerce Industry and Energy's nuclear power core technology development project in 2015.

REFERENCES

- [1] KAREI, Development of Adsorbent for C-14 Gas Trapping and Characteristics Evaluation, 2005.
- [2] F. H. Ruddy, A. R. Dulloo, J. G. Seidel, F. W. Hantz, and L. R. Grobmyer, Nuclear Reactor Power Monitoring Using Silicon Carbide Semiconductor Radiation Detectors, Nuclear Technology, Vol.140, p. 198, 2002.
- [3] F. H. Ruddy, A. R. Dulloo, J. G. Seidel, J.W.Palmour, and R. Singh, The Charged Particle Response of Silicon Carbide Semiconductor Radiation Detector, Nuclear Instruments and Methods In Physics Research, Vol.505, p.159, 2003.

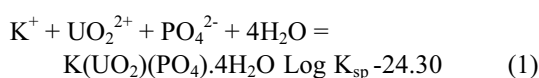
The Removal of Uranium From Aqueous Effluents via Meta-Ankoleite Formation

Richard I. Foster*, Kwang-Wook Kim, Min-Jeong Kim, Maeng-Kyo Oh, and Keun-Young Lee
Decommissioning Technology Research Division, Korea Atomic Energy Research Institute, 111,
Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

*rifoster@kaeri.re.kr

1. Introduction

It is known that many metal phosphate salts possess low solubilities. Uranium is no exception with many uranium/uranyl phosphate minerals having very low solubilities.[1] In recent years the formation of insoluble uranium phosphates, such as Meta-Ankoleite (1), have been researched for the effective sequestering of environmentally mobile uranium.[2]



In this study we have investigated the suitability of dosing mono-potassium phosphate (MKP, KH_2PO_4) for the selective removal of uranium from a complex, highly contaminated, waste effluent (Table 1). The effluent in question is generated as part of the treatment strategy for a legacy uranium catalyst waste which has been in temporary storage since 2004 in Korea.[3]

A range of both pH (3 – 11) and phosphate dosages (0.25 – 10 mM) have been screened. Precipitation kinetics has also been investigated revealing the effectiveness of phosphate addition for the rapid removal of uranium from solution in less than two hours.

The effluent was decontaminated via meta-ankoleite formation, as confirmed by XRD, to uranium levels below the compulsory release limit of 1 ppm. Final decontamination factors of the order of $\geq 10^3$ were readily achieved thus ensuring that the effluent meets the Korean release criteria for U-bearing liquid waste.

2. Experimental

2.1 Materials

All materials were used as received. MKP (KH_2PO_4 , 99.5%) was obtained from SHOWA, potassium hydroxide (KOH, 85%) from JUNSEI and nitric acid (HNO_3 , 65%) from MERCK. The effluent was produced during the treatment of the uranium catalyst and stored in 20 L plastic drums before use.[3]

2.2 Method

A known aliquot of the effluent was placed into a 50 mL beaker and stirred at 180 rpm with a magnetic stirrer bar. MKP was added to reach the required concentration (0.25 – 10 mM). KOH was added until the desired pH was reached (3.0, 5.0, 5.5, 6.0, 6.25, 6.5, 7.0, 8.0, 9.0 or 10.5) and allowed to react for two hours. For the kinetic testing solution samples were taken at 1, 2, 4, 8, 16, 30, 60 and 120 minutes and prepared for ICP-OES analysis.

All solution samples were filtered through a 0.2 μm syringe filter and acidified with 5 % HNO_3 before being analysed via ICP-OES (Analytikjena PQ9000 Elite).

Upon completion of the reaction the precipitates were collected, centrifuged and washed with de-ionised water (18.2 M Ω , 5 x 50 mL) to remove the remaining Na_2SO_4 which is present in the effluent stream (Table 1). The washed precipitates were then dried and analysed via SEM/EDX (Bruker Namo, Xflash Detector 410-M) and XRD (Bruker D2 Phaser).

Table 1. Summary of effluent parameters as received

pH	Conductivity ¹ (mS/cm)	TSS (ppm)	Turbidity (NTU)	Particle size ² (μm)	[U] (mM)	[Si] ³ (mM)	[Sb] ³ (mM)	[Ca] ³ (mM)	[Mo] ³ (mM)	[SO ₄ ²⁻] (M)	Ionic strength (M)
0.85	101.7	200	237	17.2	0.09	3.62	0.44	0.40	0.27	0.37	1.14

¹ Measured at a constant 25.0 °C, ² Mean volume measure, ³ Elements with [mM] greater than uranium (Al, Cu, Fe, Zn, B, Ni and Cr present ≤ 0.09 mM)

3. Results & Discussion

3.1 pH Screening

Fig. 1 shows the concentration of uranium remaining in solution at a range of pH values after two hours, with and without the addition of KH_2PO_4 (1 mM). The addition of phosphate is clearly effective for the removal of U when compared to no phosphate addition with an optimal pH of 6.00 – 6.50. Uranium is removed independently of silicon at the target pH.

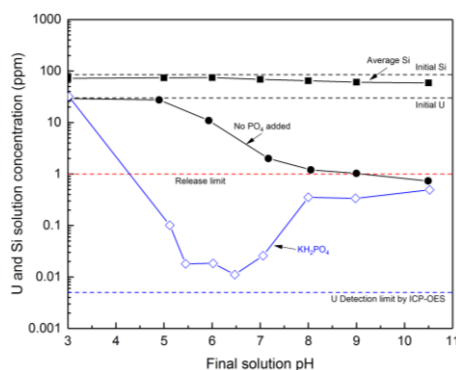


Fig. 1. Results from the pH screening tests.

3.2 Dosage & kinetics screening

Fig. 2 shows the results of the dosage screening as a function of time. A KH_2PO_4 concentration of 1 mM or greater is needed in order to achieve the required release limit with 2 mM being found as optimal.

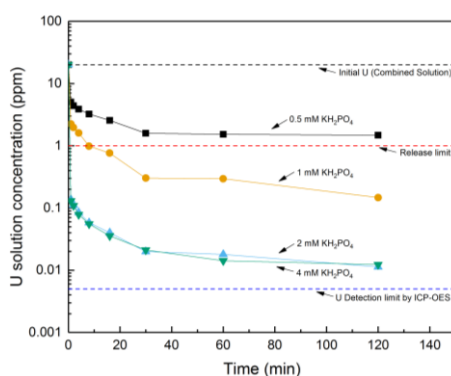


Fig. 2. Results from the phosphate dosage screening.

Further addition of phosphate does little to enhance the removal of uranium and would only lead to complications further down the process with the requirement of excess PO_4^{3-} removal.

3.3 Meta-Ankoleite characterization

The formation of the Meta-Ankoleite mineral phase was confirmed by XRD (Fig. 3). This is in agreement

with previous experimental data presented by M. Kanematsu *et al.*[2]

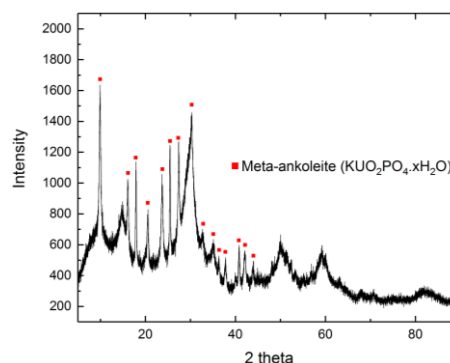


Fig. 3. XRD analysis result of the collected precipitates.

SEM analysis (Fig. 4) indicates the formation of amorphous precipitates approximately 10 – 100 μm in size. Elemental mapping indicates the presence of both Si and Sb suggesting co-precipitation is occurring. The chemical composition of Si and Sb is unknown.

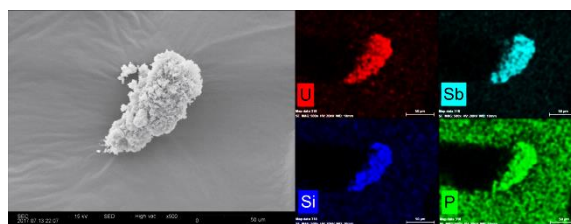


Fig. 4. SEM image with accompanying elemental map.

4. Conclusions

The addition of MKP to the effluent has been found effective for the selective removal of uranium as the Meta-Ankoleite mineral phase.

Parameter screening has enabled process optimization to a pH (6.25), dosage (2mM) and kinetics (1hr), for a final [U] of < 20 ppb.

The process will now enter bench and pilot testing phases. Residual phosphate requires a removal strategy, e.g. FeCl_3 , before final release.

REFERENCES

- [1] D. Gorman-Lewis *et al.*, "Review of uranyl mineral solubility measurements", *J. Chem. Thermodynamics*, 40, 335-352 (2008).
- [2] M. Kanematsu, N. Perdrial *et al.*, "Influence of Phosphate and Silica on U(VI) Precipitation from Acidic and Neutralized Wastewaters", *Environ. Sci. Technol.*, 48, 6097-6106 (2014).
- [3] K. W. Kim *et al.*, "The development of a process for the volume reduction of uranium catalyst waste used for production of acrylonitrile", *In Press* (2017).

Fabrication of Graphene Fiber Fabrics as Sorbents for Radioactive Iodine Gas

Minseok Lee and Ho Jin Ryu*

Korea Advanced Institute of Science and Technology, Nuclear and Quantum Engineering.

291, Daehakro, Yuseong, 34141, Republic of Korea

*hojinryu@kaist.ac.kr

1. Introduction

During a reprocessing of spent fuels, significant amount of ^{129}I are exposed into a gas stream of the facility. Since ^{129}I can affect to environment and human health, it has to be removed before release the gas. Various radioactive iodine adsorbents have been developed by impregnating silver in porous support such Ag-zeolites or Ag-silicate due to the chemical affinity of Ag for iodine [1]. Recently, graphene-based materials such as graphene-powder or graphene-aerogel have been studied for iodine sorbents, and they showed high physisorption properties for iodine gas [2]. In this study, we prepared graphene fiber fabrics for real application of iodine removal. Because fiber-form does not be compacted as powder-form, there is low pressure drop. Also, fiber has thin shape and high permeability of gas flow, adsorption kinetics are faster than granular-form [3].

2. Experiment

2.1 Preparation of graphene oxide (GO) solutions

Graphite intercalated compounds prepared from reaction of graphite flake and small quantity of sulfuric acid and oxidizing agent. After that, expanded graphite were prepared by microwave-assist expansion. Graphene oxide (GO) solution was synthesized by the modified Hummer's methods from the expanded graphite. Finally, the GO solution was washed and concentrated with centrifuge.

2.2 Preparation of graphene fibers fabrics

The graphene fiber fabrics were prepared as following the procedure [4]. The graphene oxide solution was doped on a syringe pump and injected at

the flow rate of 50 $\mu\text{L}/\text{min}$ into a fast-rotating (50 rpm) coagulation bath. The coagulation bath contained 5wt.% of CaCl_2 aqueous solution. Then as-spun graphene oxide fibers were broken by the fast rotating, and short fibers were collected in the bath. The short fibers were washed with water and collected by vacuum filtering. After drying the short fibers in vacuum oven at 60°C for a day, they were re-dispersed in mixture of water and ethanol in a ratio of 50:50 (v/v). The dispersed short fibers were collected with vacuum filtering again and dried in vacuum oven at 80°C for a day. Then the short fibers were inter-connected by rapid swallowing and drying of gel-state fibers which called "wet-fusing". Finally, the graphene oxide fiber fabrics were chemically reduced with HI at 80°C for 12 h.

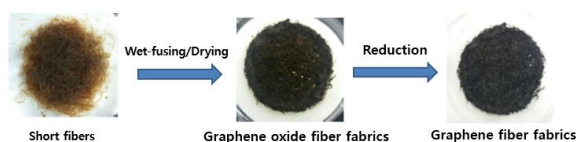


Fig. 1. Fabrication process of graphene fiber fabrics.

3. Results and discussions

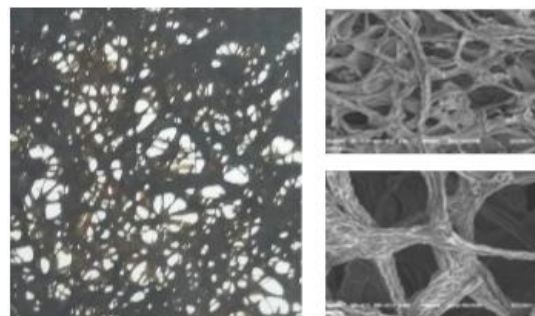


Fig. 2. The morphologies (left) and SEM image (right) of graphene fiber fabrics.

Since the graphene fiber fabrics were fabricated

from short fibers as building blocks, they have a large empty space which can accept a high flow rate of fluids. The random orientation of short fibers was due to the process of re-dispersion. The random orientation has advantages of direction-independent properties such as mechanical strength or electrical conductivity. Also, the fibers were connected strongly without binders such as resin and lots of the junction points make the fabrics resistive for fractures.

4. Conclusion

At this study, the graphene fiber fabrics were fabricated with wet-fusing method for effective removal of radioactive iodine gas. The fabrics had a randomly oriented and strong bonded of short fibers structures which have advantages of purification of flowing gas. For the future work, the characteristic of flow resistance and iodine adsorption will be investigated.

Reference

- [1] B. J. Rieley et al, Materials and processes for the effective capture and immobilization of radioiodine: A review, *Journal of Nuclear Materials*, 470, 307, 2016.
- [2] S. M. Scott et al, “Graphene-based sorbents for iodine-129 capture and sequestration”, *Carbon*, Vol. 90, p. 1, 2015.
- [3] M. Suzukiet al, “Activated carbon fiber: Fundamentals and applications”, *Carbon*, 32, 577, 1994.
- [4] Z. Li et al, “Multifunctional non-woven fabrics of interfused graphene fibres. *Nature communications*”, 7, 13684, 2016.

An Evaluation of the Glass-Ceramic Solidification Characteristics of Uranium Catalyst Waste by Green Body Pressure

Hyun-hee Sung*, Kwang-Wook Kim, Keun-Young Lee, Jimin Kim, and Bum Kyoung Seo

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

*shh414@kaeri.re.kr

1. Introduction

We have developed a process to treat a large volume of problematic radioactive waste, namely, a spent uranium catalyst which was used for production of acrylonitrile by a private company in South Korea.[1] The Korea Atomic Energy Research Institute (KAERI) is currently developing a volume reduction method for the uranium catalyst presently stored [2], for which the final solids generated in the process should be made into stable solids and satisfy the disposal conditions of solid radioactive waste in Korea.

Common immobilization methods for radioactive waste include, cement, glass, asphalt and polymers. Although using cement has advantages, such as inexpensive material cost and accumulated technology, it has a disadvantage that its chemical durability is relatively weak and its volume is larger compared with other solidifying media for the same initial waste volume. In the case immobilization of uranium waste by cement, CaO which is the main component of the cement solidification medium changes to Ca(OH)_2 when it reacts with water, and under this condition, uranium oxide is converted to uranium hydroxide ion ($\text{UO}_2(\text{OH})_x^{Y-}$), thus potentially becoming mobile and leaching out.[3] Therefore, it is necessary to develop technology for manufacturing solidified bodies with higher volume reduction factors and better physiochemical properties.

In this study, we investigated a method of sintering and immobilizing uranium-containing solids (Si-U-Sb-Fe-Al) Ox which are generated from treatment of the uranium catalyst waste. A glass solvent such as SiO_2 and Sb_2O_3 was used to form a U_3O_8 incorporated glass-ceramic structure possessing high mechanical strength and chemical stability. The solidification method and final sintered solid waste forms were characterized. It was found that the volume of the waste is reduced during the sintering

process and is made into a stable solid.

In this paper, we studied the change in physicochemical properties of sintered bodies due to changes in molding pressure during the manufacturing of Green Bodies (GB) for the production of uranium-containing glass-ceramic sintered bodies.

2. Experimental

2.1 Experimental method

Components of uranium catalyst, analysis by EDS(Energy Dispersive X-ray Spectroscopy), shows that they are glass former or flux components for soda-lime or borosilicate glasses.(Table 1)

Table 1. EDS result of Uranium waste

Element	Si	Sb	U	Fe	Al	O
Atomic wt (%)	8.77	16.9	4.40	12.2	0.37	45.0

A mass of uranium waste catalyst (3 g, 1.06 g / ml) and 0.6 ml of distilled water, as a binder, were added to a mold. The molding pressure was set at 15, 30, 60, 120, and 240 MPa pressure using a pellet press (diameter 13 mm).

The resultant GB was heated to 1100°C (heating rate 10K / min) by using a furnace, and then subjected to isothermal heat treatment for 4 hours (Fig. 1).

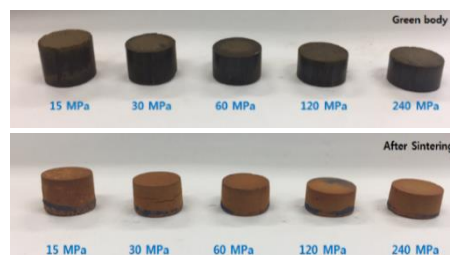


Fig. 1. Pictures of Green bodies and Sintered solids as a function of production pressure.

Several properties of density, porosity and compressive strength testing were measured.

2.2 Evaluation of sintered body characteristics

The volume and density of the sintered bodies were measured by using a hydrostatic weighing method using Archimedes principle and a gas volume meter based on Boyle's law. Bulk density and true density were obtained by two methods and porosity was measured by the difference of the values.

Final volume reduction of uranium catalyst waste by the glass-ceramic was evaluated by the difference between volumes of sintered body and initial uranium catalyst solid.

The compressive strength obtained at minimum molding pressure of 15MPa used in this work was over 3.45MPa which is the requirement for disposal of immobilized waste in Korea.(ASTM C39 Compressive Strength Standard Test)

3. Results and discussion

The volume reduction after sintering was 71.99 ~ 74.21% depending on the molding pressure used, as shown in Table 2. The difference in volume reduction due to molding pressure change was negligible when considering the high molding pressures used.

Table 2. Compression Test by Molding & Volume Reduction from original Uranium waste

Molding pressure (MPa)	After Sintering	
	Compressive strength (MPa)	Volume Reduction (%)
15	5.01	71.99
30	11.40	72.67
60	19.07	73.24
120	34.00	74.21
240	31.61	74.21

The minimum pressure of 15MPa was found that for forming a GB satisfies the compressive strength criteria of 3.45MPa, with the actual strength being 1.4 times more than indicated in the ASTM C39 standard. Table 3 shown in the porosity decreased with molding pressure, but the difference was not significant when the large difference in molding pressures used is taken into consideration. Therefore,

there was little difference in durability of glass-ceramic sintered body according to molding pressure.

Table 3. After Sintering Density and Porosity Result by molding pressure

Molding pressure (MPa)	Density (g/cm ³)		Porosity (%)
	Archimedes	Pyconometer	
15	3.78	4.61	17.84
30	3.88	4.63	16.26
60	3.96	4.59	13.71
120	4.11	4.66	11.83
240	4.11	4.61	10.88

Although using the minimum pressure produces satisfactory compressive strength results, a higher molding pressure is felt necessary owing to safety concerns during handling, transport and disposal of the waste form. However, because the cost of molding increases as the pressure of the compact increases, a molding pressure of 30 MPa was evaluated to sufficient. This pressure alone is more than four times the compressive strength criterion.

The chemical stability test of sintered body prepared at 30MPa of molding pressure was carried out, which is not present in the paper, the leaching standard for uranium test (ASTM C1285-02: PCT). It was observed to satisfy

In this works the durability and leaching characteristics of the glass-ceramic sintered body will be farther studied.

REFERENCES

- [1] K. W. Kim *et al.*, "The development of a process for the volume reduction of uranium catalyst waste used for the production of acrylonitrile", In Press, (2017).
- [2] Chung. Dong Yong, Kim. Kwang-Wook, et al., "Development of Decommissioning, Decontamination and Remediation Technology for Nuclear Facilities", KAERI report RR-4224, (2016).
- [3] Toru. Shimmura, et al., "Study on Cement solidified waste containing fuel debris", Atomic Energy Society of Japan, 2016 Fall meeting

Effects on SS304 Contents to Corrosion Properties of ZIRLO-SS304-INCONEL718-NM Alloy for Long-Term Disposal

Seung-Youb Han*, Seon-Ah Jang, and Hwan-Seo Park

Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon, Korea

*hsy@kaeri.re.kr

1. Introduction

Technologies used in solidification processing by melting spent nuclear fuel assembly materials and hulls for metal waste reduction that occurs during the pyro-processing have been studied, mainly in the United States [1-3]. A cladding hull in a spent fuel assembly to be applied to a domestic PWR is composed of a metal alloy such as Zircaloy-4, ZIRLO accounts for the highest volume and weight ratio in the generation of process waste. Recently, at KAERI, we have been conducting research on the recycling of a cladding hull as a solidification host matrix in order to solidify and handle highly radioactive anode sludge residue including noble metal (NM) that occurs from electro-refining from pyro-processing. If we produce an MWF (Metallic Waste Form) by adding and melting the alloying elements and anode sludge in the cladding hull, the effect on the volume reduction and weight loss can be maximized because the cladding hull can be 100% recycled to the solidification host matrix of MWF. According to the existing results of MWF alloys manufactured in the US, although a stable production of MWF is possible with SS-15Zr alloy, it is disadvantageous in terms of the domestic cladding hull recycling efficiency, because the Zr content is low. In addition, the mechanical stability and corrosion resistance of Zr-8SS is relatively low [3]. As a result of the new composition with Zr-Cr-Si-NM in KAERI, a stable MWF was formed without cracks. Moreover, it was possible to confirm the possibility that the cladding hull and anode sludge generated from the pyro-processing can be fabricated MWF alloy with an excellent corrosion resistance

2. Experimental Results

The ZIRLO-SS304-INCONEL718-NM composition of the three kinds of solidification host matrix compositions has the eutectic point composition, and the Zr-27Cr-3Si and Zr-17Cr-3Si compositions have a Cr composition difference of ± 5 wt.% respectively. A total of 6 specimens were prepared and evaluated by adding 8 wt.% of NM to each solidification host matrix composition and selecting three Zr-based MWF alloy compositions. All specimens were fabricated by high frequency induction heating in a silica crucible in an Ar atmosphere after forming a vacuum of 5×10^{-5} torr. To ensure the integrity of the specimen, it was held at 1600 to 1800°C for 5 minutes. After forming the melt completely, it was kept at 1350°C for about 15 minutes, furnace cooling was then applied to reduce the solidification shrinkage cracks and porosity. All prepared specimens were cut and polished, and XRD, SEM, and EDS analyses were performed for a microstructure and phase analysis.

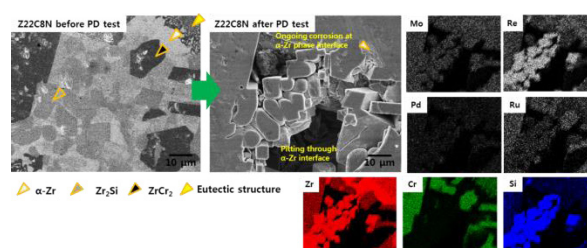


Fig. 1. SEM micrographs and EDS mapping results of the Zr-Cr-Si-NM specimen before and after potentiodynamic test.

For an evaluation of the corrosion properties, potentiodynamic (PD) and potentiostatic (PS) tests

were carried out in the acidic brine aqueous solution condition ($0.0001 \text{ mol} \cdot \text{kg}^{-1} \text{ H}_2\text{SO}_4 + 0.01 \text{ mol} \cdot \text{kg}^{-1} \text{ NaCl}$ in demineralized water, pH 4), and the leach solution was analyzed using ICP.

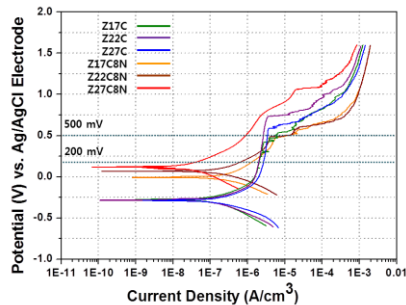


Fig. 2. The potentiodynamic polarization plots for Zr-Cr-Si-NM specimens.

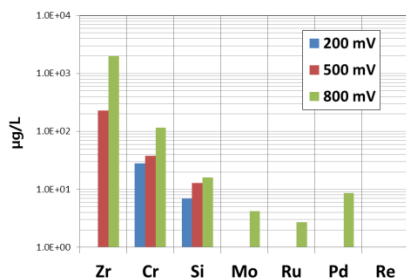


Fig. 3. ICP analysis results of electrolyte after potentiostatic test.

3. Conclusion

The main phases composing the Zr-Cr-Si-NM alloy consist of α -Zr, ZrCr_2 , and Zr_2Si . The eutectic structure and ZrCr_2 phase increased with increasing Cr content. Pd was concentrated on α -Zr, Ru on α -Zr and ZrCr_2 , Re on Zr_2Si phase. Electrochemical accelerated corrosion tests such as PD and PS were carried out on the Zr-Cr-Si and Zr-Cr-Si-NM alloys prepared in this study. The Zr-Cr-Si-NM alloys exhibited a high corrosion resistance and low corrosion rate owing to their low I_{corr} values with high E_{corr} values compared to conventional materials. As a result of the analysis of the composition of the

leachate after the PS test, there was no NM element leaching below 500 mV, or leaching of the elements excluding Re, which is a surrogate Tc at 800 mV. In particular, the Zr_2Si phase, which is encapsulating phase of Re, was very stable, and showed no shape change or leaching of Re after the corrosion test, suggesting that the Tc immobilization ability is excellent.

REFERENCES

- [1] D. D. Keiser Jr, D. P. Abraham, and J.W. Richardson Jr, "Influence of technetium on the microstructure of a stainless steel-zirconium alloy", *J. Nucl. Mater.*, 277(2-3), 333-338 (2000).
- [2] S. M. McDeavitt, D. P. Abraham, and J. Y. Park, "Evaluation of stainless steel-zirconium alloys as high-level nuclear waste forms", *J. Nucl. Mater.*, 257(1), 21-34 (1998).
- [3] W. L. Ebert and D. Kolman, Alloy Waste Form Testing Strategy Roadmap, U.S. Department of Energy Report, FCRD-SWF-2013-000226 (2013).

Behavior of Microstructure and Corrosion Properties of Zirlo-Cr-Si-NM Metal Waste Form Alloys by Change of Si Content

Seon-Ah Jang*, Seung-Youb Han, and Hwan-Seo Park

Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon, Korea

*sajang@kaeri.re.kr

1. Introduction

Pyroprocessing has been studied based on the electrochemical separation technology in molten salts for the recycling of spent fuel (SF). In the pretreatment and electrorefining processes, various metal wastes including cladding hull tubes, NFBC (Non-Fuel Bearing Components), and NMs (noble metals) in the anode sludge are generated as a process waste in the pyroprocess [1]. Among them, Zirlo cladding hulls have the highest weight ratio, which reaches about 62 wt.%, excluding the SF, whereas NFBC has the highest volume ratio. Many researchers are studying technologies for the treatment of the metal wastes by melting or compaction, followed by a solidification process. The metal waste solidified by the melting process can easily be alloyed with other elements to improve the corrosion resistance and stability for long-term disposal. However, most of the studies have focused on the alloys based on stainless steel (SS) cladding materials and NM, which are generated from the fast breed reactors [2]. Since the pyroprocessing studying at KAERI is targeting the SF from PWRs (Pressurized Light Water Reactors), it is essential to develop waste form alloys that include Zr-based cladding hull wastes with NMs. In particular, by recycling the cladding hull waste and NFBC as a solidification medium for immobilizing the NMs, we can have a great advantage not only in the volumetric reduction effect but also in the reduction of the waste

amount, and in the simplification of the waste treatment process.

In this study, Zirlo-Cr-Si-NM based alloys including all the metal waste components were fabricated by the induction melting process to evaluate the long-term disposal stability of the metal wastes generated from pyroprocess using microstructure analysis and corrosion tests.

Table 1. Composition of Metal Alloys

	S0	S1	S3	S5	S7
Zr	66.66	65.16	63.16	61.16	59.16
Cr	22	22	22	22	22
NM	10	10	10	10	10
Sn	0.56	0.56	0.56	0.56	0.56
Nb	0.78	0.78	0.78	0.78	0.78
Si	0	1.5	3.5	5.5	7.5
Total	100g	100g	100g	100g	100g

* NM(Noble Metal) : Mo, Ru, Rh, Pd, Re

2. Experiments

The induction melting process was performed using a silica crucible with 99.9% pure Zr, Cr, Si, and NM samples mounted in a vacuum arc melting furnace.

The metal waste form of about 3 x 3 cm² in size based on 100 g of the specimen was produced by melting at the temperature range of 1200°C to 1400 °C and by slowly cooling down. The various Zr-Cr-Si-NM alloys were produced by controlling the content of Silicon. The prepared specimens were evaluated for soundness and long-term stability of the waste form by examining the microstructure and by corrosion tests.

3. Results and discussion

We fabricated five waste form specimens Zr-Cr-Si alloys by controlling the content of Silicon. All the waste form specimens were fabricated in almost uniform size and shape without any apparent cracks due to a slow cooling rate.

As a result of corrosion tests, the corrosion properties of S3 and S7 specimens were excellent. E_{corr} depending on the composition of specimens were shown in Fig. 1.

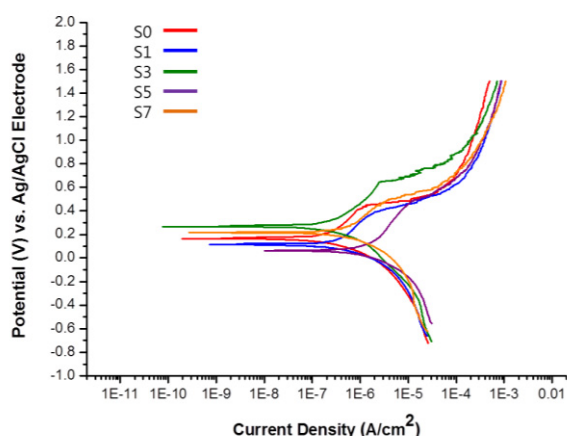


Fig. 1. Corrosion Properties.

4. Conclusion

In this study, we fabricated metal waste forms using an arc melting process and evaluated the performance thereof to recycle the cladding hull and NFBC wastes generated from pyroprocess thereby reducing the volume of the waste.

REFERENCES

- [1] K.H. Kang, C.H. Lee, M.K. Jeon, Y.L. Lee, Y.T. Choi, and G.I. Park, "Treatment Technology of Hull Wastes", KAERI/TR-5288/2013 (2014).
- [2] D. P. Abraham, S. M. McDevitt, J. Y. Park, "Microstructure and Phase Identification in Type 304

Stainless Steel-Zirconium Alloys", Metallurgical and Materials Transactions A, 27A, 2151-2159 (1996).

Study on Cs⁺ Separation by Using Ionic Liquid-Solid Extraction System

Seung-Yeon Cho^{*}, Jung-Weon Choi, Hyo-Jeong Lee, Won-Zin Oh, and Sang-June Choi

Kyungpook National University, 80, Daehak-ro, Buk-gu, Daegu, Republic of Korea

^{*}tmddus2049@knu.ac.kr

1. Introduction

In our previous study, we reported that Cs⁺ solid-liquid separation was observed when a small quantity of the ionic liquid(IL) was used for Cs⁺ extraction from liquid radioactive waste. The Cs⁺ selective separation performance of the IL extraction system in the study was evaluated by comparing with that of the conventional Cs⁺ selective adsorbents such as MWCNTs-EDA-CuFC (adsorbent-1) and mag@silica-CuFC (adsorbent-2) prepared by Lee et al. [2,3].

2. Experimental

2.1 Materials and reagents

Cesium nitrate (CsNO₃, 99%) and the crown ether extractant dicyclohexano-18-crown-6 (DCH18C6, 98%) were purchased from Sigma-Aldrich Chemical Co. (Germany). The ionic liquid 1-ethyl-3-methylimidazolium bis(trifluoro-methylsulfonyl)imide (C₂mimTf₂N) was purchased from C-TRI Co. Ltd. (Korea).

2.2 Experimental method

CsNO₃ solutions of concentrations 0.1, 1, 5, 10, 15 and 20 mM were prepared. DCH186 is dissolved to C₂mimTf₂N in the conical tube. The mixture was shaken with 40 mL of CsNO₃ solution for Cs⁺ extraction. After extraction, the Cs⁺ concentration of aqueous phase was analyzed using ICP-OES (Optima 2100 DV, PerkinElmer).

3. Results and discussion

3.1 Cs⁺ separation performance

To compare the quantity of waste with adsorption system, the ratio of removed Cs⁺ to the weight of precipitate assumed as the sorption capacity. In the Cs⁺ extraction using ILs, the precipitate was produced in almost the same molar quantity as the Cs⁺ removed by the cation exchange mechanism. These ratios have similar value regardless of Cs⁺ concentration. It means that solid-liquid extraction system is especially more efficient at lower concentration of Cs⁺ than adsorption process.

Table 1. Comparison of Cs⁺ separation performance between the IL solid extraction and the conventional adsorbents of adsorbent-1, 2

C ₀ _Cs ⁺ [mmol/L]	q _e [mmol_Cs ⁺ removed/g_waste]		
	ionic liquid Solid Extraction	adsorbent-1	adsorbent-2
0.1	1.2405	0.0423	0.0350
1	1.2685	0.4746	0.3084
5	1.2715	0.9121	0.4863
10	1.2747	1.1253	0.7585
15	1.2732	1.1770	0.6919

3.2 Industrial applicability

The conventional column adsorption process needs pretreatment and considering many factors to design column as shown in table 2. Whereas, the IL solid extraction system only needs injection of ILs and filtration after extraction. Therefore the IL solid extraction system can be considered as much simpler process and easier for industrial application.

Table 2. Comparison of the IL solid extraction system and the conventional column adsorption process

	Column adsorption	IL solid extraction system
Pretreatment	SS/Oil/Turbidity	-
Design factor	1) Particle size of adsorbent 2) Channeling 3) Throughput	1) Amount of ILs and extractant

4. Conclusion

The following conclusions are obtained by comparison of the Cs^+ selective separation performance between the IL solid extraction system and the conventional adsorbents from the liquid radioactive waste treatment point of view:

- 1) Cs^+ is extracted 1:1 molar ratio of extractant regardless of the Cs^+ concentration in the IL solid extraction system. Therefore, q_e value is constant unrelated to the Cs^+ concentration.
- 2) In adsorption system, Cs^+ was removed in equilibrium with Cs^+ concentration in the aqueous solution. Therefore, the lower the Cs^+ concentration, the lower the sorption capacity q_e value of adsorbents compared to that of the IL solid extraction system.
- 3) IL solid extraction system is much easier for industrial applications without the pretreatment of suspended solid particles and oil contamination than the conventional adsorption system.

ACKNOWLEDGEMENTS

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government (MSIP: Ministry of Science, ICT and Future Planning). (NO.2012M2B2B1055496)

REFERENCES

- [1] Su-Min Kwon et al., "A novel method for separating Cs^+ from liquid radioactive waste using ionic liquids and a selective extractant", Journal of Radioanalytical and Nuclear Chemistry, 311(3), 1605-1611 (2017).
- [2] Hyun-Kyu Lee et al., "Sorption of cesium ions from aqueous solutions by multi-walled carbon nanotubes functionalized with copper ferrocyanide", Journal of Radioanalytical and Nuclear Chemistry, 309(2), 477-484 (2016).
- [3] Hyun-Kyu Lee et al., "Copper Ferrocyanide Functionalized Core-Shell Magnetic Silica Composites for the Selective Removal of Cesium Ions from Radioactive Liquid Waste", Journal of Nanoscience and Nanotechnology, 16(6), 6223-6230 (2016).

Study on Optimization of Volume Reduction Treatment Process of Domestic Industrial Radioactive Waste

Yongho Hong*, Heekyung Kim, Hyoungmin Park, Eunsuk Choi, and Dongchul Kim
ACT Co., Ltd., 406, IT Venture Town, 3, Techno 9 Ro, Yuseong-gu, Daejeon, Korea

*hyh@actbest.com

1. Introduction

The domestic industrial radioactive waste in this study is powder type waste containing a large amount of moisture, and there is no suitable treatment method for satisfying the disposal condition. In this study, we constructed the treatment process based on the patented technology of Korea Atomic Energy Research Institute. The treatment process consisted of three steps: dissolution and solid - liquid separation, purification and solid - liquid separation, and precipitation for discharging.

In this study, the amount of radioactive waste is reduced by separating the support structure which occupies most of the volume. The radioactive concentration of the separated support structure is reduced to a level that complies with regulations for self-disposal and then discharged to the environment. Secondary wastes generated during this process are treated to be below the release criteria and discharged to the environment so that the volume of radioactive waste for disposal in a radioactive waste disposal facility is significantly reduced and the waste is to be treated to satisfy the waste acceptance criteria for radioactive waste [1].

Therefore, in order to minimize the amount of radioactive materials flowing into the support structure, this study aims to establish an efficient method of operating the solid-liquid separator by optimizing the operation method of the solid-liquid separator.

2. Operation method of solid-liquid separator

The solid - liquid separation is used in two separate processes. In a dissolution process, it is used to separate support structure and the waste. In purification and precipitation processes, it is used to treat the separated support structure for self-disposal. It is also used to treat the secondary waste to satisfy

the release criteria. The operation method of solid - liquid separator varies depending on the purpose. The volume of single channel of the filter plate of solid - liquid separator is 8.5 ℓ, and a 3-channel filter plate was prepared to fill the solids according to the purpose of the study. In the dissolution process, the filter plate was precoated with bentonite, and the purification process was operated without pre-coating [2].

In the dissolving process, the particle size of the medium injected into the solid-liquid separator is very small. The filter plate needs to be protected and smooth flow of liquid should be ensured. In the refining process, the medium injected into the solid-liquid separator was excluded from the pre-coating since the particle size was larger than the medium injected into the solid-liquid separator after the dissolving process. First, the particulate radioactive waste dissolved by the operation method of the solid-liquid separator after the dissolution was injected by opening the lower pipe switch of the dissolution tank immediately after completion of the pre-coating of the filter plate of the solid-liquid separator. The operating pressure of the solid - liquid separator was set to 5 bars, and the dissolved particulate radioactive waste was fed and subjected to dehydration and solid - liquid separation. Secondly, after the purification process, the solid that was precipitated without pre-coating was injected into the solid-liquid separator. The operation pressure of the solid - liquid separator was set to 5 bars, and the solid - liquid separation was performed after washing and dehydration.

3. Optimization of operation method of solid-liquid separator

In order to optimize the operation method of the solid-liquid separator, the main purpose of the first dissolution process is to minimize the amount of the radioactive material flowing into the purification

process after the dissolution process by injecting the undissolved solids into the solid-liquid separator as much as possible. In the second purification process, the main purpose is to minimize the amount of radioactive materials in the solid by cleaning the radioactive material absorbed by the solid material flowing into the solid-liquid separator. The optimization of the operation method of solid - liquid separator has been studied in order to satisfy these objectives and to minimize the amount of secondary waste generated during operation of the apparatus. In the first dissolution process, dissolved particulate radioactive waste was injected by opening the bottom pipe switch of the dissolution tank immediately after completion of pre-coating of the filter plate of the solid - liquid separator to optimize the operation method of the solid - liquid separator. The operating pressure of the solid - liquid separator was set at 5 bars when the dissolved particulate radioactive waste was injected.

The solution discharged from the outlet pipe of the solid-liquid separator was re-fed into the solid-liquid separator again to minimize the particulate radioactive waste support material remaining in the solution.

Fig. 1 (a) is a liquid sample after one solid-liquid separation, and (b) is a liquid sample after two solid-liquid separations. When visually confirmed, turbidity was significantly lower when the liquid was separated twice.

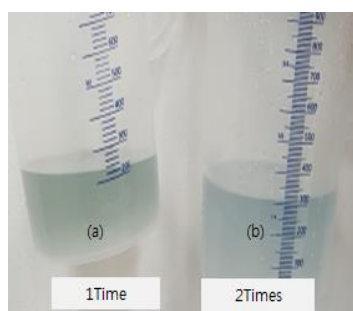


Fig. 1. Filter Press Operation Method.

In the second refining process, the sedimented solid was injected into the solid-liquid separator with the operation pressure set at 5 bars to optimize the operation method of the solid-liquid separator. In order to efficiently remove the radioactive particles remaining in the solid matter through the cleaning process after injecting the solid matter, the solid injected into the solid-liquid separator are repeatedly

washed with water and acid. After that, squeezing and air blowing processes were performed to remove moisture that contains radioactive particles existing in the solid matter. The results are shown in Fig. 2.



Fig. 2. Silica cake after filter press.

4. Conclusion

In order to optimize the operation method of the solid - liquid separator, the turbidity of the solution according to the single operation and the double operation of the solid - liquid separator was visually confirmed. After dissolving in a dissolution tank, the solution produced in the process of separating the undissolved solids from the solid-liquid separator was fed into the solid-liquid separator again. As shown in Fig. 1, there was change of turbidity. The load of the filter installed upstream of the purification process is reduced. The possibility of containing the radioactive particles in the purification process was considerably reduced.

In addition, it was confirmed that the liquid containing the radioactive particles can be sufficiently removed from the inside of the solid material through dehydration and blowing after the solid material generated in the purification process is injected into the solid-liquid separator and then repeatedly washed with water and acid.

5. Acknowledgment

This study was carried out as a project of nuclear science research and development project of Ministry of Science, ICT and Future Planning.

REFERENCES

- [1] KAERI. Patent Publication, 2013.
- [2] Korea Filter Co., Ltd. (2017) Solid-liquid Separator Manual, pp.12-15.

Preliminary Assessment of Potential Risk From Inadvertent Recycling of Disposed Activated Metal Waste and Its Implications

Wang Hyeon Lee, Ki Nam Kwon, and Jae Hak Cheong*

Kyung Hee University, 1732 Deokyoungdaero, Giheung-gu, Yongin, Gyeonggi-do, Korea

*jhcheong@khu.ac.kr

1. Introduction

Not only surface contaminated metal waste but activated metal waste is generated due to neutron activation of metallic structure around reactor core where neutron flux is maintained during the operation of a nuclear power plant. The International Atomic Energy Agency (IAEA) estimates the amount of activated metal waste is to be 650 ton from the decommissioning of a PWR of 900-1300 MWe [1].

Most of the surface contaminated metal waste can be recycled if the clearance acceptance criteria (concentration or dose) can be met with or without surface decontamination. However, much of the activated metal waste that exceeds the clearance level is to be disposed of at a radioactive waste disposal facility due to the difficulty of decontamination of volumetrically activated matrix.

For metal waste disposed of at a radioactive waste disposal facility, it is expected that the radiation effects and its consequences due to normal evolution scenarios and low probability events are not significantly different from other waste streams. However, the potential recycling of metal waste disposed of at a disposal facility cannot be excluded, since the metallic material inadvertently excavated on the ground through human intrusion for the purpose of exploring resources after the period of post-closure institutional control of disposal facilities can be considered as s valuable resources to be recycled.

Therefore, this study intends to evaluate the expected impacts of inadvertent recycling of activated metal waste after surface disposal and analyzed the implications therefrom.

2. Methodology

In this study, three types of radionuclides such as ^{59}Ni , ^{63}Ni and ^{94}Nb which are present in activated metal waste generated from nuclear power plants were selected through literature review [2]. Radionuclides with short half-lives such as ^{60}Co were not taken into account as the residual radioactivity is expected to be insignificant after hundreds of years of the period of post-closure control of the disposal facility.

The radionuclides present in scrap metal are distributed to the three phases of ingot, slag and dust filter at a certain ratio in the smelting process. Table 1 shows the mass partitioning factor (MPF) and element partitioning factor (EPF) of a metal smelting process.

Table 1. Mass and Element partitioning factor of scrap metal

	By-product	Ingot	Slag	Dust Filter
	MPF	90%	10%	1%
EPF	^{59}Ni	99%	0%	1%
	^{63}Ni	99%	0%	1%
	^{94}Nb	0%	99%	1%

The concentration of elements present in the by-product in the smelting process can be defined by the following equation:

$$C_{i,by-product} = C_{i,scrap} \cdot \frac{EPF_i}{MPF_i} \quad (1)$$

where, i is the radionuclide, $C_{i,by-product}$ is the concentration of radionuclide i in the by-product (ingot, slag, and dust filter), and $C_{i,scrap}$ is the concentration of radionuclide i in the scrap.

Therefore, when the scrap metal with initial radioactivity concentration of 1 Bq/g is subject to a

smelting process, the radioactivity concentration of ^{59}Ni and ^{63}Ni in the ingot is 1.1 Bq/g, which is insignificant, but the radioactivity concentration of ^{94}Nb in the slag is 9.9 Bq/g, which causes considerable concentration in the smelting process.

The radiation dose expected from the scenario to recycle activated metal waste can be evaluated using the RESRAD-RECYCLE Version 3.10 computer code. The code was originally developed by ANL to assess the expected radiation dose for workers in the scrap metal or waste aluminum recycling process and the general population using consumer goods and public goods by a total of 41 scenarios.

3. Results and Discussion

Table 2 shows the calculated radiation dose using the RESRAD-RECYCLE computer code when scrap metal of unit radioactivity concentration (1 Bq/g) is recycled after the post-closure institutional control period of 300 years [3].

Table 2. Results of radiation dose calculation

Radio-nuclide	Exposure dose (mSv/y)		
	Before smelting	Ingot recycle after smelting	Slag recycle after smelting
^{59}Ni	3.24×10^{-8}	2.82×10^{-8}	-
^{63}Ni	9.87×10^{-9}	8.78×10^{-9}	-
^{94}Nb	1.69×10^{-4}	-	1.63×10^{-1}

In cases of ^{59}Ni and ^{63}Ni , the workers handling with scrap metal at the smelting facility turns out to receive the highest dose. The radioactivity concentration equivalent to 1 mSv/y for human intrusion dose criterion is very high (i.e. 10^7 - 10^8 Bq/g), which is much higher compared to the disposal concentration limit (7.4×10^4 and 1.11×10^7 Bq/g, respectively) for the low-level radioactive waste repository. In other words, the potential impact of recycling of metal waste is already covered in the existing disposal safety assessment framework.

However, the radioactivity concentration of ^{94}Nb equivalent to 1 mSv/y was estimated to be 10 Bq/g, which is lower than the disposal concentration limit

(i.e. 111 Bq/g). In reviewing the calculation process, the highest radiation dose was calculated in the scenario of slag recycling. The radiation dose prior to the smelting process was calculated to be insignificant, since ^{94}Nb in scrap metal is mostly partitioned to the slag phase in the smelting process.

4. Conclusion

In this study, the potential impacts of inadvertent recycling of low-level activated metal waste disposed at a near-surface disposal facility using the RESRAD-RECYCLE computer code.

In the case of radionuclides such as ^{94}Nb , it was shown that the disposal concentration limit for the low-level radioactive waste repository may not be sufficient enough to warrant the protection of the potential intruders. More detailed studies on the recycling scenarios of activated metal waste disposed at a near surface disposal facility may be needed especially in preparation of decommissioning of nuclear power plants where large amount of activated metal waste will be generated.

REFERENCES

- [1] International Atomic Energy Agency, "Managing Low Radioactivity Material from the Decommissioning of Nuclear Facilities", IAEA Technical Reports Series No. 462 (2008).
- [2] Pacific Northwest Laboratory, "Long-Lived Activation Products in Reactor Materials", NUREG/CR-3474 (1984).
- [3] Enforcement Decree of the Nuclear Safety Act, Article 99(2) (2017).

Feasibility Study on the Development of Position Sensitive System Based on Plastic Scintillating Fiber for Detecting the Leak Position of Radioactive Waste Drum

Young Beom Song¹, Hye Jin Kim¹, Sang Hun Shin¹, and Bongsoo Lee^{1*}

¹ School of Energy Systems Engineering, Chung-Ang University, Heukseok-dong, Dongjak-gu, Seoul, Korea

*Bslee@cau.ac.kr

1. Introduction

A low and intermediate level radioactive waste has been produced from commercial nuclear power plants, research institutes and nuclear fuel manufacturing facilities [1]. This radioactive waste is stored in a drum and moved to a storage. As this radioactive waste is increasing, it is important to manage this radioactive waste. In order to handle safely this radioactive waste in the storage and efficiently reduce risks of contamination due to the radioactive materials, a monitoring system to detect a leak position of radioactive waste drum is needed [2].

In this study, we fabricated a position sensitive system (PSS) to detect the leak position of radioactive waste drum using a plastic scintillating fiber (PSF), which was cracked at regular intervals by using the blades of fiber-optic stripper.

2. Materials and Methods

2.1 Sensing material

As a sensing material to detect the leak position of the radioactive waste drum, the plastic scintillating fiber (PSF; BCF-12, Saint-Gobain Ceramic & Plastics) was used. The PSF is a cylindrical shape and composed of core and cladding, similar to a commercially optical fiber. The material of the core is polystyrene (PS) synthesized with fluorescent dopants and the cladding material is polymethylmethacrylate (PMMA). The refractive indices of the core and cladding are 1.60 and 1.49, respectively. Thus, the numerical aperture (NA) is about 0.58 by the refractive index of materials. The other physical properties of PSF are listed in Table 1.

Table 1. Physical properties of PSF

Physical properties	BCF-12
Emission color	Blue
Emission peak (nm)	435
Decay time (ns)	3.2
1/e length (m)	2.7
# of photons per Mev	~8000

2.2 Experimental setup

Fig. 1 shows the experimental setup to detect a position of radioisotope using the PSF. In order to measure the scintillating lights generated in the PSF, a photon counter system which was composed of a photon counting head (H12525-01, Hamamatsu Photonics) and a photon counting unit (C8855-01, Hamamatsu Photonics) was used. Also, we used plastic optical fiber (POF; GH-4001, Mitsubishi Rayon) for transmitting the scintillating lights produced the PSF to the light-measuring device. The diameter and length of this POF are 1 mm and 1 m, respectively.

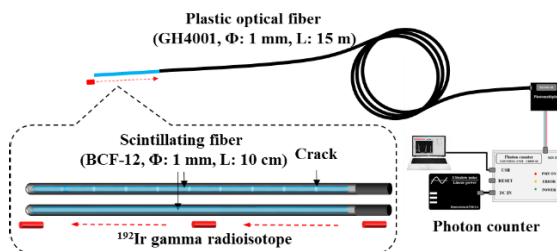


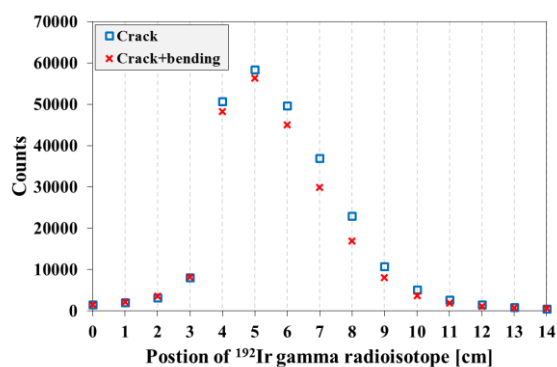
Fig. 1. Experimental setup to detect the position of the radioisotope.

To evaluate the performance of the PSF, ¹⁹²Ir gamma radioisotope was moved at intervals of 10 mm and measure the scintillation lights according to the location of ¹⁹²Ir gamma radioisotope.

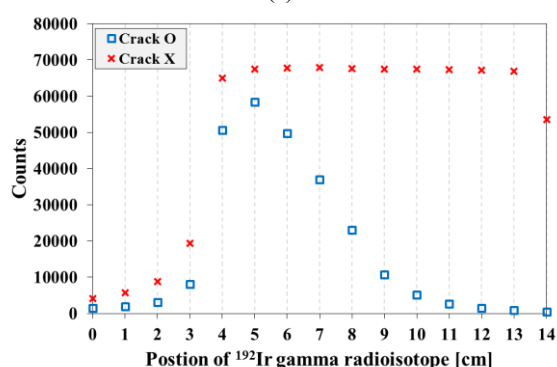
3. Results

Fig. 2 shows the counts of the scintillating lights generated on the PSF according to the position of ^{192}Ir gamma radioisotope. In Fig. 2 (a), as the position of ^{192}Ir gamma radioisotope is changed, the amount of scintillating lights was measured differently because of the increasing the number of crack. On the other hand, in the case of the PSF which doesn't have the cracks, the scintillating lights was not changed. In Fig. 2 (b), when the PSF wasn't bent, the scintillating lights were changed smaller than the bent PSF.

From the experimental results, we can detect the leak position of radioactive waste drum through the scintillating lights generated in the PSF.



(a)



(b)

Fig. 2. Counts of the PSF according to the position of the ^{192}Ir gamma radioisotope. ((a) With the existence of crack, (b) With the existence of bending)

4. Conclusion

In this study, we developed the PSS, which is composed of the PSF and the photon counter system,

to detect the leak of radioactive wastes. The experimental results show that the proposed PSS measured the different scintillating lights according to the position of ^{192}Ir gamma radioisotope. Thus, the PSS can detect the leak position of radioactive wastes by measuring the intensity of scintillating lights. It is expected that the PSS can be used as a useful monitoring system to detect a leak position of radioactive waste drum.

Further studies are planned to fabricate a PSF with the different intervals and deep of cracks for optimization of PSS. Then, we will measure the scintillating lights using the fabricated PSF and select the best PSF.

ACKNOWLEDGMENTS

This research was supported by National Nuclear R&D Program through the National Research Foundation of Korea (NRF) funded by the Ministry of Science, ICT and future Planning (No.2016M2B2B1945255) and this research supported by Basic Science Research Program through the National Research Foundation of Korea (NRF) funded by the Ministry of Science, ICT and future Planning (No.2017R1A2B2009480).

REFERENCES

- [1] Jin Beak Park, Haeryong Jung, Eun Young Lee, Chang Lak Kim, Geon Young Kim, Kyung Su Kim, Yong Kwon Koh, Kyung Woo Park, Jae Hak Cheong, Chan Woo Jeong, Jong Soo Choi and Kyung Deok Kim, "WOLSONG LOW- AND INTERMEDIATE-LEVEL RADIOACTIVE WASTE DISPOSAL CENTER: PROGRESS AND CHALLENGES", Nuclear Engineering and Technology, 41(4), 1-16, (2009).
- [2] A.Pappalardo, C.Cali, L.Cosentino, M.Barbagallo, G.Guardo, P.Litrico, S.Scirè, C.Scirè and Paolo Finocchiaro, "Performance evaluation of SiPM's for low threshold gamma detection", Nuclear Physics B-Proceedings Supplements, 215(1), 41-43 (2011).

A Conservative Approach to Estimate the Gaseous Radionuclide Migration in LILW Repository

Jung-Woo Kim^{1*}, Jaewon Lee¹, Ji-Hun Ryu¹, Jongtae Jeong¹, Dong-Keun Cho¹, and Jeheon Bang²

¹ KAERI, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Republic of Korea

² KINS, 62 Gwahak-ro, Yuseong-gu, Daejeon, 34142, Republic of Korea

* jw_kim@kaeri.re.kr

1. Introduction

The second-phase geological disposal of low- and intermediated-level radioactive waste (LILW) in Korea has been designed as a trench-type near surface disposal facility [1]. In the safety assessment of the LILW repository, the gaseous phase should additionally be considered in the radionuclide migration pathway along with the liquid phase. In fact, Hwang et al. [2] pointed out that the gaseous H₂ and CH₄ due to corrosion of concrete or iron drums and the radioactive gases due to the radiolysis are one of the main factors affecting the safety of LILW repository. In this study, the modeling studies related to the generation and migration of gaseous radionuclides in the LILW repository have been reviewed. Based on a stepwise approach which was recommended by the US NRC [3], in addition, a conservative approach using a puff release model was conducted as the first-step safety assessment of the gaseous radionuclide migration.

2. Literature Review

2.1 Modeling of Gas Generation

In 1996, Cho et al. [4] reviewed a mathematical model to predict the incidence of gas generated by metal corrosion and microbial degradation in LILW repositories. In the study, the corrosion of the metal material considered as the main factor of H₂ gas generation was classified into three stages; 1) oxidation of iron, 2) reduction of oxide film after depletion of oxygen, and 3) anaerobic corrosion of iron. And, in the gasification by microbial degradation, the gas generation rates, such as H₂, CO₂, and CH₄, were calculated according to the pH and the chemical species. The model used for the calculation was GAMMON developed by AEA Technology, UK. As a result of calculation, it was confirmed that the gas generated in LILW disposal system has the greatest amount of H₂.

In 2008, Kang et al. [5] reviewed the main mechanisms of gas generation from LILW, and evaluated the gas generation rate and the amount using SMOGG developed by Radioactive Waste Management Ltd., UK. The main mechanisms of gas

generation considered in SMOGG are metal corrosion, radiolysis, biological degradation of organics, and radioactive decay.

In a LILW repository in Finland, a field-scale experiment of gas generation was conducted, and the experimental data, such as gas generation rate, gas species, geochemical characteristics, bacteria, etc., were collected for more than 9 years [6]. The experiments were simulated by using a biogeochemical reactive transport model (Generalized Repository Model; GRM) developed by British Nuclear Fuels Ltd., UK.

2.2 Stepwise Approach

The PA Working Group of the US NRC proposed a stepwise approach to evaluate the release of gaseous radionuclides (e.g., ¹⁴C, ⁸⁵Kr, ²²²Rn, ³H, ¹²⁹I) in the safety assessment of LILW repositories, taking into account the accuracy or complexity of the conceptual model as follows [3]:

- 1st-step (selective method): As the most conservative method, it is assumed that all disposal facilities are failed at the same time and all radionuclides are released to the ground surface (= puff release) in gas phase over a short period of time. Here, for the main radionuclides contributing to the final dose, the gas phase fraction in the actual inventory is considered selectively.

- 2nd-step: Considering the distribution of the gas-liquid phases of the radionuclides reasonably, the amount of the radionuclide migrating into the gas or liquid phase is distinguished.

- 3rd-step: The gas-liquid phases radionuclide distribution is implicitly calculated taking into account the changes of the geochemical characteristics, rather than being explicitly considered like 2nd-step.

- 4th-step: As the most complex and realistic method, it considers the types and forms of various wastes, the failure mechanism of complex disposal facilities, and various mechanisms that affect gas migration.

3. A Conservative Approach

3.1 Layout of the Repository

The layout of the disposal vaults in the second-

phase geological repository of LILW in Korea is depicted in Fig. 1.

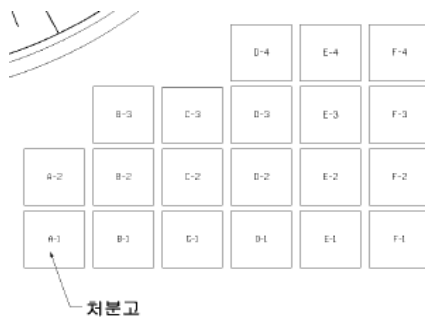


Fig. 1. Layout of the repository.

3.2 Puff Model

As the 1st-step safety assessment of the gaseous radionuclide migration in the repository, the most conservative approach was developed using a puff model. The puff model is generally based on the multivariate normal distribution function as follows:

$$f(\mathbf{x}) = \frac{1}{(2\pi)^{d/2} |\Sigma|^{1/2}} \exp \left\{ -\frac{1}{2} (\mathbf{x} - \boldsymbol{\mu})^T \Sigma^{-1} (\mathbf{x} - \boldsymbol{\mu}) \right\} \quad (1)$$

where $\mathbf{x}$ is the location vector, $f(\cdot)$ is the density function, d is the dimension of the coordinate, $\boldsymbol{\mu}$ is the mean vector, and Σ is the covariance matrix.

3.3 Conservative Assumption

- All disposal facilities are failed at a certain time after the repository is closed.
- All radionuclides are available to migrate to the ground surface in gas phase following the puff model as soon as the repository is failed.

3.4 Radionuclide Distribution at the Ground Surface

From an example result (Fig. 2), the radionuclide distribution at the ground surface could be estimated based on the arbitrarily-defined inventory distribution for each disposal vaults using the puff model (= trivariate normal distribution).

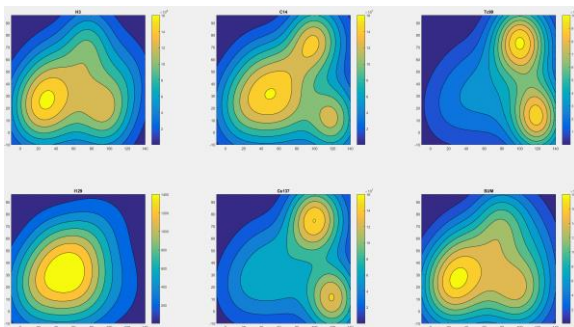


Fig. 2. Radionuclide distribution at the ground surface.

4. Conclusion

The 1st-step safety assessment conducted in this study needs to be deepened by considering the movement of the puff, the various failure time, the selection of gas phase radionuclides, and the gas phase fraction in the actual inventory. From this, the conservatives will be decreased.

ACKNOWLEDGEMENT

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. 1703005)

REFERENCES

- [1] J.H. Bang et al., "Development of Two-Dimensional Near-field Integrated Performance Assessment Model for Near-surface LILW Disposal", JNFCWT, 12(4), 315-334 (2014).
- [2] Y.S. Hwang et al., "Evaluation of Gas Generation and movement at Radioactive Waste Repository", Proc. of the KRS 2007 Fall Conference, pp. 191, 2007.
- [3] US NRC, A Performance Assessment Methodology for Low-Level Radioactive Waste Disposal Facilities: Recommendations of NRC's performance Assessment Working Group, US NRC's NUREG-1573 (2000).
- [4] C.H. Cho et al., "Assessment of gas generation in underground repository of low-level waste", Journal of the Korean Nuclear Society, 28(1), 79-92 (1996).
- [5] C.H. Kang et al., "Modeling of Gas Generation Evaluation at Radioactive Waste Repository", Proc. of the KRS 2008 Fall Conference, pp. 181-182, 2008.
- [6] J. Small et al., "Experimental and modelling investigations of the biogeochemistry of gas production from low and intermediate level radioactive waste", Applied Geochemistry, 23, 1383-1418 (2008).

5분과

제염해체 (Oral)



Laser Cutting Performances of Thick Stainless Steel Plates With a High-Power Fiber Laser for Dismantling of Nuclear Power Plants

Jae Sung Shin,* Seong Yong Oh, Hyunmin Park, Sangwoo Seon, Chin-Man Chung, Taek-Soo Kim, Lim Lee,
and Jonghwan Lee

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

*jsshin12@kaeri.re.kr

1. Introduction

Laser cutting is a widely used technology in industry. Recent advances in laser technology have led to the development of a laser capable of optical fiber delivery up to a power level of ~100 kW, and attempts have been made to apply it to facility dismantling. Particularly, in the dismantling of nuclear power plants, laser cutting has been considered as a next generation technology with many advantages. The reason for this is that since the laser is delivered by the optical fiber the small head is placed in the work space. Moreover, it is a non-contact type cutting, so that there is no reaction force. Thus, it is very advantageous for cutting through remote control which is essential because the core facilities of nuclear power plants cannot be accessed by human workers because they have high radiation. In addition to the advantage of remote cutting, the laser cutting has a narrower kerf width compared to other cutting techniques. Thus, the laser cutting can reduce the amount of secondary waste. It is very important to minimize the amount of secondary waste in dismantling nuclear facilities.

For this reason, researchers have been progressing worldwide to develop laser cutting technology. Dismantling of nuclear power plants requires cutting capability for steels with a thickness of 100 mm or more. Therefore, researchers in this field are making efforts to improve the cutting capability [1-5]. Our group has also developed the laser cutting technology. A laser cutting head capable of cutting thick steel was developed. It succeeded in cutting a stainless steel plate up to 100 mm by cutting performance test using a 6-kW fiber laser [4, 5].

In this work, a further cutting study was performed using a higher power laser of 10 kW. Stainless steel plates were cut up to a thickness of 100 mm based on

the generally 10-mm cut capability per kW, and the maximum cutting speeds against the thicknesses were obtained.

2. Laser cutting experiment

2.1 Experimental setup

Fig. 1 shows the view of the laser cutting experiment. The cutting started from the side of the specimen and proceeded with the cutting head moving from the right to the left. When the laser was irradiated on the specimen, the assisting gas was supplied simultaneously. A compressed air was used as an assisting gas for blowing off the melt. The gauge pressure was ~1MPa and the gas flow rate was measured to be 470 L/min expressed under ANR (Atmosphère Normale de Référence) condition (20°C, 101.3 kPa, 65% relative humidity).

As specimens, 60-, 80-, 100-mm thick stainless steel plates (SUS304L) were used. Cutting was performed until each line was not cut while changing the cutting speed, thereby obtaining the maximum cutting speed.



Fig. 1. View of the laser cutting experiment. The cutting started from the side of the specimen and proceeded with the cutting head moving from the right to the left.

Fig. 2 shows the maximum speeds against the thicknesses. For a thickness of 100 mm, the maximum cutting speed was measured to be 30 mm/min. Although a power of 6 kW was able to cut a thickness of 100 mm in our previous work [5], the cutting speed was very low. The maximum cutting speed was measured to be 7 mm/min for a 6-kW laser cutting. Compared to the result of 6-kW laser cutting, the maximum cutting speed was ~4.3 times higher.

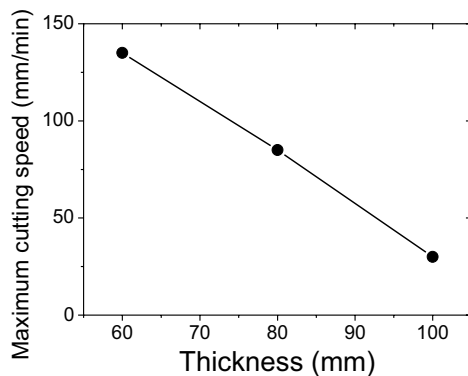


Fig. 2. Maximum cutting speeds against the thicknesses for cutting of stainless steel plates with a 10-kW fiber laser.

3. Conclusion

A cutting study was performed using a higher power laser of 10 kW. As specimens, 60-, 80-, 100-mm thick stainless steel plates were cut. And the maximum speeds against the thicknesses were obtained. For a thickness of 100 mm, the maximum cutting speed was measured to be 30 mm/min. This result can be used as a basic data for laser cutting application to dismantling of nuclear power plants.

ACKNOWLEDGEMENT

This work was supported by the national research foundation of Korea (grant no.: NRF-2017M2A8A5015146), funded by the Ministry of Science, ICT & Future Planning, Republic of Korea.

REFERENCES

- [1] K. Tamura, R. Ishigami, and R. Yamagishi, "Laser cutting of thick steel plates and simulated steel components using a 30 kW fiber laser," *Journal of Nuclear Science and Technology*, 51, 916-920 (2016).
- [2] C. Chagnot, G. de Dinechin, and G. Canneau, "Cutting performances with new industrial continuous wave ND:YAG high power lasers: for dismantling of former nuclear workshops, the performances of recently introduced high power continuous wave ND:YAG lasers are assessed," *Nuclear Engineering and Design*, 240, 2604–2613 (2010).
- [3] P. A. Hilton, and A. Khan, "Underwater cutting using a 1μm laser source," *Journal of Laser Applications*, 27(3), 032013 (2015).
- [4] J. S. Shin, S. Y. Oh, H. Park, C.-M. Chung, S. Seon, T.-S. Kim, L. Lee, B.-S. Choi, and J.-K. Moon, "High-speed fiber laser cutting of thick stainless steel for dismantling tasks," *Optics and Laser Technology*, 94, 244-247 (2017).
- [5] J. S. Shin, S. Y. Oh, H. Park, C.-M. Chung, S. Seon, T.-S. Kim, L. Lee, and J. Lee, "Laser cutting of steel plates up to 100 mm in thickness with a 6-kW fiber laser for application to dismantling of nuclear facilities," *Optics and Lasers in Engineering*, 100, 98-104 (2018).

The Preparation of NiFe_2O_4 Thin Film Prepared by E-Beam Evaporation Method for Simulated Specimen in PWR System

Mansoo Choi, Junyoung Jung, Jungsoon Park, Wangkyu Choi, Sangyoon Park, Hui-Jun Won,
and Seon-Byeong Kim*

Decontamination and Decommissioning Research Division, Korea Atomic Energy Research Institute, 111,
Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

*Corresponding author e-mail: sbkim@kaeri.re.kr

1. Introduction

Decontamination technology is necessary to reduce radioactive component and contamination of complex equipment during decommissioning of nuclear power plants. There are a number of methods for decontamination process including chemical, electrochemical, foam, and mechanical contamination technologies. Among them, decontamination foam is considered to have potential application for the cleaning of radioactive contaminants from metallic walls, overhead surfaces, and complex components [1]. For the effective decontamination, the understanding of the fundamental mechanisms responsible for corrosion and stress corrosion of austenitic and nickel base alloys is necessary. Austenitic 304L and 316L stainless steel owe their high corrosion resistance in water to the formation of a continuous oxide layer, the passive film, only a few nanometer thick at ambient temperature, strongly enriched in $\text{Cr}(3^+)$ and separating and protecting the alloy from the corrosive medium [2,3]. Many researches have been investigated the influences of various parameters such as water chemistry, materials composition and stress, on the initiation and growth of stress corrosion cracking. Most studies reported the observation of a duplex oxide layer with a Fe-rich outer layer and a Cr-rich inner layer. The outer oxide layer is composed of magnetite (Fe_3O_4) or iron-nickel spinel oxide. For the Cr-rich inner layer, composition and phase are subject to controversy.

In this study, we prepared the NiFe_2O_4 film on the stainless foil (SUS 304) using E-beam evaporation technique using NiFe_2O_4 target. Compared to the other technique, the materials can be simply deposited by E-beam evaporation and is capable of providing advantages such as large area deposition, convenient replacement of source targets, and controllability of precise deposition thickness as low as 1 nm.

2. Experimental

NiFe_2O_4 thin film was deposited on stainless foil (SUS 304, thickness 25 μm) by using an electron beam evaporation system at room temperature. NiFe_2O_4 pellet (purity 99.9%) was used as a target. Prior to deposition process, the chamber was evacuated down to 3×10^{-6} Torr. The deposition rate and thickness of NiFe_2O_4 thin film were 0.5 $\text{\AA}/\text{s}$ and about 490 nm monitored by a thickness sensor during the evaporation process. The substrates were sputter-etched with Ar ions for 5 min before deposition in order to remove any oxide layer on the SUS foil surface. For the high crystallinity of NiFe_2O_4 thin film, as-deposited sample was annealed at 700 $^\circ\text{C}$ for 1h in Ar atmosphere.

The morphology of the surface and cross-section of NiFe_2O_4 thin film was investigated by a field emission scanning electron microscopy (S-4800, Hitachi) working at 30 kV. A high resolution transmission electron microscope with energy dispersive spectroscopy (JEOL-2100F HRTEM) with a 200 kV operating voltage was also used to capture the morphology and elemental analysis of the NiFe_2O_4 sample. The thin film X-ray diffraction (Xpert PRO MRD, Philips) pattern was conducted with Cu $K\alpha$ radiation ($\lambda = 1.5406 \text{ \AA}$) operating at 40 kV and 30 mA between 10° and 90° at a scan rate of 0.01° , $2\theta/\text{min}$. Raman measurement (NTEGRA SPECTRA, NT-MDT) was conducted with a Ar laser excitation source emitting at wavelength of 514 nm. The X-ray photoelectron spectroscopy (K-alpha, Thermo VG Scientific) analysis was performed to evaluate the chemical status of the thin films and the binding energy was referenced to the C 1s peak from carbon at 284 eV.

3. Result and discussion

The crystallographic structure and phase purity of

as-prepared sample are examined by X-ray diffraction (XRD). Fig. 1 shows the XRD patterns of NiFe_2O_4 thin film after heat treatment at 700°C for 1h. The XRD pattern of as-deposited film was also shown for comparison. Three broad peaks with low intensity at 37° , 43° , and 62° in the as-deposited sample were observed, which is corresponding to the amorphous phase of the Ni and Fe oxide. The amorphous phase is could be due to the low energy for crystal formation during deposition process. However, when the sample was heated at 700°C , the film began to crystallize at high temperature as indicated by the most intense peak (311). This is assigned to those of the NiFe_2O_4 cubic spinel, which could be indexed to the inverse spinel (NiFe_2O_4 , JCPDS Card # 10-0325). The spinel oxide (440) peak indicated the spinel oxide dominated by FeCr_2O_4 . No peaks related to any other phase or impurities could be observed. After the heat treatment, the reflections become narrower with an increase of crystallite size. The impurity peak at 45° was also observed, indicating the stainless foil (denoted by star).

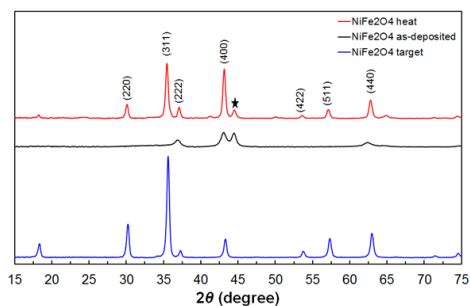


Fig. 1. XRD patterns of as deposited and heat treatment of NiFe_2O_4 and NiFe_2O_4 target.

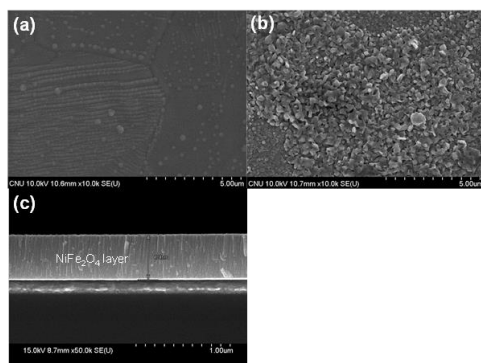


Fig. 2. SEM images of the NiFe_2O_4 thin film; (a) before and (b) after heat treatment, (c) cross-sectional view of NiFe_2O_4 thin film, respectively.

SEM images in Fig. 2a-c clearly show the surface of the as-deposited NiFe_2O_4 thin film and cross-section image, respectively. Fig. 2a and b shows the

surface morphology of NiFe_2O_4 thin film before and after heat treatment at 700°C . As shown in Fig. 2a, NiFe_2O_4 thin film present smooth surface along with base metal and a few droplets is observed due to the incomplete elimination of target splashing during deposition. After heat treatment, the NiFe_2O_4 thin film shows different morphology suggesting that the film may be undergoing change of orientation. The crystallized grains with increase in size are also observed. For the thickness measurement, the NiFe_2O_4 thin film was also deposited on indium tin oxide (ITO) glass. Fig. 2c shows the NiFe_2O_4 thin film which is vertically grown on the ITO glass without any cracking and thickness of film was about 484 nm.

4. Conclusion

Equations should be placed on separate lines and numbered:

REFERENCES

- [1] L. Nunez, L. Kaminski, Foam and gel methods for the decontamination of metallic surface, US patent, 166, 758 (2007).
- [2] V. Maurice, W.P. Yang, P. Marcus, J. Electrochem. Soc. 145, 909 (1998).

A Study of Waste Treatment in SP-HyBRID Decontamination Process Using On-Line Monitoring

J. Y. Jung^{a,b*}, S. Y. Park^a, H. J. Won^a, W. K. Choi^a, S. B. Kim^a, J. K. Moon^a, and S. J. Park^b

^aKorea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon, Korea

^bDepartment of Applied Chemistry and Biological Engineering, Chungnam National University,
99 Daehak-ro Yuseong-gu, Daejeon, Korea

*jjy87@kaeri.re.kr

1. Introduction

In general, decontamination of nuclear power plants usually generates highly radioactive wastewater. Current radioactive waste should be stored in a safe radioactive waste disposal facility, since complete removal in a short period of time is not possible. The amount of radioactive waste product can be a very sensitive issue because it must be stored in a limited processing facility. Therefore, it is one of the most important factors to minimize the amount of waste product in the wastewater treatment after decontamination. SP-HyBRID [1], decontamination process developed by KAERI, was developed to solve the problem of material damage [2] and wastewater treatment of commercial decontamination agent using organic acids. In the SP-HyBRID process, a wastewater containing sulfate, hydrazine and various metals including radioactive cobalt is formed after decontamination. Sulfate ion and metal ion are removed by precipitation by adding barium hydroxide, and hydrazine is removed by adding hydrogen peroxide. In this study, when the wastes produced after the SP-HyBRID process were removed by the above described method, the research for effective removal method of ions in wastewater and the determine the amount of additive to minimize waste were conducted. Real time process monitoring was used to minimize waste generation.

2. Experimental Methods

The wastewater produced by the SP-HyBRID process was prepared for the experiment. It was assumed that iron, chromium and nickel, the main components of the oxide film of the PWR primary system which is subject to decontamination, was dissolved. This assumption was applied to make a simulated wastewater similar to the real decontamination wastewater.

Table 1. Condition of Simulated Wastewater

SP-HyBRID	6.33 mM KMnO ₄ , 37.02 mM H ₂ SO ₄ , 57.91 mM N ₂ H ₄ , 0.5 mM CuSO ₄
Dissolved metal ions	Fe 50 ppm, Cr 50 ppm, Ni 25 ppm

In this study, the process of precipitation of wastewater ions was divided into two steps. The first step was addition of barium hydroxide for metal and sulfate ion removal. Barium hydroxide was dissolved in water and added to the solution. The second step was to remove the hydrazine by adding hydrogen peroxide. During the addition of the barium hydroxide solution and the hydrogen peroxide, the pH, ORP and conductivity of the wastewater were monitored in real time. A small amount of barium hydroxide solution and hydrogen peroxide were added periodically. The solution was sampled at each interval. The concentrations of various metal ions, sulfate ions and hydrazine in the sample solution were analyzed using ICP and UV.

3. Results

The results of monitoring recorded during decontamination waste treatment are shown in Fig. 1. As the process progressed, a change in each value was observed. Results of analyzing the components of each solution are shown in Fig. 2 and Table 2. Since the concentration difference of each ion was large, most of ions is represented by the removal rate, and barium ion is expressed by concentration unit in Fig. 2. Table 2 shows the concentration of hydrazine in the solution before and after the addition of hydrogen peroxide. Monitoring data was interpreted by each step through Fig. 2 and Table 2.

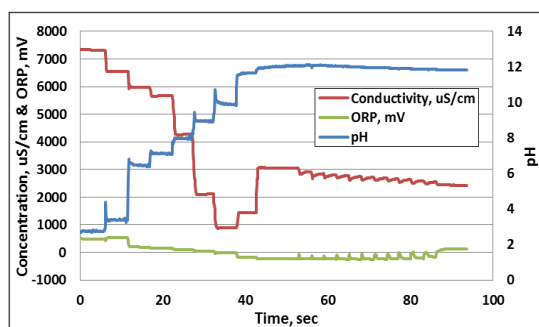


Fig. 1. Monitoring Data on pH, ORP and Conductivity during Wastewater Treatment Process.

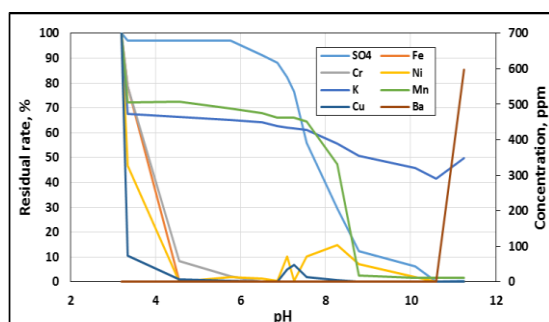


Fig. 2. Behavior of Ions in Wastewater by Addition of Barium Hydroxide.

Table 2. Concentration of Hydrazine before and after Addition of Hydrogen peroxide

Condition	Concentration, ppm	Remove efficiency, %
Before addition of hydrazine	1442	0
After addition of hydrazine	0.1	99.99

In the barium hydroxide addition step to remove metal and sulfate ions, the pH was increased by barium hydroxide. As the pH increased, most of the dissolved metal ions precipitated. pH range of the precipitated metal ions was different. It can be seen that the precipitation of metal ions depends on the solubility in each pH range[3]. But potassium was not removed in wastewater. It was assumed that potassium was not removed due to the absence of precipitation by the formation of hydroxides at high pH. To remove potassium, an ion exchange resin process should be added. And it was also confirmed that sulfate ions were precipitated to barium sulfate. Barium ions were detected in the treated solution because of excessive addition of barium hydroxide after the removal of the sulfate ions. This phenomenon is the same as the result of conductivity monitoring. In the addition step of hydrogen peroxide for hydrazine removal, the addition of hydrogen

peroxide increased the ORP. In the addition step of hydrogen peroxide for hydrazine removal, graph shows that ORP increases and decreases with each addition of hydrogen peroxide. This phenomenon appears to be due to the presence of hydrazine in solution. When hydrazine was removed by more than 99%, it was confirmed that the ORP did not decrease any more.

4. Conclusion

As a result of monitoring and analyzing solutions at each step of wastewater treatment, it can be seen the removal state of the ions dissolved from the conductivity monitor. And, since the precipitation of metal ions was dependent on the pH, it was confirmed that the metal removal interval can be predicted using pH monitoring. The residual amount of hydrazine can be deduced through ORP monitoring. In this way, it is possible to judge the optimum amount of the additives necessary for treatment of wastewater in process in real time. This is a very effective method in minimizing the amount of final waste. Therefore, it is expected that it will help to perform efficient and accurate wastewater treatment at low cost.

ACKNOWLEDGE

The authors would like to express their appreciation to the National Research Foundation of Korea (NRF) for the award of a grant funded by the Ministry of Science, ICT and Future Planning (MSIP) of the Republic of Korea, in support of the authors' work through the Nuclear Research and Development Program (NRF- 2017M2A8A5015144).

REFERENCES

- [1] H. J. Won, W. S. Lee, C. H. Jung, S. Y. Park, W. K. Choi, J. K. Moon, "A Feasibility Study on the Decontamination of Type 304 Stainless Steel by N₂H₄ Base Solution", Asian Journal of Chemistry, Vol. 26, No. 5 (2014).
- [2] D. Bradbury, "Review of Decontamination Technology Development 1977-2000", Water Chemistry of Nuclear Reactor System 8, BNES (2000).
- [3] Alison Emslie Lewis, "Review of metal sulphide precipitation", Hydrometallurgy, 104 222-234 (2010).

Reversibly Cross-Linked and Strippable Hydrogels for the Removal of Radioactive Cesium From Surface

Hee-Man Yang*, Chan Woo Park, Kune-Woo Lee, Ilgook Kim, and In-Ho Yoon
Decommissioning Research Division, Korea Atomic Energy Research Institute, 111,
Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea
*hmyang@kaeri.re.kr

1. Introduction

Radioactive materials have been extensively used in various industrial and research fields, particularly for nuclear research and industry. During their utilization, various surfaces, such as concrete, steel, and paint, in most nuclear facilities become contaminated with radioactive materials. Thus, surface decontamination is a very important technology for radioactive material utilization and even for decommissioning nuclear facilities because surface contamination can be transferred to workers by contact, and it may become airborne if not treated [1]. Meanwhile, the nuclear accident at the Fukushima Daiichi nuclear power station in 2011 unfortunately released large amounts of radioactive cesium into the environment, which also affected various urban structures, such as roofs, building exterior surfaces, and roads in a wide area [2]. Although current surface decontamination methods, including physical or chemical methods, in nuclear facilities are well established, surface decontamination for wide-area remediation after nuclear accidents is still challenging due to some drawbacks of the current surface decontamination techniques [3]. Consequently, new surface decontaminants are still desired for a facile and cost effective preparation and one-step surface decontamination.

2. Experiment

2.1 Fabrication of adsorbent

Magnetic adsorbents were synthesized by following the procedure previously reported by our group [4]

2.2 Fabrication of surface decontaminant

The surface decontaminant was prepared by mixing a Polyvinyl alcohol (PVA) solution containing PB or

adsorbent with a borax solution. First, the PVA was dissolved in an NH_4Cl solution by stirring and heating at 80°C in sealed vials to prevent water from evaporating. Then, the desired amount of PB or adsorbents were also added into the PVA solution. Finally, the aqueous borax solution was added dropwise by stirring (vortex apparatus). In addition, the molar concentration of NH_4Cl in the solution was fixed at 0.1 M.

2.3 Surface decontamination procedure

Cement was deposited on the surface of a planchet (diameter = 4 cm) and then coated with paint (Ilshin Co. Ltd., South Korea) to serve as a model surface. The painted cement was contaminated with ^{137}Cs by dropping and evaporating a known amount of ^{137}Cs solution on the surface. The contaminated surface was treated with decontamination solution. Within a few minutes, an adsorbent/hydrogel film was peeled off the surface 3 h later. The radioactivity of the painted cement before and after treatment was measured using an automatic low-background Alpha/Beta counting system. Then, the adsorbent/PVA-borate hydrogel film was added to 20 g of water, and the adsorbent was magnetically collected using an external magnet.

3. Result and Discussions

The storage modulus (G') of the PB/PVA-borate hydrogel was measured to evaluate the strippable property of the hydrogel at room temperature. Although the PB/PVA-borate hydrogel exhibited a gel-like nature with a strong elastic character, the values of G' and G'' decreased compared with those of the PVA-borate hydrogel (Fig. 3(c) and Table 1). As the wt% of PB increased in the PB/PVA-borate hydrogel, the values of G' and G'' decreased. This is probably due to the expansion of the PVA-borate network due to the electrostatic repulsion between the negatively charged

PB and the negatively charged PVA-borate network induced by the borate ion ($\text{B}(\text{OH})_4^-$) during the PVA-borate complexation [16]. Although the presence of PB in the PVA-borate hydrogel decreased the values of G' and G'' , the values are still high enough that it can be anticipated that the PB/PVA-borate hydrogel has good peeling-off properties that make it suitable for surface decontamination applications.

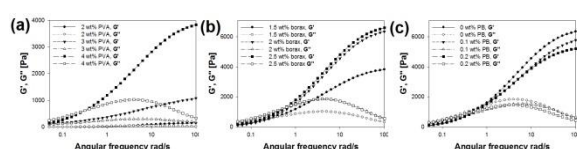


Fig. 1. Frequency sweep curves of the (a) PVA-borate hydrogel formed using 1.5 wt% borax and various wt% PVA solutions, (b) PVA-borate hydrogel formed using 4 wt% of PVA and various wt% borax solutions, and (c) PVA-borate/PB hydrogel formed by using 4 wt% PVA, 2 wt% borax and a 0.1 or 0.2 wt% PB solution.

To evaluate the decontamination ability of the PB/PVA-borate hydrogel, ^{137}Cs -contaminated stainless steel, aluminum, and painted cement were used as model surfaces. After 3 h of application of the PB/PVA-borate hydrogel onto the various contaminated surfaces, the hydrogel was removed from the surfaces. Fig. 5(a) shows that the PB/PVA-borate hydrogel was completely removed from the surfaces in one piece and did not leave any residue. As shown in scheme 1, the ^{137}Cs that was present on the contaminated surfaces was solubilized by the ion exchange of NH_4^+ for Cs^+ and was further captured by the PB, which has an adsorption selectivity for Cs in the hydrogel, until the PB/PVA-borate hydrogel was removed. The DF and R values are listed in Table 1. Surface decontamination using distilled water without the PB/PVA-borate hydrogel was also conducted using the same procedure as a control. The R values for the surface decontamination using water were 53.27% for the painted cement surface, 87.78% for the aluminum surface, and 72.54% for the stainless steel surface. The lower R value for the painted surface compared with the other surfaces is attributed to the rough surface of the painted cement, whereas the other samples had smooth surfaces. All of the R values increased when the PB/PVA-borate hydrogel was used to treat the contaminated surfaces, which indicated that the PB/PVA-borate hydrogel was more effective than conventional surface decontamination using distilled water. These results were attributed to the good surface

exchange properties of NH_4^+ for Cs^+ and the excellent Cs adsorption property of PB.

Table 1. Removal performance of the PB/PVA-borate hydrogel on ^{137}Cs -contaminated surfaces of painted cement, aluminum, and stainless steel (A_0 is the initial radioactivity before decontamination, and A_f is the final radioactivity after decontamination)

		$R = (A_0 - A_f) / A_0 \times 100$	$DF = A_0 / A_f$
Painted cement	Water	53.27	2.14
	Coating	64.91	2.85
Al	Water	83.90	6.21
	Coating	89.85	9.85
Stainless steel	Water	72.54	3.64
	coating	85.09	6.71

4. Conclusion

In this study, a hydrogel displayed a good removal performance for ^{137}Cs -contaminated aluminium and stainless steel surface, and enhanced removal efficiency for painted cement surface compared with water washing. Furthermore, the simple preparation, reusability of hydrogel make the material convenient, eco-friendly, and cost-effective. Therefore, hydrogel has good potential as a new surface decontaminant

Acknowledgement

This research was supported by the Korea Ministry of Education, Science, technology Grant (No.2017M-2A8A5015148)

REFERENCES

- [1] E. Calarese, Proc Natl Acad Sci U S A 108, 19447-19448 (2011).
- [2] N. Kinoshita, K. Sueki, J. Sasa, J. Kitagawa, S. Ikarashi, T. Nishimura, Proc. Natl. Acad. Sci. USA 108, 19526-19529 (2011).
- [3] H. N. Gray, B. Jorgensen, D. L. McClaugherty, A. Kippenberger, Ind. Eng. Chem. Res. 40 3540-3546 (2001).
- [4] Yang, H.-M., Jang, S.-C., Hong, S. B., Lee, K.-W., Roh, C., Huh, Y. S., Seo, B.-K., (2016). J. Alloy. Compd. Vol. 657, p. 387-393.

Cesium Removal From Hydrobiotite by Cationic Polyelectrolyte and Hydrogen Peroxide

Ilgook Kim*, Bo Hyun Kim, Chan Woo Park, In-Ho Yoon, Hee-Man Yang, and Kune-Woo Lee

Decommissioning Technology Research Division, Korea Atomic Energy Research Institute,

111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Republic of Korea

*ilgook@kaeri.re.kr

1. Introduction

Following the accident at Fukushima nuclear power plant in 2011, a considerable amount of research has focused on removing radioactive cesium (Cs) from soil and reducing radioactivity in contaminated area. In particular, Cs ions interact strongly and selectively with 2:1 phyllosilicate clay such as smectite, vermiculite, and illite [1]. Cs is generally adsorbed in the interlayer of these clay minerals, and irreversible adsorption takes places at the frayed edge site (FES), which is highly stable and hardly removable by conventional soil treatment methods. Among the clay minerals, hydrobiotite, interstratified layers of vermiculite and biotite, is produced during weathering of biotite to vermiculite. Vermiculitic clay is known to contain more cesium-specific adsorption sites than illitic clay and the amount of FES can increase as weathering of micaceous minerals proceeds [2].

In this study, therefore, we investigated the behavior of Cs desorption from hydrobiotite by Polyethyleneimine (PEI) as a cationic polyelectrolyte and hydrogen peroxide.

2. Experimental

To prepare Cs-sorbed clay, hydrobiotite was dispersed in an aqueous CsCl solution, and the mixtures were subjected to a horizontal shaker for one week at 25°C. The Cs-sorbed clay was collected by centrifugation at 7000 rpm followed by drying at 40°C. The concentration of Cs in the solution was measured with inductively coupled plasma mass spectrometry (ICP-MS).

For desorption test, Cs-HBT was mixed with DTAB and PEI solution in deionized water and hydrogen peroxide solution with cations. The clay was separated and washed twice with deionized water by centrifugation and was dried at 40°C. The desorbed amount of Cs was analyzed by measuring the Cs concentration in the aqueous phase by ICP-MS.

3. Results & discussion

To analyze mineralogical characteristics of hydrobiotite, the interlayer distance of the clay was investigated by X-ray diffraction (XRD) analysis. Fig. 1 represents the XRD pattern of the dried clay, and a strong peak of hydrobiotite was observed at $2\theta=7.4^\circ$ ($d=12.0 \text{ \AA}$), which is interstratified clay minerals of vermiculite and biotite.

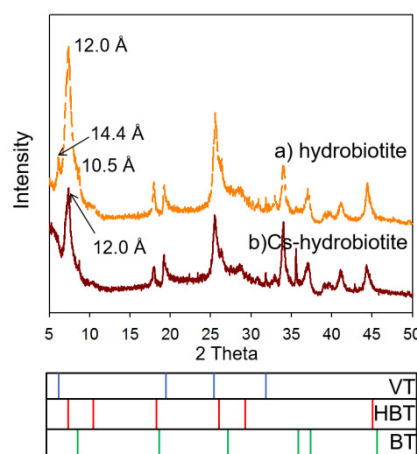


Fig. 1. XRD patterns of (a) raw hydrobiotite and (b) hydrobiotite loaded with Cs at 0.029 mmol/g clay.

To verify the effect of cationic agent on Cs desorption from HBT, the adsorption of desorption agent on clay and Cs desorption were investigated as

a function of desorption agent concentration ranging from 0.16 mmol/g clay to 0.23 mmol/g clay for 24 h. And the pH of the PEI solution was also varied from 3 to 12 (Fig. 2).

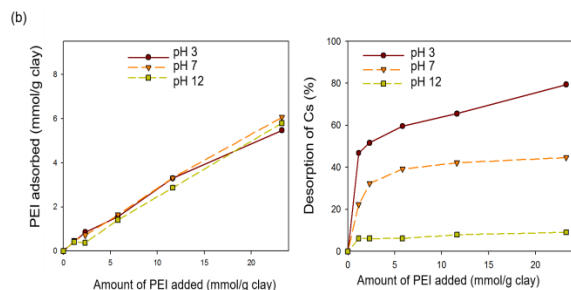


Fig. 2. Adsorption of PEI on Cs-HBT and Cs desorption with various amount of PEI at desired pH for one day.

On the other hand, in order to effect of H_2O_2 on Cs desorption, we tested H_2O_2 solution with cations (NH_4^+ , Mg^{2+}) to desorb Cs from HBT. As listed in Table 1, the percentage of the Cs desorbed from the clay tended to increase as the cations added. Also, 35% H_2O_2 solution with 50 mM Mg^{2+} showed the highest Cs removal efficiency. These results indicated that the expansion of the clay interlayer resulting from the inclusion of oxygen from H_2O_2 and the cation exchange by Mg^{2+} ions had a synergetic effect on the Cs desorption.

Table 1. Comparison of Cs removal efficiency by H_2O_2 .

	35% H_2O_2	35% H_2O_2 / 50 mM NH_4^+	35% H_2O_2 / 50mM Mg^{2+}	35% H_2O_2 / 50 mM NH_4^+ + 50mM Mg^{2+}
Cs-HBT	44.6%	46.4%	72.2%	72.8%

4. CONCLUSIONS

We investigated adsorption of Cs ions on hydrobiotite, and Cs desorption by ion exchange with cationic polyelectrolyte and hydrogen peroxide with cations. The Cs desorption efficiencies achieved from two methods were above 75% on hydrobiotite. This is because cesium ions in the interlayers were replaced by desorption agent by ion-exchange. This

comparative Cs desorption study proved that polyelectrolyte-based desorption and hydrogen peroxide methods will have great potential for remediation of Cs-contaminated soil.

REFERENCES

- [1] Motokawa R., Endo, H., Yokoyama, S., Nishitsuji, S., Kobayashi, T., Suzuki, S., and Yaita, T., "Collective Structural Changes in Vermiculite Clay Suspensions Induced by Cesium Ions." *Scientific Reports*, 4, 6585 (2014).
- [2] Kim, B., Park C.W., Yang, H.M., Seo, B.K., Lee, B.S., Lee, K.W., Park S.J., "Comparison of Cs desorption from hydrobiotite by cationic polyelectrolyte and cationic surfactant." *Colloids and Surface A*, 522, 382-388 (2017).

Weathering Effects on Fractionation Change of Cesium in Soil

Sang-Min Park*, Seunghee Kang, and Kitae Baek

Department of Environmental Engineering, Chonbuk National University, 567, Baekje-daero, Deokjin-gu, Jeonju-si, Jeollabuk-do, Republic of Korea

*sangmin@jbnu.ac.kr

1. Introduction

After nuclear accidents in Chernobyl and Fukushima, concerns about the radionuclides, strontium(Sr), iodine(I), and cesium(Cs), have increased. In particular, Cs is one of the most important dose-forming long lived radionuclides because of relatively long half-life (30 years) and its emission of gamma radiation [1]. When it is exposed to soil, Cs is strongly adsorbed on clay minerals due to the specific adsorption sites in clay minerals [2]. In general, Cs on soil exists as a residual fraction among four fractions: exchangeable, reducible, oxidizable, and residual fraction [3]. The fractionation of Cs could be changed by exposure time or repeated wet/dry conditions, that is, an ageing effect. In this study, ageing effects were investigated due to exposure time and repeated wet/dry conditions using soil samples and clay minerals.

2. Materials and Methods

2.1 Materials

In this study, 5 type soil samples, 3 type soils and 2 type clay minerals, were used. For the representative clay minerals in Korea, illite ($K_{0.8-0.9}(Al,Fe,Mg)_2(Si,Al)_4O_{10}(OH)_2$, Yong-koong illite company, Korea) and kaolinite ($Al_2O_3 \cdot 2SiO_2 \cdot 2H_2O$, Sigma-aldrich, USA), were used. CsCl (Sigma-aldrich, USA), a nonradioactive Cs (^{133}Cs), was used to artificially contaminate the soil. Potassium chloride ($\geq 99.0\%$ KCl, Sigma-aldrich, USA) was used as background solutions in wet/dry repetition.

2.2 Methods

In the experiments for the exposure time of Cs, the soil was artificially contaminated by Cs during a day, week, month, and 3 months. A mass of 3 g of soil was mixed with 30 mL Cs solution (10 mg/L) in 50 mL centrifuge tube at room temperature. In the

experiments for the repetitions of wet/dry condition, the soil was artificially contaminated by Cs during 24 hours. After artificial contamination, the soil and solution were separated by a filter paper ($< 0.45 \mu m$). After then, the soil was dried at $50^\circ C$ for 24 hours (dry condition) and the dried soil was mixed with 30 mL of 5 mM KCl solution for 24 hours (wet condition). The dry and wet according to the experimental condition were repeated to 5, 10, and 15 times, respectively. The fractionation analysis of Cs was based on the sequential extraction method called as SM&T (Standards, measurement and testing programme) method (Table 1).

Table 1. Sequential extraction condition by SM&T method

Fraction	Extractable phase	Reagents
F1	Exchangeable	0.11 M CH_3COOH
F2	Reducible	0.5 M $NH_2OH \cdot HCl$ at pH 5
F3	Oxidizable	8.8 M H_2O_2 , 1 M CH_3COONH_4
F4	Residual	Aqua regia

3. Results and discussion

In the result of exposure time test, the fractionation change was confirmed in sample 1 and 3. F1 in sample 1, 2, and 3 decreased while F4 increased. In illite and kaolinite, fractionation change was not shown (Fig. 3). However, F4 in sample 1, 2, and 3 increased according to wet/dry repetitions. It seems that a section of F1 or 2 is changed to F4 by wet/dry repetitions. In other words, Cs could be moved to the soil particle part where is difficult to be desorbed. But F4 in clay minerals, illite and kaolinite, was not increased (Fig. 4). Actually, most radiocesium in Chernobyl exists as residual fraction. However, as compared to the results of this study, it is estimated that the Cs fractionation would be changed according to wet/dry conditions.

4. Conclusions

In this study, we investigated the fractionation change of Cs in soil environments according to the aging effect, including the exposure time and wet/dry condition. According to the exposure time of Cs, F4 was slowly increased in sample 1, 2, and 3, but F4 in clay minerals were not changed. However, wet/dry

repetitions influenced the fractionation change of Cs in sample 1, 2, and 3, but F4 in clay minerals were not changed. It is estimated that F1 or F2 could be changed to F4 by wet/dry repetitions, that is, it means that Cs could be moved to the soil particle part where is difficult to desorb Cs. Therefore, we expected that the fractionation of Cs would be transformed to stable fraction from weakly bound fraction by aging.

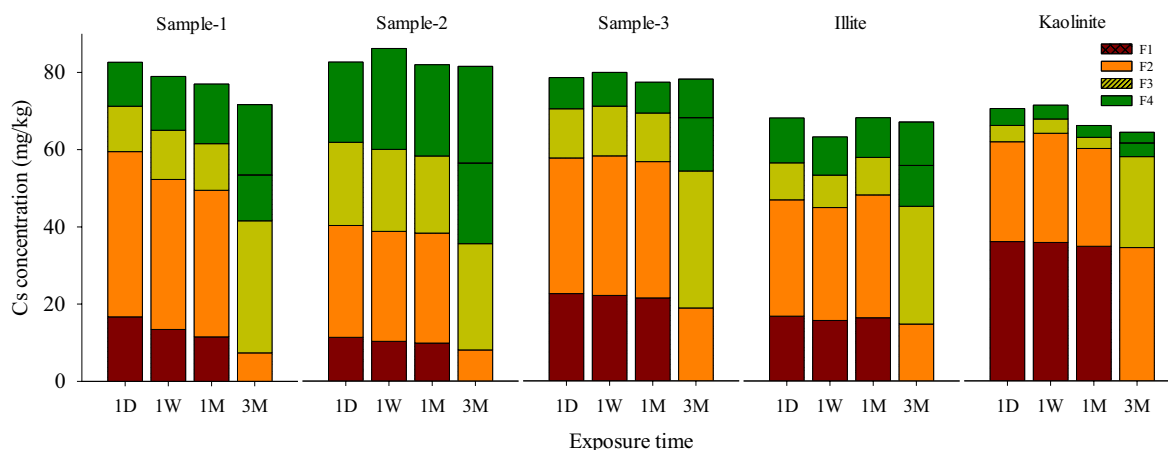


Fig. 3. Fractionation change of Cs according to the exposure time. The exposure time are 1 day, 1 week, 1 month, and 3 months which are 1D, 1W, 1M, and 3M, respectively.

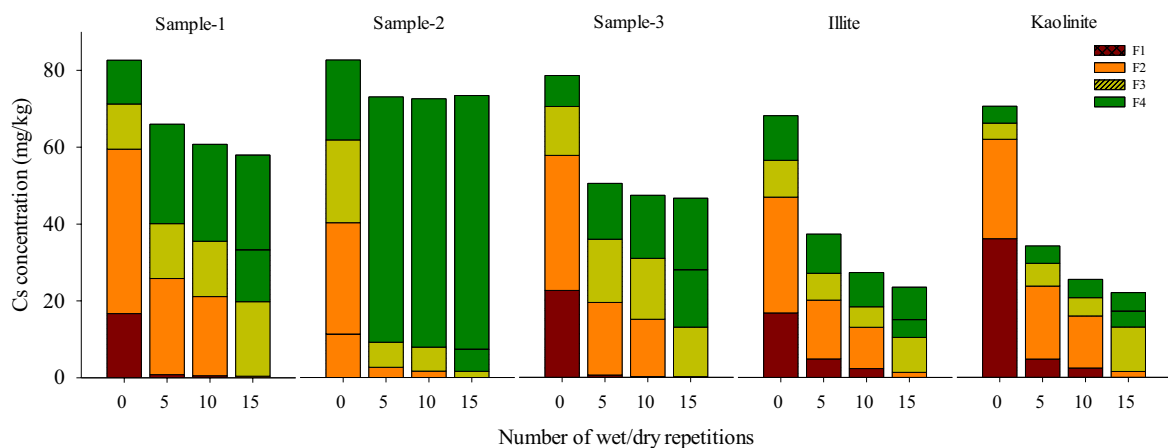


Fig. 4. Fractionation change of Cs according to the wet/dry conditions. Wet condition is 5 mM KCl solution for 24h and dry condition is 50°C for 24h.

REFERENCES

- [1] T. Saiot, H. Makino, and S. Tanaka, "Geochemical and grain-size distribution of radioactive and stable cesium in Fukushima soils: implications for their long-term behavior", *Journal of Environmental Radioactivity*, 138, 11-18 (2014).
- [2] K. Murota, T. Saito, and S. Tanaka, "Desorption kinetics of cesium from Fukushima soils", *Journal of Environmental Radioactivity*, 153, 134-140 (2016).
- [3] X.L. Hou, C.L. Fogh, J. Kucera, K.G. Andersson, H. Dahlgaard, and S.P. Nielsen, "Iodine-129 and Cesium-137 in Chernobyl contaminated soil and their chemical fractionation", *Science of Total Environments*, 308, 97-109 (2003).

Derivation of DCGLs for the Surface Soil of Kori-1 NPP by Using RESRAD Probabilistic Analysis

Ji Hyang Byon, Sang June Park, and Seokyoung Ahn *

School of Mechanical Engineering, Pusan National University, 2, Busandaehak-ro 63beon-gil, Geumjeong-gu,
Busan, Republic of Korea

*sahn@pusan.ac.kr

1. Introduction

The decommissioning related preparations of Kori Nuclear Power Plant (NPP) Unit 1 are ongoing. Safe decommissioning of nuclear power plants requires a procedure to provide guidelines for the planning, implementation and evaluation of radiological surveys being carried out to ensure compliance with the site release standards. The U.S. standard decommissioning procedure guidance manual, MARSSIM recommends derivation of DCGLs to establish safety assessment methods for site de-regulation in preliminary survey and final status survey. In the case of decommissioning of Kori-1 NPP, the derivation of DCGLs following MARSSIM guidelines are necessary. Based on less conservative but realistic ‘industrial worker scenario’ for the surface soil exposure, $DCGL_w$, the concentration of a single radionuclide that would provide 0.25 mSv/y (Domestic: 0.1mSv/yr) total effective dose equivalent (TEDE), were derived. After selecting a suite of potential radionuclides in Kori-1 NPP, Probabilistic analysis of RESRAD-ONSITE was performed to derive $DCGL_w$, with reference to the decommissioning experiences of Rancho Seco NPP and Zion NPP in the U.S.

2. Potential Radionuclides of Concern

Due to the limited source information of Kori-1 NPP, the method used at Rancho Seco NPP for determining radionuclides of concern (ROC) which include DandD and ORIGEN codes implementation is not appropriate so the method used at Zion NPP was used to derive ROC for Kori-1 NPP case. In conclusion, nevertheless in different ways of approach, ROCs from both Rancho Seco and Zion NPPs are very similar.

The list of ROC can be listed based on NUREG/CR-3474, NUREG/CR-4289 and WINCO-1191 as shown in Table 1 [1, 2, 3].

Table 1. Theoretical Radionuclides of Concern [1]

Radionuclides			
NUREG/CR-3474			
³ H	⁷⁹ Se	⁹⁹ Tc	⁶³ Ni
¹⁴ C	⁸¹ Kr	^{108m} Ag	²⁰⁵ Pb
³⁶ Cl	⁸⁵ Kr	^{121m} Sn	^{166m} Ho
³⁹ Ar	⁹⁰ Sr	¹²⁹ I	²³³ U
⁴¹ Ca	^{92m} Nb	¹³³ Ba	^{178m} Hf
⁵³ Mn	⁹³ Zr	¹³⁴ Cs	¹⁵⁸ Tb
⁵⁵ Fe	⁹³ Mo	¹³⁷ Cs	¹⁵¹ Sm
⁵⁹ Ni	⁹⁴ Nb	¹⁴⁵ Pm	¹⁵² Eu
⁶⁰ Co	^{239/240} Pu	¹⁴⁶ Sm	¹⁵⁴ Eu
⁶³ Ni	²³³ U	¹⁵¹ Sm	¹⁵⁵ Eu
WINCO-1191		NUREG/CR-4289	
¹⁴⁷ Pm		²³⁸ Pu	
²⁴¹ Pu		²³⁷ Np	
¹²⁵ Sb		²⁴¹ Am	
		²⁴⁴ Cm	

There are no sample analysis yet performed at Kori-1 NPP so the method used at Zion Station TSD-11-001 was used to exclude radionuclides which contributes a relative fraction of 0.0001 (0.01%) or less [4]. As a result, except for the inert gas Ar-39, the remained ROCs are shown in Table 2.

Table 2. Radionuclides Relative Fractions Equal To, or Greater than, 0.0001

Radionuclides			
³ H	⁵⁹ Ni	⁶⁰ Co	¹²⁵ Sb
¹⁴ C	⁶³ Ni	⁹⁴ Nb	¹³⁴ Cs
⁵⁵ Fe	⁹⁰ Sr	⁹⁹ Tc	¹³⁷ Cs
²⁴¹ Am	²³⁸ Pu	²³⁹ Pu	²⁴⁰ Pu
²⁴¹ Pu	^{233/244} Cm		

By integrating Tables 1 and 2, the radionuclides with active concentrations of less than 0.0001 (0.01%) compared to the activity concentrations of Co-60 and Ni-63 which are dominant in activity contribution are excluded. The excluded radionuclides were ³⁶Cl, ⁴¹Ca, ⁵³Mn, ⁷⁹Se, ^{92m}Nb, ⁹³Zr, ⁹³Mo, ^{108m}Ag, ^{121m}Sn, ¹²⁹I, ¹³³Ba, ¹⁴⁵Pm, ¹⁴⁶Sm, ¹⁵¹Sm, ¹⁵⁵Eu, ¹⁵⁸Tb, ^{166m}Ho, ^{187m}Hf, ²⁰⁵Pb and ²³³U. As a result of calculating dose contribution of standardized radionuclides by multiplying dose factor

from Zion TSD-14-019 by source term [5], Five radionuclides were selected as the main radionuclides to determine the soil DCGLs. ^{60}Co , ^{63}Ni , ^{90}Sr , ^{134}Cs and ^{137}Cs accounted for more than 99.5% of the dose.

3. Derivation of DCGL_w

Site-specific DCGL_w values can be derived as follow: [6]

$$\text{DCGL}_w = \frac{\text{Regulatory dose limit} - \text{Potential dose}}{\text{Peak of the mean dose}} \quad (1)$$

Probabilistic dose modeling should use the "Peak of the mean" to demonstrate compliance with 10 CFR Part 20, Sub E [7]. To perform the parameter sensitivity analysis, various site-specific parameter values with 0.037 Bq/g (1 pCi/g) of each detectable radionuclide with a simplified hydrologic and geologic model of Kori-1 were used as deterministic and probabilistic inputs. The results are shown in Table 3. Part Rank Correlation Coefficient (PRCC) was used to estimate a nonlinear and monotonic relationship which provides a unique contribution of the input parameters to the resulting dose as recommended in NUREG/CR-6692. If the absolute value of the PRCC is greater than 0.25, then the parameters were classified as sensitive, the parameter value at either the 75% quartile or the 25% quartile was assigned to calculate total effective dose equivalent (TEDE) [8].

Table 3. Assigned values of RESRAD-ONSITE from parameter sensitivity analysis

Parameter	PRCC	Quartile	Assigned value
External gamma shielding factor	0.89	75%	0.397
Density of contaminated zone	0.64	75%	1.673
Kd of ^{60}Co in contaminated zone	0.45	75%	1283.31

4. Results & Discussion

Table 4. Single radionuclide DCGL_w values for Industrial Worker Scenario

Radionuclide	<i>Rancho Seco</i> DCGL_w [Bq/g] (0.25mSv/yr)	KORI DCGL_w [Bq/g] (0.25mSv/yr)	KORI DCGL_w [Bq/g] (0.1mSv/yr)
^{134}Cs	0.829	0.888	0.355
^{137}Cs	1.954	2.108	0.843
^{60}Co	0.466	0.497	0.199
^{90}Sr	240	160.32	64.13
^{63}Ni	562400	591935	236774

5. Conclusion

As recommended in MARSSIM, the DCGLs were derived by using RESRAD-ONSITE with probabilistic analysis. The derived DCGL can then be used as an evaluation criterion of the MARSSIM survey unit and FSS design for Kori-1 NPP. The methodology based on MARSSIM but differently applied to Rancho Seco NPP and Zion NPP were analyzed and applied to Kori-1 NPP case. The DCGL results are very similar to the Rancho Seco case which used similar 'industrial worker scenario' of Kori-1 NPP case.

ACKNOWLEDGEMENT

This research was supported by a Korea Foundation of Nuclear Safety (KoFONS) grant funded by the Nuclear Safety and Security Commission (NSSC) (No. 1305009-0416-CG100).

REFERENCES

- [1] NRC, NUREG/CR-3474, "Long-Lived Activation Products in Reactor Materials", 1984.
- [2] NRC, NUREG/CR-4289, "Residual Radionuclide Contamination Within and Around Commercial Nuclear Power Plants", 1986.
- [3] Westinghouse Idaho Nuclear Company Inc., WINCO-1191, "Radionuclides in United States Commercial Nuclear Power Reactors", 1994.
- [4] Zion Solutions Inc., TSD 11-001 Rev.1, Technical Support for Potential Radionuclides of Concern During the Decommissioning of the Zion Station, 2015.
- [5] Zion Solutions Inc., TSD 14-019, Radionuclides of Concern for Soil and Basement Fill Model Source Terms, 2014.
- [6] United States Nuclear Regulatory Commission, "Rancho Seco License Termination Plan", 2014.
- [7] NRC, NUREG/CR-1757, Vol.2, Rev.1, Consolidated Decommissioning Guidance Characterization, Survey, and Determination of Radiological Criteria, 2006.
- [8] NRC, NUREG/CR-6692, Probabilistic Modules for the RESRAD and RESRAD-BUILD Computer Codes, 2000.

Development of Measurement and Segregation System for Radioactive Soil

Jae-Mun Han*, Sang-Chul Lee, Min-Jung Kim, and Jin-Hee Kim

HYUNDAI E&C, 17-6, Mabuk-ro 240beon-gil, Giheung-gu, Yongin-si, Republic of Korea

*jmhan@hdec.co.kr

1. Introduction

The decommissioning of the nuclear power plant will proceed to the D&D design, decontamination, demolition, waste disposal and site restoration. Site restoration is a process of removing the contaminated soil and returning the site to a green field before the installation of a nuclear power plant. The removed radioactive contamination soil is disposed of as radioactive waste, and a high amount of site restoration costs is generated from disposal of waste. Therefore, the site restoration cost can be minimized by minimizing the radioactive waste generated at site restoration [1]. This study will demonstrate the measurement and segregation system that can minimize the generation of radioactive waste through the process of active separation of real-time radioactive contamination.

2. Active real-time radiation measurement and separation system

The soil sorting system is technology that measures the radioactivity of soil that is transported by conveyor belts, and mechanically separates radioactive contaminated materials into clean and contaminated waste streams [2].

2.1 Soil measurement and segregation system

This is accomplished by passing the soil on a conveyor belt under an array of sensitive and rapidly reacting, radiation detectors that measure radionuclide concentrations. The contaminated soil above the desired cleanup limits is automatically

diverted into a separate waste stream. As shown in Fig. 1, contaminants are removed by soil measurement and segregation system. Thus, it can significantly reduce the overall amount of contaminated soil requiring disposition as radioactive waste.

Typical radionuclides that can be measured by soil measurement and segregation system including Cs-137 and Co-60. The main factors for measurement on this system depend on the ambient radiation background, conveyor belt speed, thickness of the soil layer on the conveyor, and contaminant gamma ray energy and abundance [3].

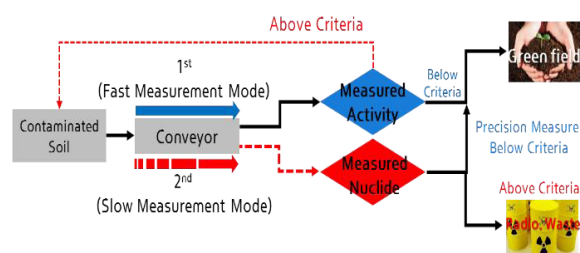


Fig. 1. Process diagram of soil measurement and segregation system.

2.2 Configurations of segregation system

The soil measurement and segregation system contained two detectors which are thallium-doped sodium iodide NaI(Tl) and four PVT scintillators housed in an environmentally controlled box for temperature stabilization and background radiation reduction. Gamma spectra in pre-defined energy ranges are collected over a fixed distance interval (60 cm) using a Multi-Channel Analyzer (MCA). The system includes a controller for conveyor belt speed and sensors for conveyed material depth, detector temperatures, belt speed, and reversing belt direction.



Fig. 2. Real-time measurement and segregation system.

2.3 Operation of soil measuring system

Prior to measuring by the system, excavated soil is pre-separated by method of wet multi particle separation through a vibrating screen.

The sized feed soil is loaded into the hopper. The soil height on survey conveyor are regulated by a “strike-off” bar which maintained a maximum belt fill depth of 10 cm. Depending on the operating conditions and data requirements, the soil move at typical conveyor speeds between 0.67 ~ 2 cm/s beneath the suspended NaI detector and PVT. The specific activity is automatically acquired for $60 \times 60 \times 10$ cm of soil and compared to the segregation criteria 0.1 Bq/g.

Once the soil reached the end of the conveyer belt, a reversing conveyor divert the soil to either the above or below criteria depending on its specific activity.

3. Conclusion

The soil measurement and segregation system is a volume reduction and waste minimization process in which the particles that contain contamination are removed from the remaining relatively clean soil. The pilot test of this system isn't performed for a real radioactive contaminated soil. After the pilot test of a real radioactive contaminated soil, the correct separation efficiency for radioactive contaminated soil can be evaluated.

REFERENCES

- [1] IAEA, “Remediation of Sites with Mixed Contamination of Radioactive and Other Hazardous Substances”, Technical Reports Series no. 442 (2006).
- [2] EPA, “Technology Reference Guide for Radioactively Contaminated Media”, 402-R-07-004 (2007).
- [3] Alejandro U. Lopez, and Jeffrey W. Lively, “Application of soil Segregation Technology: Reducing Uncertainty and Increasing Efficiency at an NRC test reactor Decommissioning site”, WM2011 Conference (2011).

5분과

제염해체 (Poster)



Considerations in Unit Cost Factor Development for Nuclear Power Plant Decommissioning Project

Wanil Jung*, Myung Duck Yang, and Young Jin Yang

KEPCO E&C, 269 Hyeoksins-ro, Gimcheon-si, Gyeongsangbuk-do, Korea

*wanil@kepco-enc.com

1. Introduction

The accurate cost estimate is one of the basic elements for the successful execution of the projects. Following the permanent shutdown of Kori Nuclear Power Plant unit 1, touched in June 2017, the decommissioning work is emerging as one of the hottest social issues. In preparation for it, this paper surveys the concept of Unit Cost Factor (UCF) and its application example in foreign decommissioning cases. Finally this paper suggests the key points to be considered during the development of UCFs for the domestic decommissioning projects.

2. Main Discussion

2.1 Concept of UCF

The UCF method is widely used in cost estimating to simplify the estimation for a large number of plant inventory items (pipe, valves, pumps, tanks, etc). UCFs are developed in terms of hours of the mixed labor crew to perform an activity (such as removal) per unit of measure (such as each, ton, m², etc). Then, with the given hourly rates of the crew and equipment, the total cost for the unit of measure can be estimated [1].

2.2 Examples of UCFs in domestic general construction projects

In domestic construction projects, UCFs are referred to as “Ilwidaega” and the formalized tables with detailed input of the required material, laborers, their quantities and prices are referred to as “Ilwidaegapyo” as shown in Table 1.

Table 1. Example of UCF in domestic construction projects

Item	Spec.	Unit	Quantity	Unit Price	Amount
1. Materials	Ziin Interior Film	FW Grade	m ²	1.1	15,000
	sub-total				16,500
2. Subsidiary Materials	Primer	PM02	kg	0.08	20,000
	Thinner	Samwha A-7	kg	0.16	7,000
	Putty	Poly Putty	kg	0.08	5,000
	sub-total				3,120
3. Laborer	Helper	Priming	MD	0.07	80,000
	Handyman	Cutting	MD	0.07	100,000
	Technician	Filming	MD	0.07	130,000
	sub-total				21,700
Total					41,320

2.3 Examples of UCFs in foreign decommissioning projects

In the decommissioning cost analyses for the nuclear power plants in United States, about 150 UCFs have been used. The report lists those UCFs as extracted in Table 2 and describes the detailed elements of a sample UCF as shown in Table 3.[2] The sample UCF of “Removal of Contaminated Heat Exchanger smaller than 3,000 lbs” presents 6 sub-activities, their duration, work difficulty factors, required laborers, equipment and consumables for the unit measure of the work.

Table 2. UCF List (part) used in US decommissioning cost analysis

Unit Cost Factor	Cost/Unit (\$)
Removal of Contaminated Pump Moter, 300-1000 pound	738.15
Removal of Contaminated Pump Moter, 1000-10,000 pound	2,374.95
Removal of Contaminated Pump Moter, > 10,000pound	5,364.75
Removal of Contaminated Heat Exchanger < 3000 pound	3,414.57
Removal of Contaminated Heat Exchanger > 3000 pound	9,912.93

Table 3. Detailed Elements of a UCF used in US decommissioning cost analysis [2]

Example: Unit Factor for Removal of Contaminated Heat Exchanger < 3,000 lbs.				
1. SCOPE				
Heat exchangers weighing < 3,000 lbs. will be removed in one piece using a crane or small hoist. They will be disconnected from the inlet and outlet piping. The heat exchanger will be sent to the packing area.				
2. CALCULATIONS				
Activity Description		Critical Duration (minutes)		
Install contamination controls, remove insulation, and mount pipe cutters		60		
Disconnect inlet and outlet lines, cap openings		60		
Rig for removal		30		
Unbolt from mounts		30		
Remove contamination controls		15		
Remove heat exchanger, wrap in plastic, and send to packing area		60		
Critical Duration		255		
Work Adjustments (Work Difficulty Factors)				
+ Respiratory Protection (25% of Critical Duration)		64		
+ Radiation/ALARA (20% of Critical Duration)		51		
Adjusted Work Duration		370		
+ Protective Clothing (30% of Adjusted Work Duration)		111		
Productive Work Duration		481		
+ Work break adjustment (8.33 % of Productive Work Duration)		40		
Total Work Duration		521		
*** Total Work Duration = 521 minutes or 8.683 hours ***				
3. LABOR REQUIRED				
Crew	Number	Duration (hours)	Rate (\$/hr)	Cost
Laborers	3.00	8.683	\$47.52	\$1,237.85
Craftsmen	2.00	8.683	\$61.55	\$1,068.88
Foreman	1.00	8.683	\$65.29	\$566.91
General Foreman	0.25	8.683	\$68.99	\$149.76
Fire Watch	0.05	8.683	\$47.52	\$20.63
Health Physics Technician	1.00	8.683	\$36.57	\$317.54
Total labor cost				\$3,361.57
4. EQUIPMENT & CONSUMABLES COSTS				
Equipment Costs				none
Consumables/Materials Costs				
Universal Sorbent 50 @ \$0.43/sq ft (1)				\$21.50
Tarpaulins (oil resistant/fire retardant) 50 @ \$0.32/sq ft (2)				\$16.00
Gas torch consumables 1 @ \$8.19/hr x 1 hr (3)				\$ 8.19
Subtotal cost of equipment and materials				\$45.69
Overhead & sales tax on equipment and materials @ 16.00%				\$7.31
Total costs, equipment & material				\$53.00
TOTAL COST: Removal of contaminated heat exchanger <3000 pounds:				\$3,414.57
Total labor cost:				\$3,361.57
Total equipment/material costs:				\$53.00
Total craft labor man-hours required per unit:				63.386

2.4 Considerations in UCF Development for domestic decommissioning projects

A systematic and elaborate approach is needed for the UCF development as the domestic nuclear power plant decommissioning task and the cost estimate for it are the critical challenges. The key points to be considered during the UCF development are listed below.

- Review of the UCF Classification
 - Can the existing UCFs cover the whole scope of the decommissioning project?
 - Can the existing UCFs be applied to domestic decommissioning project?
 - Is there any need of adding or re-classifying of the existing UCFs?
 - What about and how to develop the bases and premises of the re-classification of UCFs?
- Review of sub-activities of each UCF
 - Are the existing sub-activities of each UCF are proper, feasible, applicable, and cost-effective in Korea?

- If not, what are the alternatives?
- What are the proper labor crews for each UCF?

- Review of duration of each sub-activity and UCF
 - Collection of and analysis on reference data in US decommissioning cost analyses as many as possible
 - Survey on the existence and productivity rate of domestic standard means of same or similar work type of each sub-activity
 - Study on site characteristics and work difficulty factors for domestic decommissioning site
 - Study of adjustment of existing duration considering the difference between US and domestic labor productivity rates.
 - Set up of our own productivity rate and duration of each sub-activity
- Pilot test program to set up the UCF
 - Pilot test on the specific supposed technics with relevant contractors to survey the feasibility and the productivity
 - Acquisition of test equipment, laboratory and contaminated test samples

3. Conclusion

Through the surveys on the existing UCF and its application example in foreign decommissioning cases, this paper suggests the key points to be considered during the development of UCFs for the domestic decommissioning projects

As the accurate cost estimate is critical in preparing the project, it is highly expected to develop the UCFs of our own through the thoughtful and systematic approach by relevant parties [3].

REFERENCES

- [1] RSCS, KEPCO E&C, Decommissioning Cost Analysis for the Korean pressurized Water Nuclear Power Plant, Final Report, 2009.
- [2] TLG Services, Inc., Decommissioning Cost Analysis for the Vermont Yankee Nuclear Power Station, 2012.
- [3] Wanil Jung, et. al., Development of Level 1 Schedule for Nuclear Power Plant Decommissioning Project in Korea, Korean Radioactive Waste Society 2017 Spring Conference, 2017.

Laser Cutting Experiment of Thick Stainless Steel for Nuclear Decommissioning: Effect of Nozzle Types

Seong Y. Oh*, Jae Sung Shin, Sangwoo Seon, Taek Soo Kim, Hyunmin Park, Lim Lee, Chin-Man Chung,
and Jonghwan Lee

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

*Corresponding author: syoh73@kaeri.re.kr

1. Introduction

Laser cutting technology coupled with fiber laser can be suitable dismantling tool at a nuclear decommissioning site due to the production of a small amount of secondary waste and remote controllability.[1-2] The mechanism of laser cutting technology is described that high-powered laser beam heats up the local area of steel plate over melting points. The resultant molten steel is blown outside steel plate by injecting assist gas with high speed. The high speed assist gas is generated through that high pressured gas passes through the nozzle with small diameter. Notably, the geometric configuration of the nozzle significantly influences the flow behavior of assist gas emerging from the nozzle tip, which is closely related to melt removal efficiency in the laser cutting. [3]

In this study, two types of nozzles, subsonic and supersonic nozzle [3], were used in comparatively evaluating the cutting performance with the focus on the melt removal efficiency according to variation of stand-off distance. Stand-off distance refers to distance between nozzle tip and specimen surface.

2. Experiment

2.1 Experimental Setup

Laser cutting system associated with 6kW fiber lasers (IPG, YLS 6000) contains cutting head and 25-m-long process fiber. The high-powered laser beam is fed into cutting head through the process fiber with a core diameter of 100 μ m. The cutting head serves to focus the laser beam onto the specimen and discharge an assist gas with high pressure into cutting-kerf. In detail, the laser beam was focused by two optics in the cutting head, a collimation lens (f: 160mm) and parabolic focusing mirror (f: 600mm). The nozzle

coupled with the cutting head is connected to the high-pressure gas pipe so that compressible air of 10.0 bar gauge pressure is introduced from the outside and injected into the specimen. The throat diameters of convergent and convergent-divergent nozzles used in the experiment were 2mm and the flow rates was about 400L/min. X-Y-Z axis stage adjusts relative motion of cutting head. The dust collector sucks dust generated in the laser cutting experiment and exhausted to the outside after passing through the filter. The throat diameters of convergent and convergent-divergent nozzles was 2mm and the flow rates was about 400L/min. The 60-mm-thick stainless steel blocks (SUS 304L) was used in the cutting experiment.

3. Result and Discussion

3.1 Effect of subsonic nozzle on cutting performance

The cutting performance of stainless steel plate (SUS 304L) with 60 mm thickness was evaluated using subsonic nozzle. The subsonic nozzle has a convergent shape. Laser cutting was carried out from left side of specimen toward right side and its entire cutting length was 40mm for The cutting speed was initially set at 20 mm/min until 10 mm cutting, after which, 60 mm/min cutting speed was set to the reminder of 30mm cutting. Complete cutting was observed in the range of 1-mm and 3-mm stand-off distances, whereas incomplete cutting was observed in the case of 4.0 mm stand-off distance. Fig. 1 presents the incomplete laser-cut specimen using a subsonic nozzle under the condition of 4-mm stand-off distance. As shown in Fig. 1(b) and 1(c), despite sufficient provision of laser energy, kerf is not generated along the rear side of the steel block in the early stages of laser cutting. This implies that the increase of stand-off distance in the subsonic nozzle leads to the degradation of melt removal efficiency.

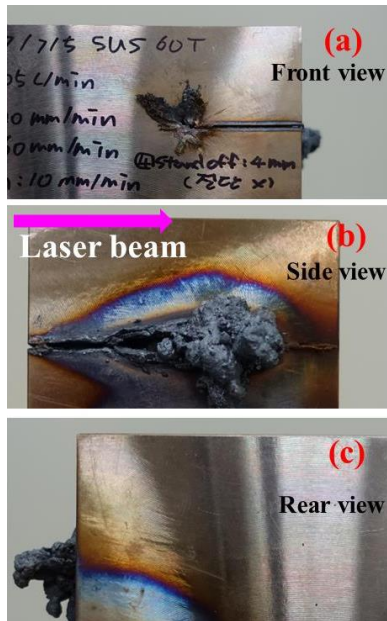


Fig. 1. Stainless steel after cutting ($t=60$ mm) using subsonic nozzle at 4mm standoff distance. Front face(a), Side face(b), Rear face(c).

3.2 Effect of supersonic nozzle on cutting performance

For the case of supersonic nozzle (convergent-divergent shape), same experimental conditions as in the case of subsonic nozzle were employed except standoff distance. The standoff distance was steadily increased up to 15mm. Complete cutting was observed until the condition of 10mm standoff distance.

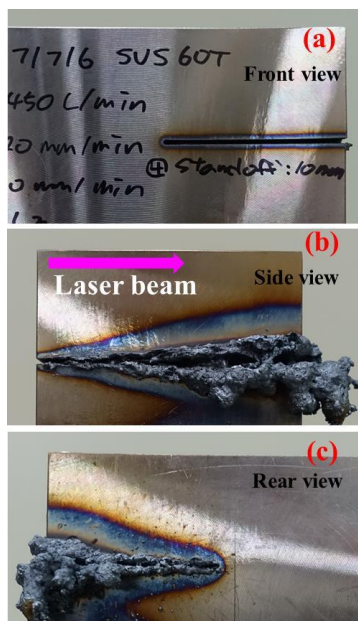


Fig. 2. Stainless steel after cutting ($t=60$ mm) using supersonic nozzle at 10mm standoff distance. Front face(a), Side face(b), Rear face(c).

Fig. 2 presents the specimen after laser cutting at 10mm standoff distance. It implies that supersonic nozzle is more efficient than subsonic nozzle in aspect of melt removal rate.

4. Conclusion

It was shown that the cutting performance according to the variation of standoff distance is influenced by geometric configuration of nozzle. In practical use, the supersonic nozzle is desirable because sufficient stand-off distance is necessary to provide easier remote operation.

REFERENCES

- [1] K. Tamura, R. Ishigami, and R. Yamagishi, "Laser cutting of thick steel plates and simulated steel components using a 30 kW fiber laser", *Journal of Nuclear Science and Technology*, 53, 916-920 (2016).
- [2] C. Chagnot, G. de Dinechin, and G. Canneau, "Cutting performances with new industrial continuous wave ND:YAG high power lasers For dismantling of former nuclear workshops, the performances of recently introduced high power continuous wave ND:YAG lasers are assessed laser cutting of thick steel plates and simulated steel components using a 30 kW fiber laser", *Nuclear Engineering and Design*, 240, 2604-2613 (2010).
- [3] H. C. Man, J. Duan, T. M. Yue, "Analysis of the dynamic characteristics of gas flow inside a laser cut kerf under high cut-assist gas pressure Laser cutting of ", *Journal of Physics D: Applied Physics*, 32, 1469-1477 (1999).

Accident Case Analysis for Safety Assessment of Decommissioning

Kyungmin Kim¹, Jaeyong Lee¹, Je-keun Jun², and Yong-soo Kim^{1*}

¹Department of Nuclear Engineering, Hanyang University, 222 Wangsimni-ro, Seongdong-gu, Seoul, Korea

²UjuEnertech Co., 222 Wangsimni-ro, Seongdong-gu, Seoul, Korea

*yongskim@hanyang.ac.kr

1. Introduction

In order to evaluate the safety assessment for decommissioning, it is necessary to analyze and investigate possible accidents/hazard during decommissioning. In Korea, according to Nuclear Safety and Security Commission Notice No. 2015-8, "Provisions on Preparation of Nuclear Power Plants Decommissioning Plan, etc.", "abnormal accidents" are expected to occur more than once during the decommissioning process. In addition, that the decommissioning activities and work are not proceeding as planned [1]. According to the "IAEA Safety reports series No. 77 Safety Assessment for Decommissioning", "hazard" is classified as a cause of radioactive and non-radioactive cause, which is a potential influence on human health and the environment due to facilities, It means [2]. In this study, the definition of abnormal accidents and the cases of accidental decommissioning of nuclear power plants were investigated.

2. Types of Accidents that Occur When Decommissioning Activities

2.1 Nuclear and Radiation Accidents

2.1.1 Criticality. In a permanently shutdown nuclear power plant, nuclear fuel is completely removed from the storage facility, so that a critical accident does not occur. However, there is a possibility of occurrence of criticality in fissile material treatment plants or waste storage tanks used in nuclear fuel fabrication plants, spent fuel reprocessing plants, nuclear fuel enrichment plants,

2.1.2 Exposure. The presence of contaminants such as radioactive materials and contaminants in the event of an accident occurring during decommissioning increases the likelihood of external exposure. For example, workers working around the radioactive stainless steel of a reactor pressure vessel may be

exposed to external exposure. If radioactive materials are present as surface contaminants, internal exposure due to inhalation and ingestion by workers or residents may occur. When working in decommissioning areas, caution should be exercised for the risk of inhalation of radionuclides that release alpha rays.

2.1.3 Release of Radioactive Materials. Because of the nature of nuclear decommissioning work, it includes cutting and decontamination processes, so the wastes generated here have different characteristics from the wastes generated during operation of nuclear power plants. Liquid wastes are generated more than in nuclear power plants due to decontamination process, but gaseous wastes occur less than in nuclear power plants. Concrete, metal, etc. can reach limit when they reach below the decontamination standard, so there is a possibility that some materials exceeding the decontamination standard are included and are erroneously released.

2.2 Non-radioactive Accidents

2.2.1 Combustible and flammable materials. The material used in the cutting and decontamination process may cause a fire, and the instantaneous heating of the combustible material may increase the risk of fire. Explosions may also be caused by solutions or flammable materials used during demolition operations.

2.2.2 Hazardous Materials. Toxic and harmful materials emitted during decommissioning of nuclear power plants include thermal insulators, asbestos, lead paint, electrical insulators, PCBs, decontamination solutions, etc., which must be treated in a special way.

2.2.3 Physical Hazards. There may be a collision due to the use of temporary structures or structures during the demolition work, and there is a possibility of collision of heavy equipment, injury due to sharp materials, and equipment fall at high level, and human accidents during equipment use.

2.2.4 Natural Hazards. Natural disasters such as typhoons, earthquakes and floods have a higher risk of decommissioning than during operation of nuclear power plants.

3. Accidents Case of the Decommissioning

Table 1 presents the highest doses in each of four categories of radiological accidents as obtained from licensing-basis documents. The highest doses result from postulated fuel-related accidents and radioactive-material-related accidents. All accidents that were reviewed used conservative assumptions to calculate the offsite dose [3].

Table 1. Highest Offsite Doses Calculated for Postulated Accidents in Licensing-Basis Documents [3]

Accident Description	Offsite Whole-Body Dose, rem
Fuel-Related Accidents	
Shipping cask or heavy load drop into fuel element storage well	0.186
Loss of pre-stressed concrete reactor vessel shielding water	0.035
100% fuel failure	0.027
Simultaneous failure of fuel assemblies	0.016
Spent fuel handling accident	0.013
Fuel assembly drop	0.0026
Radioactive Material-Related Accidents (Non-Fuel)	
Spent resin handling accident	0.96
Explosion inside vapor container	0.44
Radioactive liquid waste system leaks and failure	0.23
Materials-handling event	0.16
Fire in intermodal container of waste	0.1
Decontamination events	0.039
External-Events Initiated Accidents	
Natural disaster, tornado	0.0001
Physical security breach	<0.000001
Offsite Transportation Accidents	
Reactor pressure vessel railroad accident and fire	0.00014
Transportation accident	<0.000001

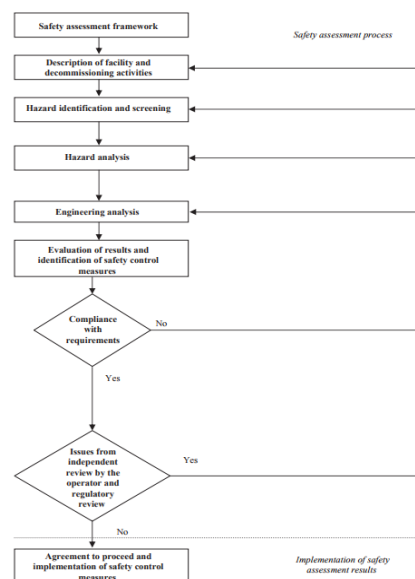


Fig. 1. Main steps of safety assessment [2].

4. Conclusions

In this study, we conducted a survey on accident cases, which is one of the important factors for nuclear safety evaluation. As a result of the study, detailed data were constructed for each accident case in countries with nuclear decommissioning experience. However, in Korea, which is about to decommissioning Kori Unit 1, there is a lack of data on accident investigation and legal regulations. Research is required for successful nuclear decommissioning and accurate safety assessment.

ACKNOWLEDGEMENT

The Nuclear Safety Research Program funded by the Nuclear Safety and Security Commission (No. 1305009) supported this research.

REFERENCES

- [1] NSSC, Nuclear Safety and Security Commission Notice No. 2015-8, "Provisions on Preparation of Nuclear Power Plants Decommissioning Plan, etc.", 2015.
- [2] IAEA, Safety Assessment for Decommissioning, "IAEA Safety reports series No. 77, 2013.
- [3] NRC, Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities: Regarding the Decommissioning of Nuclear Power Reactors, 2002.

Preliminary Evaluation of DCGLs for Site Release After Decommissioning of Korean Research Reactor 1 & 2

Tae Gwan Do¹, Jin Ho Park¹, Geun Ho Kim^{1,2}, and Kwang Pyo Kim^{1*}

¹ Kyung Hee University, 1732 Deokyoung-daero, Giheung-gu, Yongin-si, Gyeonggi-do, Republic of Korea

² Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon gi, Yuseong-gu, Daejeon, Korea

*kpkim@khu.ac.kr

1. Introduction

The decommissioning project of Korean Research Reactor (KRR) 1&2 was introduced in the preparation stage for site release in 2017. Korean government legislated NSSC (Nuclear Safety and Security Commission) Notice No.2016-33 in 2016. Article 4 of the Notice states that dose criteria for site reuse should not exceed 0.1 mSv per year. DCGL (Derived Concentration Guideline Levels) should be assessed to demonstrate that dose criteria are met. Article 2 of the Notice states that not only commercial reactor but also research reactors should be evaluated. Therefore, we performed preliminary evaluation of DCGLs for site release after decommissioning of KRR 1 & 2.

2. Material and Methods

2.1 Evaluation of DCGLs

Korea Institute of Nuclear Safety (KINS) suggested a method to evaluate DCGLs. Radiological evaluations according to exposure scenarios are performed for each nuclide and thus deriving DCGL values. The DCGL evaluation procedure is as follows: (1) selection of radionuclides in a site, (2) selection of representative exposure scenarios, (3) assessment of radiation dose per unit radioactivity (1 Bq/g) for each radionuclide, (4) derivation of DCGLs by dividing personal dose (assessed in the 3 step) by dose criteria (0.1 mSv/year), and (5) confirmation whether radiation dose according to each scenario satisfies dose criteria.

2.2 Selection of radionuclides at KRR-1&2 sites

Radionuclides at KRR-1&2 sites were selected based on NUREG-3474, NUREG-4289, NUREG-

0130, ORIGEN code, and historical site assessment (HSA) [1]. The NUREG documents describe the activation products for the commercial reactors in the US. Since radioactivity concentration varies depending on reactor type and the operation period of the nuclear facility, ORIGEN code was used to evaluate the radioactivation. Among the radionuclides derived from the procedure, radionuclides with a half-life less than 3 years were excluded because KRR-1&2 was permanent shutdown 22 years ago. Table 1 shows radionuclides that are considered in this study.

Table 1. Radionuclides considered in this study

Radio-nuclide	Half-life (years)	Radio-nuclide	Half-life (years)
¹⁴ C	5.73×10^3	¹³³ Ba	10.5
³⁶ Cl	3.01×10^5	¹³⁷ Cs	3.02×10^5
⁶⁰ Co	5.27	¹⁵² Eu	13.6
⁷⁹ Se	1.13×10^6	¹⁵⁴ Eu	8.59
⁹⁰ Sr	28.6	¹⁵⁸ Tb	1.80×10^2
^{108m} Ag	4.18×10^2	^{178m} Hf	30.0
^{121m} Sn	5.00	²³⁷ Np	2.14×10^6
¹²⁹ I	1.57×10^7		

2.3 Investigation of the RESRAD input data

Personal radiation doses were calculated using RESRAD program. For RESRAD program calculation, we used preferentially domestic data if the data were available. If the domestic data were not available, we used international data or RESRAD default value. The volume of contamination in the site was set at 7,112 m³, derived by multiplying the land area of 47,417 m² and the surface soil thickness of 0.15 m. The evaluation period for the radiation dose was set at 1,000 years according to NSSC Notice No.2016-33.

3. Results and Discussion

Radiation doses by calendar time after site release are given in Fig. 1. The radiation dose was calculated assuming unit radioactivity (1 Bq/g). Radiation doses initially increased with time, reached the maximum in 3 years after site release, and then decreased rapidly with time. The maximum radiation dose was about 13 mSv/year assuming radioactivity of 1 Bq/g.

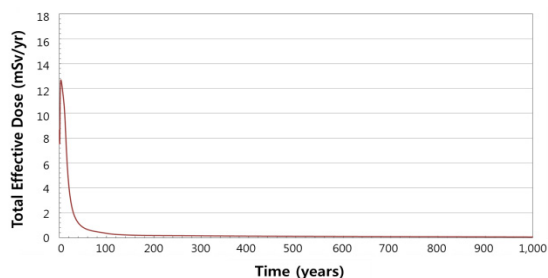


Fig. 1. Radiation doses by calendar time after site release.

Fig. 2 shows radiation dose by radionuclide when the radiation dose is the maximum (3 years after site release). The iodine-129 contributed the highest radiation dose (about 4.3 mSv/year). It was followed by ^{178m}Hf , ^{60}Co , ^{14}C , etc.

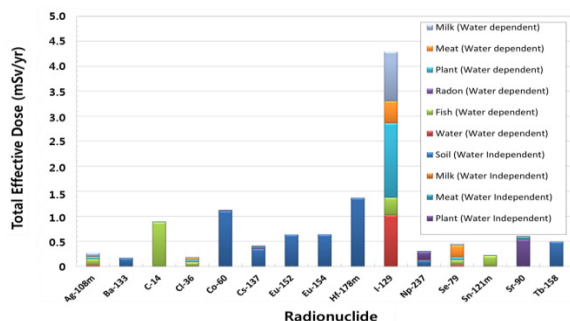


Fig. 2. Radiation doses by radionuclide when the radiation dose is the maximum.

Table 2 shows DCGLs derived based on the radiation dose assessment. DCGLs were 0.241 Bq/g for ^{137}Cs and 0.0888 Bq/g for ^{60}Co .

Table 2. DCGLs derived in this study

Radio-nuclide	DCGL (Bq/g)	Radio-nuclide	DCGL (Bq/g)
^{14}C	0.11	^{133}Ba	0.60
^{36}Cl	0.52	^{137}Cs	0.24
^{60}Co	0.089	^{152}Eu	0.16
^{79}Se	0.22	^{154}Eu	0.16
^{90}Sr	0.17	^{158}Tb	0.20
^{108m}Ag	0.37	^{178m}Hf	0.073
^{121m}Sn	0.45	^{237}Np	0.33
^{129}I	0.023		

4. Conclusion

In this study, DCGLs in the KRR-1&2 sites were evaluated for site release after decommissioning. Specific input data for nuclide selection and RESRAD input factor are insufficient because site release for KRR-1&2 is still in preparation stage. Therefore, we performed preliminary evaluation using the currently available data. Based on the preliminary study, DCGLs for ^{137}Cs and ^{60}Co were 0.241 Bq/g and 0.0888 Bq/g, respectively. The procedures and methods in this study can be applied to derive DCGLs for not only research reactors but also commercial reactor decommissioning.

ACKNOWLEDGEMENTS

This research was supported by grant from “Establishment of site release method for KRR 1 or 2 applying MARSSIM” carried out by Korea Atomic Energy Research Institute.

REFERENCES

- [1] Pengfei Zhao, Yeo Ryeong Jeon, Yongmin Kim, Jong Seh Lee, Seokyoung Ahn, “A Radionuclides Suite Selection for Site Characterization and Final Status Survey in the U.S. NPPs”, Journal of Nuclear Fuel Cycle and Waste Techonology, 14(3), 267-277 (2016).

pH and Electric Conductivity of Cement Solidification of Metal Hydroxide Waste

Daeseo Koo*, Hyun-Hee Sung, Sang Bum Hong, and Bum-Kyoung Seo

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989 beon-gil, Yuseong-gu, Daejeon, Korea

*ndskoo@kaeri.re.kr

1. Introduction

A lot of uranium soil waste from a uranium conversion facility has been stored in KAERI since 2010. The study of decontamination for uranium soil waste has been conducted [1-2]. Radioactive material (concentrated liquid waste, waste resin, waste sludge) generated from the decontamination process has to solidify and pass the integrity test of KORAD (Korea Radioactive Agency) for permanent disposal. For permanent disposal, a study on cement solidification of the radioactive waste is requested. Several studies on cement solidification of the radioactive waste have been carried out [3-7].

In this study, to study the characteristics of cement solidification on metal hydroxide waste for permanent disposal, the experiment for the solidified cement was performed. The pH and electric conductivity of cement solidification, which was immersed in demineralized water for a specified time, were measured and analyzed on sampling time.

2. Experiment and result

2.1 Experiment and measurement

2.1.1 Sample preparation. Table 1 shows test conditions of cement solidification. According to Table 1, water, Portland 1 species, and waste were homogeneously mixed using a Mortar Mixer (HJ-1150). The mixing material is put into a polyethylene mold (ID 50 mm, H 120 mm) and solidified for 4 weeks. After solidifying the mixing material covered with vinyl for 4 weeks, cement solidification was fabricated.

Samples were prepared (diameter 50 mm, height 100 mm) by cutting the solidified cement with micro cutter. The samples were immersed in demineralized water for 90 days.

Table 1. Test conditions

Specimen	Waste (g)	Cement (g)	Water (g)
C-2.0-60	2.0(106.95)	1(53.481)	1.80(96.253)
C-2.0-70	2.0(97.560)	1(48.786)	2.10(102.44)
C-2.0-80	2.0(89.687)	1(44.849)	2.40(107.62)
C-1.8-60	1.8(103.56)	1(57.541)	1.68(96.657)
C-1.8-70	1.8(94.435)	1(52.471)	1.96(102.83)
C-1.8-80	1.8(86.788)	1(48.221)	2.24(108.00)
C-1.6-45	1.6(116.60)	1(72.890)	1.17(85.270)
C-1.6-50	1.6(110.30)	1(68.970)	1.30(89.640)
C-1.6-60	1.6(99.620)	1(62.270)	1.56(97.130)

2.1.2 Measurement. The purpose of the immersion test for cement solidification is to evaluate durability of cement solidification at a permanent disposal site. The pH and electric conductivity of cement solidification, which was immersed in demineralized water for a specified time (1 day, 3 days, 7 days, 14 days, 47 days, 90 days), was measured.

2.2 Result and discussion

Fig. 1 shows the pH of cement solidification. The pH of demineralized water was about 5. The pH of the solidified cement, which was immersed in demineralized water, had some fluctuation from day 1 of sampling until day 47 of sampling. It approached about 13 after 40 days of sampling. The immersion test on cement solidification indicates a strong alkali property.

Fig. 2 indicates the electric conductivity of cement solidification. The electric conductivity of demineralized water was about 0.5 mS/m. The electric conductivity of cement solidification, which was immersed in demineralized water, shows a rapid increase from day 1 of sampling until day 7 of sampling. After 47 days of sampling, a saturation trend is shown. The solidified cement, which contains many cement components, shows a bigger electric conductivity, while the solidified cement that contains many waste components shows a smaller

electric conductivity. It was determined that the amount of cement has a strong influence on the electric conductivity compared with the waste amount.

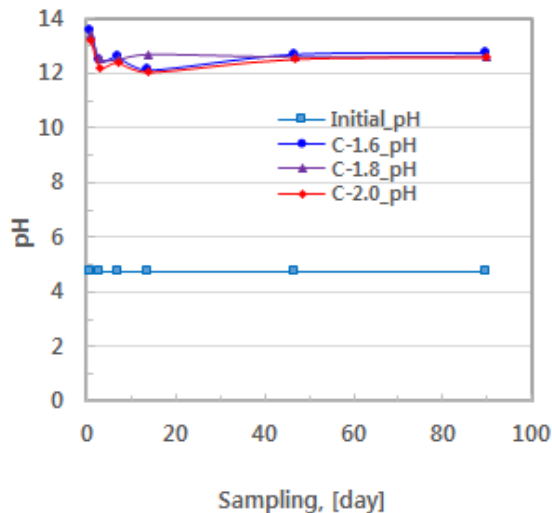


Fig. 1. pH of cement solidification.

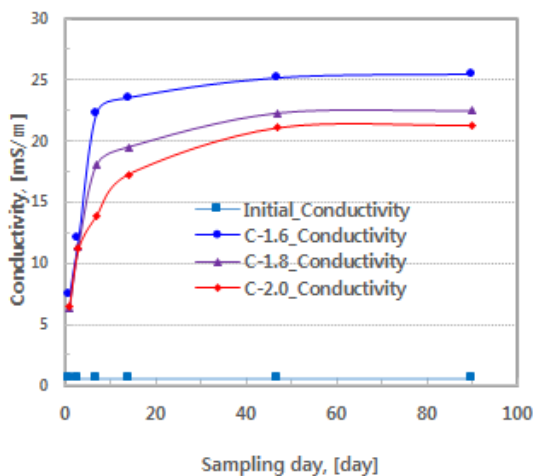


Fig. 2. Electric conductivity of cement solidification.

3. Conclusions

To study the characteristics of cement solidification on metal hydroxide waste for permanent disposal, the pH and electric conductivity of solidified cement was measured and analyzed. The pH of solidified cement immersed in demineralized water indicates some fluctuation from day 1 of sampling until about day 40. It approaches about 13 after 40 days of sampling. The immersion test on cement solidification indicates a strong alkali property. The electric conductivity of

solidified cement immersed in demineralized water shows a rapid increase from day 1 of sampling until day 7. After 47 days of sampling, a saturation trend is shown. The solidified cement that contains a large cement component shows greater electric conductivity, while the solidified cement that contains many waste components shows smaller electric conductivity. It was determined that the amount of cement has a strong influence on the electric conductivity compare with the waste amount.

REFERENCES

- [1] G.N. Kim, S.S. Kim, H.M. Park, W.S. Kim, J.K. Moon, J. H Hyeon, "Development of Complex Electrokinetic Decontamination Method for Soil Contaminated with Uranium", *Electrochimica Acta*, 86, 49-56 (2012).
- [2] G. Kim, H. Won, W. Oh and C. Jung, A Study on Aging Effect in Removal of Radionuclides from Soil by Electrokinetic Method, *J. Korea Society of Waste Management*, Vol. 21, No.3, pp. 243-252, (2004).
- [3] G.N. Kim, H.J. Won, C.H. Jung, J.H. Yoo, W.Z Oh, "Effect of pH on Soil Remediation by Electrokinetic", *Journal Korean Solid Wastes Engineering Society*, 17, 781-788 (2000).
- [4] G.H. Jeong, K.J. Jung, S.T. Baik, U.S. Chung, K.W. Lee, S.K. Park, D.G. Lee, H.R. Kim, Solidification of Slurry Waste with Ordinary Portland Cement, Report. KAERI/RR-2194/(2001).
- [5] S.W. Shin, K.D. Kang, H.S. Kim, J.K. Son, Y.J. Choi, Studies on Test and Assessment of Immobilization and Clearance Methodology for Soil and Uranium Wastes, Report. KAERI/CM-758/(2003).
- [6] J. H Kim, B. G. Ahn, H. Y Kim, H. H Park, A Study on the Solidification of Borate Wastes using Cement, *J. of Waste Management* 6(2), 161-166, (1989).
- [7] Y.J. Lee, K.W. Lee, B.Y. Min, D.S. Hwang, J.K. Moon, "The Characterization of Cement Waste Form for Final Disposal of Decommissioning Concrete Wastes", *Annals of Nuclear Energy*, 77, 294-299 (2015).

Polymeric Nanofiltration Membranes for Desalination & Nuclide Separation

Hyung-Ju Kim*, Keun-Young Lee, and Bum-Kyoung Seo

Decommissioning Technology Research Division, Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Republic of Korea

*hyungjukim@kaeri.re.kr

1. Introduction

Within several decades, global interest in clean water has been significantly increased.¹ In nuclear engineering field, water (or seawater) is used for moderator or coolant, and this is because every nuclear power plant is built close to coastal area.² Since nuclear energy is extremely dangerous and unknown territory considering common sense, wastewater treatment coming from nuclear facility is very important and highly focused area.

Among several techniques, membrane based separation has been rising technology due to low energy intensity, high special efficiency, and pressure driven process.³ To overcome other technology, membrane materials used for separation should possess high permeability (throughput) and high selectivity (process efficiency).⁴

There are several membrane based wastewater treatment techniques, such as microfiltration, ultrafiltration, nanofiltration, reverse osmosis, and forward osmosis. With these various techniques, selection of appropriate operation and membrane is the most important issue in membrane separation community.

In this work, we describe the nanofiltration technique for commercial polymeric membranes to apply desalination and aqueous nuclide separation. Nanofiltration of three different aqueous nuclide solutions is studied to understand the permeation characteristics of the commercial polymeric membranes. We demonstrate that the nanofiltration membranes display decent permeation properties, highly rejecting nuclides considering their salt rejection at room temperature conditions.

2. Main title

Nanofiltration membrane performance was evaluated with a lab-made crossflow test unit shown in Fig. 1.

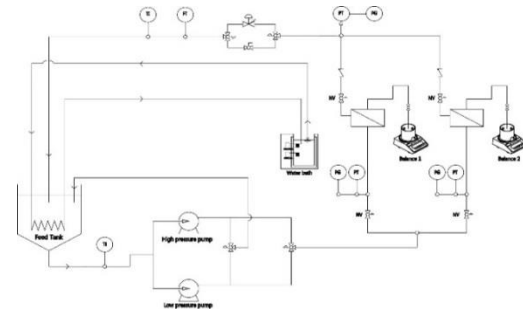


Fig. 1. Schematic diagram of the nanofiltration system. (TI: temperature indicator, FT: flow transmitter, PT: pressure transmitter, PG: pressure gauge, and NV: needle valve)

Fig. 2 shows desalination data for 35,000 ppm of NaCl aqueous solution under 30 bar at room temperature. For the feed reservoir, the seawater is modeled which indicates the 35,000 ppm (3.5wt%) of NaCl aqueous solution. In Fig. 2(a), the conductivity values are well coincided with the designated value 54.7 mS/cm. After passing nanofiltration membranes (NF90 and NF270), the conductivity values are significantly decreased to 19 mS/cm and 38 mS/cm which are corresponding to 12,400 ppm and 24,300 ppm, respectively. With regard to rejection, they are 82% and 37%, and these are not qualified rejection value as desalination membranes, and much lower rejection comparing to the general reverse osmosis membranes. Flux is calculated by dividing the volumetric rate with membrane area to obtain membrane-area-normalized value. Due to the usual trade-off property coming from the separation membranes, NF90 shows 160 LMH and NF270 shows 210 LMH. Even if the NF270 has high flux, NF90, which shows high rejection, is more appropriate membrane for nuclides separation considering the nuclear engineering area.

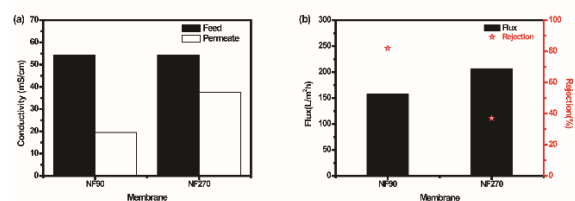


Fig. 2. Desalination results regarding (a) conductivity and (b) flux & rejection from NF90 and NF270 under 30 bar at room temperature.

Fig. 3 shows nuclide separation results for aqueous nuclide solution containing 10 ppm of Cs, Sr, and Co using NF90 nanofiltration membrane under 30 bar at room temperature. Through the solution-diffusion mechanism, the nuclides are rejected from 10 ppm to 1.6 ppm for Cs, 2.2 ppm for Sr, and 3.4 ppm for Co. This significant downgrade of concentration is attributed to the nuclides rejection by polyamide chains and their high affinity to water. The concentration downgrade is corresponding to the rejection of 83%, 78%, and 64%. Similar to the desalination result, nuclides separation also has trade-off property between flux and rejection. Among three different nuclides solutions, aqueous Cs solution describes most stable flux and rejection with narrow error bars. This represents that polyamide based nanofiltration membrane is well-applied to the Cs comparing to other main nuclides. Even higher rejection can be anticipated tuning the feed pressure, and membrane surfaces.

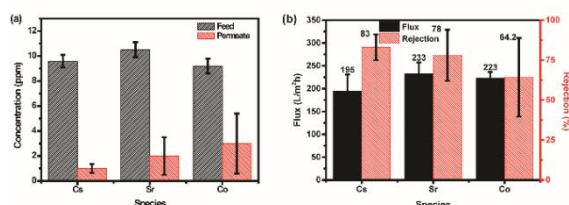


Fig. 3. Nuclide separation results regarding (a) concentration downgrade and (b) flux & rejection from NF90 under 30 bar at room temperature.

Table 1 summarizes desalination and nuclide separation results for NF90 at room temperature.

Table 1. Performance summary of NF90 membrane

	Flux (L/m ² h)	Feed conc. (ppm)	Permeate conc. (ppm)	Rejection (%)
NaCl	158	35,000	12,600	82
Cs	195	9.6	1.6	83
Sr	233	10.5	2.2	78
I	223	9.2	3.4	64

3. Conclusion

This work demonstrates that commercial polymeric membranes, NF90 & NF270, could apply to separation of nuclides. The commercial polyamide based polymeric membranes show high rejection for model seawater and aqueous nuclides solutions (Cs, Sr, and Co) at room temperature. This work on polymeric membranes for nuclide separation leads to an understanding of the behavior of polymeric membranes. Future work using radioactive nuclides and mixed nuclides solution is warranted.

REFERENCES

- [1] T. Hillie, M. Hlophe, "Nanotechnology and the Challenge of Clean Water," *Nature Nanotechnology*, 2, 11, 663-664 (2007).
- [2] S. Otake, T. Kobayashi, "Sedimentation and Remobilization of Radiocesium in the Coastal Area of Ibaraki, 70 Km South of the Fukushima Dai-Ichi Nuclear Power Plant," *Environmental Monitoring and Assessment*, 185, 7, 5419-5433 (2013).
- [3] H. Strathmann, "Membrane Separation Processes: Current Relevance and Future Opportunities," *AIChE Journal*, 47, 5, 1077-1087 (2001).
- [4] B. D. Freeman, "Basis of Permeability/Selectivity Tradeoff Relations in Polymeric Gas Separation Membranes," *Macromolecules*, 32, 2, 375-380 (1999).

Development of 3D Programs for Efficient Radiation Dose to Data Management

SangBok LEE*, Seung-gun AN, Yeo-jib CHOI, Jeong-soo PARK, Tae-ryong KIM, and Seung-jin CHO

Sunkwang T&S Co, Ltd. Sunkyung Officetel 20F #3, Gongwon-ro, Guro-gu, Seoul, Korea

*Lsbguy87@gmail.com

1. Introduction

In these days, the dismantling of nuclear power plants in Korea is a hot issue because of the new government launching the rule out the nuclear power plants. At present, there are eight nuclear power plants that will be obsolete. (Excluding Gori Unit 1, construction standards of the 1980s) All of these are expected to be dismantled by 2030. According to this, the development of technologies for dismantling is urgent in addition to data management is too. Because it is very important part in dismantling.

The basis of decommissioning nuclear power plants is residual radiation measurement. This part is very important because residual radiation measurements must be completed to prove that the waste is harmless to humans. For this reason, we have developed a 3D program that simplifies the management of residual radiation data. This 3D program is a very simple and convenient to manage residual radiation measurement data.

In this paper, we discuss the usefulness of the 3D program that can efficiently manage the data of scan survey among the decommissioning of nuclear power plants.

2. Main title

2.1 Background

This study was carried out in 2016 to study the residual radiation and activity effect for the rescission of the site regulation by the TRIGA Mark II & III. Residual radiation measured value was measured by two ways. One is direct method such as scan survey (LB-124, Radeye G-10), and the other is indirect method (Smear method). The data were collected and applied to all programs.

2.2 Development of 3D program modeling

The modeling structure of the 3D program is shown in the following.

Table 1. 3D program modeling structure

No	Session
1	Selection of survey building
2	Grid segmentation and perform scan survey
3	Building investigation and drawing
4	Development to 3D program
5	Apply dose value to 3D program
6	Check the result and make use of program

For scanning, spaces were divided by the grid of 1x1(m) (Fig. 1) and the total number of the measurement data in the research was 7,502 pieces. Based on the scan survey, the building was actually completed the drawing of the building as shown in Fig. 2. The completion period to the drawing was about two months. Based on this, we completed the 3D program.



Fig. 1. Grid segmentation for scan survey.



Fig. 2. Create drawing for 3D programs.

2.3 Application and effects of 3D program

Based on the drawing of TRIGA Mark II & III simulated in 3D, we applied dose information that we surveyed to each grid segmentation area. Dose information was entered for each building, room and grid segmentation. An example of grid segmentation areas is shown to Fig. 3. After entering all the results, we were able to see all of the dose results in the 3D program by clicking the section that we want to see. When you click the section, the input results are displayed as shown in Fig. 4.

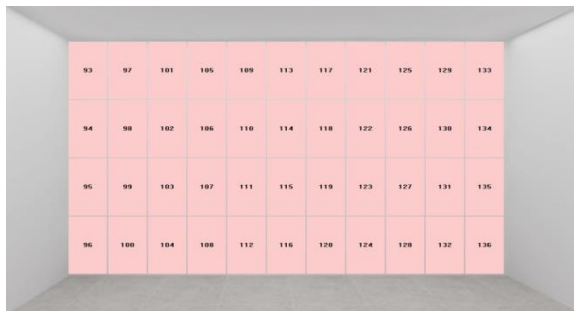


Fig. 3. Classification by 3D program grid segmentation.

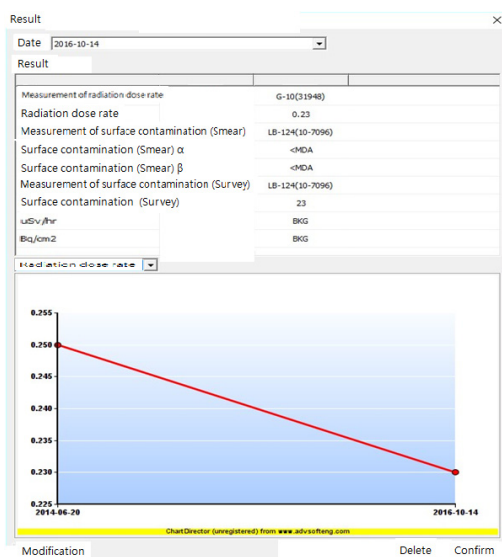


Fig. 4. Radiation dose view in grid segmentation units.

Completed design of TRIGA Mark II & III is as follows in Fig. 5. This is also the first screen displayed in the program. It is possible to search all dose information of all grid segmentation area in the building.



Fig. 5. Classification by 3D program grid segmentation.

3. Conclusion

Gori nuclear power plant unit 1 was stopped and waiting for its demolition. The dismantling of nuclear power plants involves a lot of work including many construction methods, decontamination and waste disposal and it is necessary to systematically manage the output from these processes. It is essential to collect and manage information of areas that we want to decommission such as dose information and history of the facility. This program will be very effective in managing large amounts of information and can easily see implicit information. Therefore, this 3D program development in this aspect will be very useful for dismantling nuclear power plants in the future.

REFERENCES

- [1] KAERI, "Evaluation of Residual Radiation and Radioactivity Level of TRIGA Mark II, III Research Reactor Facilities for Safe Decommissioning", 1999.
- [2] Sunkwang T&S, "For derestriction of site regulation by TRIGA Mark II, III, radiation & radioactivity survey and classification to radioactive waste", 2016.
- [3] KHNP web site, "General Status of Nuclear power plants".

Extraction of Actinide and Lanthanide From Nitric Acid Medium Using Ionic Liquid

Iksoo Kim*, Dongyong Chung, Keunyoung Lee, and Bum-Kyoung Seo

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

*niskim@kaeri.re.kr

1. Introduction

Ionic Liquids (ILs) are receiving increased attention in the area of nuclear fuel cycle and waste management because they have several attractive properties suitable for aqueous and non-aqueous reprocessing applications. ILs are being evaluated as an alternative to the conventional diluent in aqueous reprocessing and as a substitute to the traditional high-temperature molten salts in the non-aqueous reprocessing of spent nuclear fuels [1, 2]. In this study, the application of ILs, 1-alkyl-3-methyl imidazolium bis(trifluoromethylsulfonyl) imide ($C_n\text{mimTf}_2\text{N}$, $n = 4, 6, 8$), in the extraction of Am(III) and Eu(III) ions from nitric acid medium using *n*-octyl(phenyl)-*N,N*-diisobutylcarbamoyl methyl phosphine oxide (CMPO) and tri-*n*-butyl phosphate (TBP) as extractants was investigated. The distribution ratios of Am(III) and Eu(III) in CMPO-TBP/ $C_n\text{mimTf}_2\text{N}$ were measured as a function of various parameters such as the concentrations of nitric acid, CMPO, and TBP. Extraction experiments were also carried out at varying temperatures to determine the influence of temperature on the extraction process

2. Experimental

2.1 Chemicals and equipment

All the chemicals and reagents used in the study were of analytical grade and used without further purification. Ionic liquid, $C_n\text{mimTf}_2\text{N}$, was procured from C-TRI, Korea. CMPO and TBP were obtained from Merck. $^{241}\text{Am(III)}$ and $^{152}\text{Eu(III)}$ were obtained from IPL. To remove the water and volatile impurities, $C_n\text{mimTf}_2\text{N}$ was dried under a reduced pressure at 373 K, and the residual water concentrations measured using a Karl Fisher moisture content meter (Metrohm 831 KF Coulometer) were below 100 ppm.

2.2 Experimental method

All the extraction studies were carried out at 298 K. The organic phase (1 mL) was equilibrated with the same volume of aqueous phase for 3 hr. Extraction of Am(III) and Eu(III) as a function nitric acid was studied by equilibrating the organic phase with aqueous phase containing desired concentrations of CMPO and nitric acid spiked with $^{241}\text{Am(III)}$ and $^{152}\text{Eu(III)}$ tracer. After 3 hr of equilibration, the radioactivity of $^{241}\text{Am(III)}$ and $^{152}\text{Eu(III)}$ distributed between organic and aqueous phases was measured using MCA (Canberra DSA-1000, HPGe detector). The distribution ratio (D_M) and separation factor (SF) were determined using Eq. (1) and (2).

$$D_M = [M]_{\text{org}}/[M]_{\text{aq}}, M = ^{241}\text{Am or } ^{152}\text{Eu} \quad (1)$$

$$\text{SF} = D_{\text{Am}}/D_{\text{Eu}} \quad (2)$$

3. Results and discussion

The distribution ratio of $^{241}\text{Am(III)}$ and $^{152}\text{Eu(III)}$ in a solution of 0.05 M CMPO-1.1 M TBP/ $C_n\text{mimTf}_2\text{N}$ as a function of concentration of nitric acid is shown in Fig. 1. It also shows the distribution ratio obtained in a solution of 0.05 M CMPO-1.1 M TBP/*n*-dodecane (*n*-DD). In cases where ILs ($C_n\text{mimTf}_2\text{N}$) are used as the diluent, increasing aqueous phase acidity causes a decrease in the extraction of Am and Eu on the contrary to the case of using *n*-DD as the diluent. Extraordinary extractions of Am and Eu are observed when ILs are used as diluents. D values of ILs are much greater than those of *n*-DD. The use of ILs enhances the extraction of Am and Eu. D decreases with increase in the chain length of alkyl group of the cationic part of the ILs, which is attributed to the hydrophobicity of the ILs. Higher aqueous solubility of the butyl form is responsible for the higher metal ion extraction.

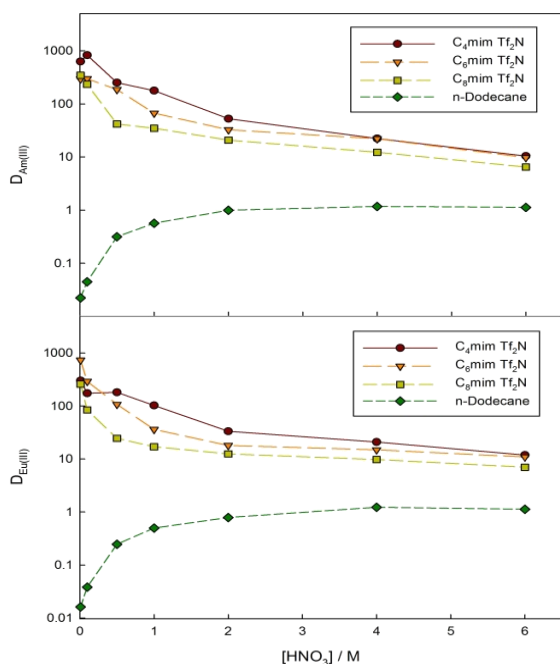


Fig. 1. Distribution ratio of Am and Eu with $[\text{HNO}_3]$.

The number of CMPO molecules involved in the extraction can be determined by slope analysis method. This was obtained by measuring the distribution ratio of Am and Eu as a function of CMPO concentration in ionic liquid phase, and plotting the logarithmic value of the measured distribution ratio against CMPO concentration and the results are shown in Fig. 2. Linear regression analysis of the data results in a slope of 2.5 for Am

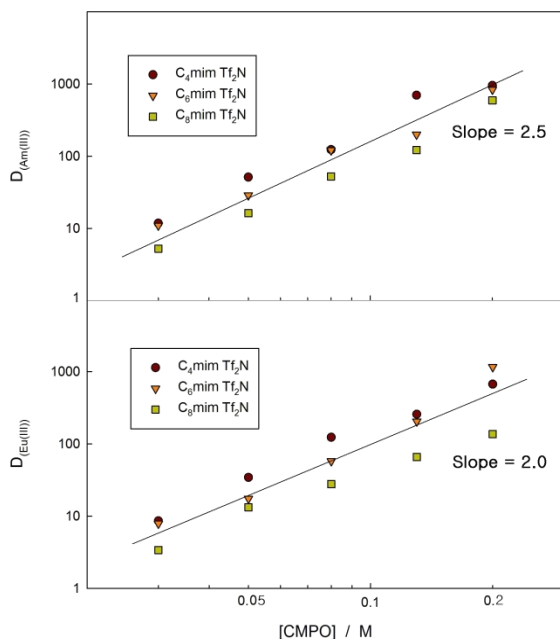
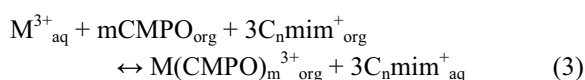


Fig. 2. Distribution ratio of Am and Eu with $[\text{CMPO}]$.

and 2.0 for Eu, indicating that 2 or 3 molecules of CMPO are involved in the extraction step.

A change of the chain length of the alkyl group of the RTILs did not influence the stoichiometry of the extracted complex. The results suggest that the extraction of Am and Eu by CMPO in RTILs from nitric acid medium proceeds through the following cation-exchange mechanism.



Extraction experiments were also carried out at varying temperatures to determine the influence of temperature on the extraction process, and the thermodynamic parameters were calculated. The distribution ratios were found to decrease with increasing temperature, indicating that the extraction process is exothermic.

4. Conclusion

Feasibility of ILs as novel media for the extraction of Am and Eu was examined. CMPO dissolved in IL showed much high extraction ability compared to in an ordinary diluent, n-DD. Partitioning of the metal cations into the IL with CMPO proceeds through cation exchange mechanism, which is different from that of the conventional solvent extraction system. The results suggest that ILs are a promising medium for Ln/An elements separation.

References

- [1] P. N. Pathak, "N,N-Dialkyl amides as extractants for spent fuel reprocessing - an overview", *Journal of Radioanalytical and Nuclear Chemistry*, 300, 7–15 (2014).
- [2] T. Prathibha, K.A. Venkatesan, B. Robert Selvan, M.P. Antony and P.R. Vasudeva Rao, "Anomalous extraction behavior of americium(III) in some diglycolamide isomers present in ionic liquid medium", *Radiochim. Acta* 100, 907-913(2012).

Case Study of Interview for HSA at Decommissioned NPPs in the US

Donghee Lee

Central Research Institute Korea Hydro and Nuclear Power Co. LTD, 1312-70 Yuseon-daero, Yuseong-gu,
Deajeon, Korea
donghee.lee@khnp.co.kr

1. Introduction

With regard to decommissioning and site reuse of nuclear power plant, the IAEA [1] identifies the site release from regulatory control as the final step in decommissioning.

Generally, the site release can be achieved provided the level of residual radioactivity at the site is demonstrated to meet the site release criteria. This demonstration is performed by implementing "Radiation Survey and Site Investigation (RSSI) process."

Historical Site Assessment (HSA) which is the first step of the RSSI assesses a history of the radiological contamination from the start of the site activity to the present time. By investigating documents produced during operation period and interviewing the persons concerned to collect undocumented information.

Therefore, this paper tries to study cases of interviews which have been performed the HSA at decommissioned NPPs in the US.

2. Purpose & type of interview

2.1 Purpose of interview

The purpose of the interviews or personnel questionnaires is to obtain "tacit knowledge". Tacit knowledge is undocumented, and consists of unique capabilities, skills and knowledge. Interviews with current or previous employees are conducted to collect first-hand information from the site or facility. Interviews cover general topics such as radioactive waste handling procedures. Results of early interviews are used to guide subsequent data collection activities.

2.2 Type of interview

Several different interviewing approaches may be used to gather information. These approaches should be tailored to best meet the site needs.

One common approach is a single interviewer and a single interviewee. A positive to this approach is that it is less intimidating to the interviewee and may be viewed as a more confidential approach to gathering information. A negative to this aspect is that the interviewer must continually split focus between asking questions and documenting responses.

Another approach is to have two interviewers and a single interviewee. This allows one interviewer to ask questions and a second to be the scribe. Although eliminating the potential concerns with recording interviews, the interviewee may view two interviewers with some concern, hence potentially resulting in less openness and potentially less information.

A third approach is group interviews with multiple interviewees at one time. In this case, it is essential that at least two interviewers gather the information. Information may be provided at a quick enough pace that one interviewer could not gather the data and provide the questions.

3. Overseas cases of interview

3.1 Connecticut Yankee [2,3]

Personnel interviews of CY were conducted with current and former employees and contractors involved in operations, maintenance and radiation protection activities at the site. In order to assure confidentiality for the interviewees, plant security

conducted the interviews. Also vendors and contractors were informally interviewed to verify, provide or clarify data used to develop the HSA document. Consequentially, 47 formal interviews, based initially on questionnaires filled out by employees regarding contamination issues, were conducted.

3.2 Maine Yankee [2,4,5]

Maine Yankee also interviewed terminating employees for any recollection of such events. Informal interviews occurred with all personnel. During their final exit from the site, security personnel asked if the individual was aware of any radiological or other spills which may have occurred on-site. Between August 2001 and December 2002, approximately 150 observations were noted based on the questionnaire program. These observations represented a combination of current and past employees, primarily from the operations and radiation protection departments.

3.3 Zion [2,6]

Two types of interview questionnaires were used in the conduct of the Zion HSA. The first type of questionnaire was designed for permanent site production personnel. The second type of interview questionnaire was used during the personnel exit process (permanent and contractor staff). Typically, individuals were provided the questionnaire as a part of the standard station exit process. These exit interviews/questionnaires were obtained more as a quality check on information obtained during permanent staffing interviews. During the conduct of the Zion HSA, over 300 personnel interviews occurred with current and previous Zion personnel both utility and long term contractor personnel.

through a total of 47 interviews. In Maine Yankee, informal interviews of all employees as well as about 150 formal interviews were performed. In Zion, over 300 separated by incumbents and retirees personnel interviews were performed.

According to permanent shutdown of Kori-1, the pre-work to decommission the plant is currently underway by Korea Hydro & Nuclear Power CO., LTD (KHNP).

The HSA as a part of decommissioning is essential to investigate a history of contamination. Furthermore, interviews with the persons concerned are need to confirm a history of undocumented contamination. Therefore, all these works will be carried out for the decommissioning of Kori-1 with reference to this study.

REFERENCES

- [1] IAEA, "Safety Guide No. WS-G-5.1", 2006.
- [2] EPRI, "Capturing Historical Knowledge for Decommissioning of Nuclear Power Plants", 1009410, 2004.
- [3] NRC, "Haddam Neck Historical Site Assessment Supplement", ML012430208, 2001.
- [4] NRC, "Maine Yankee Historical Site Assessment", ML012830192, 2001.
- [5] NRC, "Maine Yankee Historical Site Assessment Supplement", ML021130343, 2002.
- [6] Rock E. Aker, CHP, "Zion Station Historical Site Assessment", ComEd, 1999.

4. Conclusion

This paper assesses interview cases which have been performed for the HSA at decommissioned NPPs in the US. In Connecticut Yankee, information about contaminated site and facilities was collected

Segmentation and Packaging Plan for the Reactor Pressure Vessel

Hyun-Jae Yoo and Byung-Sik Lee*

Dankook University, 119, Dandae-ro, Dongnam-gu, Cheonan, Chungnam, 31116, Korea

*bslee@dankook.ac.kr

1. Introduction

One of the most challenging tasks during plant decommissioning is the removal of highly radioactive internal components from the reactor pressure vessel (RPV) in the Kori unit 1. It is also expected to belong to the most difficult activities, because these must be segmented underwater due to the severe radiological conditions of the RPV internals [1]. In order to support this activity, the systems with cleaning of the water in the reactor cavity need to remain operational. It is therefore recommended that the reactor internals are removed as early as possible in the plant dismantling sequence, so that these water systems and their associated support systems can be released for decommissioning. This minimizes the costs of maintaining these systems in operation after permanent plant shutdown.

Moreover, after spent fuel, reactor internals constitute the next significant contributor to the radiological inventory of the decommissioned NPP. Therefore, the spent fuel removal and the early removal of the reactor internals significantly reduce the total site radiological hazard.

2. Segmentation and Packaging Plan

2.1 General Procedures to Develop the Plan

The segmentation and packaging plan is necessary to define in detail the sequential steps required to segment, separate, and package each individual component of RPV, based on an activation analysis and component characterization study. The plan is to describe the strategy for dismantling the components so that they can be disposed of in the most cost effective manner which can be governed by many factors such as waste container selection, disposal costs, transportation requirements, etc. [1]. The plan is prepared early in the planning phase before the dismantling activity considering the following sequences and processes [2],

- Collection of various types of design and operating data of the NPP
- Development of a 3D model of the reactor based on the collected data for the comprehensive understanding of the reactor
- Activation analysis of the reactor vessel and internals to define waste characterization and classification of the reactor components
- Evaluation of disposal options depending on waste characterization and the waste acceptance criteria
- Conceptual tooling development based on the activation analysis, waste characterization, and disposal option.
- Segmentation and packaging plan contains optimization packaging efficiency while considering segmentation schedule. And the plan defines type and quantity of waste containers and defines location and number of cuts per waste container

Based on the dismantling experience of the RPV internals, however, actual developing process of the segmentation and packaging plan is usually to start considering the end of the process such as handling of the containers, the waste acceptance criteria, what type and size of containers are available for the disposal options [1].

2.2 Disposal Requirements

In 2013, radioactive waste classification system in Korea was revised reflecting the international standard recommended by International Atomic Energy Agency (IAEA) [3], and specifies Low and Intermediate Level Waste (LILW) further into the Intermediate Level Waste (ILW), Low Level Waste (LLW) and Very Low Level Waste (VLLW) depending on its specific radioactivity [4].

The only disposal option currently available in Korea is the Gyeongju Centralized LILW Disposal Facility of cavern type repository. This repository is

designed to accept Very Low Level Waste (VLLW), Low Level Waste (LLW) and Allowable Intermediate Level Waste (AILW)*, in accordance with the corresponding Waste Acceptance Criteria (WAC), issued by KORAD as owner and operator of the facility.

Per the WAC for Gyeongju, the radioactive waste may be classified as VLLW, LLW or AILW. These classifications are based on the content of both long- and short-lived radionuclides in accordance with National Notice [4]. According to the WAC, the VLLW, LLW and AILW require immobilization and encapsulation inside the packages.

2.3 Waste Disposal Packages

In accordance with the segmentation and packaging strategy Non-AILW is stored in the specially designed canisters (SDCs), which are then placed into a temporary storage building or in an ISFSI [4]. The customer will supply all waste packages for the segmentation activity in compliance with appropriate KORAD WAC.

Until now waste containers are classified with steel drums, concrete containers, HIC, and polyethylene containers in Gyeongju. The steel drums are classified with 200L and 320L drums, and the concrete containers are classified with circular concrete containers and rectangular concrete containers [5]. LILWs packed in 200L drums are disposed of in a 16-pack concrete disposal container. The extra compressed 200L drums are packed again in 320L drums. The nine 320L drums are disposed of in a 9-pack concrete disposal container. The physical characteristics of the disposal container for the 200L drums and 320L drums are described in Table 4. The other concrete containers are disposed as itself in the repository.

2.4 Selection of a Cutting Technology

Segmentation of the RPV internals can be performed using thermal and mechanical cutting techniques and the best technique chosen will depend on the application and item to be cut.

Selection of the proper cutting technology in Kori unit 1, however, will be made considering the following factors in detail based on the past experience,

- Occupational safety and optimization for

radiation protection

- Secondary waste minimization
- Process safety and simple operation
- Reliability and maintainability
- Cutting capacity

3. Conclusion

It is important to well prepare the dismantling of a RPV internals in advance. A detailed study of the optimum dismantling scenario must be done upfront, considering the available plant systems and infrastructure.

3D models help complete these tasks as well as for determining the logistics of component placement and movement in the reactor cavity. With optimized planning and predictable results, these complicated tasks can be performed while maintaining occupational exposure and controlling project costs.

The amount of cutting should be optimized with respect to the number of cuts and the size of the final waste packages. The cutting plan should minimize the number of cuts, produce waste pieces that can be easily packaged, and minimize the final waste volume. Segmentation should begin with the least irradiated components in order to gain experience with the tooling before applying it to highly radioactive components, where tooling failure could significantly increase personnel exposure.

REFERENCES

- [1] Josept Boucau, et.al., Best Practices for Preparing Vessel Internals Segmentation Projects, PREDEC 2016; Title of the Communication, February 16~18, 2016, Lyon, France.
- [2] Michael Anderson and Steve Garner, Segmentation Planning- Integration of Waste Management Issues in Decommissioning, IAEA, Madrid Spain, 2016.
- [3] IAEA, Classification of Radioactive Waste, General Safety Guide No. GSG-1, 2009.
- [4] NSSC, Regulation on Classification of Radioactive Waste and Self Disposal, 2014.
- [5] Jin Beak Park, et.al., Wolsong Low and Intermediate Radioactive Waste Disposal Center: Progress and Challenges, Nuclear Engineering and Technology, Vol.41 No.4, 2009.

Study on Induction Melter Technology to Treat Decommissioning Metallic Radwaste Generated From Kori Unit 1

Sung-Hoon Hong*, Seokju Hwang, and Cheon-Woo Kim

KHNP CRI, 70, 1312-gil, Yuseong-daero, Yuseong-gu, Daejeon 34101, Republic of Korea

*sunghoon.hong@khnp.co.kr

1. Introduction

Development of metallic radwaste melting technology has been pursued actively since the 1960's and has become the technology for processing of metallic radwastes generated at the time of operation and decommissioning of nuclear facilities with the construction of commercial scale facilities for melting of metallic radwastes since the 1990's [1-2]. The majority of the radwastes generated at the time of the decommissioning of Kori Unit 1 are metals with approximately 86%. Among metallic radwaste, about 59% will be very low level (VLL) that can be recycled in the nuclear industry. In particular, there is a need for enhancement of the economic feasibility of the decommissioning industry through the reduction of disposal volume through recycling in the case of VLL metallic radwaste. In this study, the current status of the induction melter technology that is being applied most widely among the diverse range of melter technologies for the processing, volume reduction and recycling of the metallic radwaste was examined and the possibility of application of this technology to the decommissioning industry for Kori Unit 1 was considered.

2. Induction melter technology

2.1 General principle

Induction melter melts the metals inserted into the ceramic body by generating Joule heat with induced current generated through current flowing in the induction coil that surrounds the ceramic body.

Induction melter is composed of ceramic body, induction coil, discharge outlet (spout), hood and tilting device, etc. as shown in Fig. 1.

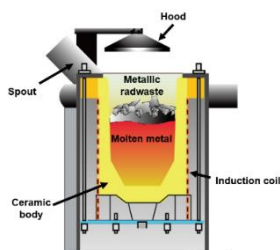


Fig. 1. Schematic diagram of induction melter.

2.2 Advantages and disadvantages of induction melter

Advantages of the induction melter includes ease of discharge since the tilting method is applied after having melted metal, ability to minimize the generation of exhaust gas in comparison to other melting methods as well as spontaneous heating and stopping. Disadvantages include difficulties in treatment of metal with a high level of radiology since the sealing of the top portion of the melter is not easy, ability to melt only metals and concerns for steam explosion if the metal being melted contains moisture. Moreover, if hard bridging is created due to the accumulation of slag on top of the molten metal, there is possibility of leakage of melted substance and explosion of the smelting furnace due to the loss of control because of the presence of accumulated slag.

2.3 Status of overseas and domestic

The USA, Germany, Sweden, England, France, Russia and Korea have been operating metallic radwastes processing facilities for various metallic radwastes including carbon steel (CS), stainless steel (STS), aluminum and copper since 1992. Among these countries, Germany and Sweden are known to operate their facilities exemplarily. KNF in Korea has been collecting the radioactive substance through slag by melting metallic radwastes and executing clearance level disposal for the remaining molten metal since 2011. The performances of the treatment by facilities in overseas countries and Korea are summarized in the Table 1.

Table 1. Treated amounts at each facility

Owner/Facility (Country, Starting year)	Target wastes	Capacity (ton/batch)	Products
EnergySolutions (USA, 1992)	CS, STS, Aluminum	20	Ingot, shielding block, waste container
CARLA (Germany, 1989)	CS, STS, Aluminum, Copper, Lead	3.2	Ingot, shielding block, waste container
STUDSVIK (Sweden, 1987)	CS, STS, Aluminum, Brass, Copper, Lead	3.5	Ingot, shielding block
Capenhurt (England, 1994)	Aluminum, Brass, Copper	4	Ingot
CENTRACO (France, 1999)	CS, STS, etc.	4	Ingot, tube
ECOMET-S (Russia, 1994)	CS	2.5	Ingot
KAERI (Korea, 2013)	CS, STS	0.35	Ingot
KNF (Korea, 2011)	CS, STS	0.25	Ingot

*CS : carbon steel, **STS : stainless steel

3. Consideration of application

In the event of considering the installation of induction melter for the processing of metallic radwastes generated at the time of decommissioning of Kori Unit 1, various aspects including operating conditions, key components and capacities, and layout plans, etc. were considered.

3.1 Condition of operation

The condition of operation for the induction melter is as follows (Table 2).

Table 2. The condition of operation

Throughput	Melting for 45min, drainage for 15min
# of melters	2ea
Operation method	One is in operation and the other one is charging and heating mode
Power supply	Dual type inverter
Drainage method	Tilting type
Required area for installation	About 100m ²
Power requirement	About 1.5MW

The evaluated power requirement is estimated based on the below information (Table 3).

Table 3. The evaluated power requirement

Caloric requirement for raising of temp. to 1,550°C for CS	About 354,500kcal/ton (i.e., 412kWh/ton, 1kWh=860kcal, 1kcal=0.00116279kwh)
Applied energy transformation efficiency	About 65%
Melter transformation efficiency	412kWh/ton/0.65=634kWh/ton
Required electricity capacity	634kWh/tonx1.5ton/h x60min/45min=1.27MW (about 20% surplus consideration : 1.5MW)

3.2 System configuration

Floor plan and frontal view of the layout plan for

1.5MW induction melter for the processing of metallic radwastes at the time of decommissioning of Kori Unit 1 are illustrated in the Fig. 2.

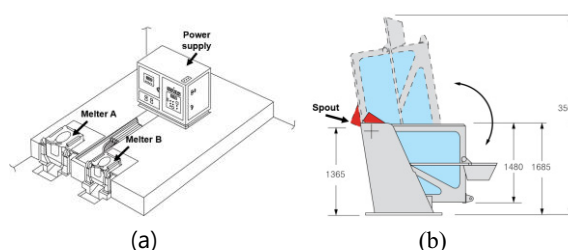


Fig. 2. (a) System arrangement for dual induction melter and (b) tilting type induction melter.

4. Conclusions

Evaluation was made on the 2 units of induction melter equipped with the dual type inverter power supply capable of processing 1.5 tons of carbon steel per hour, and the method of their application was established to process metallic radwastes generated at the time of decommissioning of Kori Unit 1. Induction melter is being used most widely throughout the world and its performances have been proven. Accordingly, it is most appropriate for processing of metallic radwastes. In the case of Kori Unit 1, it is necessary to process metallic radwastes decommissioning within approximately 5 years. Accordingly, it is desirable to install dual units in order to elevate the processing rate. However, although it is determined that there would not be any difficulty in the case of very low level radwastes, it is necessary to determine in the near future whether it can be applied to low level radwastes with a relatively higher level of radioactivity.

REFERENCES

- [1] KHNP CRI, "Study on Commercialization Measures for Treatment of Large Metallic Radwaste", Technical Report (2013).
- [2] KHNP CRI, "Feasibility Study on Treatment and Recycling for Radioactive Metal Waste", Technical Report (2014).

Magnetic Clay Microspheres Prepared by Spray-Drying for Adsorption of Cs

Chan Woo Park^{1*}, Gyeong Jun Chung², Juri Hwang^{1,2}, Hee-Man Yang¹, and In-Ho Yoon¹

¹Decontamination & Decommissioning Research Division, Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Republic of Korea

²Department of Chemical Engineering, Chungnam National University, 99, Daehak-ro, Yuseong-gu, Daejeon, Republic of Korea

*chanwoo@kaeri.re.kr

1. Introduction

Natural clay minerals have high cation adsorption properties because they have an anionic layered structure. Since the price of clay is very low, in addition, the clay is widely used as an adsorbent for hazardous materials. Smectite, vermiculite, and illite are general 2:1 type clay minerals, and each clay mineral shows different Cs adsorption properties. Expendable clay minerals, such as smectite and vermiculite, have high adsorption capacity for Cs. Even though illite has a low Cs adsorption capacity, it has a frayed edge site which has high selectivity for Cs ion.

According to the soil classification based on particle size, clay is classified as less than 2 micrometers. In order to remove the radionuclides in the waste water or contaminated ground water by using clay as an adsorbent, a column process is generally utilized. However, because of the small size of the clay, the permeability is low and thus the decontamination rate is low. In order to solve such a disadvantage, small clay particles must be processed to make large particles. It is also preferred that the adsorbent has magnetism so that it can be recovered by the magnet after use for easy recovery of the adsorbent used.

Among the various microparticle preparation methods, the spray drying method easily provides a microsphere. Various morphology and size can be controlled according to manufacturing conditions.

In this study, we prepared microsphere type adsorbent by spray drying clay mineral and magnetic iron oxide nanoparticles. Montmorillonite, vermiculite, and illite were used as clay minerals and poly (vinyl alcohol) was used as a binder. The physicochemical properties of the prepared microparticles were analyzed and the adsorption characteristics of the adsorbents to Cs ions were evaluated.

2. Experimental methods

2.1 Preparation of clay microspheres

Clay mineral and iron oxide (~ 50 nm) were dispersed into an aqueous solution of 3% poly (vinyl alcohol) (PVA) using a homogenizer. The dispersion was pumped into a spray dryer and the dispersion was sprayed at an inlet temperature of 160. The collected microspheres were observed by optical microscope and scanning electron microscope.

3. Results and Discussion

3.1 Preparation of clay microspheres

When the clay microsphere was prepared by spray drying, illite and montmorillonite formed spherical particles, whereas vermiculite did not. Even though the vermiculite used in the experiment was crushed into small particles, irregular shape of the powder

was not suitable for the preparation of microspheres.

Montmorillonite and illite microspheres were formed by aggregation of clay platelets, and pore spaces between clay particles formed porous microstructures. The size of the particles was 10 to 30 micrometers.

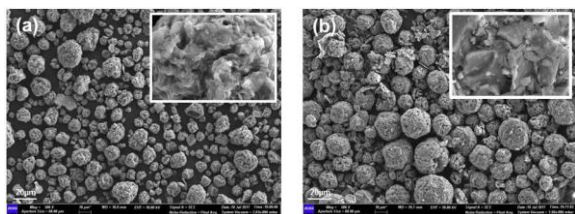


Fig. 1. Scanning electron microscopy images of (a) montmorillonite and (b) illite microspheres.

3.2 Magnetic separation of clay microspheres

In order to recover the adsorbent after adsorption of radionuclides, 50 nm sized iron oxide particles were mixed with clay to prepare microsphere. As shown in the figure, the microparticles dispersed in water can be easily recovered by the external magnet.



Fig. 2. Magnetically collected clay microspheres after adsorption of Cs.

4. Conclusion

Montmorillonite or illite-based clay microsphere for Cs adsorption was prepared by spray drying method. After spray drying, a spherical microspheres with a porous structure were obtained. Clay microspheres containing iron oxide were easily

recovered by external magnetic field after adsorption of ions.

REFERENCES

- [1] M. J. P. Ribeiro, J. A. Labrinch, "Control of geometrical parameters of clay particles processed by spray-drying with relevance in water filtration beds", *Industrial Ceramics*, 18(2), 74-79 (1998).

Development of Milling Machine to Decontamination Radioactive Metal Waste

Dong-Kyun Ko^{*}, Geon-Hwa Lee, Tae-Heon Kim, Sung-Jun Hong, and Eui-Dong Lee

Hana Nuclear Power Engineering, 804, Hanam-daero, Hanam-si, Gyeonggi-do, Republic of Korea

^{*}kodk07@naver.com

1. Introduction

Decommissioning of nuclear power plant has become one of the most hot issues recently in Korea and abroad, so does the importance of technologies/techniques involved in dismantling and decontamination of materials and components of decommissioned nuclear power plant, especially for large component such as steam generator, reactor head, etc.

These large components for decontamination for disposal purposes, unlike decontamination of component in service in an operating nuclear power plant, selecting physical decontamination technique with a relatively high efficiency could be advantageous because decontamination process can be done aggressively without concerns for mechanical integrity of the component. However, in applying physical decontamination technique, many other factors should be considered such as the amount of secondary wastes, proper radiation protection and worker's radiation safety.

In this study, we developed milling decontamination equipment for metal surface decontamination with multiple radiation protection features.

2. Main subject

2.1 Milling

Water compartment channel head of replaced steam generator was chosen for decontamination by milling process. The specimen was taken for decontamination from the channel head which has a large curvature shape and it is quite a big specimen which dimension is 500x250x150 mm.

In consultation with IAEA report on decontamination technologies [1], target level of decontamination process was decided to remove its surface by milling operation sufficiently enough to cover 16 μ m thickness of radioactive crud of steam generator.

Sequence of testing decontamination process as follows; The specimen is secured at the lathe properly. Then, the heights of its fine elements drawn on the surface are measured element by element through the work probe to numerically express the shape of the specimen. The result is loaded to the control unit of the lathe. Now, the measured value is the basis. Enter the thickness you want to milling here to complete the data setup. By making the first measurement with the work probe, this will secure uniformed milling operation on curved or various formed surface of the component, thus providing uniformed decontamination and preventing unwanted excessive production of secondary wastes. After the specimen setting is completed, milling cutter rotates and perform its milling operation along its path in accordance with numerical input data representing the specimen's shape.

Cutting surface is slightly overlapped to eliminate blind spots.

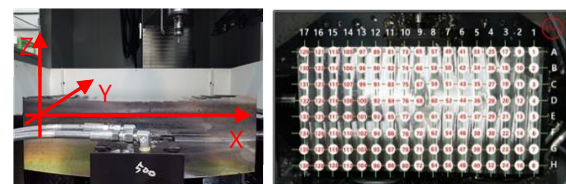


Fig. 1. Measurement [Work probe (left), point (right)].

2.2 Results of Decontamination process

Fig. 2 shows the average of each X-axis and showing the heights difference of each element before and after milling operation. Although there are some deviations due to overlapping operation, it is confirmed that cutting has been made more than the input value over all sections.

According to the above results, it was confirmed that the numerical curvature surface can be decontaminated almost uniformly through the curvature, with a range of the input values from 0.01 to 1 mm. Decontamination target can be reached without much difficulty.

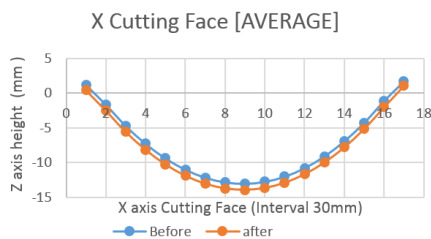


Fig. 2. Before / after X axis milling [Line 1 to 17].

2.3 Radiation protection

Radioactive dusts generated during milling operation cause worker and work space contaminated. Therefore, milling operation was done in a box-shaped closed structure. This container was equipped with ventilation system with HEPA filter on top of it and inside of the container was maintained negative pressure in order to prevent dust dispersal. Control of decontamination process by milling operation was performed outside of the container by remote control and radioactive chips from milling operation were not handled by worker but directly transferred to the waste drums through a conveyor belt.

Since cutting oil commonly used in milling operation generates a large amount of secondary wastes, misty-type oil injection was adopted. This type only needed a small amount of oil injecting to the cutting portion and most of it was vaporized due to heat generated during milling operation and was treated by HEPA filters, thus, significantly reduced the amount of secondary wastes.

In addition, the automatic tool changing system was adopted so that tools were changed automatically during a set of operation without worker's intervention, thereby reducing the risk of workers' exposure and contamination.



Fig. 3. Milling Equipment.

In consideration of radiation safety of the equipment operator, the process was controlled

remotely by an operator at a distance of around 10 m. The whole process was fully monitored through internal CCTV, allowing the operator to easily watch and follow up if necessary without the risk of exposure.

3. Conclusion

We have developed milling equipment for decontaminating metal surface which enables to remove radioactive surface of large component of nuclear power plant such as steam generator, reactor head, etc. uniformly without being affected by curvature of the surface of the component. It can make it possible to reduce excessive amount of secondary wastes due to overcutting the metal surface. This milling machine can be set up in the range of 0.01 ~ 1 mm(1,000 μ m), i.e., it can remove metal surface up to the thickness of 1,000 μ m, far deeper than 16 μ m known as the thickness of radioactive crud of steam generator. So, complete decontamination of metal surface of the component by milling machine we develop is possible.

In addition, by adopting various radiation protection features, minimum radiation exposure to workers can be expected.

In the future, this type of decontamination equipment with high-tech controllers to cope with various sizes and shapes in conjunction with various form of radiation protection features can handle decontamination process more effectively with minimum secondary wastes.

Acknowledgements

This research was supported by the Korea Institute of Energy Technology Evaluation and Planning funded by the Ministry of Trade, Industry and Energy.

REFERENCES

- [1] IAEA, Lawrence, E. Boing, 10. 2006, "Decommissioning of nuclear facilities – Decontamination technologies.
- [2] Final report on Application of Decontamination & Dismantling Technology for Old Steam Generator.

Case Study on Safety Assessment Approach in Yankee and Trojan Decommissioning Plan

Kwangho Jo*, Taesik Yun, Donghee Lee, Younggook Kim, Hyejin Jung, and Jae Yong Oh
Korea Hydro & Nuclear Power Central Research Institute, 70, Yuseong-daero 1312beon-gil,

Yuseong-gu, Daejeon, Republic of Korea

*kwangho.jo@khnp.co.kr

1. Introduction

Decommissioning of nuclear facilities has to be accomplished by assuring the safety of workers, the public and the environment because the decommissioning activities are under high radioactivity and significant potential accident. A number of documents must be prepared to help ensure that the decommissioning process is carried out in a safe and efficient manner. There is, however, no experience of safety assessment applied to decommissioning of commercial facilities in Korea.

The safety assessment of a decommissioning plan is applied to identify potential (radiological and non-radiological) hazards and risks [1]. The YNPS (Yankee nuclear power station) decommissioning plan presents an accident analysis that could occur during the period from approval of the decommissioning plan through completion of the final radiation surveys. The TNP (Trojan Nuclear Plant) decommissioning plan presents the results of evaluation and analyses of postulated decommissioning events and evaluates the potential for adverse effects on public health and safety.

The aim of this paper is to illustrate safety assessment methodology to a nuclear facility and the comparison with its hazards expected to result from the YNPS&TNP decommissioning activities.

2. Decommissioning plan in Korea

The FDP (Final Decommissioning Plan) document is consists of twelve chapters in NSSC Notice (The guideline to write FDP). The safety assessment contents for decommissioning are listed below:

- ✓ Principle & Criteria, Exposure scenario, Dose assessment, Residual radioactivity, Abnormal occurrence, Hazard

3. YNPS decommissioning plan

The decommissioning plan has been prepared in accordance with the requirements of 10 CFR 50.82(b) using guidance.

3.1 Accident analysis

This section of the decommissioning plan presents the accident analysis that assesses impacts of decommissioning on both occupational and public health and safety. The accident analysis considers decommissioning events, fuel storage events, and external events. Analysis of decommissioning events included all phases of decommissioning activities: decontamination, dismantlement, packaging, storage, and radioactive material handling.

3.2 Event identification process

The process includes development of a logic diagram to evaluate all phases of decontamination, dismantlement, and fuel management activities as well as to identify non-radiological events.

- ✓ Events affecting workers, including
 - Radiological and non-radiological events
- ✓ Off-site events affecting public
- ✓ Non-radiological events affecting public
- ✓ Radiological events affecting public
 - Decommissioning activity events
 - Loss of support system events
 - Fire and explosion events
 - External events
 - Spent fuel storage events

3.3 Event affecting workers analysis

Implementation of radiation protection & industrial safety program ensures that potential hazard events are sufficiently minimized.

3.4 Event affecting public analysis

Compliance with requirements (Quality assurance plan, Waste management program) ensures that both the probability of occurrence and the consequences of an off-site event do not significantly affect the public health and safety. The ELSA computer program was used to calculate the Exclusion Area Boundary doses relative to the radioactivity released [2].

4. TNP decommissioning plan

The Event analysis methodology is generally based on the methodology used in NUREC/CR-0130. In TNP decommissioning plan presents the types of activities that could occur during decommissioning and the accident scenarios for these types of activities.

4.1 Potential radiological events

The event analyses with the potential for airborne radioactive releases were classified into four events which are the same events as the radiological events affecting public in YNPS.

4.2 Non-radiological events

In the decommissioning plan, there is no decommissioning events identified that would be initiated from non-radiological sources that could significantly impact public. The hazardous materials and wastes will be controlled by Hazardous Material

Control Program and the Chemical Work Permit Program [3].

5. Conclusion

The safety assessment methodology was reviewed by comparing with YNPS and TNP decommissioning plan. The accident analysis assessed the impact of decommissioning on workers, the public and environment. The evaluation of events that could affect worker's health and safety indicated that implementation of the radiation protection program ensures that these events are sufficiently minimized. Analysis of events that could affect public health and safety indicated that there were no events that could significantly affect public health and safety in both company decommissioning plan. The categorization of a similar chapter is shown in Fig. 1 for the reference.

REFERENCES

- [1] Standard Format and Content for Safety related Decommissioning documents, IAEA Safety Reports Series No. 45.
- [2] Yankee nuclear power station decommissioning plan, Revision 0.0, Yankee Atomic Electric Company.
- [3] Review of PEG's decommissioning plan for the TROJAN nuclear plant, Oregon department of energy.

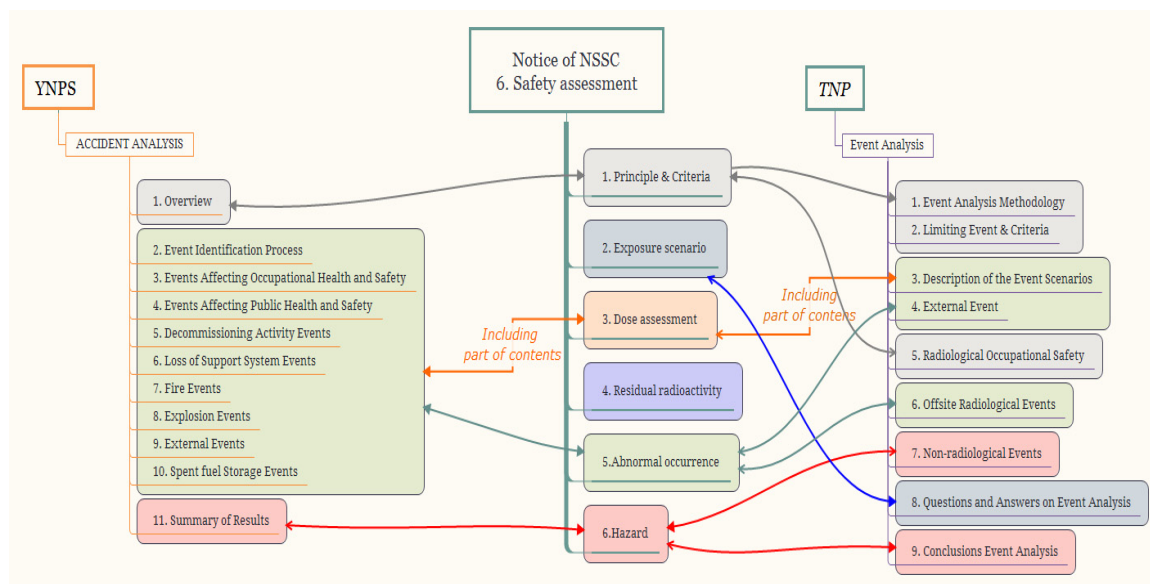


Fig. 1. Similar chapter in YNPS and TNP with NSSC Notice.

An Analysis on Internal Exposure Evaluation Code for Radioactive Aerosols

Hakyun Lee, SunIl Kim, and Jongsoon Song*

Chosun University, 309 Pilmun-daero, Dong-gu, Gwangju, Korea

Leehakyun1@naver.com

1. Introduction

Radioactive aerosol generated in the course of decommissioning of nuclear power plants is the main factor of internal exposure to radiation of the workers. Radioactive aerosol is mainly generated during cutting of concrete, cutting of metal and melting of metal. The level of risk to the decommissioning workers is determined by the nuclide contained in the aerosol. There have been studies on the exposure to radiation caused by radioactive aerosol generated due to critical accidents. This study comparatively analyzes the internal radiation exposure evaluation codes with the size distribution of aerosol as a factor in order to evaluate the risk of radioactive aerosol to the workers during decommissioning of nuclear power plants.

2. Main title

2.1 Aerosol-based internal exposure scenario

Internal exposure means exposure of tissues or organs of a human to the radionuclide infiltrated into human body by way of inhalation, intake or dermal penetration, and adsorbed and disintegrated in various tissues and organs according to physiological flow. When evaluating the internal exposure dose, it is very important to understand the behavior of radionuclide within a body. To evaluate the internal exposure dose, it is required to understand the type of radionuclide element taken, and distribution of the radionuclide elements in a human body according to physical and chemical state.

The amount of aerosol adsorbed in respiratory organs of a worker varies considerably depending on the size of particles. In general, particles of with the diameter of 10um or larger tend to deposit in nasal mucosa. The particles between 2.5 um and 10 um move to the upper tissues of the respiratory organs, affecting the exposure dose and risk depending on type and concentration of radionuclide. Those of 2.5 um or smaller can infiltrate into the blood vessel

via alveoli and lung sac, causing exposure of entire body. Fig. 1 and Table 1 show the deposition rate and deposition fraction by particle sizes suggested by ICRP. ET1 is the front of nasal cavity. ET2 is the rear of nasal cavity, pharynx and larynx. BB is windpipe, main bronchus and bronchial tubes. bb is bronchiole and terminal bronchiole. AI is respiratory bronchiole, alveoli and lung sac.

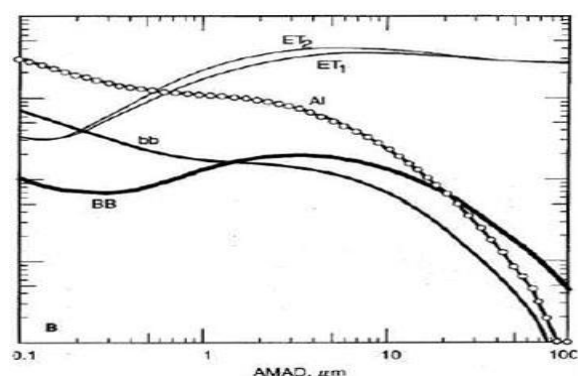


Fig. 1. Particle deposition rate by respiratory area according to AMAD.

Table 1. HRTM deposition fraction (AMAD=5μm)

Area	Deposition fraction (%)
ET ₁	33.85
ET ₂	39.91
BB	1.78
bb	1.10
AI	5.32
Total	81.96

2.2 Internal exposure evaluation code

2.2.1 BiDAS (Bioassay Data Analysis Software).

This code uses ICRP-66 respiratory organ model, ICRP-30 digestive organ model, and ICRP-30/56/67/69/71 biomechanics model. The code for inhalation and ingestion is subdivided into 0.01 um, 0.03 um, 0.1 um, 0.3 um, 1 um, 3 um, 5 um and 10 um, and depending on the type of absorption of chemical (Type F,M,S).

2.2.2 KIDAC (KHNP's Internal Dosimetry Assessment Code).

KIDAC is the code developed by KHNP for evaluation of internal exposure dosage. The code uses ICRP-66 respiratory organ model, ICRP-30 digestive organ model, and ICRP-56/67/69 metabolism model. The code evaluates CED based on the amount of intake and by using the dosage conversion factor. The code can calculate the radionuclide of 1um, 3um, 5um and 10um.

2.2.3 IMBA (Integrated Modules for Bioassay Analysis).

This code uses ICRP-66/30 gastrointestinal tract model, NCRP wound model and ICRP-78 biometrics model. For respiratory model, digestive organ model and metabolism model, the code evaluates the exposure dose with ICRP-68 standard worker as the recommended model. The size of the radionuclide supported for calculation is 5um as recommended by ICRP. The user can change the factors.

2.2.4 MONDAL (Monitoring to Dose Calculation).

The code uses ICRP-66 respiratory organ model, ICRP-30/56/67/69 metabolism model and ICRP-30 digestive organ model. The sizes of the radionuclide supported for calculation are 0.1 um, 0.3 um, 1 um, 3 um, 5 um and 10 um.

2.3 Estimation of size distribution of particles for decommissioning of nuclear power plant

Typical processes conducted during decommissioning of a nuclear power plant are cutting, milling and grinding. In order to estimate the size distribution of aerosol generated during decommissioning work, the change of size of aerosol generated during milling of Fe alloy has been analyzed. The test shows that longer the milling work, the average particle size gets smaller. The particle size varies widely from 0.09 um to 40 um. In general particles of 10 um or larger are filtered by mask.

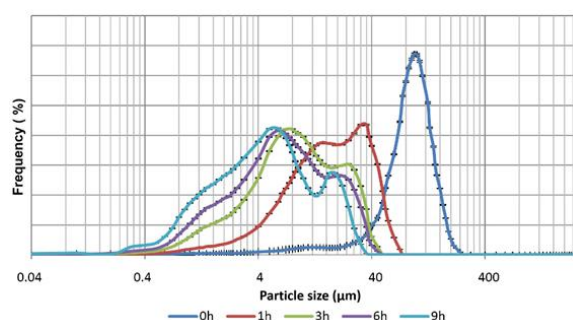


Fig. 2. Size distribution of aerosols generated by Milling.

3. Conclusion

Aerosol generated during decommissioning of a nuclear power plant can be in the form of submicron particle. The submicron aerosol particles deposit in lung sac, causing exposure of the entire body through blood, and hence, more critical than aerosol of larger particle. Therefore, BiDAS is the most appropriate code for evaluation of internal exposure of decommissioning workers as it calculates small aerosol particle of 0.03 um. The code is also proven for its reliability through IMIE. Therefore, it can be used as an important index for evaluation of internal exposure dose of the workers in future decommissioning of nuclear power plants.

REFERENCES

- [1] Werner Hofmann, "Modelling inhaled particle deposition in the human lung", 2011.
- [2] KAERI, "Technology Development for Evaluation of Operational Quantities in Radiation Protection", Final Research Report, 2003.
- [3] Tea-Yong Lee, Jong-Kyung Kim and Jong-Il Lee and Si-Young Chang, "Bioassay Data Analysis Software for Evaluating a Radionuclide Intake and Dose", Journal of the Korean Radioactive Waste Society, 2(2), 113-124 (2004).
- [4] KOARA, "Development of Internal Exposure Dose Assessment Techniques for Domestic Workers Using Radiation and Radioisotope", Final Research Report, 2014.
- [5] Eun-Ju Kim, Hee-Geun Kim, Gak-Hyun Ha and Hyung-Suk Lee, "Comparison of Internal Dose Assessment due to Intake of I-131", Proceeding of the Korean Radioactive Waste Society, 1(2), November (2003).

An Analysis of Characteristics of Radioactive Aerosol by Cutting Methods Available for Decommissioning of Nuclear Power Plants

SunIl Kim, Hakyun Lee, and Jongsoon Song*

Chosun University, 309 Pilmun-Daero, Dong-Gu, Gwangju, Republic of Korea

kingdom17c@naver.com

1. Introduction

Most of the studies on radioactive aerosol have been focused on critical accidents. Radioactive aerosol is also the biggest factor that affects the internal explosion to radiation of the workers during the decommission process. Therefore, it is required to analyze the characteristics of radioactive aerosol generated before starting decommissioning of nuclear power plants. The sources of radioactive aerosol during decommissioning of nuclear power plants include the situations, such as cutting, dissolution, generation of dust and incineration. This study provides scope of cutting for decommissioning of nuclear power plants, cutting time per cutting method, and the size of radioactive aerosol generated.

2. Application of cutting technology for decommissioning of nuclear power plant

The cutting process during decommissioning of nuclear power plant applies to Secondary System tube, Concrete Iron plate, facility, Primary Cooling pipe, Concrete Bioshield, and Concrete wall.

3. Generation of radioactive aerosol

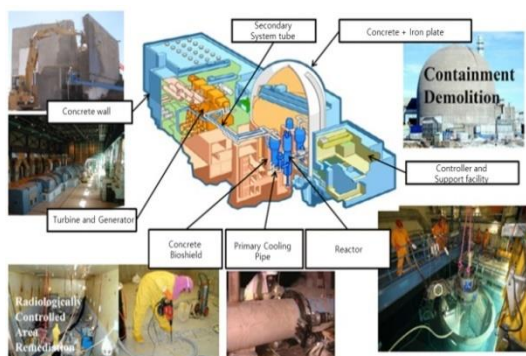


Fig. 1. Cutting range when decommissioning nuclear power plant.

3.1 Critical accident

In accidents of nuclear power plant, aerosol is generated mainly due to nucleation of supersaturated steam. Radioactive steam evaporated from hot core remains spreads from the core remains. The radioactive steam gets supersaturated due to low temperature, and the supersaturated steam molecules agglomerate with each other or with other heterogeneous particles, generating aerosol particles.

3.2 Decommissioning of nuclear power plant

During decommissioning of a power plant, nuclear aerosol is generated due to cutting or melting the pipes and concrete in the primary system.

4. Cutting methods and characteristics

In order to understand the speed of cutting process and the size of aerosol by cutting methods, we have analyzed the BR-3 pipe cutting case in Belgium. In analysis of cutting speed by thickness, Plasma cutting and Flame cutting are found to be superior to other technologies. For the influence of pipe thickness on the aerosol particle, it was found that Flame cutting and Arc saw method generate relatively large radioactive aerosol particles. Most of the radioactive aerosol particles of 5~10 μm are filtered by nose hair or mouth. Those of 1~5 μm are adsorbed in bronchial tubes and lungs. However, the radioactive aerosol particles of 0.1~1 μm move to and adsorbed in alveoli. The alveolus is the organ where oxygen in blood is exchanged. Radioactive aerosol particles adsorbed in alveoli causes exposure of the whole body to radiation.

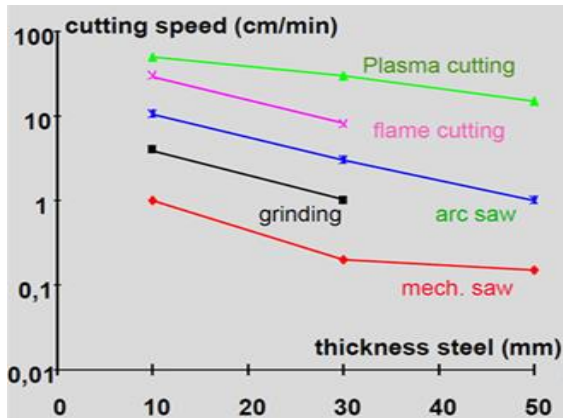


Fig. 2. Cutting speed by cutting method.

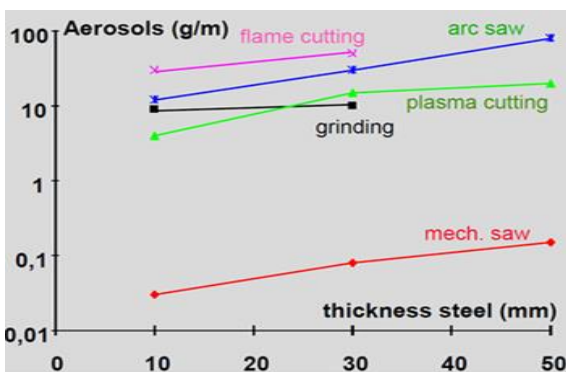


Fig. 3. Size of aerosol particles by cutting method.

REFERENCES

- [1] S.G Park, H.R Kim, D.G Lee, U.S Jung and K.J Jung "A State of the Art on The Dismantling Techniques for the KRR-1 & 2 Decommissioning", KAERI/AR(2001).

5. Conclusion

During decommissioning of nuclear power plants, Plasma cutting and Flame cutting are most appropriate as they require short process time and generate relatively large radioactive aerosol particles. The best way of the cutting process for decommissioning of nuclear power plants is to calculate the volume of radiation to be exposed by the workers, and to estimate the optimum work time. In preliminary estimation of the volume of radiation exposed, it is difficult to measure in some pipes and facilities due to complex structure of nuclear power plants. It is possible to establish the radiation exposure scenario through the computer codes based on the experiment result, and to utilize the data for future decommissioning of nuclear power plants.

Preliminary Considerations for Preparing a Domestic Decommissioning Radiological Environmental Impact Assessment

Sang-Ho Lee and Chang-Lak Kim*

KEPCO International Nuclear Graduate School, 658-91Heamaji-ro, Seosaneng-myeon, Ulju-gun,
Ulsan, 689-882, Korea
jjj1sh2@naver.com

1. Introduction

As Kori Unit1 is permanently shut down in June, 2017, permanent suspension and decommissioning of domestic old aging nuclear power plants are anticipated. Domestic nuclear power plants should conduct a radiation Environmental Impact Assessment (EIA) based on the construction, operation and accident of the power plant as required by the Nuclear Safety Act. But there is no detailed regulation on decommissioning radiation EIA in Korea.

In this paper, it reviewed the issues that should be considered in advance for the preparation of the decommissioning radiation EIA.

2. Methodology

2.1 Review of whether Final Decommissioning Plan includes Radiological Environmental Impact Assessment or Not

Major nuclear power nations such as the United States, Canada, the United Kingdom, and France are operating in an integrated form of EIA and radiation EIA. In other words, radiation EIA is included in the EIA. In Korea, there is no detailed regulation on the evaluation of decommissioning radiation environment impact. So, it reviewed the domestic nuclear regulation whether there is a need to separately prepare the NPP decommissioning radiation EIA in Korea.

There are two options to consider when making a nuclear decommissioning radiation environment impact assessment report as below:

- ① Include radioactive environmental impact assessment in the decommissioning plan
- ② To prepare separate disassembly radiation environmental impact assessment report

2.2 Domestic Atomic Energy Law

Table 1. Decommissioning Environmental Impact Assessment related Atomic Energy Law

Nuclear Safety Act
Article 103 (Feedback from residents). ① Provide a draft of the radiation environmental impact assessment report or make public hearings. ② Provide a draft of the decommissioning plan or make public hearings.
Regulations on technical standards such as reactor facilities
Article 85-16 (Decommissioning Environmental Impact Assessment). ① The operator shall evaluate the expected radiation environmental impact assessment due to the decommissioning of the reactor facility. ② The operator shall establish and implement an environmental monitoring plan before and during decommissioning of the reactor facilities.

Based on the above laws and regulations, it is a decision to be included in the decommissioning plan rather than requiring the creation of a radiological EIA at the time of decommissioning. This is not a concept to prepare separate radiological EIAs.

2.3 Environmental Impact Assessment Items Considering Decommissioning

There are three types of EIA (environmental monitoring before decommissioning, environmental monitoring during decommissioning, impact on nearby residents) in decommissioning plan. The environmental monitoring before decommissioning is performed in the same as the environmental monitoring during operation. Survey plans, analysis items, detection target values, data processing and evaluation of environmental monitoring during decommissioning is performed in the same as the environmental monitoring before decommissioning. However, the decommissioning radiation environment impact assessment report should additionally describe the issues caused by decommissioning. Table 2 shows the EIA items due to decommissioning activities.

Table 2. Environmental Impact from Decommissioning Activities

1. Remove Fuel	• Transfer fuel to spent fuel pool • Drain primary system • Process liquid
2. Organization Changes	• Reduce staff • Employ contractor or additional staff • Adjust site training
3. Stabilization	• Drain and flush system • Isolate systems, structures and components that are no longer required • Rewiring of site to eliminate unneeded electrical circuits
4. Post-Shutdown Surveys	• Baseline surveys for the decontamination work • Continual surveys
5. Create Nuclear Island	• Install electrical power to SFP • Move old or install new security-related power
6. Chemical Decontamination of Primary Loop	• Cutting, chemicals in, chemicals out, cleanup
7. Large Components Removal	• Remove reactor vessel and internals intact or cut up • Steam generator and outer large components removed intact or cut up
8. Decontamination and Dismantlement Phases of DECON	• Chemical decontamination (surface/specific components) • Decontamination pipes in walls • High-pressure water sprays • Remove contaminated soil • Preventive and corrective maintenance on SSCs
9. System Dismantlement	• Cut out radioactive piping • Remove large and small tanks or other radioactive components from the facility
10. Structure Dismantlement	• Rubblization • Remove structures that were necessary for plant operation
11. LLW Packaging and Storage	• LLW Packaging and Storage
12. Transportations	• Large components • SF transportation from pool • LLW/VLLW transportation to disposal facility
13. License Termination Activities	• Complete final radiation survey • Partial site release
14. Newly Installed Decommissioning Waste Treatment	• Melting related accidents (fire, human injury)
15. Impact from Nuclear Power Plant Operation	• Evaluation of exposure dose (residents, animals, plants)

3. Conclusion

This paper discusses considerations for creating a decommissioning radiation environment impact assessment. As a result, the final decommissioning plan included the evaluation of the decommissioning radiation environment impact, and it was judged that it was not necessary to prepare it separately. In addition, the decommissioning radiation environmental impact assessment is the same as the operating radiation environmental impact assessment, but the additional items arising from the decommissioning should be described.

4. Acknowledgement

This research 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. 1305009).

REFERENCES

- [1] Nuclear Safety and Security Commission Notice, "Regulations on the Preparation of Nuclear Decommissioning Plans", July, 2015.
- [2] NUREG-0586, "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities", November, 2002.

Preliminary Study on Environment Mapping for Nuclear Reactor Dismantling

Sungmoon Joo*, Jonghwan Lee, Ikjune Kim, Dongjun Hyun, Shinyoung Kang,

Kwanseong Jeong, and Byungseon Choi

Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon, Korea

*smjoo@kaeri.re.kr

1. Introduction

Prediction of time and cost for dismantling of nuclear reactor facilities is critical in the preparation phase of decommissioning. Existing methodologies using prior data are useful for evaluating initial estimates, but these estimates need to be updated using as-built data.

Simulation-based evaluation for the dismantling scenario of the reactor vessel at Kori Unit 1 has been proposed [1]. The simulation-based evaluation potentially yields more realistic estimates, provided realistic as-built data is used. One example is reactor vessel upper internal (RVI) structure cutting (Fig. 1); usually design data (e.g., CAD model) is used in the simulation, however, to make the segmentation and cutting plan more realistic, as-built data is necessary.

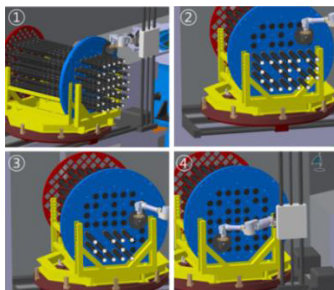


Fig. 1. Cutting Simulation of RVI [1].

This study introduces an algorithm for building as-built environment model using point clouds, and preliminary result is presented.

2. Background Information

This section provides background information for 3D modeling and mapping using point clouds.

2.1 Point Cloud

Point cloud is a set of data points in some coordinate system. In a 3D coordinate system, these points are

usually defined by X,Y, and Z coordinates, and often are intended to represent the external surface of an object. Point clouds may be created by 3D sensors such as 3D laser scanners.

2.2 Registration

Often the point clouds are acquired from different view-points. The (point set) registration finds the relative pose (position and orientation) between views in a global coordinate frame, such that the overlapping areas between the point clouds match as well as possible. The overall objective registration is to align individual point clouds and fuse them to a single point cloud.

2.3 Mapping and Localization

Registration algorithms are used in applications such as 3D mapping and localization. The reconstructed map is represented by a set of point clouds which are aligned by means of registration. If the poses where two point clouds were taken from, the result of registration yields a merged point cloud referenced to a global coordinate frame. However, when the poses are unknown, the registration involves the process of finding out the poses, so called localization. The localization requires a loop closure, the problem of recalling revisited (or seen again) scene.

3. Real-time Appearance-Based Mapping

Real-time Appearance-Based Mapping (RTAB-Map) was developed to provide an appearance-based localization and mapping solution independent of time and size, to achieve online loop closure detection for long-term operation in large environment [2]. The idea resides in only using a limited number of locations for loop closure detection so that real-time constraints can be satisfied, while still gain access to locations of the entire map whenever necessary.

The main feature of RTAB-Map is its memory management which enables real-time mapping; memory management, which has been used in robot localization to increase recognition performance in dynamical environment or to limit memory used, is used for online localization and mapping, where new locations are dynamically added over time.

Fig. 2 shows the main concept of RTAB-MAP and its memory management model [2]. The perception module acquires an image and sends it to sensory memory (SM). SM evaluates the image signature to reduce data dimensionality and to extract useful features for loop closure detection. Then SM creates a new location with image signature and sends it to short-term memory (STM). STM updates recently created locations through a process referred to as weight update.

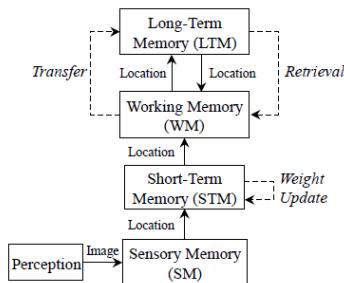


Fig. 2. RTAB-Map Memory Management Model.

Algorithm 1 illustrates the overall loop closure detection process [2].

4. Implementation & Test Results

RTAB-Map algorithm is available as an open source for test and development. Fig. 3 is a map of author's desk area represented by a colored point cloud. Implementation and tests were performed on a PC with Ubuntu 14.04 using Kinect V1 sensor.



Fig. 3. Mapping of Desk Area using RTAB-Map.

Table 1. Algorithm 1

Algorithm 1 RTAB-Map	
1: $time \leftarrow \text{TIMENOW}()$	$\triangleright \text{TIMENOW}()$ returns current time
2: $I_t \leftarrow$ acquired image	
3: $L_t \leftarrow \text{LOCATIONCREATION}(I_t)$	
4: if z_t (of L_t) is a bad signature (using T_{bad}) then	
5: Delete L_t	
6: else	
7: Insert L_t into STM, adding a neighbor link with L_{t-1}	
8: Weight Update of L_t in STM (using $T_{\text{similarity}}$)	
9: if STM's size reached its limit (T_{STM}) then	
10: Move oldest location of STM to WM	
11: end if	
12: $p(S_t L^t) \leftarrow$ Bayesian Filter Update in WM with L_t	
13: Loop Closure Hypothesis Selection ($S_t = i$)	
14: if $S_t = i$ is accepted (using T_{loop}) then	
15: Add loop closure link between L_t and L_i	
16: end if	
17: Join $trash$'s thread	$\triangleright$ Thread started in TRANSFER()
18: RETRIEVAL(L_i)	$\triangleright \text{LTM} \rightarrow \text{WM}$
19: $pTime \leftarrow \text{TIMENOW}() - time$	$\triangleright$ Processing time
20: if $pTime > T_{\text{time}}$ then	
21: TRANSFER()	$\triangleright \text{WM} \rightarrow \text{LTM}$
22: end if	
23: end if	

5. Conclusion

This study introduces RTAP-Map algorithm for nuclear reactor facility environment mapping to build as-built models, and a preliminary modeling result is presented. The method can be applied to build as-built models of reactor facilities for dismantling, and the as-built data can contribute to more realistic simulation-based evaluation.

Acknowledgements

This work was supported by the nuclear research and development program through the national research foundation of Korea funded by the Ministry of Science and ICT.

REFERENCES

- [1] Kim, I., Choi, Hyun, D., Moon, J., Lee, Jeong, K., Kang, S., "Integration of the Unit Modules of the Nuclear Facility Dismantling Simulation System", Proc. of the KRS 2017 Spring Conference, pp. 297-298, May 24-26, 2017, Busan.
- [2] M. Labbe, F. Michaud, "Appearance-Based Loop Closure Detection for Online Large-Scale and Long-Term Operation", IEEE Transactions on Robotics, Vol. 29, No. 3, pp. 734-745, 2013.

Design Characteristics for Ease of Dismantling of Research Reactor

Kook-Nam Park^{1*}, Seung-Kook Park¹, Yun-Dong Choi¹, Young-San Choi¹, and Do-Hyung Kim²

¹ Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yeseong-gu, Daejeon, Korea

² ORiON E&C, #305, 37, Seongsui-ro 22-gil, Seongdong-gu, Seoul, 04798, Korea

*knpark@kaeri.re.kr

1. Introduction

In order to obtain permission for the construction and operation of nuclear reactors and related facilities, an initial decommissioning plan must be submitted pursuant to Article 10 and 11 of the Nuclear Safety Act. Article 5, Clause 5 of the Nuclear Safety Commission Notification Regarding the Preparation of Nuclear Power Plants Decommissioning Plans, etc., describes the characteristics of the design considering the ease of dismantling and the measures for dismantling during construction or operation [1]. This paper describes the initial decommissioning plan for the design characteristics and measures for ease of dismantling of Hanaro and related facilities that have been operating normally since 1995.

2. Design characteristics

Hanaro was designed to be easy to dismantle and minimize the amount of radioactive waste generated. It minimizes the dose exposure for workers and has design characteristics that can provide accessibility and sufficient working space for dismantling [2].

2.1 Design characteristics considering ease of dismantling in accordance with ALARA principle

The research reactor was designed and constructed by applying the ALARA design concept to facilitate dismantling work during dismantling and to minimize the dose exposure to dismantling workers. To this end, during normal operation, the process systems are designed to keep the reactor safe and to prevent unnecessary radioactive leakage and to minimize the dose exposure.

The process system simplifies testing and maintenance by minimally activating devices. It was designed to be easy to access and reliable when needed.

2.2 Design characteristics applied to minimize the possibility of leakage of radioactive materials in structures, systems and parts

HANARO was designed to minimize leakage of radioactive materials from structures, systems, and parts through the concept of negative pressure at inside the buildings and multi-barrier walls. Major equipment such as piping, and the heat exchanger during operation minimized the possibility of leakage of radioactive materials through seismic analysis, vibration evaluation, fabrication test, commissioning test, and in-service inspection. The chimney, which is a reactor structure, mixes the cooling water passing through the flow tube and the water tank rising along the bypass channel to the cooling water outlet. In addition, a hot water layer, shown in Figure 1, is placed in the upper part of the pool to suppress the radioactive materials in the cooling system.

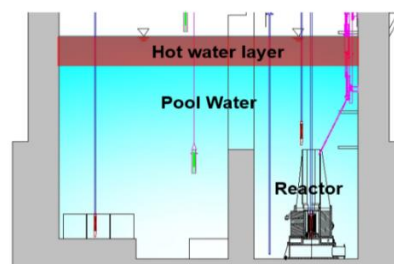


Fig. 1. Hot water layer in the upper part of the pool.

2.3 Design characteristics applied to minimize radioactive contamination in the planned and uncontrolled state and to minimize the generation of radioactive waste during operation

As an engineering safety facility, the reactor containment system is designed to minimize the leakage of radioactive material from nuclear reactor buildings through penetration isolation, building airtightness and negative pressure maintenance. The reactor building is a radiation controlled area, which adopts the concept of semi-containment building to the nuclear reactor building structure to prevent diffusion and leakage of radioactive material, and keeps the inside of the building at negative pressure. The design criteria for environmental impacts during normal operation of Hanaro are in accordance with the provisions of the Nuclear Safety Commission

Regulation No.17, “Regulations on technical criteria for reactor facilities, etc.” [3].

2.4 Design characteristics that can demonstrate optimization of facility layout for accessibility and demolition of workers

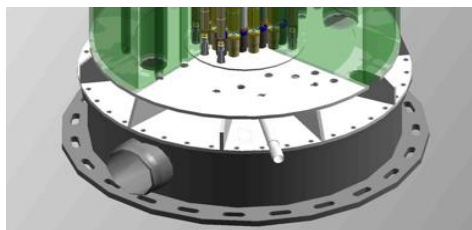


Fig. 2. Reactor structure with bolt holes.

A ceiling crane with a capacity of 30 tons is installed in the reactor hall and can be used for decommissioning activity. The reactor structure, which is the main equipment of the research reactor, is composed of the inflow cavity, the lower grid plate, the flow tube, the reflector tank, and the chimney along the circulation path of the cooling water, and is locked in the reactor pool. The reactor structure is assembled with bolts, as shown in Figure 2, so that each part of the reactor can be separated and disassembled if easily them cutting.

3. Action Plan

3.1 Measures to minimize leakage, radioactive contamination and radioactive waste generation during operation

To minimize the leakage of radioactive materials during operation and the occurrence of radioactive contamination and radioactive waste through it, Hanaro was designed to prevent leaks and take measures through immediate leakage detection in case of radioactive material leakage.

The leak detector for detecting leakage is as follows. Where there is a potential for leaks in the primary cooling system, heat exchangers and pumps are commonly used in nuclear reactors. Even if there is leakage through the gasket, there is a drain groove connected to the liquid waste treatment system at the bottom of the equipment room under the heat exchanger so that the leaked cooling water collects in the drainage groove. The leak detector installed in the home detects a leak and an alarm is sounded in the control room OWS (Operator Workstation). If the alarm sounds, the operator immediately stops the reactor operation [4].

3.2 Management plans and methods for key records of design, construction and operation that can affect dismantling

Major records on the design, construction, and operation that may affect the dismantling of Hanaro are registered with the Advanced Nuclear Safety Information Management System (ANSIM) at KAERI. It is organized by the task. In document management, the design documents, design drawings, procedures, and manuals of HANARO and utilization facilities, as well as operation experience reports are registered.

4. Conclusion

Hanaro is a research reactor that was designed and constructed over a period of 10 years from 1985, and has been operated to date. In this research reactor, design characteristics consisting of four design characteristics are considered, from design to consideration of ease of dismantling in accordance with the ALARA principle. First, measures were taken to minimize leakage, radioactive contamination and radioactive waste generation during operation. Second, management plans and methods for key records on the design, construction and operation were built into the system. The ease of dismantling is based on the safety analysis report.

REFERENCES

- [1] U.S. Nuclear Regulatory Commission, Regulatory Guide 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning”, June 2008.
- [2] Korea Atomic Energy Research Institute, “Safety Analysis Report for HANARO” KAERI/TR-3898/2016, Rev. 6, Nov. 2016.
- [3] Nuclear Safety Commission, Regulations on the Preparation of Nuclear Facilities Decommissioning Plan, etc. Notice 2015-8.
- [4] Seung-Kook Park, Kook-Nam Park, etc., “Analysis of Design Characteristics for Ease of Decommissioning of HANARO and Related Facilities”, Technical Report, KAERI/TR-6677/2016, pp 23-36.

Study on the Precipitation Behavior of Metal Ions in the Chemical Decontamination Waste by Using Carbonates

Jin-Hee Kim*, Hyun-Kyu Lee, June-Hyun Kim, Won-Zin Oh, and Sang-June Choi

Kyungpook National University, Dae-Hak ro 80, Dae-Gu, Korea

*Kimjinhee33@knu.ac.kr

1. Introduction

The accumulation of radioactive materials in the corrosion oxide film on the surface of the NPP system causes worker's exposure to radioactivity during maintenance or decommission of NPP, and decontamination is done through chemical methods or physical methods to reduce worker's exposure. Especially, in the case of chemical decontamination, chemicals are used to dissolve the corrosion oxide film, from which various metal ions are dissolved. The HP/CORD is widely used for the decontamination of NPP system and it consists of oxidation by HMnO_4 , reduction by oxalic acid and decomposition by $\text{UV}/\text{H}_2\text{O}_2$.

Kim et al. [1] simulated the corrosion oxide film in NPP system, and through comparison with other literature [2,3] confirmed that the simulation was similar to actual corrosion oxide film of NPP.

In this study, we investigated the precipitation behavior of metal ions that can be retained after decontamination of the corrosion oxide film and the metal precipitate was formed through pH control. The remaining metal ions were selected as Fe, Cr and Mn by reference to study on simulation of the corrosion oxide film. Na_2CO_3 was used for pH control, and the precipitation behavior of each metal ion was evaluated depending on pH.

2. Experiments

In this study, 1 mM Fe^{2+} , Fe^{3+} , Cr^{3+} , Mn^{2+} were selected as metal ion targets for the precipitation, and for this, $\text{Fe}(\text{NH}_4)_2(\text{SO}_4)_2$, FeCl_3 , CrCl_3 and MnCl_2 were used, respectively. Na_2CO_3 was injected into the initial solution to set pH to 4, 5, 6, 7, 9, 11 and the precipitation behavior was evaluated by

measuring the metal ions concentration using ICP-OES, after filtering.

3. Results

3.1 Precipitation of Fe^{2+} and Fe^{3+}

The Precipitation behavior of Fe^{2+} and Fe^{3+} depending on pH condition is shown in Fig. 1. The result shows that Fe^{2+} was completely removed in pH 4 to pH 5 range and Fe^{3+} in pH 5 to pH 6 range.

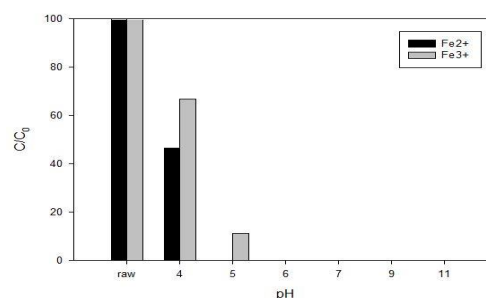


Fig. 1. Precipitation of Fe^{2+} and Fe^{3+} depending on pH.

3.2 Precipitation of Cr^{3+}

The Precipitation behavior of Cr^{3+} depending on pH condition is shown in Fig. 2. The result shows that most of Cr^{3+} was removed in pH 6 region.

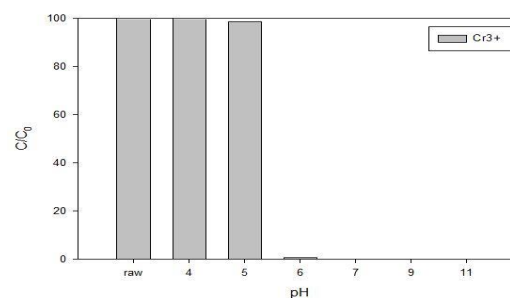


Fig. 2. Precipitation of Cr^{3+} depending on pH.

3.3 Precipitation of Mn^{2+}

The Precipitation behavior of Mn^{2+} depending on pH condition is shown in Fig. 3. The result shows that Mn^{2+} was completely removed in pH 9 to pH11 range.

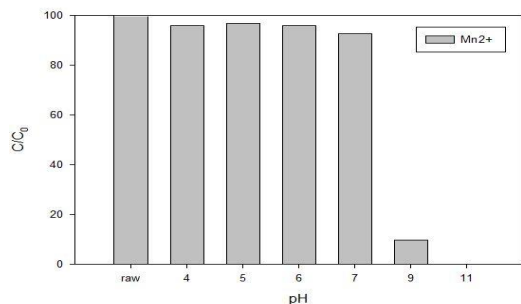


Fig. 3. Precipitation of Mn^{2+} depending on pH.

4. Conclusion

The following conclusion was obtained about the precipitation behavior of metal ions from the NPP chemical decontamination in the experimental ranges: The pH control by carbonates is the key factor to precipitate the metal ions in the form of metal carbonates or hydroxide and the increase of pH up to 11 is necessary to precipitate Mn^{2+} ions.

5. Acknowledgement

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. 20141510300310).

REFERENCES

- [1] Dong-Yeon, Kim et al., “Simulation of the corrosion oxide film in PWR system for the development of the decontamination technology”, Korean Radioactive Waste Society ATUMN, 13(2), 377-378 (2015).

- [2] J.Robertson, “The Mechanism of high temperature aqueous corrosion stainless steels”, Corrosion Science, 443-465 (1991).
- [3] J.H.Carter, “Corrosion and passivation on PWR primary materials to reduce corrosion product activity, Nucl. Energy, 45-52 (1985).

Computational Model of Copper Catalyzed Metal Oxide Dissolution in Reducing Acidic Solution

Seon-Byeong Kim*, Hui-Jun Won, Sang-Yoon Park, Mansoo Choi, Wangkyu Choi, and Bum-Kyoung Seo
Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon, Korea

*sbkim@kaeri.re.kr

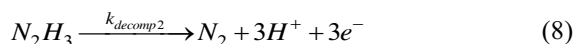
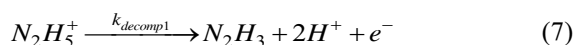
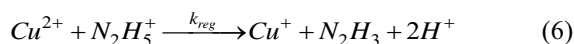
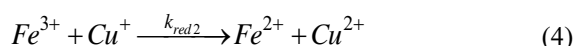
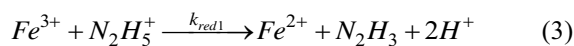
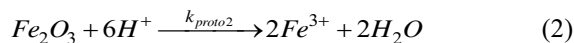
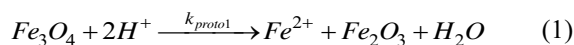
1. Introduction

Redox reaction is the main function of the metal component dissolution from the oxides. Insoluble Fe(III) existing in magnetite, for an example, can be reduced in the form of Fe(II) ion by reduction of electron and thus soluble into the solution. On the other way, insoluble Cr(III) could be released by the oxidation from chromite as Cr(VI) which is the most soluble chromium compound in water.

There have been tons of studies to examine the dissolution of metal oxides in terms of dissolution kinetics, type of reactants, geometry, etc [1-3]. However, most of previous studies is the observation of macroscopic dissolution characteristics and might not provide the atomic scale characteristics of dissolution reactions. In this study, computational models of dissolution mechanisms for metal oxides, specifically for magnetite and nickel-ferrite in the present study. The relevant reaction pathways are postulated with the reaction rate constants obtained by batch experiments and provided with the plausible pretexts. The approximated dissolution model and its computation are expected to alternate the time-consuming experimental dissolution tests and sophisticated interpretation of dissolution mechanism with limited experimental data. The dissolving solution we used in the present study is composed of hydrazine as a strong reducing agent and sulphuric acid as a hydrogen donator. This solution is prepared in lieu of oxalic acid or EDTA based organic acid solution to prevent the further treatment problems by the use of those chemical agents. In addition, the dissolving ability of the solution is catalyzed by the addition of copper ions.

2. Dissolution mechanisms of metal oxides

Metal oxides undergo both acidic and reductive dissolutions in N_2H_4/H^+ solution. Ample hydrogen ion is offered in acidic mood initially and may be additionally provided during the dissolution by oxidation of $N_2H_5^+$. Also copper ions lead more favorable reduction process by reducing ferric ion to ferrous ion. The lower activation energy of the $N_2H_4/H^+/Cu^{2+}$ system could dissolve metal oxides faster than systems with no copper ions. The overall mechanism is described as follows.



In deterministic model, interpreting the microscopic behavior between solutes and reactants relying on the surface geometry is delegated by macroscopic ODE reaction rate equations derived by continuous concentration changes. The law of mass action as an empirical rule of thumb was used to determine the equations. Using MATLAB non-stiff ODE solver, all reaction rate equations were solved simultaneously and shown in Fig. 1.

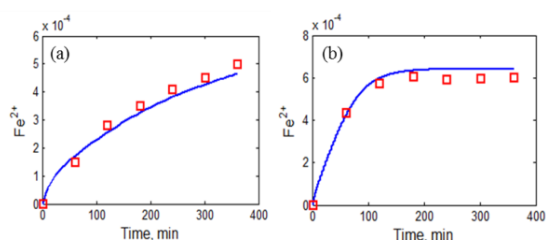


Fig. 1. Profiles of chemical reagents in Fe_3O_4 dissolution w/ N_2H_4 and H_2SO_4 solution for (a) w/o Cu^{2+} and (b) w/ Cu^{2+} .

3. Effects of Concentrations of Hydrazine and Copper Ions

In acidic solution, the dominant existence of hydrazine is hydrazinium ion, N_2H_5^+ , after the protonation in ample hydrogen condition. The oxidation of hydrazine to N_2H_4^+ ion and further to nitrogen in the solution where a reduction half-cell reaction presents is described in equations (7) and (8). For the dissolution of magnetite, a favorable reduction process takes place by reducing ferric ion to ferrous ion. However, the paths of hydrazine oxidation could differ when the solution is added by cupric ions. This was depicted in both Figs. 2 and 3.

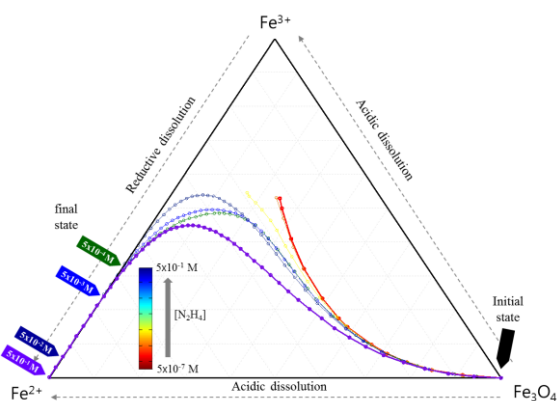


Fig. 2. Composition triangle of Fe_3O_4 , Fe^{3+} , Fe^{2+} after 6 h of the magnetite dissolution model by the increase of N_2H_4 , but $[\text{Cu}^{2+}]$ is 50 mM for everywhere.

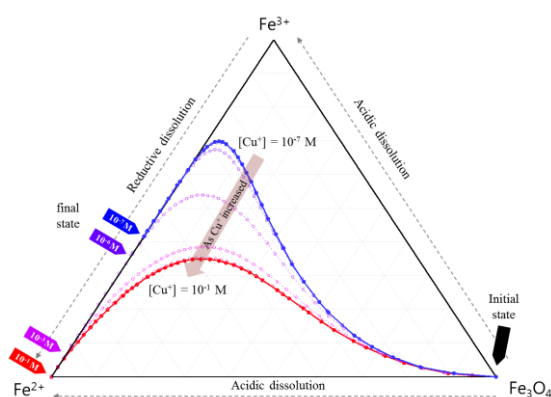


Fig. 3. Composition triangle of Fe_3O_4 , Fe^{3+} , Fe^{2+} after 6 h of the magnetite dissolution model by the increase of Cu^+ , but $[\text{N}_2\text{H}_4]$ is 0.05M for everywhere.

4. Conclusion

From this study, it was suggested that hydrazine as a reducing agent facilitated the dissolution of magnetite to a certain degree. The addition of cupric ion in hydrazine-based acidic solution was suggested to make the dissolution more effective, and accordingly gave rise to significantly enhanced dissolution rate. This study also shows that a well-established computational model can be a useful tool to cope with the detailed estimation of performance for the decontamination of large facility.

REFERENCES

- [1] R. M. Cornell and R. Giovanoli, "Acid dissolution of hematites of different morphologies", *Clays and Clay Minerals*, 28, 223-232 (1992).
- [2] Brown, E. B., Dollimore, D., and Galwey, A. K., "Reactions in the solid state", *Comprehensive chemical kinetics*, 141-156 (1980).
- [3] Sidhu, P. S., Gilkes, R. J., Cornell, R. M. Posner, A. M., and Quirk, J. P., "Dissolution of iron oxides and oxyhydroxides in hydrochloric and perchloric acids", *Clays and Clay Minerals*, 29, 269-276 (1981).

Development of Waste Characterization System for the Life-Cycle of Decommissioning Nuclear Facilities

Inhye Hahm*, Daeseo Koo, Kook-Nam Park, Seung-Kook Park, Sang Bum Hong, and Bum-Kyoung Seo
Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Republic of Korea

*ihhahm@kaeri.re.kr

1. Introduction

Since a large quantity of radioactive and nonradioactive waste is generated during the decommissioning of nuclear facilities, waste management is one of the crucial parts of effective planning and cost estimation. Therefore, in countries with decommissioning experience, decision making methodology and related technologies have been developed. However, existing systems, which are usually designed from the costing point of view, have limitations in terms of waste management.

To overcome these shortcomings, in the present study, waste management system including efficient waste tracking module from point of waste generation to final disposal were designed.

2. Technical status

The existing waste management function implement just simple activities like transferring waste drums to disposal from its generation or cost calculation at final disposal stage. On the other hand, decommissioning wastes typically handles various processes such as segmentation, repackaging, and decontamination for volume and radioactivity reduction. Also the physical, chemical and radiological characteristics change continuously. Therefore, to optimize waste management, it is essential to develop a technology to trace waste history at final stage. Below is an introduction to the software that implements some of the related functionality.

2.1 eOMEGA

The eOMEGA code is a decommissioning costing software with fully implemented International Structure for Decommissioning Costing (ISDC) structure, methodology and ISDC costing approach [1].

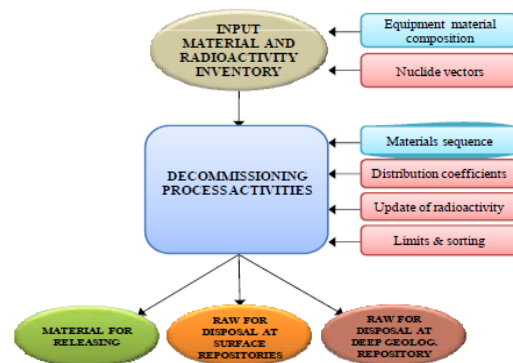


Fig. 1. Principle scheme of waste management in the eOMEGA code.

Because waste management costs have a large impact on cost estimates, eOMEGA has system for simulating material and radioactive flow where physical and nuclide resolved radiological charactersitics of each material item are linked together [2].

2.2 DEXUS

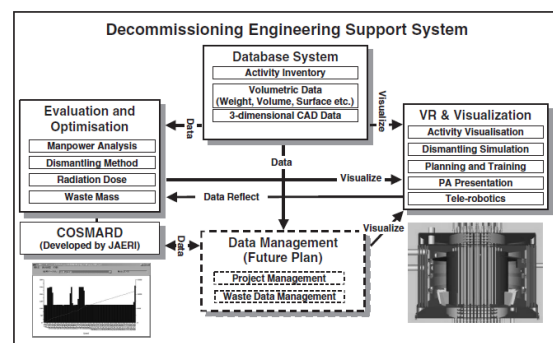


Fig. 2. Conceptual Design of DEXUS [3].

DEXUS, an engineering support system for decommissioning, has been developed to make an efficient decommissioning plan by adopting new information technologies such as 3-dimensional computer aided design systems and virtual reality systems. Although DEXUS only provides functions such as calculation of the exposure dose at present, it is planning to build effective waste management system based on these 3D functions.

3. System design requirements

Waste characterization system is designed to improve decommissioning safety and reduce waste. Waste is managed based on the waste packages as a basic unit in the system. Waste packages and waste containers are managed separately. Waste containers provide general information for waste management such as the facility, date of occurrence, type of waste and serial number.

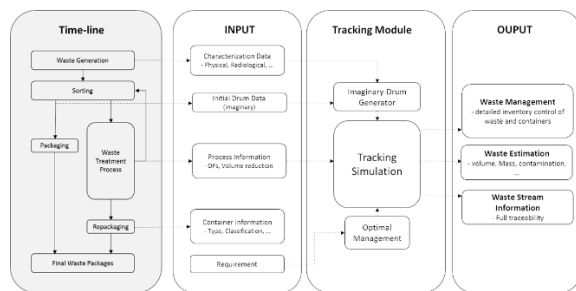


Fig. 3. Schematic of waste management module.

Information of decommissioning waste packages changes continuously during the process such as segmentation, repackaging, and decontamination. Therefore a key part of the system design is to track the history of physical, chemical and radiological information and ultimately to provide the information required for the disposal. In addition, the tracking process does not need to continuously input/output all the information related to the packages, but only inputs/outputs changed data.

When the system user outputs information related waste generation, processing and disposal, system provides all the integrated packages information and tracks waste stream in the container.

4. Conclusions

The waste characterization system for the life-cycle of decommissioning nuclear facilities is designed for

the integrated management and evaluation of the radiological characteristics of the various types of waste generated during decommissioning process.

REFERENCES

- [1] IAEA, "Innovative and Adaptive Technologies in Decommissioning of Nuclear Facilities, IAEA-TECDOC-1602 (2008).
- [2] Matej Zachar, et al., "Development of Methodology for Cost-Effectiveness Analysis of Decommissioning Waste Management Scenarios, International Conference on the Safety of Radioactive Waste Management (2016).
- [3] Yukihiro Iguchi, et al., "Development of Decommissioning Engineering Support System (DEXUS of the Fugen Nuclear Power Station), Journal of Nuclear Science and Technology. (2012).

Correlations Between Safety Assessment and WDFs in NPP Decommissioning

Jae Yong Oh^{*}, Kwangho Jo, Ji-hwan Yu, and Younggook Kim

Decommissioning Technology Team, CRI, KHNP, 70, Yuseong-daero 1312beon-gil, Yuseong-gu, Daejeon, Korea

^{*}jaeyongoh@khnp.co.kr

1. Introduction

Decommissioning is the last phase of the nuclear power plant (NPP) life cycle followed by the site release. OECD/NEA defines the decommissioning as all of the administrative and technical actions associated with early planning for cessation of operations through termination of all licenses and release of the site from nuclear regulatory control [1]. Korea Hydro and Nuclear Power (KHNP) made a decision for permanent shutdown of Kori-1 on June 18th, 2015, which, finally, has been shut down since June 19th, 2017.

To date, Korea has no decommissioning experiences on a commercial nuclear reactor but on a research reactor. Accordingly, KHNP has begun to develop necessary decommissioning technologies which helps reduce uncertainties and avoid risky events with regard to hazards and safety. Hence, this paper mainly deals with correlations between safety assessment and Work Difficult Factors (WDFs) in decommissioning. In addition, we try to suggest systematic application of those correlations for the effective decommissioning of Kori-1 NPP.

2. Methodologies & Results

Safety assessment is closely related to work difficulties of decommissioning workers, which has an immediate impact on decommissioning costs. Due to the correlation between safety assessment and work difficulties, as a matter of fact, OMEGA, one of the decommissioning cost estimation programs, has the function to consider occupational dose in calculating decommissioning costs. It has become essential that safety assessment is a key process to estimating decommissioning costs.

2.1 Safety Assessment for Decommissioning

Safety assessment considers consequences and frequencies of the failure and events based on hazard analysis. Due to the characteristics such as activation and contamination of structure, system and components (SSCs) in decommissioned NPP, both industrial (non-radiological) and radiological hazards should be taken into account.

2.1.1 Industrial Hazards. Non-radiological and industrial hazards are obviously identified taken place or not. Thus, they are analyzed into the complex results by consequence (severity) exemplified in Table 1 and frequency (probability, likelihood) shown in Table 2. Fig. 1 illustrates the complex results of industrial hazards as a matrix of consequence and frequency [2]. Main industrial hazards are represented as fire, explosion, flooding, toxic, electrical, physical, human errors, etc.

Table 1. Consequence of industrial hazards

Level	Consequence (Severity)
1	No loss (No effect)
2	Lost day work (1 day ~ 1 week)
3	Injury (1 week ~ 3 months)
4	Disabilities (3 months ~ 1 year)
5	Fatalities (More than 1 year, death)

Table 2. Frequency of industrial hazards

Level	Frequency (Probability/Likelihood)
1	Rare (Less than 10 %)
2	Unlikely (10 % ~ 25 %)
3	Possible (25 % ~ 50 %)
4	Likely (50 % ~ 75 %)
5	Almost certain (More than 75 %)

Frequency Level	Almost Certain	Very Low	Medium	High	Very High	Very High
	Likely	Very Low	Medium	High	High	Very High
	Possible	Very Low	Low	Medium	High	High
	Unlikely	Very Low	Low	Low	Medium	Medium
	Rare	Very Low	Very Low	Very Low	Very Low	Very Low
		No loss	Lost day work	Injury	Disabilities	Fatalities
		Consequence Level				

Fig. 1. Risk matrix of industrial hazards.

2.1.2 Radiological Hazards. Since decommissioning workers often conduct in radiological environments, they could be frequently exposed to the environment of radiological hazards [3].

Radiological hazards also can be classified by their consequences. Table 3 shows the categorization of radiological hazards by consequence and frequency.

Table 3. Categorization of radiological hazards

Frequency \ Consequence	Consequence		
	Low	Moderate	High
$< 10^{-6}/\text{yr}$	Class 4	Class 4	Class 3
$10^{-6} \sim 10^{-4}/\text{yr}$	Class 4	Class 3	Class 2
$10^{-4} \sim 10^{-2}/\text{yr}$	Class 3	Class 2	Class 1
$10^{-2} \sim 10^{-1}/\text{yr}$	Class 3	Class 1	Class 1

2.2 Work Difficulty Factors

WDFs are weighted values of durations and costs for decommissioning activities. Table 4 demonstrates the representative WDFs such as accessibility, ALARA (radiation condition), respiratory protection measures, personal protection clothing etc. [4]. WDFs could be additionally defined if another factor is needed to more accurately consider the work environment.

Table 4. Composition of Work Difficulty Factors

WDF	Description
Accessibility	Structural complexity in workspace
ALARA	Radiation condition and training
Respiratory	Respirator protection equipment
Prot. Cloth	Wearing radiation protective clothes

2.3 Application of Safety Assessment to WDFs

Results of safety assessments are able to a part of WDFs according to their characteristics of the

corresponding hazards. For instance, physical hazard could change the accessibility factor; exposure dose rate alter the ALARA factor, the measures of radiation protection turn into respiratory protection and personal protection clothing factors. Especially, radiological hazard is dependent on anticipated occupational dose. In some cases, the total WDF value would be maximized to 356% as demonstrated in Table 5 [3].

Table 5. Total WDF values by dose rate phases

Dose rate phase	Total WDF values			
	Case 1	Case 2	Case 3	Case 4
Phase 1	1	1.32	1	1
Phase 2	1.2	1.8	1.55	1.21
Phase 3	1.4	1.92	1.67	2.2
Phase 4	1.6	2.28	2.02	2.57
Phase 5	2.4	2.47	2.66	3.56

3. Conclusions

Safety assessment is one of the most important processes for the safety of on-site occupational workers and nearby residents. Therefore, WDFs reflecting the results of safety assessment will highly influence decommissioning activity costs. However, accurate safety assessment prevent more drastic increase of costs occurred by severe accidents. From this study, we find that there are deep correlations between safety assessment and decommissioning activities, which eventually leads to rational, economical and safe decommissioning.

REFERENCES

- [1] Nuclear Energy Agency, Improving Nuclear Regulation, NEA No. 6275, OECD (2009).
- [2] K.-S. Jeong; B.-S. Choi; J.-K. Moon; D.-J. Hyun; J.-H. Lee; G.-H. Kim; H.-S. Hwang; S.-Y. Jeong and J.-J. Lee, Risk reduction approach to decommissioning hazards of nuclear facilities, *Annals of Nuclear Energy*, 63, 382-386 (2014).
- [3] IAEA, Safety Assessment for Decommissioning, Safety Reports Series No. 77, IAEA (2013).
- [4] Thomas S. L., etc., Guidelines for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates, AIF/NESP-036, Volume 1 (1986).

Dismantlement of Surveillance Capsule for Hanul Unit 3

Kisoo Heo^{*}, Bounghok Yoo, Seungjae Baik, Gilsoo Kim, and Youngjun Kim

Korea Atomic Energy Research Institute, 989-111, Daedeok-daero, Yuseong-gu, Daejeon, Korea

^{*}heokisoo@kaeri.re.kr

1. Introduction

Reactor pressure vessels(RPVs) in nuclear power plants are irradiated by neutrons during reactor operation, which have been hardened and embrittled due to changing the mechanical properties. To evaluate safety and life-time of RPV, surveillance test must be performed with the same specimens as RPV material compositions. At the beginning of reactor operation, a surveillance capsule is installed by RPV, then it is withdrawn at the end of operating cycle [1]

In this paper, dismantling procedure of surveillance capsule in hotcell is introduced and developed for effective working.

2. Dismantlement and Result

Unlike previous manual withdrawal, we developed the automatic specimen withdrawal machine for efficient working process and undamaged specimen. In order to prevent specimen loss, dismantlement of surveillance capsule is performed by 2 members.

2.1 Capsule Preparation

Surveillance capsule is carried into M2 hotcell through the pool in IMEF(Irradiated Material Examination Facility) in KAERI(Korea Atomic Energy Research Institute). Fig. 1 shows Surveillance capsule in M2 hotcell.



Fig. 1. Surveillance capsule.

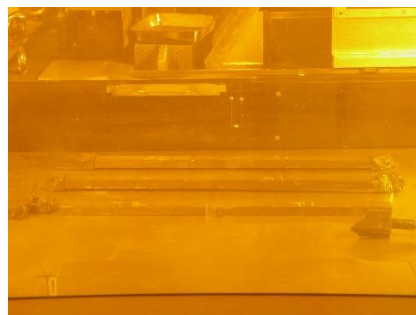


Fig. 2. Marked capsule.



Fig. 3. Wedge plug cutting.

2.2 Marking

Surveillance capsule consist of total 6 sections. In order to specimen traceability, Each section is marked identification number. Fig. 2 shows marked capsule.

2.3 Removal of Wedge Plug

In order to specimens withdrawal, Wedge plugs are removed by universal cutting machine in M2 hotcell. Fig. 3 shows cutting of Wedge plug by universal cutting machine.

2.4 Specimens Withdrawal

The specimens are withdrawn by automatic specimen withdrawal machine. This machine is

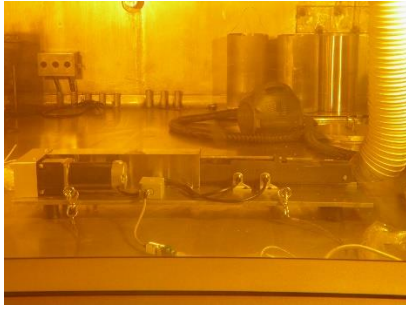


Fig. 4. Automatic specimen withdrawal machine.



Fig. 5. Dismantled capsule and specimens.

operated by DC motor and finish working per 1 section in 30 seconds. Fig. 4 shows installed automatic specimen withdrawal machine in hotcell.

2.5 Result

As a result, we obtained undamaged surveillance test specimen and performed surveillance test successfully. Fig. 5 shows dismantled capsule and surveillance test specimens.

3. Conclusion

Surveillance test is very important to estimate lifetime of RPV in PWR. Neutron bombardment changes mechanical properties of reactor structural components including vessel. Critical point for LBB(Leakage Before Break) must be decided and discussed with various reactor operating conditions. In this paper, dismantlement of surveillance capsule in hotcell is introduced and shown by machines with modifications which were developed to do hotcell work effectively.

Surveillance tests have been performed in every year and done by IMEF in KAERI.

REFERENCES

- [1] B. C. Kim, et al, "FINAL REPORT FOR THE 4TH SURVEILLANCE TEST OF THE REACTOR PRESSURE VESSEL MATERIAL(CAPSULE Y0 OF ULCHIN NUCLEAR POWER PLANT UNIT 2" (2003).

Consideration of Treatment and Disposal for Decommissioning Waste

Younggook Kim*, Taesik Yun, Hyejin Jung, Kwangho Jo, and Jae Yong Oh

Decommissioning & Spent Fuel Technology Center, KHNP CRI, 70, Yuseong-daero 1312beon-gil, Yuseong-gu, Daejeon, Republic of Korea

*kim.090909@khnp.co.kr

1. Introduction

On June 19, 2017, Korea's first commercial nuclear power plant, Kori Unit1, was permanently shut-down for decommissioning. After about 5 years of transition period, the spent fuel is defueled to the off-site and then actual dismantling could begin.

The Kori Unit1 decommissioning waste that is expected to be generated in the dismantling of one nuclear power plant is approximately 80,000 drums. However, we are aimed at reducing to 14,500 drums by applying a volume reduction technique (1drum = 200ℓ) [1].

In this study, we are going to define the sources of decommissioning waste and analyze the proper processing and disposal plan in Korea.

2. Decommissioning Waste

2.1 Decommissioning Waste Generation

Decommissioning wastes generated during the dismantling process are characterized by the generation of a large amount of waste in short period and various source terms.

Some are produced primarily through decontamination and dismantling process of contaminated SSCs, and the other are during the decontamination process whose objective is to reduce the radioactivity level to a goal for recycling or clearance waste. A portion of wastes could be generated in the form of mixed waste with non-radioactive or radioactive and hazardous materials mixed. The waste disposal cost of dismantled waste accounts for approximately 40% of the total decommissioning cost [2].

2.2 Solid Waste

Solid wastes are classified as activated metal waste (RVI, RPV), radioactive concrete waste (Bioshield

concrete), and soil waste (Excavated soil exceeding DCGL). In addition, wet wastes (spent resin, filter, sludge) generated from the liquid radioactive waste management system or facility. And also dry waste (air cleanup filter, dry active waste) from HVAC and decontamination process. The others are classified as mixed wastes in which radioactive and non-radioactive materials with harmful material are mixed.

2.3 Liquid Waste

Liquid radioactive wastes are decontamination liquid wastes generated by the decontamination SSCs, system water from fuel pool cooling and filtering system and laundry waste by workers' clothing washing and personal decontamination.

2.4 Gaseous Waste

Gaseous wastes would be produced during decontamination and cutting operations. Types of them consist of radioactive particulates suspended in the air of buildings and exhaust gases generated during the melting process. In order to manage radioactive dusts that could be generated during dismantling work, separate air purification equipment such as temporary pollution control equipment and dust collector is needed.

2.5 Comparison of Generated Decommissioning Waste

Table 1. Comparison generated waste by Classification

Waste Classification	Generated Decommissioning Waste (m ³)			
	EU	Maine Yankee	Rancho Seco	KORI
Class A	2,911	90,650	17,244	5,759
Class B/C	2,459	570	93	5,990
GTCC	109	-	11	21
Sum	5,479	106,610	17,348	11,770

3. Consideration for Decommissioning waste processing and disposal

Dismantled wastes should be considered for the disposal according to the material and form, apart from the level of radioactive contamination.

The purpose of this section is to describe the requirements for the radioactive waste generated from dismantlement by the situation in Korea. The following are important considerations in the disposal of decommissioning waste.

3.1 ILW, LLW Radioactive waste Acceptance criteria

The total radioactivity at the KORAD disposal facility is $1.08\text{E}+16\text{Bq}$. But the radioactivity calculated through the source term evaluation of Kori Unit1 reactor vessel internal is $7.65\text{E}+16\text{Bq}$ which, therefore, cannot be disposed at KORAD disposal facility.

If intermediate wastes generated from Kori Unit1 are not met the acceptance criterion of KORAD, additional waste storage on site is required. This problem causes an increase in decommissioning cost and worker's exposure.

3.2 Development for waste disposal container

Up to date, the container for disposal of decommissioning waste is limited to 200ℓ drums in Korea. When decommissioning, wastes of various shapes and sizes are generated. In order to pack into the 200ℓ drum containers, more cutting process should be needed and resulted in a large amount of secondary wastes and increased worker exposure.

Developing disposal waste containers is needed to minimize worker's radiation exposure and to optimize decommissioning costs.

3.3 GTCC disposal methodology

GTCC, such as reactor vessel internal, is currently not disposable in KORAD. In case of Connecticut Yankee Nuclear power station, RVI was disposed by using ISFSI (Independent Spent Fuel Storage Installation) disposal container.

If RVI is able to be disposed of in container that used for Spent Fuel dry storage, it could be disposed of without any additional safety or sensitivity assessment.

3.4 Design of Decommissioning Waste Processing Facility

The wastes generated during the dismantling process should be treated by using the existing waste management system.

If there is a large amount of wastes exceeding the capacity of the existing facilities or having unacceptable characteristics, a comprehensive disposal facility for disposal waste should be designed.

Therefore, the Comprehensive Disposal Facility would ensure processing performance and capacity based on the expected type, characteristics, amount generated, treatment level, technology level and regulatory condition which are on the basis of evaluation of the radiological characteristics and developing the technology of D&D. Furthermore the Comprehensive Disposal Facility should be designed considering these results.

4. Conclusion

Decommissioning strategy of KHNP is to immediately dismantle Kori Unit1. In order for the decommissioning schedule to proceed in an efficient manner, the hurdles of disposal waste treatment must be solved first. Otherwise, the decommissioning schedule would be delayed which results in increasing the decommissioning cost and the radiation exposure of the workers.

Before the actual dismantling work begins, to solve the considerations mentioned above, it will be necessary to concentrate on R&D and standardization criteria for disposal waste during the transition period.

REFERENCES

- [1] Jong Soon Song, Young-Guk Kim, Sang-Heon Lee, "A Pre-Study on the Estimation of NPP Decommissioning Radioactive Waste and Disposal costs for Applying New Classification Criteria", Journal of Nuclear Fuel Cycle and Waste Technology, Vol. 13 No.1 (2015).
- [2] Notice of MOTIE (Ministry of Trade, Industry and Energy) 2015-132 "Notice of cost estimation standard for radioactive waste management and spent fuel management (2015).

6분과

방사선환경 및 안전 (Oral)



Aerial Radiation Measurements as an Effective Means of Collecting Emergency Information

Byoung-Jik Kim

Korea Institute of Nuclear Safety, 62, Gwahak-ro, Yuseong-gu, Daejeon, Republic of Korea

bjkim@kins.re.kr

1. Emergency information and ARM

1.1 Protective measure in a radiological emergency

In situations such as nuclear accidents where radioactive releases occur, the necessary measures are to reduce the risk of radiation by protecting people who may be exposed to radioactivity or to move them from the area of higher level of radiation to of lower. For the purpose, we can take urgent protective actions and other response actions such as evacuation, restrictions on the food chain and on water supply, prevention of inadvertent ingestion, restrictions on the consumption of food, milk and drinking water and on the use of commodities, decontamination of evacuees, control of access and traffic restrictions in a nuclear or radiological emergency [1]. These actions shall be carried out according to very careful decisions based on reliable information which is referred to as emergency information here.

1.2 Emergency information

Emergency information is the location of people, the use of protective drugs, and the distribution of radioactivity. Based on this information, it is a protective action to keep people away from radiation or weaken radiation risks.

Assuming a more realistic emergency situation, it is not easy to identify and control the position of a person. However, if you accurately grasp the radioactivity distribution and publish it quickly, you can lead people away from radiation.

In the nuclear accident, the surroundings (even over 100 km distance from the accident) will be in extreme confusion. If information such as the distribution of radioactive contamination is not adequately provided, people who do not need to evacuate will try to move from a safe place to another, which will make the situation worse. Perhaps the non-radiological effect will cause more damage.

1.3 Aerial radiation measurements (ARM)

Precise measurements (for example, sample & analysis) are not suitable for conducting radioactive contamination surveys over a large area in a short time (within a few hours) when radioactive material releases such as the accident of Fukushima Dai-ichi Nuclear Power Plant (FDNPP). Moving measurements are inevitable even if somewhat inaccurate, and aerial measurements using aircraft are the most effective and efficient means thus far.

The FDNPP accident and the response underlined the importance of ARM technology, which enabled emergency decision makers to be aware of the state of radioactive contamination [2, 3]. This technique, which has been widely used in relation to the FDNPP accident, mainly focuses on calibration to convert counts in the air to the ground radiation dose or radioactivity concentration and mapping to project the dose or the concentration back to the ground.

2. ARM technology overview

2.1 Definition

Radiation measurement over the ground while flying aircrafts with radiation instruments

2.2 Purpose

To identify the radioactive deposition on the ground surface

2.3 Normal procedure

- i. Measurement by system (radiation coefficient, position, altitude)
- ii. Correction methods such as excluding radiation not coming from the surface
- iii. Calibration with radiation attenuation according to the altitude

- iv. Corresponding measurement results after calibration to a radiation value according to the ground position

2.4 Elements of the technology

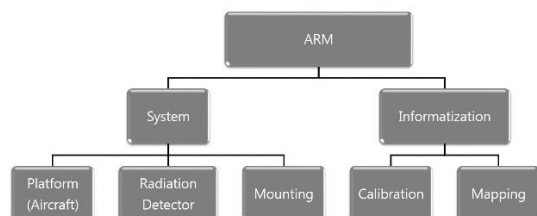


Fig. 1. The classification chart of the technical elements.

ARM technology can be divided into system technology and information technology. System technology can be broken down into platform technology such as selecting or remodeling the aircraft, radiation detector technology and mounting technology. Information technology is classified into calibration technology and mapping technology.

3. Current status

On November 3 and 4, 2015, measurements were carried out in the contaminated area around the FDNPP. Measurements were carried out in an area within 10 km from FDNPP, where an accident occurred due to the March 2011 earthquake in eastern Japan, and where the contamination remains and the access is controlled (Fig. 2(a)). Calibrations to convert the coefficients from the detector in air to the radiation dose or radioactivity concentration on the surface were carried out in areas previously surveyed as flat and evenly contaminated (Fig. 2(b)). Explorations for contamination mapping were conducted in river basins, with contaminants appearing in different water and soil contours, which were expected to make intuitive comparisons easier for the mappings (Fig. 2(c)).

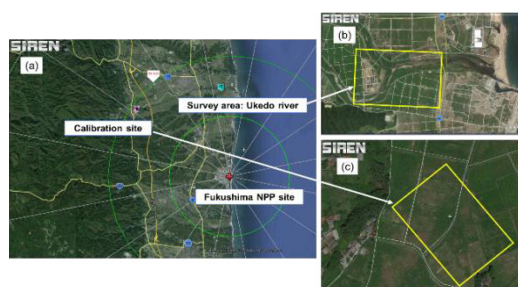


Fig. 2. (a) Measurements were performed in an area within 10 km from FDNPP. (b) Survey flights were conducted in river basins. (c) The calibration site was located in a flat area.

The aircraft flew through the designed route under the control of a ground station. Specifically, KINS instrumentation was controlled via a wireless internet connection and was monitored at the headquarters of KINS, located in Daejeon, Korea (Fig. 3).

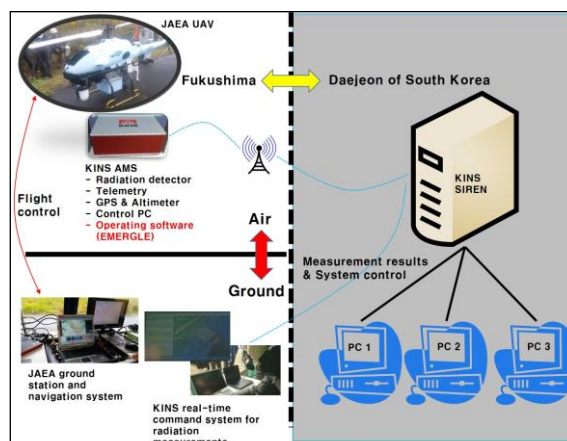


Fig. 3. KINS instrumentation was controlled and monitored via a wireless internet connection.

4. Future challenges

One of the most important emergency information, the radiation distribution map, is the result of ARM. Efforts are being made to improve the quality of this result. Especially, improvement of spatial resolution of the map is expected to advance the response against a radiological emergency.

In addition, technical standards for applications should be established. Because the performance varies according to various combinations of technical elements, it seems necessary to set technical requirements that meet the application's purpose.

REFERENCES

- [1] IAEA, Safety Standards, "Preparedness and Response for a Nuclear or Radiological Emergency", General Safety Requirements, No. GSR part 7 (2015).
- [2] Y. Sanada and T. Torii, "Aerial radiation monitoring around the Fukushima Dai-ichi nuclear power plant using an unmanned helicopter", J. Env. Rad. 139, 294-299 (2015).
- [3] Y. Sanada, T. Orita, T. Torii, "Temporal variation of dose rate distribution around the Fukushima Daiichi nuclear power station using unmanned helicopter", Appl. Rad. and Isot. 118, 308-316 (2016).

Beta-Ray and Gamma-Ray Detection Technique for the Soil Surface Using Plastic Scintillators

Jun Woo Bae*, Uk Jae Lee, and Hee Reyoung Kim

Department of Nuclear Engineering, Ulsan National Institute of Science and Technology,
50, UNIST-gil, Eonyang-eup, Ulju-gun, Ulsan Metropolitan City, 44919, Republic of Korea

*skypia12@unist.ac.kr

1. Introduction

Monitoring of residual radioactivity in the decommissioning site is one of the important technology included in the remediation step of the site. The conventional gamma-ray detector is hard to detect beta-ray because of protecting material to protect scintillator from moisture and oxidation. Because Geiger-Müller counter has no energy resolution, it is impossible to discriminate the type of particle. To overcome these weaknesses of conventional detector, a phoswich detector which used combination of scintillator and pulse shape analysis (PSA) was developed but it had problem of overlap between beta and gamma-ray [1].

In this study, beta-ray and gamma-ray discrimination technique using plastic scintillators with different thickness. The previous study was carried out using combination of two scintillators, plastic scintillator and cadmium tungstate [2]. The study showed the ability of discrimination but the energy calibration between two scintillators was very hard because light outputs of the scintillator are different. Thus, the same scintillators, plastic scintillator, were used in the study except for the thickness.

2. Materials and Methods

2.1 Concept of sensitivity ratio

For a gamma-ray, linear attenuation coefficient of a medium is highly related to the gamma-ray detection efficiency. By the way, detection efficiency of beta-ray is not significantly depending on the medium because the range of beta-ray is short enough where the range of 2 MeV beta-ray was less than 10 mm in the plastic medium. If the thicknesses of scintillators are different, the detection efficiency of gamma-ray would be significantly different while the sensitivity of beta-ray would not.

2.2 Theory and calculation

One scintillator with thickness of 10 mm (Thick) and the other one with thickness of 1 mm (thin) were used in this study. The detection efficiency of each scintillator was experimentally characterized. The sensitivity ratios for the gamma-ray (R_{gamma}) and beta-ray (R_{beta}) were defined as following equations.

$$R_{\text{gamma}} = \frac{\varepsilon_{\text{Thick, gamma}}}{\varepsilon_{\text{thin, gamma}}} \quad (1)$$

$$R_{\text{beta}} = \frac{\varepsilon_{\text{Thick, beta}}}{\varepsilon_{\text{thin, beta}}} \quad (2)$$

where $\varepsilon_{a,b}$ was detection efficiency for the type of radiation (b) with different thickness of scintillator (a). The counting rate ratio between two scintillators was utilized to determine the contribution of the beta-ray in the measured counting rate.

$$R_{\text{meas}} = \frac{C_{\text{Thick}}}{C_{\text{thin}}} \quad (3)$$

where C_{Thick} and C_{thin} were counting rates measured by thick and thin scintillators, respectively. The portion of beta-ray contributed counting rate (C_{cal}) could be determined by equation (4).

$$C_{\text{cal}} = C_{\text{meas}} \cdot X_{\text{meas}} = C_{\text{meas}} \cdot \frac{R_{\text{gamma}} - R_{\text{meas}}}{R_{\text{gamma}} - R_{\text{beta}}} \quad (4)$$

where C_{meas} was counting rate of the sample measured by thick scintillator, X_{meas} was portion of beta-ray contribution which had value between 0 to 1. Related uncertainties were carefully calculated according to the principle of error propagation for the factors causing errors R_{gamma} , R_{meas} , R_{beta} , C_{meas} and background counting rate [3].

2.3 Experimental setup

5 source sets were used to verify the theory. The sources were combination of disc sources and they were described in Table 1. Single ^{60}Co and ^{90}Sr sources were used to characterize the detection efficiency. Combination sources were used to verify the theory to discriminate the beta-ray and gamma-ray contributed counting rate.

Table 1. Information of source sets

Source ID	Composition
Co	Single ^{60}Co (27,209 Bq)
Sr	Single ^{90}Sr (3,456 Bq)
SrCo	One ^{90}Sr and one ^{60}Co .
SrCoCo	One ^{90}Sr (3,456 Bq) and two ^{60}Co (27,209, 17,746 Bq).
SrSrCo	Two ^{90}Sr (3,456, 3,274 Bq) and one ^{60}Co (27,209 Bq)

The prototype detector of was described at Fig. 1. Each plastic scintillator was optically coupled with photomultiplier tube (PMT) and correlated electronic signal processing devices. The soil surface would be scanned by each detector. In this experiment, the disc source sets were departed from the detector by 10 cm and measured by 3 times.

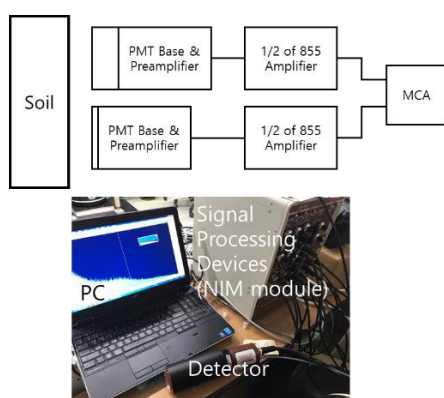


Fig. 1. Scheme of detector composition and its picture.

3. Results and Discussion

The background counting rates for thick and thin scintillator based detector were 803 ± 12 and 398 ± 8 cpm, respectively. Table 2 shows the result of measured counting rates and determined beta-ray

contribution. The uncertainty shown in the Table 2 includes uncertainties caused in measurement. The relative uncertainty was increased by 13 times in average during extracting of beta-ray contributed counting rate.

Table 2. Abstract of measurement results

Scintillator type	Source ID	Net Average (cpm)	X_{thick}	Extracted Beta-ray (cpm)
Thick	Co	$3,399 \pm 48$	0	0
	Sr	$6,068 \pm 26$	1	$6,068 \pm 228$
	SrCo	$9,557 \pm 14$	0.632	$6,040 \pm 250$
	SrCoCo	$12,366 \pm 22$	0.468	$6,021 \pm 270$
	SrSrCo	$16,284 \pm 32$	0.77	$12,000 \pm 471$

4. Conclusion

Method of selective beta ray detection using the sensitivity difference between thick and thin plastic scintillation was suggested. The ability of discrimination between beta-ray and gamma-ray was tested and successfully proven. Subsequent application to low-radioactive sources and additional evaluation of energy-dependent gamma-ray sensitivity should be performed.

REFERENCES

- [1] S. Yamamoto and J. Hatazawa, "Development of an alpha/beta/gamma detector for radiation monitoring" Review of Scientific Instruments, 82, 113503 (2011).
- [2] J. W. Bae, H. R. Kim, "Selective Beta and Gamma-ray Discrimination by CdWO_4 and PlasticScintillator" Proc. of the KARP 2017 Spring Conference, 258-259, April 13, 2017, Gunsan.
- [3] G. F. Knoll, Radiation detection and measurement, 4th ed., Hoboken, N.J.: Wiley, (2010).

The Real-Time Monitoring of Dose Rate and Radioactivity Using an Environmental Radiation Monitor Based on the NaI(Tl) Detector

Young-Yong Ji^{1*}, Chang-Jong Kim¹, Kyo-Sun Lim¹, Hyon-Sock Chang², Taehyung Lim², and Wanno Lee¹

¹ Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Republic of Korea

² SI Detection Co. Ltd., 21, Yuseong-daero 1628beon-gil, Yuseong-gu, Daejeon, Republic of Korea

*yyji@kaeri.re.kr

1. Introduction

In general, the high pressure ion chamber (HPIC) and GM (Geiger Muller) counter are widely used as a radiation detector for monitoring the ambient dose rate in the environment as well as the nuclear facility. These detectors have a strength about the reliable and stable measurement of the ambient dose rate. However, the HPIC and GM counter can only give the result of the measured ambient dose rate without the information of the artificial and natural radiation. To overcome this drawback of the HPIC and GM counter, the spectrometric determination of the ambient dose rate is on an increasing trend using a NaI(Tl) scintillation detector. The dose conversion algorithm by the detector used should be prepared in advance to convert the measured energy spectrum for counts into the ambient dose rate.

In this study, an environmental radiation monitor (ERM) based on a 3"x3" NaI(Tl) detector (EFRD3300, SI Detection Co. Ltd., Korea) was applied to perform real-time monitoring of the dose rate and radioactivity for detected gamma nuclides around an ERM by using the dose rate spectroscopy [1-3]. First, the dose rate and radioactivity for detected natural radionuclides using an ERM based on a 3"x3" NaI(Tl) detector were experimentally verified by comparing with them of in situ gamma-ray spectrometry results obtained by a portable HPGe detector and the analysis of samples taken around an ERM. Finally, an ERM with dose rate spectroscopy was used for the real-time monitoring of two variables around the ERM by reporting them every 15 minutes.

2. Method and Results

2.1 Dose rate spectroscopy

The dose rate spectroscopy is first introduced to calculate the individual dose rate for detected gamma nuclides from the environment as well as radioactive materials. This method was expended to infer the radioactivity from the calculated individual dose rate for detected gamma nuclides by using the dose-to-curie (DTC) conversion method. To use the dose rate spectroscopy, the measured energy spectrum for count rate from an ERM was converted into one for dose rate by multiplying the count rate and dose conversion factor, such as a $G(E)$ in the unit of nGy/h/cps, as shown in Fig. 1.

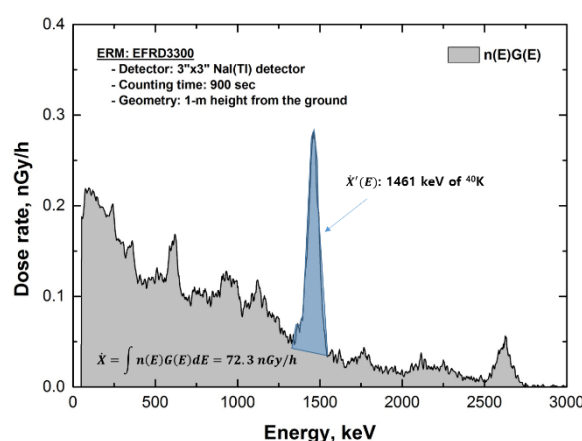


Fig. 1. The energy spectrum for dose rate measured from an EFRD3300.

2.2 Mathematical expression

In the case of a nuclide with a single photon energy, its individual dose rate can be shown in Eq. (1). If a nuclide has several gamma-rays and the contribution ratio of one gamma-ray to the individual dose rate is

calculated, its individual dose rate can be calculated as Eq. (2). Finally, the radioactivity in the ground can be estimated by dividing the calculated individual dose rate into the dose rate per unit curie, d_i , at 1 m above the ground due to the homogeneous distribution of natural radionuclides, as shown in Eq. (3).

$$\dot{X}(E) = \int_{E_n}^{E_m} \frac{n'(E)G(E)W^{-1}(E)}{DP(E)} dE \quad (1)$$

$$\dot{X}_i = \frac{\dot{X}(E)}{CR_i(E)} \quad (2)$$

$$A_i = \frac{\dot{X}_i}{d_i} \quad (3)$$

where, E_n and E_m are the lower and upper bounds of a peak, $X(E)$ is the dose rate induced from a gamma-ray with an energy E , $DP(E)$ is the peak-to-total ratio in the energy spectrum for the dose rate, $W(E)$ is the angular correction factor for the G-factor, X_i is the individual dose rate of nuclide i , $CR_i(E)$ is the contribution ratio of one gamma-ray to the individual dose rate of nuclide i , and A_i is the radioactivity in the ground.

2.3 Experimental verifications

The calculation results of the radioactivity using the ERM were verified by comparing with results from a portable HPGe detector and the sample analysis results taken from the soil around the ERM, as shown in Fig. 2. The used portable HPGe detector uses in situ objective counting system (ISOCS) software (ver. 4.2.1) to calculate the theoretical efficiency in accord with the geometry in the in situ measurement.

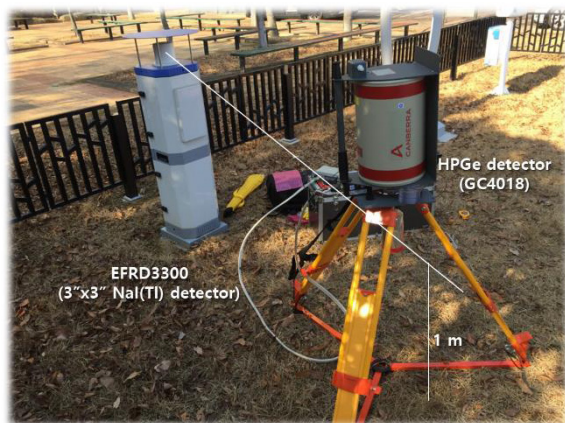


Fig. 2. The in situ gamma-ray spectrometry using a

portable HPGe and ERM.

Four natural radionuclides, such as ^{214}Bi , ^{228}Ac , ^{208}Tl , and ^{40}K , were analyzed using an ERM based on a 3"x3" NaI(Tl) detector, by applying the interference correction. All analysis results of the radioactivity of four natural radionuclides in the ground obtained by an ERM using dose rate spectroscopy were within a 20 % difference from those of the analysis sample taken around the ERM.

3. Conclusion

Dose rate spectroscopy was applied to an ERM based on a 3"x3" NaI(Tl) detector to conduct real-time monitoring of the ambient dose rate as well as the individual dose rate and radioactivity for detected gamma nuclides around an ERM. The experimental verification was conducted at two different sites with installed ERMs, and the results were compared to those obtained using a portable HPGe detector and sample analysis. A good agreement between an ERM and sample analysis was achieved in the analysis results of four natural radionuclides in the ground. As a result, it could be concluded that an ERM with a gamma-ray spectrometer can give reliable and stable monitoring results of ambient dose rate as well as individual dose rate and radioactivity of detected gamma nuclides around the ERM.

REFERENCES

- [1] Ji YY, Chung KH, Lee W, Park DW, Kang MJ, "Feasibility on the spectrometric determination of the individual dose rate for detected gamma nuclides using the dose rate spectroscopy", *Radiat. Phys. Chem.* 97, 172-177, (2014).
- [2] Ji YY, Chung KH, Kim CJ, Kang MJ, Park ST, "Application of the dose rate spectroscopy to the dose-to-curie conversion method using a NaI(Tl) detector", *Radiat. Phys. Chem.* 106, 320-326, (2015).
- [3] Ji YY, Kim CJ, Chung KH, Choi HY, Lee W, Kang MJ, Park ST, "In situ gamma-ray spectrometry in the environment using dose rate spectroscopy", *Radiat. Phys. Chem.* 119, 90-102, (2016).

Design Requirements of a Large-Area Radiation Measuring Instrument for Decommissioning of Nuclear Power Plants

Kidoo Kang*, Yuho Won, Moonhyung Cho, and Jaeun Lee

Central Research Institute, Korea Hydro & Nuclear Power Co., Ltd, 70, Yuseong-daero 1312beon-gil, Yuseong-gu, Daejeon, Republic of Korea

*kidookang@gmail.com

1. Introduction

There are many kinds of radiation measuring instruments in nuclear power plant for the purpose of radiation protection of workers, radioactivity analysis of radwaste and radiation monitoring of the environment. Korea's first commercial nuclear power plant Kori-1 shut down permanently in June 2017. The actual decommissioning will begin in 2022, and then various measuring instrument are needed. Particularly, in order to inspect the contamination of wide area such as the floor of the radiation zone or the site of the power plant, the customized equipment is required to overcome the limited application of current floor contamination monitor system.^[1]

2. Design requirements of the radiation measuring instrument

2.1 Background

KHNP-CRI established centralized radiation calibration laboratory in 2010 to enhance the management of radiation instrument. It includes the procurement, maintenance and management of portable radiation instruments, calibration and related research. The calibration laboratory was accredited by KOLAS authorized by ILAC. The total number of the accredited items is 11, so far.

Recently, KHNP has developed a new clearance monitor to determine whether the metal scraps meet the regulatory criteria or not. The special features of the clearance monitor are; 1) 12 plastic scintillators and 2 NaI scintillators enable nuclide identification 2) The MDA was lowered to around 1/100 of the allowed clearance level 3) the chamber volume was increased by two times. Integrated operation and performance tests of the clearance monitor were performed. The results of the measurement of metal

scraps in which standard sources were placed (approximately 1/100 of the clearance level) showed a good agreement with the source activity within error of -16 % ~ -12 %.^[2]

Based on the infrastructure of the radiation lab and experiences with development of radiation measuring instrument, KHNP-CRI is developing 'Large-Area Radiation Measuring Instrument for Decommissioning of Nuclear Power plants' as a part of the mid- to long-term research plan.

At the decommissioning stage of nuclear power plants, it is required to measure various types of buildings, structures, and systems. The measuring results are widely used for a decommissioning plan and other work process.

The radiation survey is done by confirming contamination of material, its quantitative evaluation, and analyzing the radionuclide and its concentration. The method consists of scanning method, static method, and sampling method which are summarized in Table 1.

Table 1. Comparison of radiation survey for decommissioning

Category	Scanning method	Static method	Sampling method
Usage	contamination distribution analysis	quantitative evaluation	radionuclide analysis
Sample representative	very high	high	low
Accuracy	low	high	very high
Equipment	surface contamination monitor	- surface contamination monitor - in-situ gamma spectrometer	- alpha spectrometer - liquid scintillation counter - gamma spectrometer
Measuring target	floor, site	system, structure, wall	concrete, soil, groundwater

2.2 Objectives of CRI's prototype instrument

The target of the development of 'Large-Area Radiation Measuring Instrument for Decommissioning for Nuclear Power Plants' is to develop prototype instrument and method for measuring radioactivity of flat surface such as floor of the radiation zone or the site of the power plant. To get the satisfactory result, several sub-targets were considered as shown in Table 2.

Table 2. Sub-targets of large-area radiation measuring instrument

Classification	Unit	Performance goal
1. size of detector	cm	over 120(W) × 15(D)
2. MDA	Bq/cm ²	10%~50% of derived concentration guideline level
3. detecting efficiency	%	30% ≥ ⁹⁰ Sr/ ⁹⁰ Y
4. measuring time	min.	2min./m ² ~ 5min./m ²

The size of the detector will be determined so as to overcome the limitations of existing surface contamination monitors including MDA, economic efficiency, and maintainability.

The minimum detectable activity (MDA) should be determined at the level of 10% to 50% of the derived concentration guideline level, based on the Scan MDC calculation formula presented in MARSSIM.^[3]

The channel interference between alpha and beta is designed within less than 10% .

Measuring time is defined as the time taken to measure the MDA at 1 m² area. It takes 222 minutes for a pancake type(detector size 15 cm²) and 33 minutes for a flat type(detector size 100 cm²) The prototype instrument with measuring time of 5 minutes will save time by 85 ~ 95%.

2.3 Additional Required Features

Some additional features are; 1) mapping of measured value based on position-sensitive detector 2) implementation of real time MDA 3) implementation of statistical algorithms based on sign test/WRS test.^[3]

Moreover, it must have a function to select the automatic and manual modes to prevent human error.

3. Conclusion

As a part of the company's mid- to long-term research and development plan, KHNP is carrying out a R&D 'development of large-area radiation measuring instrument for decommissioning of nuclear power plants'. Its purpose is to develop position sensitive radiation measuring device.

The MDA of the new device will be at the level of 10 to 50% of the residual radioactivity concentration, while the detection efficiency is over 30%, and measuring time is about 2~5 minutes for 1 square meter. The measuring time can be drastically shortened(over 85%) comparing to the existing surface contamination monitor.

In addition, the minimum detectable concentration (MDC) will be displayed in real time with statistical verification algorithm, which will greatly enhance reliability of measurement for decommissioning survey.

REFERENCES

- [1] M.H Cho, "A Feasibility Study on Radioactivity Evaluation Technologies for Structures and Materials having Various Shapes and Sizes", KHNP Technical Report, 31-45 (2016).
- [2] K.D Kang, "Development of Clearance Monitor for NPP's EM Wastes and its Performance Test", Proc. of the 2017 Waste Management Conference, March 5-9, 2017, Phoenix.
- [3] NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual" (2000).

Radioactivity Characteristics and Safety Management of Steel Scarp

Yeoungsuk Go^{1*}, Eun Lee¹, Suk K. Sim¹, Sun Dong Ju², and Sang-uk Oh²

¹Environment & Energy Technology, Inc., 213, R&D Building 2, 99, Gajeong-ro, Yuseong-gu, Deajeon, Republic of Korea

²Korea Foundation of Nuclear Safety, 49, Deawangpangyo-ro 644beon-gil, bundang-gu, Seongnam-si, Gyeonggi-do, Republic of Korea

*ysgo@en2t.com

1. Introduction

After the Fukushima nuclear accident in 2011 safety management of NORM (Naturally Occurring Radioactive Material) became one of the key issues of nuclear safety in Korea and ‘Safety Management Law of the NORM [1]’ has been implemented and enacted in 2012. Steel scrap is classified as inadvertent radioactive materials and under regulatory control by the “Safety Management Law of the NORM” To protect the radiation exposure of the general public as well as the field workers, steel industry in Korea installed and operates the radiation monitoring system at the entrance of the steel scrap trucks. In this study, the radiation characteristics of the steel scrap with inadvertent radiation detected recently has been analyzed and suggests some regulatory guides for the management, disposal and reuse as well as the safety management of the steel scrap after reviewing Ukrainian regulatory and safety management of the steel scrap since Chernobyl nuclear accident in 1986.

2. Radioactivity Characteristics of the Recent Inadvertently Detected Steel Scrap

According to the “Safety Management Law of the NORM” the steel scrap load in the truck should be passed through the radiation detection system installed at the truck entrance to the steel manufacturing industry. As shown in Table 1, measured radioactivity and nuclides have been measured for 8 times for 4 steel companies in Korea. It was shown in Table 1 that the radiation levels of the inadvertently detected radioactive steel scrap exceed the general public dose limit of 1 mSv/yr

recommended by IAEA. For the field detection, portable survey meter was used for the radiation measurement and radioactive nuclide identification. Detailed radiation analysis has been performed following the initial field detection and analysis. In Korea, domestic steel scrap is classified as imported steel scrap, NORM and orphan steel scrap. For the imported steel scrap, the inadvertently detected steel scrap is returned to its origin port. However, for the steel scrap with NORMs or the orphan steel scrap, it is difficult to trace back and identify the origin of the steel scrap due to various route of the transportation for the steel scrap.

Table 1. Steel Scrap Field Detection and Radiation Characteristics

	Dose (mSv/yr)	Nuclide
Company A 1st	10.51	Ra
Company A 2nd	35.04	Th
Company A 3rd	37.41	Co
Company A 4th	131.4	Co
Company B 1st	47.30	Th, Rn
Company B 2nd	120.01	-
Company C	3.066	-
Company D	118.26	Th, ³ H

3. Ukrainian Steel Scrap Regulation and Safety Management

Ukrainian government implemented the radiation safety management regulation for the steel scrap industry since the Chernobyl nuclear accident in 1986 [2]. By the regulatory guides, steel scrap dealers should take the radiation safety and regulation training for the Radiological

Certificate from the government. In Ukraine, there are almost 2,400 certified steel scrap dealers. Ukrainian regulation for the Radiological Certificate classifies the sale of the steel scrap in three categories based on the radiation level at 1m above the steel scrap surface; domestic use and export, only domestic use, and no use even in Ukraine. Ukraine also established multi-barrier system to control steel scrap; steel scrap dealers, border and Exclusive Zone (Chernobyl) and export consignment. The licensing through the certificate and supervision extend to the steel scrap dealers for regulation of the raw materials and products, periodic monitoring of the work place and storage facility, and also the steel scrap export and sales control by issuing Radiological Certificate.

4. Conclusion

Currently, the inadvertent radiation detections of the steel scrap is increasing in Korea. However there is no regulatory clearance level for the artificial nuclides of the steel scrap in the “Safety Management Law of the NORM” Therefore, for the safety control and management of the steel scrap, the criteria of clearance level as well as the regulatory guides for the management, disposal and reuse of the steel scrap should be established through reviewing international safety standards and criteria of the steel scrap. Fixed radiation monitoring system is also required for the small and medium steel scrap dealers to trace and identify the origin of the inadvertently detected steel scrap. Regulatory initiative is needed to establish regulatory guides in dilution and disposal such as permanent disposal storage for the NORM and inadvertently detected steel scrap.

REFERENCES

- [1] Safety Management Law of the Naturally Occurring Radioactive Materials, 2012.
- [2] Prevention, Detection and Response System for Inadvertent Radioactive Material in the Scrap Metal in Ukraine. 2008.

Radiological Assessment of Spent Resin Treatment Device for Disposal Safety

UkJae Lee*, Woo Nyun Choi, and Hee Reyoung Kim

Department of Nuclear Engineering, Ulsan National Institute of Science and Technology,

Banyeon-ri, Eonyang-eup, Ulju-gun, Ulsan 44919, Korea

*dldnrwp@unist.ac.kr

1. Introduction

At a nuclear power plant (NPP), resins are used to purify liquid radioactive waste generated from operating NPP. In the case of pressurized heavy water reactor, resin is used in deuterium and deuterium, liquid waste disposal system etc. The spent resin is stored in a resin storage tank and processed in accordance with the decommissioning plans. Resins generated from pressurized heavy water reactors contain various radioactive nuclides and are classified as an intermediate level radioactive waste (ILW), especially due to high concentration of long-lived radioactive C-14 (half-life of 5,730 years). In case of Wolsong unit 1,2,3,4, disposal costs of about 80 billion Won are required to dispose without treating the spent resin [1]. To solve this problem, a spent resin treatment device has been developed which able of recovering and recycling the C-14 from spent resin. In this paper, VISIPLAN code was used to evaluate the external exposure dose [2].

2. Method

The spent resin will be treated through a heat desorption process using a treatment device such as a microwave.

The VISIPLAN exposure dose assessment code was used to confirm the radiation safety of radiation workers on the spent resin treatment device.

2.1 Modeling of spent resin treatment device

The actual spent resin treatment device is modeled based on the basic structure of box, cylinder, sphere, cap, cone etc. supported by VISIPLAN code. The space with the device was defined as a concrete wall with a length of 4 m and the device was located at the center of the space. The device was modeled with stainless steel material. In the case of the main

process part, the sphere cap model was used for the upper and lower structures, and the tube model was used for the middle part. For the tube connecting the main process part, the tube model was used as the basis and bending part was modeled using the torus model. The cone model was used to model the connection between the upper part of the process part for desorbing C-14 in the resin and the transfer pipe using a microwave.

2.2 The source information of the spent resin treatment device

The radioactivity of the radionuclides in the resin was checked to evaluate the exposure dose to the spent resin treatment device. As shown in Table 1, based on the radioactivity concentration in the resin derived from the experiment, the maximum value of the radioactive concentration for each nuclear species was used, and the radioactivity values corresponding to 1,000 g of the input dose at the time of the device operation was derived.

Table 1. Radioactivity concentration of resin and radiation of the spent resin treatment device

Nuclide	Radioactivity Concentration (Bq/g)	Radioactivity (Bq)
H-3	3.75E+04	3.75E+07
C-14	2.23E+05	2.23E+08
Co-57	2.91E+01	2.91E+04
Co-60	4.94E+02	4.94E+05
Cr-51	2.58E+02	2.58E+05
Cs-134	1.57E+01	1.57E+04
Cs-137	9.09E+03	9.09E+06
Mn-54	2.67E+01	2.67E+04
Nb-95	4.39E+01	4.39E+04
Sb-125	4.25E+02	4.25E+05
Zr-95	2.75E+01	2.75E+04
Eu-152	5.12E+02	5.12E+05
Eu-154	4.33E+01	4.33E+04

2.3 Exposure scenario for radiation worker

Based on the worker scenario, the radiation exposure dose of the workers was calculated. In order to operate the spent resin treatment device, it is assumed that the time required to put the spent resin into the device and operate the device is 1 hour. The deviation of the working time was set to 10 minutes, and the dose was evaluated for the working time of 50 minutes and 70 minutes. The position of the worker was set at a distance of 20 cm from the center surface of the device and the height was set at 120 cm, which is the height from the ground to the worker's chest. The main internal exposure pathway was considered as respiration and ingestion. The 1 % outflow from treatment device was assumed.

3. Result and discussion

3.1 Dose rate distribution by the spent resin treatment device

As shown in Fig. 1, based on the structural modeling of the spent resin treatment device and the definition of the source, the radiation dose rate distribution in the space where the device is located was confirmed.

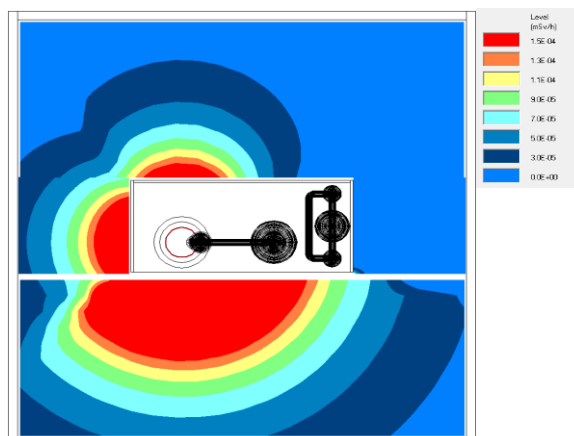


Fig. 1. The distribution of the exposure dose rate due to the spent resin treatment device derived by VISIPLAN (Z-view).

For the dose distribution rate, the highest value was found to be $1.1\text{E}-03$ mSv/h. It was confirmed that the nuclide which has the largest influence on the dose distribution rate was Cs-137.

3.2 Dose assessment of radiation worker

As shown in Table 2, for 1 hour of working time, the received dose was calculated as $5.3\text{E}-05$ mSv/h. In order to derive the annual worker dose based on the received dose per hour, 2,000 hours which is the annual working hours of a typical radiation worker was used. The derived dose value is 0.106 mSv. It can be checked that the annual dose received from the spent resin treatment device doesn't reach the 100 mSv limit for 5 years and 50 mSv limits for 1 year in maximum, the annual worker dose. The internal dose was evaluated as 3.22 mSv in terms of committed effective dose.

Table 2. Dose assessment result of radiation worker

Working Time (min)	Exposure Dose (mSv)
50	$4.4\text{E}-05$
60	$5.3\text{E}-05$
70	$6.1\text{E}-05$

4. Conclusion

The radiological assessment for radiation worker of spent resin treatment process was conducted based on VISIPLAN. The internal dose was evaluated about 30 times higher than the external dose. It is thought to be due to beta-emitting nuclides such as C-14. This result shows that the annual dose due to spent resin treatment device does not reach the annual worker's dose limitation. Therefore, the radiological safety of the workers in the spent resin treatment process is secured.

REFERENCES

- [1] Ministry of Trade, Industry and Energy, No. 2014-165. "Regulation on the calculation criteria for management costs of radioactive wastes and spent nuclear fuel et al."(2014).
- [2] Vermeersch, F. "VISIPLAN 3D ALARA planning tool, User's guide." *SCK CEN, Mol Belgium* (2005).
- [3] ICRP, 1991. 1990 Recommendations of the International Commission on Radiological Protection. ICRP Publication 60. Ann. ICRP 21 (1-3).

Inhalation Dose Assessment at Phosphate and Potassium Industries

Jung Hwan Jang¹, Seung Woo Ji¹, Jin Hee Jeon¹, Jae Kook Lee^{1,2}, and Kwang Pyo Kim^{1*}

¹Kyung Hee University, 1732 Deokyoungdaero, Giheung-gu, Yongin, Gyeonggi-do, Republic of Korea

²Korea Institute of Nuclear Safety, 62 Gwahak-ro, Yuseong, Daejeon, Republic of Korea

*kpkim@khu.ac.kr

1. Introduction

Phosphate and potassium are naturally occurring radioactive materials (NORM). Therefore, the workers in the phosphate and potassium industries are subjected to internal exposure due to inhalation of airborne particulates containing radioactive materials. International Atomic Energy Agency (IAEA) suggested industry sectors as being required some form of regulation consideration [1].

Inhalation dose to workers varies with particulate properties including airborne particulates concentration and radioactive concentration. International Commission on Radiological Protection (ICRP) recommends that site-specific information on particulate properties should be measured and then used for inhalation dose assessment [2]. Therefore, the actual measurement data of particulate properties is required for accurate dose assessment.

The objective of this study was to assess inhalation dose based on the actual measurement data of airborne particulates. Assessment of inhalation dose was implemented for phosphate and potassium industries in Korea.

2. Materials and Methods

The properties of airborne particulates were characterized at major processing area. The particulate properties included concentration in the air, size distribution, mass density, shape, and radioactive concentration. In this study, we selected two fertilizer facilities. One handles mainly phosphate materials and the other handles mainly potassium materials.

Cascade impactor was used to collect airborne particulates (Fig. 1). The measurement of the radioactivity concentration of materials taken from the major process was performed using HPGe detector (Fig. 1). The mass density and physical shape of airborne particulates were analyzed using pycnometer and scanning electron microscopy respectively.

Inhalation dose coefficients were derived from the ICRP-66 human respiratory tract model and particulate properties. Inhalation doses to workers were calculated using inhalation dose coefficients and exposure scenarios at fertilizer facilities.



Fig. 1. Cascade impactor and HPGe detector.

3. Results and Discussion

Fig. 2 shows airborne particulate concentrations collected in two NORM facilities. Airborne particulates concentration widely ranged 0.075-203.32 µg/L. Airborne particulate concentration was the highest for phosphate rock warehouse. Lots of airborne particulates were scattered at the phosphate rock warehouse while on the work.

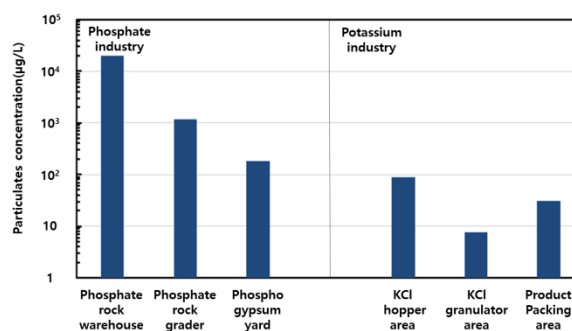


Fig. 2. Airborne particulates concentration in phosphate handling facility and potassium handling facility.

Fig. 3 shows the radioactivity concentrations. Radioactivity concentrations of the materials ranged 33-866 Bq/kg for uranium series, 0-2 Bq/kg for

thorium series, and 0-15,400 Bq/kg for K-40. Radioactivity concentrations were the highest in phosphate rock for uranium decay series and phospho-gypsums for thorium decay series. Potassium chloride had the highest K-40 concentration.

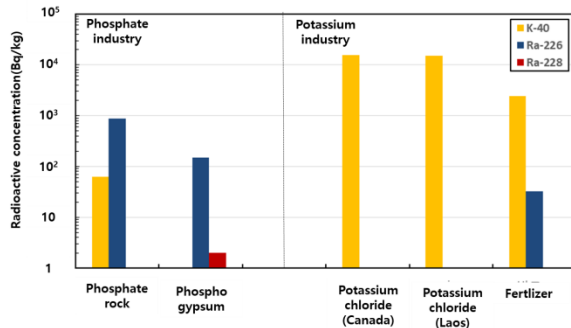


Fig. 3. Radioactivity concentrations of NORM materials in phosphate handling facility and potassium handling facility.

Mass densities were about 3.40 g/cm³ for phosphate rock, 2.1 g/cm³ for phospho-gypsum, 1.97 g/cm³ for KCl, and 2.0 g/cm³ for fertilizer. The airborne particulates appeared to be spherical in most processing area. Therefore, unity was assigned as shape factor for internal dose assessment.

Inhalation doses were calculated using the particulate properties and exposure scenarios. Inhalation doses widely ranged 3.95×10^{-9} – 4.02×10^{-1} mSv/y. Inhalation dose was the highest at phosphate rock warehouse as 0.402 mSv/y. Airborne particulate concentration at the warehouse was up to 20 times higher than the other areas. The difference in radiation doses was mainly due to airborne particulate concentration and radioactivity concentration.

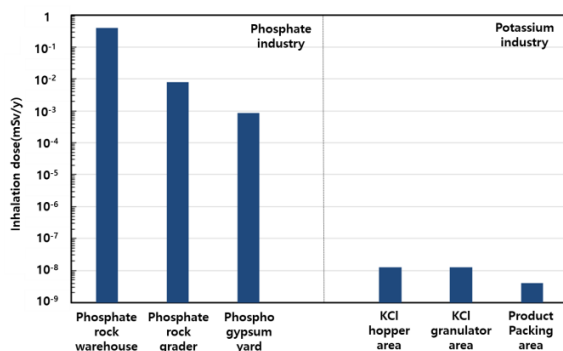


Fig. 4. Inhalation doses in phosphate handling facility and potassium handling facility.

4. Conclusion

We assessed inhalation doses to the workers phosphate handling facility and potassium handling facility. Inhalation doses widely varied with airborne particulates properties. Therefore, optimization of radiation protection in NIORM industry is necessary. The results of this study can be used for characterizing radiation exposure at Korean NORM industries.

ACKNOWLEDGMENTS

This work was supported by a grant from “Establishment of Technical Basis for Implementation on Safety Management for Radiation in the Natural Environment” carried out by Korea Institute of Nuclear Safety.

REFERENCES

- [1] IAEA, Assessing the Need for Radiation Protection Measures in Work Involving Minerals and Raw Materials, SRS 49 (2006).
- [2] ICRP, Recommendations of the International Commission on Radiological Protection, ICRP 66 (1994).

6분과

방사선환경 및 안전 (Poster)



Improved PRIDE Access Control System

Seonho Noh*, Youngkuk Jang, Hui-seok Gang, and Ilje Cho

Korea Atomic Energy Research Institute, 989-111 Daedeokdaero, Yuseung-gu, Daejeon, Korea

*nohsh@kaeri.re.kr

1. Introduction

There is a facility named PRIDE in KAERI for developing Pyroprocessing technology. In PRIDE depleted Uranium feed material and a depleted Uranium mixed with some surrogate material are used for performing engineering scale Pyroprocessing.

When entering PRIDE, every people had to use the RFID(Radio Frequency Identification) system. But the RFID system had a lot of disadvantages. To get more specific radiation dose and data, we improved PRIDE's access control system.

2. Improvement

2.1 Pre-System

RFID system was PRIDE's existing system. It detects Radio Frequency when a person tries to enter the facility. After the RFID system detected the frequency, the entering person has to type some information to open the gate. And even if the person wants to exit the facility, the person has to type some information again.



Fig. 1. PRIDE pre-system.

This system had a lot of disadvantages. Some people miss read their personal dosimeter and write

the wrong value in the system. And also some people can just write '0' in the system because they are annoyed to read their personal dosimeter. So we can tell that the system is very weak on radiation protection perspective.

2.2 New system

Fig. 2 is the new access control system which is installed in the PRIDE fitting room.



Fig. 2. PRIDE new access control system.



Fig. 3. System gate.

As you can see there is a gate in the new system. The gate is to prevent people who is not authorized to enter PRIDE. Also there is a hole which you can see in Fig. 4. The hole is to put the personal dosimeter in. When the dosimeter is in the hole, the computer

which is linked to the hole reads the personal radiation dosimeter automatically. It reads the radiation value and the entering time in real time. After reading the system sends the data to the main server which is managed by the ‘Security Division’



Fig. 4. New system's hole.

3. Conclusions

We can now strictly manage the radiation-controlled area by improving the access control system. Also it can collect all the person's radiation level easily than the pre access system in real time

This system will be a beneficial useful in many areas or facility which has to be strictly managed.

REFERENCES

- [1] ANL-7959 Hot Fuel Examination Facility /North Facility Safety Report, February 1975, Argonne National Laboratory pp. 42-53.
- [2] The EBR-II Fuel Cycle Story, Charles E. Stevenson, American Nuclear Society pp. 16-25.

Tritium Inventory Estimation for HCCR-TBS at PD-1 Phase

Hyung Gon Jin ^{a*}, Dong Won Lee ^a, Jae Sung Yoon ^a, Suk Kwon Kim ^a, Eo Hwak Lee ^a, Seong Dae Park ^a,
Dong Jun Kim ^a, Yunjeong Hong ^a, and Seungyon Cho ^b

^aKorea Atomic Energy Research Institute, 111 Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Republic of Korea

^bNational Fusion Research Institute, 169-148 Gwahak-ro, Yuseong-gu, Daejeon, Republic of Korea

*jhg@kaeri.re.kr

1. Introduction

Korea has developed a Helium Cooled Ceramic Reflector (HCCR) TBM to be tested in the ITER [1-4]. It consists of two major loops, which are HCS (Helium Cooling System) and TES (Tritium Extraction System) (Fig. 1). Tritium is one of the most highly permeable molecule on earth, therefore tritium permeation takes place from TES to HCS in the TBM. Permeated tritium migrates along the system pipes, thus tritium inventory should take into account with respect to entire HCCR TBS. This paper presents assumption of the estimation, boundary condition and summary of the results.

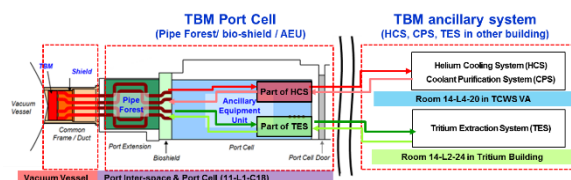


Fig. 1. Schematics of HCCR-TBS.

2. Assumptions

To estimate the tritium inventory of HCCR TBS, followings were assumed.

- Tritium production is given by neutronics calculation. (No extra tritium flux from plasma)
- Continuous back to back plasma pulse with duty factor 0.25.
- Steady state condition (instantaneous tritium diffusion to the purge gas isotropically)
- Averaged tritium generation rate (constant over the time and uniform in the TBM volume)
- Transport mechanism through structural materials is bulk diffusion.
- Tritium in NAS tube is accumulated over time. (No service vacuum system)
- Tritium concentration in space is constantly zero to

estimate conservative release rate.

- Averaged temperature and pressure for each region

3. Material Properties

Material properties were given by benchmark guideline [5], therefore, Eurofer's properties were used for TBM steel instead of Korea RAFM steel (ARRA)'s one, and AISI-316L were chosen for estimation of diffusivity, solubility and permeability of hydrogen isotopes in the pipes. For the benchmark exercise Li_2SiO_4 residence time was used.

Table 1. Material properties

Parameter	Value/correlation	unit of measurement
Sieverts' constant Eurofer/T	$2.25 \cdot 10^{-2} \cdot \exp(-15100/RT)$	$\text{mol} \cdot \text{m}^{-3} \cdot \text{Pa}^{-0.5}$
Diffusivity Eurofer/T	$4.57 \cdot 10^{-7} \cdot \exp(-22300/RT)$	$\text{m}^2 \cdot \text{s}^{-1}$
Sieverts' constant AISI-316L/T	$1.47 \cdot \exp(-20600/RT)$	$\text{mol} \cdot \text{m}^{-3} \cdot \text{Pa}^{-0.5}$
Diffusivity AISI-316L/T	$7.66 \cdot 10^{-8} \cdot \exp(-42500/RT)$	$\text{m}^2 \cdot \text{s}^{-1}$
Tritium residence time in Li_2SiO_4	$1.28 \cdot 10^{-5} \cdot \exp(9720/T)$	h

4. Boundary Conditions

Equations should be placed on separate lines and numbered:

Table 2. Summary of boundary condition

Tritium Production Rate	25.9 mg/day (continuous back to back with duty 0.25)
PI/PC Volume	280.264 m^3
TCWS VA Volume	500 m^3 (considering occupied by HCCR HCS/CPS)
TES Pipe Thickness	3.68 mm
HCS Pipe Thickness	8.56 mm (Vertical shaft pipe thickness 8.56 mm)
BZ Pipe Thickness	4.0 mm
N-DS at PI/PC	40 m^3/h
HVAC at TCWS	1 Vol/hr, i.e. 500 m^3/h

5. Geometrical Information

Fig. 2 is pipe length and operational temperature information of each room which HCCR-TBS is allocated. Design of HCCR-TBS is evolving and this data reflects recent update of PD-1 phase.

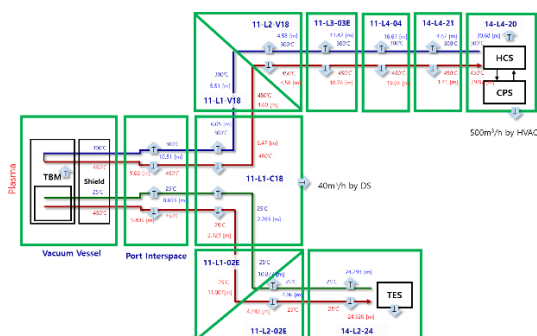


Fig. 2. Pipe length and temperature of HCCR-TBS.

6. Inventory Estimation Results

This result is one of outcomes of TF-TPSD (Task Force TBM Program Safety Demonstration), which includes tritium release, inventory and benchmark analysis. It has been provided for CP-SDD (Connection Pipe System Design Description) as an input data set of HCCR-TBS. Table 3 is a summary the results which consists of four parts (TBM box steel, TES steel, HCS steel and ceramic breeder) and tritium partial pressure of each sub system as an information.

Table 3. Summary of tritium inventory results

Tritium partial pressure in HCS (Pa)	0.0104
Tritium partial pressure in TES (Pa)	0.202
Tritium solubilized in the TBM box steel (mg)	5.25E-01
Tritium solubilized in the TES steel (mg)	6.00E-01
Tritium solubilized in HCS steel (mg)	4.46E+01
Tritium inventory in ceramic Breeder (mg)	2.07E+00

7. Conclusion

Tritium inventory of the PD-1 phase HCCR-TBS has been estimated. One important thing for this result is degree of conservatism/margin of estimated tritium inventory is huge due to oxide layer of the surface, isotope effect caused by existence of hydrogen in coolant and operational scenario.

Tritium inventory issue in fusion facility is critical and has huge uncertainty to get realistic release rate. Practically it is expected that tritium amount in human access area for fusion reactor is negligible or manageable, however, dynamic and multi-dimensional tritium inventory analysis including operational scenario could be a solution for detail estimation

8. Acknowledgement

This work was supported by the R&D Program through the National Fusion Research Institute (NFRI) funded by the Ministry of Science and ICT of the Republic of Korea (NFRI-IN1703).

REFERENCES

- [1] D.W. Lee, et. al., “Current Status and R&D Plan on ITER TBMs of Korea,” Journal of Korean Physical Society, 49 S340-S344 (2006).
- [2] D.W. Lee, et. al., “Preliminary Design of a Helium Cooled Molten Lithium Test Blanket Module for the ITER Test in Korea,” Fusion Eng. Des. 82, 381-388 (2007).
- [3] D.W. Lee, et. al., “Helium Cooled Molten Lithium TBM for the ITER in Korea,” Fusion Sci. and Tech. 52, 844-848 (2007).
- [4] D.W. Lee, et. al., “Design and Preliminary Safety Analysis of a Helium Cooled Molten Lithium Test Blanket Module for the ITER in Korea,” Fusion Eng. Des., 83, 1217-1221 (2008).
- [5] I. Ricapito, TBM Project team, F4E “Guideline for the Benchmark Exercise on Tritium Modelling of TBSs”, TM9J3Q (2016).

External Dose Assessment to Workers in a NORM Industry Applying External Dose Rate Map

Seung Woo Ji¹, Jung Hwan Jang¹, Do Yeon Kim¹, Jeong Hwan Yoon¹, Jae Kook Lee^{1,2}, and Kwang Pyo Kim^{1*}

¹ Kyung Hee University, 1732 Deokyoung-daero, Giheung-gu, Yongin-si, Gyeonggi-do, Republic of Korea

² Korea Institute of Nuclear Safety, 62, Gwahak-ro, Yuseong-gu, Daejeon, Republic of Korea

*kpkim@khu.ac.kr

1. Introduction

Naturally occurring radioactive materials (NORM) contain natural radionuclides. Therefore the workers in the NORM industries are subjected to radiation exposure. A new law of Natural Radiation Safety Management Act was enacted in the Korea to protect workers against ionizing radiation from NORM.

Potassium is one of the NORMs. Natural potassium contains 0.0118% of ⁴⁰K which is natural radionuclide [1]. It decays to ⁴⁰Ca by emitting a beta particles and ⁴⁰Ar by emitting gamma radiation. The strong gamma radiation emitted from ⁴⁰K may result in radiation exposure to workers in potassium handling industries [2]. However, the radiation exposure to workers in potassium industry has not been adequately addressed. The objective of this study was to assess external dose to workers at potassium industry applying external dose rate map.

2. Materials and Methods

The external dose rates were measured at major potassium handling facilities using portable survey meters. At each facility, site was divided in to several grids. External dose rates were measured for 3 times at center of each mesh and averaged. Each grid was classified into four groups depending on the measured dose rates. Table 1 shows major three potassium industries and their processing area.

Table 1. Major potassium industries and processing area

Facility	Major processing area
I	Warehouse, Packing area A, B
II	Warehouse, Mixing area, Packing area
III	Input area, Drying area, Packing area

Target workers for external dose assessment were selected based on the external dose rate map. Most of the target workers spent time within the grid with relatively high external dose rate. Exposure scenario was analyzed by interview with target workers. We calculated external dose to the target workers using measured external dose rate and exposure scenario.

3. Results and Discussion

Fig. 1 shows external dose rate map at whole site of facility II. The external dose rates at most of potassium facilities were background level.

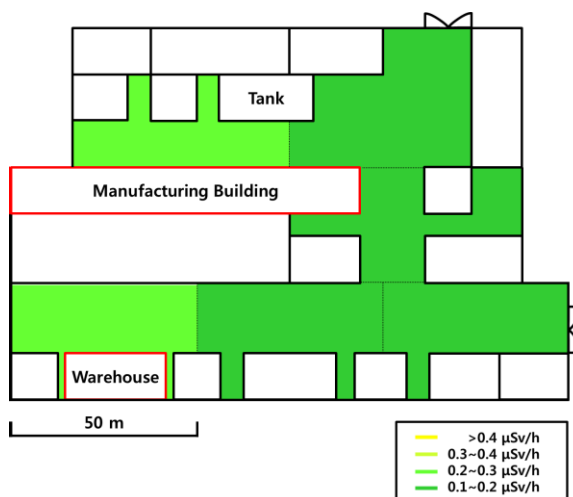


Fig. 1. External dose rate map at whole site in the facility II.

Fig. 2 shows external dose rate map at warehouse of facility II. Relatively high external dose rate was found at some sites such as warehouse in the facility II. At the some sites such as warehouse, a large amount of potassium compounds were handled. Based on the external dose rate map, 9 processing area were selected for external dose assessment at potassium facilities.

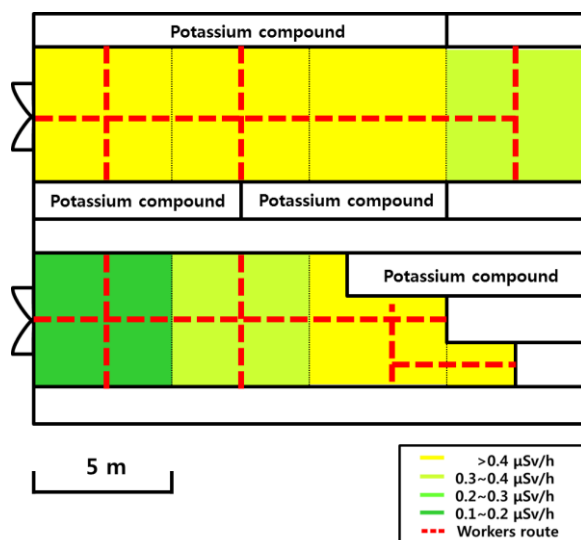


Fig. 2. External dose rate map at warehouse in facility II.

Fig. 3 shows annual radiation dose to the workers at potassium industries. Annual dose varied with workers, widely ranging 0.011 - 0.368 mSv/y. Radiation dose was the highest to worker at packing area A in the facility I. It could be contributed that working time at packing area in the facility I was higher than other processing areas.

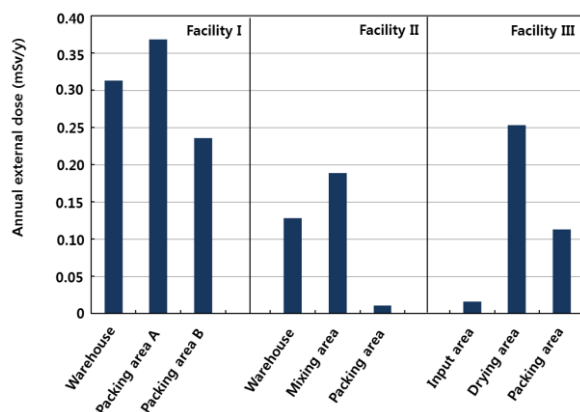


Fig. 3. Annual external dose to workers at potassium industries.

4. Conclusion

External radiation dose to workers in potassium handling industries were assessed in this study. External dose rates were directly measured and dose rate map was generated at 3 potassium facilities for the external dose assessment. Based on the external dose rate map methods given in this study, target workers can be selected effectively. External dose

rate map also can be used for management of radiation exposure to workers. The results of this study can contribute to protect workers in NORM industries against ionizing radiation.

ACKNOWLEDGMENTS

This work was supported by a grant from “Establishment of Technical Basis for Implementation on Safety Management for Radiation in the Natural Environment” carried out by Korea Institute of Nuclear Safety.

REFERENCES

- [1] FPTRPC, Canadian Guidelines for the Management of Naturally Occurring Radioactive Materials, Canadian NORM Working Group of the Federal Provincial Territorial Radiation Protection Committee (2011).
- [2] K. Emumejaye, Determination of ^{40}K Concentration in Some Powdered Milk Samples Consumed in Delta State, Nigeria, IOSR Journal of Applied Physics, 2(3): 8-12 (2012).

Improvement of Remote Control of Storage Vault Lid O-Ring Replacement in ACPF

Yunmock Jung*, Jongkwang Lee, Sunseok Hong, and Eunyoung Choi

Korea Atomic Energy Research Institute, Daedeok-daero 989-111, Yuseong-gu, Daejeon, 34057, South Korea

*jym@kaeri.re.kr

1. Introduction

When performing a pyroprocessing experiment, Uranium / molten salt ingot which are by-products are generated and need a place to store them. We have called this place a Storage Vault which is located in the ACPF (Advanced spent fuel Conditioning Process Facility). The O-ring inserted in the lid, one of the components of the storage vault was designed without considering the replacement at the time of first production. However, insertion and replacement of the O-ring is necessary for the continuous and safe use of the storage vault. And in a special environment called a hot cell, there should be a way to replace the O-ring with MSM (Master and Slave Manipulator). Therefore, the purpose of this research is to improve the remote control of storage vault lid O-ring replacement.

2. Experimental Procedure

2.1 Improvement of SV lid

The storage vault is divided into vault A and Vault B, and their specifications are shown in Fig. 1.

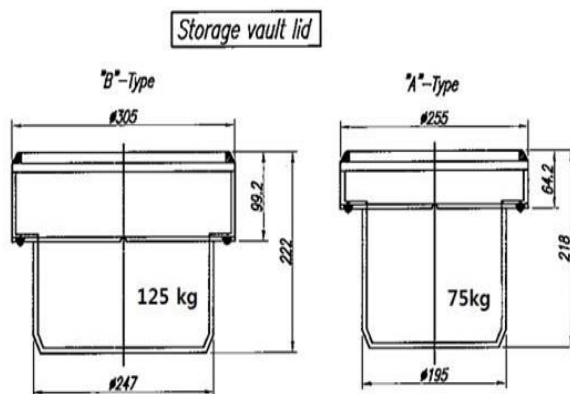


Fig. 1. Specification of storage vault.

The existing O-ring groove of storage vault was manufactured to standard size. However, Continuous use has resulted in damage to the interface surface of the O-ring/storage vault container, requiring measures to maintain airtightness. Due to the specificity of the hot cell, the surface of the vault couldn't be processed directly so the O-ring grooves were machined. Therefore, a part of side surface of the O-ring grooves were cut and proceed so that 3.2 to 3.5 mm the O-ring protrudes out of the O-ring grooves. Four remote change holes ($\phi 8$) were drilled at 90° intervals in the storage vault circular O-ring groove and O-ring replacement tool was manufactured (Fig. 2).



Fig. 2. O-ring holes and changing tool.

2.2 Design of storage vault lid rotator

In order to replace the O-ring of the storage vault lid, the storage vault lid must be rotated but it is impossible to operate with MSM because storage vault lid weight is about 125kg (B type). Therefore, a device which can rotate the storage vault lid was required and designed.

The M8 cell in ACPF where this rotating device is to be installed is very narrow. To make 'a' type and 'B' type lids available at the same time, a space ring which has difference of height is created on the bottom of storage vault lid, so that it can be conveniently replaced with a MSM. In addition, the

bottom surface is equipped with a 'slip pad' to prevent the slipping of storage vault lid. In the upper fixture and bolts of the rotating structural part, the hexagonal part was 19mm and vox socket was used a standard product of dodecagon. And it could be made operable with ratchet wrench. The base plate of the rotating device was made rigid so that the lid didn't move when it rotates, and the center of gravity was considered. The rotating part was configured at a worm/screw method and the ratio was about 40:1. The screw was assembled by inserting a thrust bearing and the hex 19mm was machined on the upper/lower part of the screw.

3. Conclusions

In the past, the O-ring inserted in the lid, one of the components of the storage vault was designed without considering the replacement. However, the remote control of the O-ring replacement was required for continued use. The following two parts are improved to give the remote control to the replacement of the O-ring.

- (1) Improvement of SV lid
- (2) Design of storage vault lid rotator and make a dedicated tool

We able to test the remote control of the replacement of the O-ring and replace the O-ring without any difficulty.

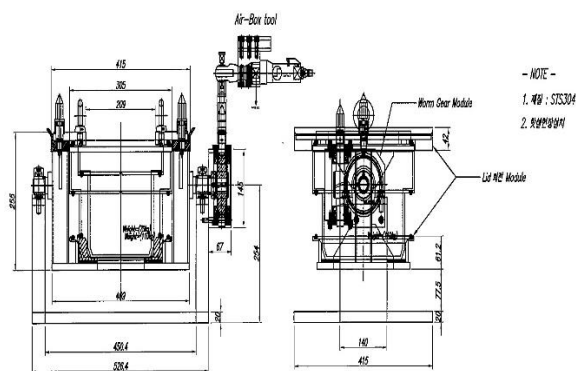


Fig. 3. Drawing of the rotator.

Analysis of Natural Radionuclides Concentration in Jeju Atmospheric PM₁₀ Aerosols During the Period of Asian Dust in 2015

Chung-Hun Han^{1*}, Duk-Woo Kim¹, and Jae-Woo Park²

^{1*}Institute for Nuclear Science and Technology, Jeju National University, 102, Jejudaehak-ro, Jeju-si, Jeju-do, Korea

²Department of Nuclear & Energy Engineering, Jeju National University, 102, Jejudaehak-ro, Jeju-si, Jeju-do, Korea

*tang007@jejunu.ac.kr

1. Introduction

In this study, we have measured, using the ICP-DRC-MS, radioactivity concentrations of ⁴⁰K, ²³²Th and ²³⁸U contained in the atmospheric PM₁₀ aerosols which were collected at the Gosan during the period of Asian Dust in 2015.

2. Materials and methods

2.1 Air sampling locations

Air sampling for aerosols was conducted at the Gosan of Jeju Island, Korea. The samples have been collected using PM₁₀ sampler at 24 hour during the period of Asian Dust in 2015. On collecting samples, the air flow rate was kept to about 16.7 L/min, and total air flow was calculated from the flow rate and running time.

2.2 Sample Analysis

For the analysis of PM₁₀ elemental species, the aerosol samples were decomposed with acids using a microwave digestion system 10 mL acid solution (5.55% HNO₃/16.75% HCl). The vessel was heated at 180°C for 15 minutes with 1000 W microwave radiation to digest the PM₁₀ aerosols [1]. The number of elements determined by ICP-DRC-MS instruments was 3 species such as ³⁹K, ²³²Th and ²³⁸U. For the elemental analysis, the instrumental detection limits (IDL) of ³⁹K, ²³²Th and ²³⁸U were 27.12 µg/L, 2.98 ng/L and 3.23 ng/L, respectively.

The radioactivity concentrations of the isotopes ⁴⁰K, ²³²Th and ²³⁸U in the collected fine dust were calculated using the following Eq. (1) (Ref. 2):

$$A_i = \frac{\ln 2}{T_{1/2}} \times \frac{\rho_i \times m_e}{M_i} \times N \quad (1)$$

where A_i , $T_{1/2}$, ρ_i , m_e , M_i , and N are the radioactivity concentration (Bq/m³), half-life time (s) of isotope i , isotopic ratio (natural abundance) of isotope i , the mass concentration of element e corresponding to isotope i (g/m³), atomic mass (g/mol), and the Avogadro's number, respectively. The isotopic ratios for ⁴⁰K, ²³⁸U and ²³²Th were 0.000117, 0.99275 and 1.0, respectively.

3. Results and discussion

We have measured, using the ICP-DRC-MS, radioactivity concentrations of ⁴⁰K, ²³²Th and ²³⁸U contained in Jeju atmospheric PM₁₀ aerosols during the period of Asian Dust in 2015 (Fig. 1).

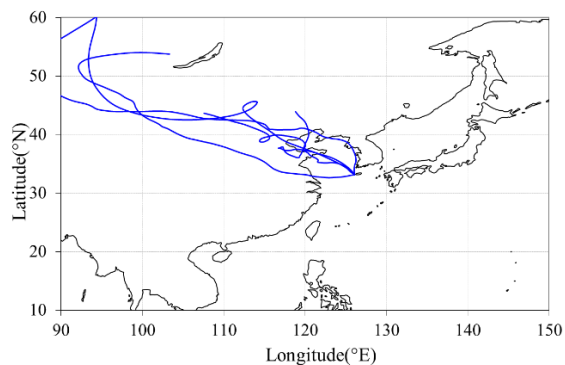


Fig. 1. 5-Day backward trajectories at the Gosan Site of Jeju Island during the study period.

Table 1. The natural radionuclides concentrations in Jeju PM₁₀ aerosols during the period of Asian Dust in 2015

Date	PM ₁₀ (μg/m ³)	⁴⁰ K (pg/m ³)	²³² Th (pg/m ³)	²³⁸ U (pg/m ³)
02-22	198.86	134.60	651.80	165.30
02-23	194.34	167.41	895.93	176.83
03-21	191.37	153.05	336.94	115.82
03-22	124.79	123.28	364.59	101.10
06-12	114.74	95.93	259.51	112.40
Average	164.82 ±41.37	134.85 ±27.57	501.75 ±265.78	134.29 ±34.25

Table 2. The radioactive concentrations in Jeju PM₁₀ aerosols during the period of Asian Dust in 2015

Date	⁴⁰ K (μBq/m ³)	²³² Th (μBq/m ³)	²³⁸ U (μBq/m ³)
02-22	35.65	2.64	2.04
02-23	44.34	3.63	2.18
03-21	40.53	1.37	1.43
03-22	32.65	1.48	1.24
06-12	25.41	1.05	1.38
Average	35.71±7.30	2.03±1.08	1.65±0.42

The mean mass concentration of PM₁₀ was 164.82 μg/m³ (Table 1). During the study period, the mean concentrations of ⁴⁰K, ²³²Th and ²³⁸U were 134.85, 501.75 and 134.29 pg/m³, respectively.

During Asian Dust periods (5 samples), the radioactive concentrations of ⁴⁰K, ²³²Th and ²³⁸U were 35.71, 2.03 and 1.65 μBq/m³, respectively (Table 2). The ²³²Th/²³⁸U activity ratio of Asian Dust was 1.23.

4. Conclusions

The atmospheric PM₁₀ aerosols were collected at Gosan of Jeju Island, which is one of the natural background sites of Korea, during the period of Asian Dust in 2015. This study analyzed using ICP-DRC-MS the concentrations of potassium, thorium and uranium. The mean radioactive concentrations of ⁴⁰K, ²³²Th and ²³⁸U were 35.71, 2.03 and 1.65 μBq/m³,

respectively. The ²³²Th/²³⁸U activity ratio of Asian Dust was 1.23.

Acknowledgements

This research was supported by Basic Science Research Program through the National Research Foundation of Korea (NRF) funded by the Ministry of Education (No. NRF-2010-0020077).

REFERENCES

- [1] A. Mainey and T. William, "Compendium of methods for the determination of inorganic compounds in ambient air (Method IO-3.1; Selection, Preparation and Extraction of Filter Material)". *US Environmental Protection Agency* EPA/625/R-96/010a, 1-27 (1999).
- [2] J. Magill and J. Galy, "Radioactivity, Radionuclides, Radiation." Springer, Berlin (2005).

Enhanced Maintenance of the Supply Piping for L-Ar

*Youngkuk Jang, Seonho Noh, Hui-seok Kang, and Ilje Cho

Korea Atomic Energy Research Institute, 989-111 Daedeokdaero, Yuseong-gu, Daejeon, Korea

*jangyk@kaeri.re.kr

1. Introduction

Korea Atomic Energy Research Institute (KAERI) has a Pyro-processing Integrated Inactive Demonstration facility (PRIDE) for the development of pyro-technology. In essence, the PRIDE enable integrated pyro-systems testing at engineering scales using depleted uranium or surrogates for depleted nuclear fuels. The PRIDE must maintain an inactive (argon) atmosphere due to the characteristics of processes that take place in it, such as electrochemical reduction, electrochemical refining, and electrochemical smelting. In concentrating impurities at the facility, oxygen and moisture must be lower than 50 ppm.

This paper describes the improvement in maintenance of the supply piping of liquid argon(L-Ar) in the PRIDE facility.

2. Main

2.1 Configuration of L-Ar supply piping

The L-Ar supply piping to control impurities in argon is composed of argon tanks – vaporizer – Ar cell.

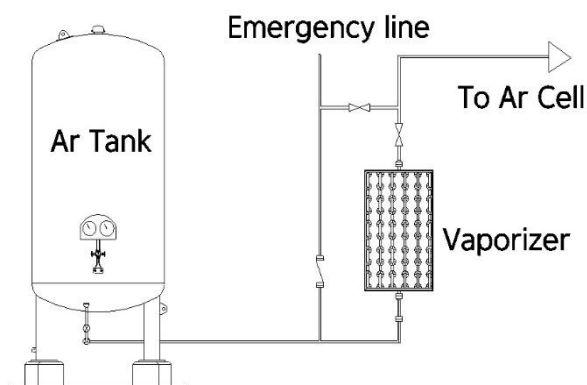


Fig. 1. Schematic diagram of L-Ar supply piping.

2.2 Enhanced maintenance of L-Ar supply piping

2.2.1 Problems. Existing installed L-Ar supply pipes are installed with regular gasket type.

This facility uses liquid argon as needed, and the characteristics of the liquid argon are frequently caused by the degradation of the gasket and deformation of the piping as shown in Fig. 2.



Fig. 2. Damaged gasket and piping.

2.2.2 Contents of improvement. To minimize the flow of oxygen and moisture into argon cell when performing maintenance on argon supply piping.

(1) Installation of additional valve

In order to minimize the flow of oxygen and moisture in the argon cell, the two valves shall be additionally installed in the end of the vaporizer to purge the piping and vaporizers of the vaporizer, as shown in Fig. 3, to minimize the inflow of oxygen and moisture from the argon cell.

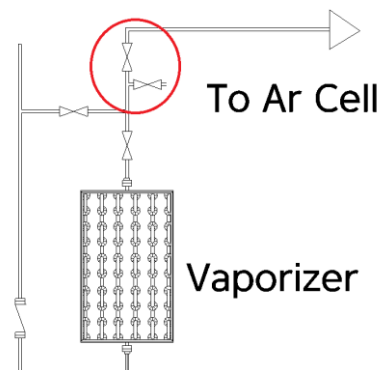


Fig. 3. Installation of additional valves.

(2) Adiabatic work of piping

To cope with rapid changes in temperature, insulate the pipes connecting the argon tank to vaporizer with heat insulation.

(3) Use extremely low temperature gaskets

Replace with the extremely low temperature gaskets for specification in Table 1.

Table 1. Gasket specification

Temperature (°C)	Nominal thickness (mm)	Pressure (MPa)	size (mm)
-200~200	3.0	4.0	1270

3. Conclusion

It is expected to facilitate maintenance of the supply piping without increasing the degree of contamination (oxygen and moisture concentration) of the argon cell, the insulation of the piping between argon tanks and vaporizer of the argon supply piping, and the additional valves installation. It is also expected that he will create a quality inert air test atmosphere in the argon cell.

REFERENCES

- [1] ANL-7959 Hot Fuel Examination Facility/North Facility Safety Report, February 1975, Argonne National Laboratory pp.75-78.
- [2] The EBR-II Fuel Cycle Story, Charles E.Stevenson, American Nuclear Society pp. 42-53.

Study on Nuclear Test Scheme and Test Results for Safety Neutron Measurement System in Research Reactor

Sanghoon Bae*, Young-ki Kim, and Hanju Cha

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989, Yuseong-gu, Daejeon, Republic of Korea

* shbae@kaeri.re.kr

1. Introduction

Nuclear test of safety class neutron measurement system (NMS) in research reactors (RR) has been recognized key data during commissioning stage. Most of test results hardly tell the difference between test result and commissioning outcome. However, several parameters such as linearity and sensitivity need to pay more attention to the test data even though it is not significant. In particular, taking account for the importance of nuclear test, it will be introduced in this paper to verify how the acceptance criteria was determined, if test methodology was suitable, and what the deviation of the detector's sensitivity influenced whole performance.

2. Nuclear Test Setup for Safety NMS in RR

2.1 Nuclear Test Setup for Safety NMS

Nuclear test are involved in the type test sequence for safety NMS in RR which generally proceeds in following order as shown in Fig. 1.

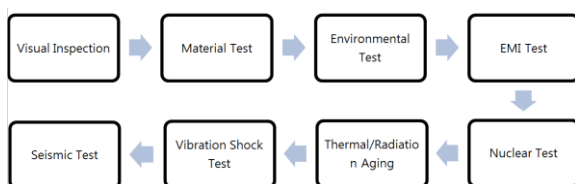


Fig. 1. Type Test Sequence of Safety System.

Before the nuclear test, every process is almost identical to the other case of safety system. Nuclear test was performed with a Wide Range Fission Chamber (WRFC) which is used as a detector of NMS in targeting research reactor. The test specimen is prepared for continual test with being connected via the field cables to the signal processing unit.

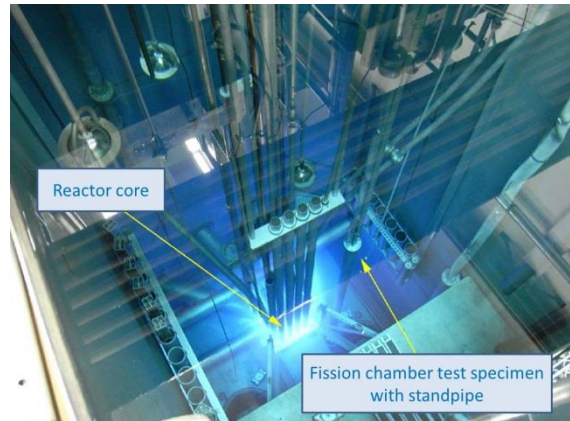


Fig. 2. Nuclear Test Setup of NMS in RR.

The WRFC was mounted electrically isolated in a dedicated movable standpipe in the reactor pool as shown in Fig. 2 with the detector axis parallel to the standpipe axis, ensuring the same vertical position with respect to the reactor core for all measurements. WRFC was placed in the reactor pool at two different locations, referred to position 1 (low neutron flux, far from the reactor core, SE-Pool1) and position 2 (high neutron flux, close to the reactor core, NE-Pool2), to achieve the required neutron fluxes. This scheme was employed to always operate the reactor at power levels above 2.5E-3%FP to be able to stabilize the reactor at the chosen power level and to improve the accuracy of the reactor power measurement by the reference instrumentation.

2.2 First Nuclear Test Result for Safety NMS

Data was obtained at two different positions within the reactor pool, corresponding to reactor power levels of 9.8E-6%FP to 100%FP. In addition to the parameterization, the linearity of the electronics response was tested as a function of the reactor power. Fig. 3 shows the pulse signal, mean square variance signal, and the combined wide range log power signal measured by as a function of the reactor power level.

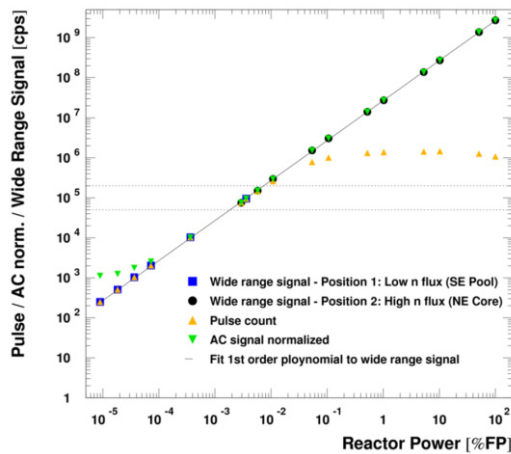


Fig. 3. First Test Result of NMS with curve fitting.

Mean square variance signal is converted AC signal normalized and the dotted lines mark the start and end of the overlap region in which the wide range signal is interpolated between the pulse count and the AC signal normalized. A first order polynomial was fitted to the data obtained with the modified NMS electronics. The residuals of the wide range signal to the 1st order polynomial fit as a function of the reactor power. The error lied within the tolerable range under statistical and systematic uncertainties which is acceptance limit of 1%FS on the logarithmic analog outputs. For the linear neutron power analog output current, calculated value from the measured detector DC current signal with the nominal transfer function of the signal processing electronics met the requirement showing little deviation between the calculated output current and the expected output current corresponding to the reactor power level.

3. Second Test Result for NMS and System Modification

For the first nuclear test which was performed in 2015 evaluated the performance of the fission chamber with the integrated system. The test results of the NMS electronics response and the system integration aspects showed the deviation attributed to the anomalous fission chamber response. The investigation of the test specimen concluded that it was filled with different gas mixture from what it used to be. A re-evaluation of the nuclear data indicated that the DC sensitivity for the linear power signal among the NMS outputs deviated from the specification by a factor 2 to 3. The DC sensitivity of a fission chamber

means the sensitivity of the DC current signal used for linear power signal. At any rate, such the large deviation provides a chance of excess on the inbuilt margins in the NMS and consequently impaired its performance at high neutron fluxes above 5.9×10^9 nv to 1.0×10^{10} nv to the extent of nonconformance to the requirement.

The circuit modification is aimed at the extension of the DC current output, from 2 mA to 3 mA as feasible as it can. This modification allows the full measurement not to be saturated before the maximum level of DC current. The modification of the signal conditioning modules in the NMS constitutes as followings; High voltage module and DC current input module. These modules were modified as shown in Fig. 4 and then tested in the field with the relevant procedure. The nominal output current of the high voltage module was increased from 2 mA to 3 mA, which requires the replacement of concerned components on the modules. At the same time, the input range of the DC current was adjusted from 0-2 to 0-3 mA, in order to fit the current that is supplied by the high voltage module to the fission chamber.

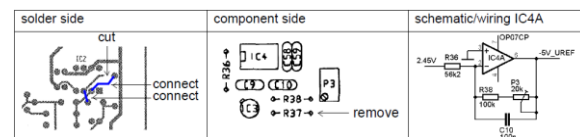


Fig. 4. Modification of NMS Electronics Circuit.

4. Conclusions

Thus far, the issue on nuclear test for the NMS in RR has been summarized with field experience and this practical approach was suggested based on test data. The nuclear test finally ended up the circuit modification and retesting, however it sheds on the light this case should be handled with precarious attention factoring in the severity of the linear power signal.

REFERENCES

- [1] S.S. Kapoor, V.S. Ramamurthy, Nuclear Radiation Detectors. New Age International, (1986) 34.
- [2] C.G. Clayton, Nuclear Geophysics: Selected Papers on Applications of Nuclear Techniques in Mineral Exploration, Mining and Process Control, (1983), 261-263.

Analysis of Off-Site Dose on the Public due to the Discharge of Gaseous Tritium From Wolsong NPPs

Dae-Sun Son*, Jong kuk Lee, Kwan-hee Lee, Sung-il Kim, and Sangmyeon Ahn

Korea Institute of Nuclear Safety, 62, Gwahak-ro, Yuseong-gu, Daejeon, Republic of Korea

*k735sds@kins.re.kr

1. Introduction

Tritium (referred to as H-3) is a radionuclide that is mainly handled in terms of radiation safety management of nuclear power plants (referred to as NPPs) for its high activity. This paper summarizes data of gaseous H-3 from Wolsong NPPs in commercial operation between 2007 and 2016.

KHNP (Korea Hydraulic & Nuclear Power Co.) has operated 6 nuclear power plants in Wolsong site. The type of reactor of Wolsong Unit 1 to 4 is PHWR (Pressurized Heavy Water Reactor), and Shinwolsong Unit 1 and 2 is PWR (Pressurized Water Reactor). The gaseous H-3 that has been discharged from Wolsong NPPs occupied a large portion of the radiation exposure dose to the public. Therefore, it is one of the important concerns for nuclear safety regulation to analyze the radiological hazard of H-3 by evaluating the emission and the dose contribution to the public radiation exposure accordingly.

2. Analysis on discharged H-3 and its dose calculation

2.1 Technical standards and regulatory requirements for radioactive effluents from NPPs

NPPs in commercial operation must be ensured to be operated in compliance with standards and regulatory requirements. It includes environmental protection standards in accordance with relevant law, and technical standards to maintain radiation doses from as low as reasonably achievable (ALARA). Discharge control standard of ventilation on Gaseous H-3 should be under $3.0\text{E}+03 \text{ Bq/m}^3$ at the Exclusion Area Boundary (EAB), and annual dose should be lower than 0.25 mSv/year at EAB in the operation of multi-nuclear reactor facilities at one site [1].

2.2 Generation and properties of H-3

H-3 diffuses into the coolant from the damaged fuel cladding after ternary fission of U-235, and is generated by neutron capture of deuterium or (n, α) reaction of Li-6 (${}^6\text{Li} (n, \alpha) {}^3\text{H}$). H-3 has a half-life of 12.3 years and is converted to a stable nuclide of He-3 by beta decay. The maximum energy of the emitted beta particles is 18.6 keV and the average energy is about 5.7 keV. Since H-3 is a beta-emitting radionuclide with a weak permeation energy, the radiation hazard due to external exposure is negligible, but it is easily absorbed into the human body through oral and skin, which is a major cause of internal exposure.

2.3 Analysis on the emission of gaseous H-3

PHWR generates much more H-3 than PWR because it uses heavy water as a coolant which has high probability of H-3 generation due to neutron reaction. Fig. 1 shows the discharged gaseous H-3 from Wolsong NPPs during the last 10 years (2007 to 2016)[2],[3]. Sinwolsong Unit 1 and 2 which are PWRs started operation in 2012, and it is confirmed that the amount of discharged H-3 is much less than in Wolsong Unit 1 to 4, which are PHWRs.

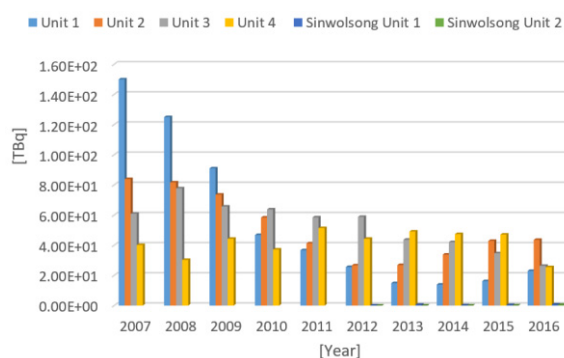


Fig. 1. Amount of discharged gaseous H-3 at Wolsong NPPs per unit.

As part of efforts to reduce the discharge of H-3, KHNP has been operating TRF (Tritium Removal Facility) in Wolsong NPPs since July 2007. As a result, as shown in Fig. 2, the total amount of discharged gaseous H-3 decreased gradually from 3.35E+02 TBq in 2007 to 1.20E+02 TBq in 2016.

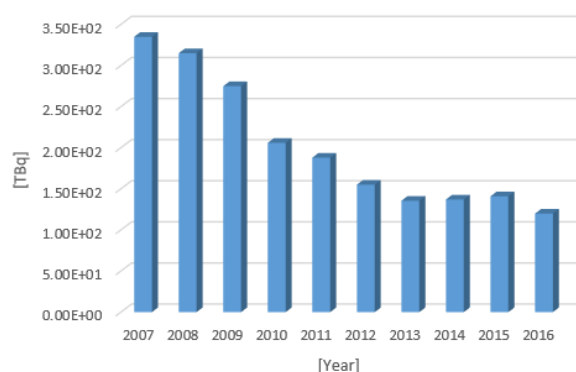


Fig. 2. Total amount of gaseous H-3 discharged from Wolsong NPPs.

2.4 Offsite dose contribution of H-3 to the public

Radioactive gaseous effluents are diffused into the atmosphere first and then they form radioactive clouds and they are deposited on the surface. Thus, the body is exposed externally by the radioactive sources, and internally by respiring the air and digesting intakes of plants and animals contaminated by radioactive materials. The environmental behavior and exposure pathways of gaseous effluents follow the US NRC's Regulatory Guidelines [4], and the concept and system for evaluating the offsite dose conform to the ICRP-60 Recommendation and the IAEA BSS-115. Estimated offsite dose from gaseous H-3 from Wolsong NPPs for the last 10 years (2007 to 2016) is shown in Fig. 3.[5],[6].

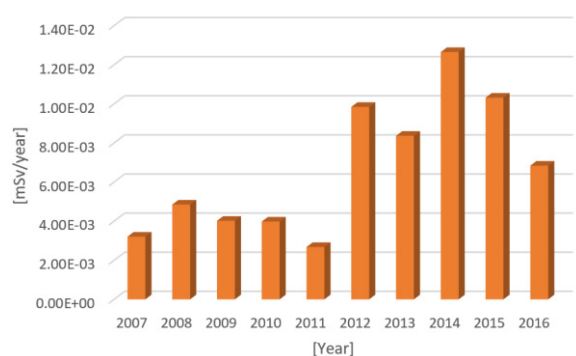


Fig. 3. Estimated offsite dose from gaseous H-3 at Wolsong NPPs.

Although the total gaseous effluents decreased, the offsite dose increased in 2012 more than before. There are a few dominant factors for offsite dose calculation. In this study, change of atmospheric dispersion factor value is one of the major causes of dose increase.

3. Conclusion

In this paper, analysis on discharged gaseous H-3 which is the main radionuclide from the Wolsong site and dose contribution accordingly was conducted during the recent 10 years. Total amount of discharged gaseous H-3 decreased gradually from 3.35E+02 TBq in 2007 to 1.20E+02 TBq in 2016. Although the total gaseous effluents decreased, the offsite dose increased in 2012 more than before. The change of atmospheric dispersion factor value is one of the major causes of dose increase.

4. Acknowledgment

This study is the result of the nuclear safety research project which was funded by the Nuclear Safety Commission and supported by the Korea Nuclear Safety Foundation. (No. 1305004)

REFERENCES

- [1] Nuclear Safety Act.
- [2] The annual report on the environmental radiological surveillance and assessment around the nuclear facility, KHNP, 2007~2016.
- [3] KINS Waste Comprehensive Information Database, 'http://wacid.kins.re.kr'.
- [4] US NRC. R.G.1.109, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR PART50, Appendix. I (1977).
- [5] ICRP60, Recommendations of the International Commission on Radiological Protection (1990).
- [6] IAEA BSS-115, International Basic Safety Standards for Protection Against Ionizing Radiation and for the safety of Radiation Sources (1996).

7분과

방사화학 (Oral)



Liquid-Liquid Extraction Combined With Capillary Electrophoresis for the Determination of Rare Earth Elements in a High Saline Matrix (LiCl-KCl Molten Salt)

Jihye Kim^{a*}, Kihwan Choi^b, and Tae-Hong Park^a

^a Nuclear Chemistry Research Division, Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon, Korea

^b Division of Metrology for Quality of Life, Korea Research Institute of Standards and Science, 267, Gajeong-ro, Yuseong-gu, Daejeon, Korea

*jihyekim21@kaeri.re.kr

1. Introduction

Rare earth elements (REEs) have been widely applied to many fields of geology, pharmacy, and agriculture. The determination of REEs has been reported using various analytical techniques such as inductively coupled plasma optical emission spectrometry (ICP-OES), inductively coupled plasma mass spectrometry (ICP-MS), ion chromatography (IC), capillary electrophoresis (CE). Generally, ICP-MS detection has been described as a powerful technique for the analysis of REEs in environmental samples because of low detection limits and high selectivity. However, the sensitive instrument is not widespread due to the running costs. No matter which analytical technique is adopted, sample pretreatment steps are required before analysis. Especially, the determination of REEs in highly saline sample matrix is difficult due to occurrence of matrix effects.

Liquid-liquid extraction (LLE) as the most common sample pretreatment technique has been used for sample cleanup and preconcentration of REEs. LLE is simple, robust, and compatible with many analytical techniques such as ICP-OES, ICP-MS, and CE. Recently, to analyze the REEs, CE has gained considerable interest owing to fast separation, high separation efficiency, and low sample requirement.

Here, we developed a new method of LLE coupling with CE for the determination of REEs from highly saline sample. The parameters influencing the LLE and CE separation of REEs were investigated. The proposed method was successfully applied to the analysis of REEs in high salts of LiCl-KCl solution.

2. Materials and methods

2.1 Liquid-Liquid extraction

REEs sample solution (donor phase) was prepared in the molten salt (44-56 wt% LiCl-KCl). A 4 mL of donor solution containing REEs was transferred into the vial. A 2 mL of organic phase (0.29 M 2-ethyl hexylphosphate as a carrier in heptane) was added on the top of the donor solution. Extraction was done by vortexing vigorously for 10 min. After extraction, the phases were separated by centrifugation at 3000 g for 5 min and the donor phase was discarded by a pipet. To decrease the complexation of the REEs and carriers, 2 mL of octanol was added into the organic phase (extract). Back extraction was achieved with 2 mL of 0.1 M HNO₃. In the process of back extraction, the sample was mixed for 10 min and then separated by centrifugation at 3000 g for 5 min. After centrifugation, two organic phases were discarded and the aqueous phase was removed using a pipet. After extraction, the aqueous phase was transferred into a 200 μ L vial and introduced into the capillary. The LLE procedure is shown in Fig. 1.

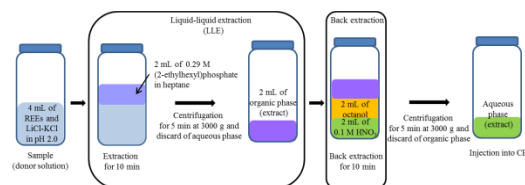


Fig. 1. LLE procedures for the determination of REEs from highly saline sample.

2.2 Electrophoretic conditions

CE was performed on a CESI 8000 system from Beckman (Fullerton, CA, USA) (Fig. 2) with

preprogrammed Karat software. Analytes were monitored using a UV detector at 214 nm. The run buffer was 10 mM 4-methylbenzylamine (4-MB), 4 mM 2-hydroxyisobutyric acid (HIBA), 0.4 mM malonic acid, and 0.1% hydroxypropyl cellulose (HPC), adjusted to pH 4.8 with 2-ethylbutyric acid. Electrophoresis was carried out with a voltage of 25 kV. The capillary cartridge temperature was set to 25°C. A new fused silica capillary of 50 µm ID and 360 µm OD and effective/total length of 50/60 cm (Polymicro Technologies, Phoenix, AZ, USA) was washed with 0.1 M NaOH for 5 min, water for 5 min, and a run buffer for 5 min, each at 50 psi. Between runs, the capillary was treated by rinsing it with 0.1M NaOH for 2 min, water for 2 min, and the run buffer for 3 min at 50 psi.

4-MB was used as a UV absorbing probe for indirect UV detection. HPC was employed as EOF modifier. HIBA and malonic acid acted as a complexing agents for sufficient selectivity of REEs.

2.3 Instrument



Fig. 2. CESI 8000 system from Beckman.

2.4 Principle of capillary electrophoresis

Capillary electrophoresis is an analytical technique that separates ions based on their electrophoretic mobility with the use of an applied voltage. The electrophoretic mobility of analyte is dependent upon the charge of the molecule, the viscosity, and the atom's radius.

3. Results and discussion

3.1 Comparison of standard sample and REEs extract after LLE in capillary zone electrophoresis mode

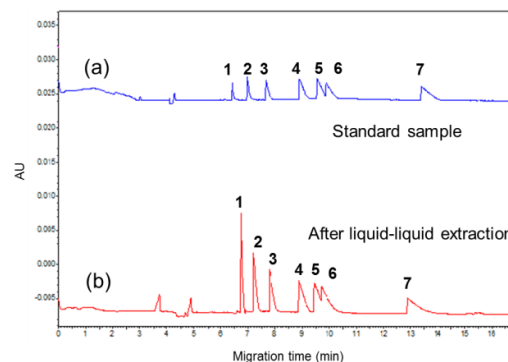


Fig. 3. Electropherogram of (a) the standard sample (b) sample extract by LLE. Sample solution was prepared in 44-56 wt% LiCl-KCl solution. Peak identification: (1) La, (2) Ce, (3) Nd, (4) Sm, (5) Eu, (6) Gd, (7) Er.

Fig. 3 described the sample cleanup was achieved by liquid-liquid extraction from highly saline sample. When the only LiCl-KCl solution was injected hydrodynamically at 0.5 psi for 5 s into the CE, Li^+ ion and K^+ ion were detected within 4 min with same CE conditions. However, after LLE, those ions were not detected in CE electropherogram. In fact, high amounts of Li^+ and K^+ ions removed from the sample matrix during LLE because they did not complex with a carrier (2-ethyl hexylphosphate).

4. Conclusions

LLE coupled to CE has been proposed for the determination of REEs in highly saline sample (44-56 wt% LiCl-KCl). The advantages of the proposed method are good extraction efficiency (70-98%) and efficient separation with relatively low running costs. Furthermore, the proposed method is applicable to determination of trace amounts of REEs in spent fuel cell containing molten LiCl-KCl salt.

5. References

- [1] C.W. Noack, D. A. Dzombak, and A.K. Karamalidis "Determination of rare earth elements in hypersaline solutions using low-volume, liquid-liquid extraction", *Environmental Science & Technology* 49, 9423-9430 (2015).
- [2] Z. Xu, K. Kawahito, X. Ye, A. R. Timerbaev, T. Hirokawa "Electrokinetic supercharging with a system-induced terminator and an optimized capillary versus electrode configuration for parts-per-trillion detection of rare-earth elements in CZE", *Electrophoresis*, 32, 1195-1200 (2011).

Development of Spectrophotometric On-Line Monitoring System for Groundwater Near Silo Repository

Tae-Hyeong Kim^{1*}, Sol-Chan Han², Jong-Il Yun², and Hong Joo Ahn¹

¹Nuclear Chemistry Research Division, Korea Atomic Energy Research Institute, 111 Daedeok-daero-989, Yuseong-gu, Daejeon, Republic of Korea

²Department of Nuclear and Quantum Engineering, KAIST, 291 Daehak-ro, Yuseong-gu, Daejeon, Republic of Korea

*thkim@kaeri.re.kr

1. Introduction

Low- and intermediate-level radioactive wastes (LILW) are currently being stored in silo type repositories in Gyeongju. In order to prevent the release of radionuclide to the environment, a multi-barrier system, including natural geological barrier, and an engineered barrier system (EBS), has been designed. Since radionuclide migration from the repository to the geosphere occurs through the groundwater, monitoring of the groundwater system is necessary to evaluate the performance of disposal system after the closure.

For that purpose, we developed an on-line monitoring system based on the UV-Vis absorption spectrometry coupled with liquid waveguide capillary cell (LWCC) in order to detect iron, cobalt, and nickel ions, which are short half-life radionuclides existing in the groundwater.

2. Experimental

2.1 Performance assessment model

The performance assessment model for LILW repository was developed using the GoldSim (Ver.11) of the Monte Carlo based program. The silo repository and trench type repository, which make up the LILW, were developed in separated sub-modules. Details applied in the model are identical to the previous work [1].

2.2 UV-Vis absorption spectrophotometry

All absorption spectra was measured with the UV-Vis absorption spectrometer (MCS601 UV-NIR-C, Carl Zeiss) coupled with LWCC (LWCC-4100, World Precision Instruments). 2-(5-bromo-2pyridylazo)-5-(N-propyl-N-3-sulfopropylamino)phenol (Br-PAPS) (Sigma-Aldrich) was used as chelating reagent. Solvent degassing unit (AG-32, Gastorr) was applied to remove existing micro-bubbles, which may interfered with the signal.

3. Result and Discussion

Sorption is the main retardation process for the transport of radionuclide from the repository to geosphere and is quantified by the distribution coefficient (K_d). The distribution coefficient is strongly dependent on the geochemical conditions, such as pH, temperature, composition of soils and etc [2].

The annual dose rate was calculated with different distribution coefficient in two scenarios as seen in Fig. 1: normal and abnormal.

In normal scenario, it is hypothesized conservatively that the silo is fully saturated with groundwater and waste drum loses its credibility right after the closure. While, in abnormal scenario, it is hypothesized that the distribution coefficients are considered to be decreased by one thousandth. As represented in Fig. 1, in abnormal scenario, the

annual dose rate becomes dominated by short half-life radionuclides such as ^{55}Fe , ^{60}Co , ^{63}Ni , ^{90}Sr , and ^{137}Cs , unlike the normal scenario.

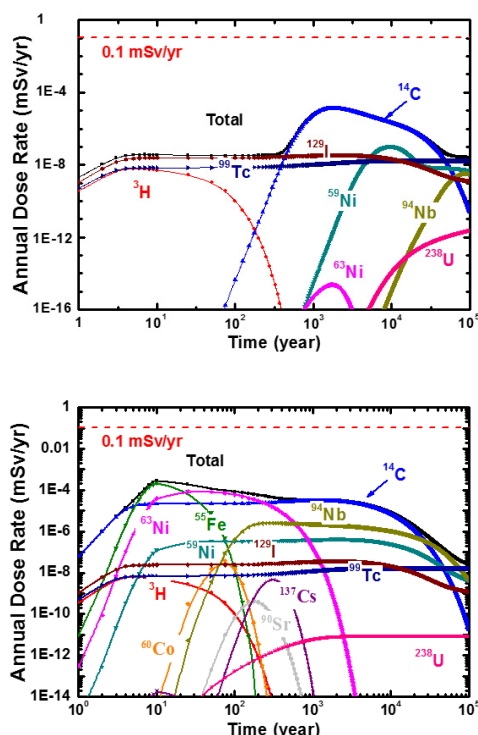


Fig. 1. Annual dose rate (a) normal scenario (top) and (b) abnormal scenario (bottom).

The absorbance of metal Br-PAPS chelate was measured in the metal concentration range from 10^{-9} mol/L to 10^{-8} mol/L, which is equivalent to approximately 0.05 ppb to 0.5 ppb, respectively, as shown in Fig. 2. The analytical absorbance was evaluated at its second maximum position after background subtraction and detection limits for iron, cobalt and nickel were determined using 3 rule as 0.021 ppb, 0.013 ppb, and 0.008 ppb, respectively.

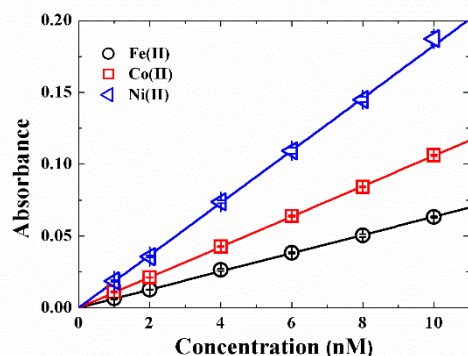


Fig. 2. Calibration curves of metal Br-PAPS chelates ([Fe, Co, Ni] = 10^{-7} M, [Br-PAPS] = 2.5×10^{-7} M, pH=9.0).

4. Conclusions

According to the results from the safe assessment model, short half-life radionuclides become dominated when the distribution coefficient decreases. In this case, the on-line monitoring system developed in this study can be applied to detect such metal ions to evaluate the performance assessment of the repository.

5. Acknowledgement

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. 1305032).

REFERENCES

- [1] S. C. Han and J.-I. Yun, "Sensitivity Analysis for Major Input Parameters in LILW Repository Safety Assessment Model", Proc. Of the KRS 2016 Autumn Conference, 14(2), October 12-14, 2016, Jeju.
- [2] A. Poteri, H. Nordman, V.-M. Pulkkanen and P. Smith, "Radionuclide Transport in the Repository Near-Field and Far-Field", Posiva Report 2014-02, Posiva Oy, Helsinki, Finland, 2014.

Kinetic and Mechanistic Studies on Phosphoester Cleavage by Tetravalent Uranium Species

Hyejin Cho* and Wansik Cha

Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon, Republic of Korea

*hcho921@kaeri.re.kr

1. Introduction

In deep groundwater systems, tetravalent uranium (U(IV)) is a dominant redox form of uranium because of an anaerobic condition. Thus, it is very important to understand the behaviors of U(IV) interacting with organic molecules in order to assess the geological migration of actinide species in the groundwater environment [1].

Although many lanthanide and transition metal cations have been investigated as catalysts for the hydrolysis of phosphoesters, soluble and insoluble U(IV) species has not. This prompted us to try to generate the series of U(IV) species to explore their catalytic activities and hydrolytic mechanisms.

2. Results and Discussions

The rate constants of phosphoester hydrolyses catalyzed by U(IV) species in aqueous solutions were determined using UV-vis spectroscopy. A variety of forms of U(IV) species ($U^{4+}(aq)$, $UO_2(cr)$ -nanoparticles (NPs), U(IV)-oxalate(ox)₂, U(IV)-NPs-polyvinylpyrrolidone (PVP)) were prepared to assess their catalytic activity toward p-nitrophenyl phosphoester (NPP) cleavage at a wide range of pH (3-10).

In order to quantitatively evaluate U(IV) relative to Ce(IV) promoters of phosphoester cleavage, we have determined the reactivity of U(IV) toward NPP and additional substrates (bis(p-nitrophenyl) phosphate (BNPP), O,O-Dimethyl O-(4-nitrophenyl) phosphate (paraoxone-methyl)).

Substrates were hydrolysed readily in the present U-

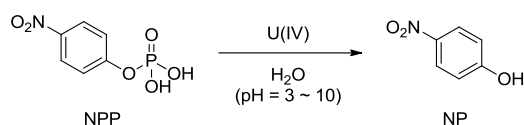


Fig. 1. Hydrolysis of NPP catalyzed by U(IV).

(IV) catalysts. Reaction conditions comprised 50 μ M substrates and 100 μ M U(IV) catalysts [2] in 0.1 M $NaClO_4$ solution at pH 3-10 at both 25 and 50 $^{\circ}C$.

We employed a 2-fold excess of catalysts over substrates which is much smaller amount relative to previous papers [3-5]. In the previous works, a 20-fold excess of catalysts was added to speed up the hydrolysis. The high concentration of metal catalyst, however, causes the precipitation of catalyst at high pH ranges.

Kinetics were followed spectrophotometrically, monitoring the disappearance of the substrates (290 nm) and the appearance of p-nitrophenol (NP) (317 nm) or p-nitrophenolate (400 nm). Rate constants, k for NPP hydrolyses mediated by U(IV) species, appear in Table 1, where kinetic data for comparable hydrolyses catalyzed by cerium(IV) ammonium nitrate (Ce(IV)-AN) are also collected.

Table 1. Kinetics of the U(IV) and Ce(IV) catalyzed hydrolysis of NPP

Catalyst	T ($^{\circ}C$)	$k_{obs}/10^{-4} s^{-1}$			
		pH 3	pH 5	pH 7	pH 9
U(IV)-NPs	25	1.18	1.38	1.72	0.761
	50	30.8	35.5	27.4	2.22
U^{4+}	25	0.71	0.61	0.50	0.273
	50	30.8	35.5	27.4	2.22
U(IV)-NPs-PVP	25	1.05	1.38	1.52	0.982
	50	28.3	34.6	35.4	2.40
U(IV)- ox_2	25	0	0	0.463	0.227
	50	0	0	1.50	0.68
Ce(IV)-AN	25	15.3	23.8	19.0	16.1
	50	89.4	128	102	89.4

It is revealed that U(IV) catalysts mediate the cleavage of substrates is at least $ca.10^5$ times greater

than the for the uncatalyzed reaction ($k_{\text{obs}} = 3.3 \times 10^{-8} \text{ s}^{-1}$). Moreover, the U(IV) catalyzed hydrolysis is also *ca.* 3 - 4 times faster than Eu^{3+} and Tm^{3+} catalyzed cleavage in the similar conditions. Interestingly, U(IV)-NPs and U(IV)-NPs-PVP are comparable to Ce(IV), and they are generally more active than $\text{U}^{4+}(\text{aq})$ and U(IV)-ox_2 . Such high catalytic activity can be explained by the solid-phase reaction mechanism. The reaction seems to undergo via strong binding of substrates to the surface of nanoparticles which possess the hydroxide ions on the surface.

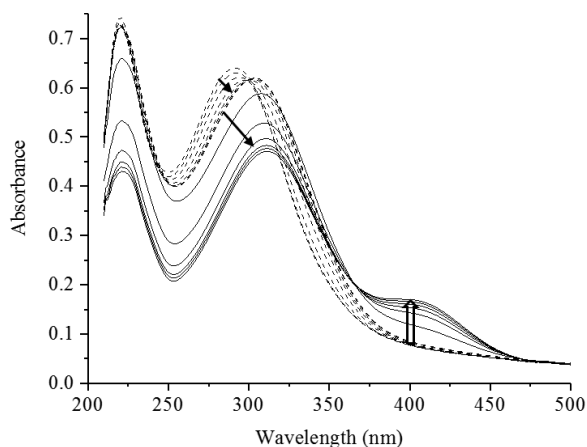


Fig. 2. UV-Vis absorption spectra of the NPP hydrolysis catalyzed by U(IV)-NPs at 50 °C monitored for 600 s (dash lines : 2-30 s, solid lines : 100-600 s).

In fact, formation of metal-phosphoester complexes occurs at the initial stage of the hydrolysis reactions, which is more evident at higher pHs. The UV-Vis absorption spectra in Fig. 2 shows that the U(IV)-NPP complexes are likely produced in the very beginning of the reaction, in 30 s (dash lines). Such complexation allows further increasing of the efficiency of the hydrolysis and suggests the possibility of two-step reaction mechanism: the formation of the complex (fast) and the cleavage of phosphoester bond (rate determining step).

3. Conclusion

Various U(IV) species were prepared to study the reaction kinetics and mechanism of the phosphoester cleavage. The prepared U(IV) species mediate the hydrolysis of NPP in aqueous solutions with rate enhancements compared to the uncatalyzed reactions. Interestingly, U(IV)-NPs is revealed as the most efficient catalyst for the hydrolysis of NPP among

the other forms of U(IV) catalysts at pH 3-10. The proposed mechanism in this study concludes that these catalytic reactions are mediated by the solid phases or NPs dispersed in the reaction medium.

ACKNOWLEDGMENTS

This study is supported by the Nuclear Research and Development program of the National Research Foundation of Korea (Grant code: 2017M2A8A-5014719)

REFERENCES

- [1] J.-I. KIM, "Significance of Actinide Chemistry for the Long-term Safety of Waste Disposal", Nuclear Engineering and Technology, 38, 459 (2006).
- [2] W. Cha, S. T. Kim, D. B. Seo, H.-R. Cho and E.C. Jung, "Formation Mechanism and Characteristics of U(IV) Nanoparticles Under Anaerobic Aqueous Conditions", Mid-Term ThermAc Workshop, December 1-2, Dresden, Germany (2016).
- [3] K. O. Aileen Chin and J. R. Morrow, "RNA Cleavage and Phosphate Diester Transesterification by Encapsulated Lanthanide Ions: Traversing the Lanthanide Series with Lanthanum(III), Europium(III), and Lutetium(III) Complexes of 1,4,7,10-Tetrakis(2-hydroxyalkyl)-1,4,7,10-tetraazacyclo dodecane", Inorganic Chemistry, 33, 5036 (1994).
- [4] K. Bracken, R. A. Moss and K. G. Ragunathan, "Remarkably Rapid Cleavage of a Model phosphodiester by Complexed Ceric Ions in Aqueous Micellar Solutions", Journal of the American Chemical Society, 119, 9323 (1997).
- [5] H. Schneider, J. Rammo and R. Hettich, "Catalysis of the Hydrolysis of Phosphoric Acid Diesters by Lanthanide ions and the Influence of Ligands", Angewandte Chemie International Edition, 32, 1716 (1993).

UV-Vis Absorption Properties of Uranyl-Nicotinic Acid Complexes in Aqueous Medium

Seonggyu Choi and Jong-Il Yun*

Department of Nuclear and Quantum Engineering, KAIST, 291, Daehak-ro, Yuseong-gu, Daejeon, Korea

*jiyun@kaist.ac.kr

1. Introduction

The safety of a disposal facility for high level radioactive waste (HLW) is strongly dependent on the migration of released actinides through the natural water, such as groundwater and seawater. Therefore, a detailed understanding of the complexation reaction between actinides and natural organic matters (NOM) is required for the reliable safety assessment of the disposal facility since additional complexation leads to an increase in the solubility of actinides. Humic acids and fulvic acids are principal components of NOM, and those organic acids have carboxylic functionalities in their chemical structure. In addition, it was found that the organic acids also contain a small amount of nitrogen as amine functionalities or heterocyclic compounds arisen from humification [1]. In this study, nicotinic acid was selected as a representative simple N-modified model ligand of NOM, and its complexation with uranyl (UO_2^{2+}) was observed by means of UV-Vis absorption spectrophotometry.

2. Experimental

2.1 Chemical reagents

All sample preparations and experiments were conducted under ambient aerobic condition at a temperature of $23 \pm 2^\circ\text{C}$. Aqueous solutions were prepared using deionized water (Millipore, Milli-Q/RiOs). Perchloric acid solutions were prepared by diluting 70wt% perchloric acid (Sigma-Aldrich, ACS reagent grade) with deionized water. Uranyl stock solutions were prepared by dissolving $\text{UO}_2(\text{NO}_3)_2 \cdot 6\text{H}_2\text{O}$ (Fluka, 98.0~102.0%) in perchloric acid solution, and nicotinic acid stock solutions were prepared by dissolving powdered nicotinic acid (Sigma-Aldrich, $\geq 99.5\%$) in deionized water. The pH and ionic strength of samples were adjusted using perchloric acid solution, dissolved

sodium hydroxide solution (Sigma-Aldrich, 99.99%), and dissolved sodium perchlorate solution (Sigma-Aldrich, ACS reagent grade).

2.2 UV-Vis absorption measurement

The UV-Vis absorption measurements were performed using a ZEISS MCS601 UV-NIR C spectrometer, a ZEISS CLH600 light source, and Hellma QS quartz cuvettes with 1.00 cm pathlength.

3. Results and discussion

3.1 Hydrolysis behavior of U(VI)

The hydrolysis behavior of U(VI) was assessed as a function of pH in order to exclude hydrolyzed uranyl species in uranyl-nicotinic acid complexation experiments. The species distribution of U(VI) was calculated using the OECD NEA-TDB and the specific ion interaction theory (SIT), and the corresponding result is given in Fig. 1.

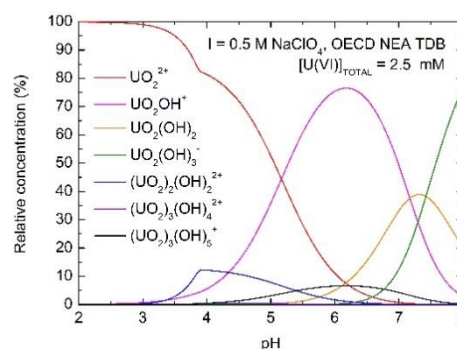


Fig. 1. Species distribution of U(VI) as a function of pH.

The species distribution shows that the hydrolyzed species are occupied less than 1% of 2.5 mM total concentration of U(VI) at pH less than 3.0, and $(\text{UO}_2)_2(\text{OH})_2^{2+}$ is the most predominant hydrolyzed species in mild acidic condition.

This U(VI) hydrolysis behavior was also verified by means of UV-Vis absorption spectrophotometry.

Fig. 2 shows the UV-Vis absorption spectra of U(VI) at different pH values. The molar absorptivity and absorption bands of U(VI) did not change at pH up to 2.9. However, the molar absorptivity was increased and the UV-Vis absorption bands showed a red-shift tendency from pH 3.5. The change in the UV-Vis absorption properties is well adapted to the UV-Vis absorption properties of UO_2^{2+} and $(\text{UO}_2)_2(\text{OH})_2^{2+}$ species reported in [2].

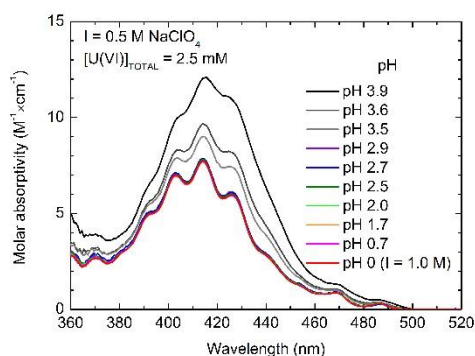


Fig. 2. UV-Vis absorption of U(VI) at different pH.

3.2 UV-Vis absorption properties of uranyl-nicotinic acid complexes

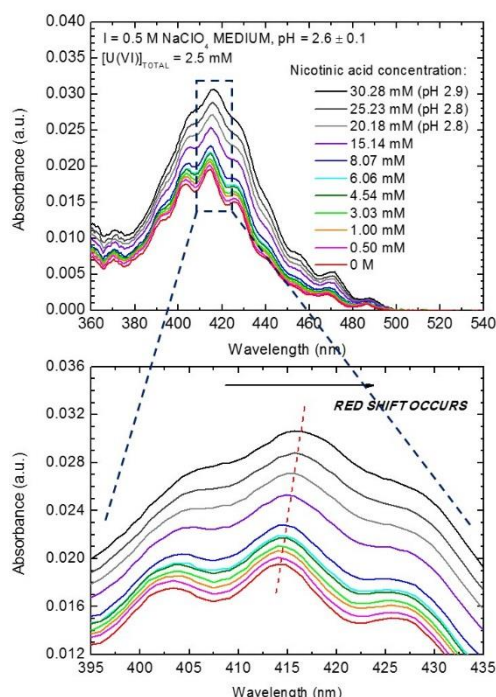


Fig. 3. UV-Vis absorption of uranyl-nicotinic acid complexes.

The uranyl-nicotinic acid complexation experiments were conducted with a constant total concentration of U(VI) at 2.5 mmol/L and various concentrations of

nicotinic acid. Fig. 3 shows the measured UV-Vis absorption of U(VI) as the concentration of nicotinic acid increases. The pH was controlled not to exceed 3.0, so the hydrolyzed species were surely excluded. The UV-Vis absorbance of U(VI) was gradually enhanced and the absorption bands exhibited the red-shift tendency by 2 nm when the concentration of nicotinic acid increases from 0 to 30.28 mmol/L. Since there was no significant UV-Vis absorption of nicotinic acid observed in the wavelength range 360~520 nm, these UV-Vis absorption changes indicate the complexation between uranyl and nicotinic acid.

4. Conclusion

The hydrolysis of U(VI) was evaluated by calculation using both reported thermochemical data and experimental method, and the complexation between uranyl and nicotinic acid, which is one of the major simple N-modified model ligands of NOM, was observed by means of UV-Vis absorption spectrophotometry.

ACKNOWLEDGEMENT

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-2017M2A8A-5014801)

REFERENCES

- [1] A. Vairavamurthy, S. Wang, "Organic Nitrogen in Geomacromolecules: Insights on Speciation and Transformation with K-edge XANES Spectroscopy", *Environ. Sci. Technol.*, 36, 3050-3056 (2002).
- [2] G. Meinrath, "Aquatic Chemistry of Uranium: A Review Focusing on Aspects of Environmental Chemistry", *Freiberg On-line Geoscience Vol. 1*, TU Bergakademie Freiberg (1998).

Variation of Compositions at the Interface Between $\text{U}_{0.97}\text{Th}_{0.03}\text{O}_2$ and Zr From High-Temperature Annealing

Young-Sang Youn^{1*}, Jeongmook Lee¹, Jandee Kim¹, and Jong-Yun Kim^{1,2}

¹Nuclear Chemistry Research Division, Korea Atomic Energy Research Institute: 111,
Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon 34057, Republic of Korea.

²Radiochemistry & Nuclear Nonproliferation, University of Science & Technology, Gajeong-ro 217,
Yuseong-gu, Daejeon, 34113, Republic of Korea

*ysyoun@kaeri.re.kr

1. Introduction

The study on the interaction between uranium dioxide and zirconium has been intensively performed to understand the interfacial phenomenon between oxide fuel and Zircaloy cladding in fuel rods [1-5]. Under actual irradiation, a variety of fission products were produced, and they influenced the chemical properties of uranium dioxide. Therefore, the interaction between fission product doped uranium dioxide and zirconium has to be investigated to confirm the effect of fission product on their interaction. Among various fission products, we selected thorium as a representative with +4 oxidation state in actinide elements.

Herein, we will describe the variation of compositions at the interface between thorium doped uranium dioxide and zirconium from annealing at 1700°C for 18 h using X-ray photoelectron spectroscopy (XPS), X-ray diffraction (XRD), and Raman spectroscopy.

2. Experimental Details

Pellets of thorium doped uranium dioxide, $\text{U}_{0.97}\text{Th}_{0.03}\text{O}_2$, were fabricated using UO_2 and ThO_2 powders. In addition, Zr samples were produced by cutting a Zr metal rod. The $\text{U}_{0.97}\text{Th}_{0.03}\text{O}_2$ -Zr system was heated at 1700°C for 18 h in an alumina tube furnace under H_2 condition. After the annealing, the interface between $\text{U}_{0.97}\text{Th}_{0.03}\text{O}_2$ and Zr samples was measured using XPS, XRD, and Raman spectroscopy.

XPS data were recorded in a high vacuum chamber installed using a VG Scientific ESCALAB 220i-XL system with an Al K_α X-ray source (1486.6 eV) at a base pressure below 1.0×10^{-8} Torr. The binding energies of all measured spectra were calibrated relative to that of adventitious C 1s (284.6 eV).

XRD spectra were collected in the range of 20-120° with a step size of 0.02° for 0.2 s using a Bruker D8 ADVANCE system with a Cu K_α X-ray source.

Raman results were obtained using a ANDOR shamrock SR500i Raman spectrometer with a helium-neon laser of 632.8 nm wavelength operating at 8 mW. The spectra were acquired in the wavenumber range of 400 to 1200 cm^{-1} .

3. Results

In XRD and Raman spectra obtained at room temperature, the peaks corresponded to those of cubic UO_2 crystal structure. In addition, XPS data acquired at room temperature exhibited that the surface of thorium doped uranium dioxide was oxidized to hyperstoichiometric $\text{U}_{0.97}\text{Th}_{0.03}\text{O}_{2+x}$ containing U(IV), U(V), U(VI), and Th(IV). In Zr sample at room temperature, the diffraction patterns of Zr metal and ZrO_2 were displayed in XRD data, and a ZrO_2 peak was obtained in the XPS spectrum.

After the annealing of $\text{U}_{0.97}\text{Th}_{0.03}\text{O}_2$ -Zr system at 1700°C for 18 h, XRD and Raman data obtained from the $\text{U}_{0.97}\text{Th}_{0.03}\text{O}_2$ pellet were similar to those before the annealing, while XPS spectrum of the

$U_{0.97}Th_{0.03}O_2$ pellet showed signals related to U(IV) and Th(IV). In the case of Zr sample after annealing, the peaks of ZrO_2 were obtained in XRD and Raman spectra.

“Temperature-dependent variations of the interface between UO_2 and Zr”, Bulletin of the Korean Chemical Society, 36, 2068-2072 (2015).

4. Conclusion

Change of compositions at the interface between $U_{0.97}Th_{0.03}O_2$ and Zr before and after the annealing was researched using XPS, XRD, and Raman spectroscopy. The $U_{0.97}Th_{0.03}O_2$ and Zr samples at room temperature were oxidized to hyperstoichiometric $U_{0.97}Th_{0.03}O_{2+x}$ and ZrO_2 with a small amount of Zr metal, respectively. After the annealing of the $U_{0.97}Th_{0.03}O_{2+x}$ -Zr system, the reduction of uranium and zirconium in each sample occurred in our experiments.

ACKNOWLEDGMENTS

This research was supported by the National Research Foundation of Korea (NRF) funded by the Korean government (MSIP) (grant No. 2017M2A8A-5014754).

REFERENCES

- [1] P. Hofmann, and C. Politis, “The kinetics of the uranium dioxide-Zircaloy reactions at high temperatures”, Journal of Nuclear Materials, 87, 375-397 (1979).
- [2] D. R. Olander, “The UO_2 -Zircaloy chemical interaction”, Journal of Nuclear Materials, 115, 271-285 (1983).
- [3] M. Miyake, “A study of the reaction between UO_2 and Zr”, Journal of Nuclear Materials, 154, 123-127 (1988).
- [4] P. Hofmann, “Current knowledge on core degradation phenomena, a review”, Journal of Nuclear Materials, 270, 194-211 (1999).
- [5] Y.-S. Youn, J.-G. Kim, S. D. Park, and Y.-K. Ha,

Electrochemical Behaviors of Nickel Hexacyanoferrate-Graphene Oxide Nanocomposite for Cesium Ion Separation

Dongchul Choi*, Sang-Eun Bae, and Tae-Hong Park

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

*cdc777@kaeri.re.kr

1. Introduction

Recently, environmental concerns related to nuclear power energy are one of the most important issues. After the Fukushima Daiichi nuclear power plant accident, the oceans and soils were severely contaminated with radionuclides including radioactive cesium (^{137}Cs). A great deal of efforts have been ongoing to remove the radionuclides [1]. Especially, the removal of ^{137}Cs from contaminated environments has become more highlighted since the Fukushima nuclear disaster [2]. In this study, we fabricated an electrode with nickel hexacyanoferrate-graphene oxide (NiPB-GO) nanocomposites and improved ion selectivity and long term stability for electrochemical cesium ion separation over a pristine NiPB electrode.

2. Experimental

2.7 g of $\text{Ni}(\text{OAc})_2 \cdot 4\text{H}_2\text{O}$ was dissolved in a mixture of 175 mL of H_2O and 25 mL of DMF. (Solution A). 4.8 g of $\text{Na}_4[\text{Fe}(\text{CN})_6 \cdot 10\text{H}_2\text{O}]$ and 7 g of NaCl were dissolved in 175 mL of H_2O (Solution B). Solution A was added dropwise to Solution B with stirring at room temperature. The precipitates were separated by centrifugation and then washed with MeOH. The greenish solid was dried in an oven at 60°C for overnight. Finally, nano sized NiPB was obtained.

9.6 mg of NiPB and 0.4 mg of GO (manufacture?) were suspended into a 0.01wt % poly vinyl alcohol aqueous solution. The mixture was irradiated by ultrasonic wave for 10 min. A NiPB-GO nanocomposite modified indium tin oxide (ITO) electrode (10×30 mm) was prepared by drop-casting the NiPB-GO nanocomposite solution (0.2 mL, 0.5 mg/mL). A NiPB-modified ITO electrode as a control was also prepared by the same method. All

electrochemical experiments were performed in a three-electrode system using a potentiostat/galvanostat: the NiPB-GO modified ITO Ag/AgCl, and platinum mesh electrodes were used as working, reference and counter electrodes respectively. The morphology of NiPB, NiPB-GO nanocomposite was observed using a field-emission scanning microscopy (FESEM: JEOL JSM-7000F). The X-ray diffraction patterns were obtained by Bruker D8 advance.

Table 1. Conditions of electrochemical experiment

Working electrode	NiPBGO/ITO
Reference electrode	Ag/AgCl
Counter electrode	Platinum mesh
Potential window	0 – 1 V
Scan rate	5 mV/s
Electrolyte	1 M LiNO_3 , 1M NaNO_3 , and 1 M CsNO_3

3. Results and Discussion

3.1 Confirmation of NiPB and NIPBGO

The NiPB-GO nanocomposite was characterized by Raman spectroscopy (Raman), powder X-ray diffraction (XRD), and scanning electron microscopy (SEM) analyses. In the NiPB-GO nanocomposite, two prominent D and G bands from GO were observed at 1321 and 1590 cm^{-1} , respectively. The XRD analysis of NiPB-GO shows peaks at $2\theta = 17.4, 24.6, 34.9, 38.4, 42.4, 49.1, 53.6$, and 56.8° , which were indexed as the face-centered cubic structure of NiPB with a lattice of $10.28\text{\AA}$. The morphologies of NiPB and NiPB-GO nanocomposite are observed by FESEM. NiPB shows an aggregated form of spherical shape with an average diameter of 18 nm

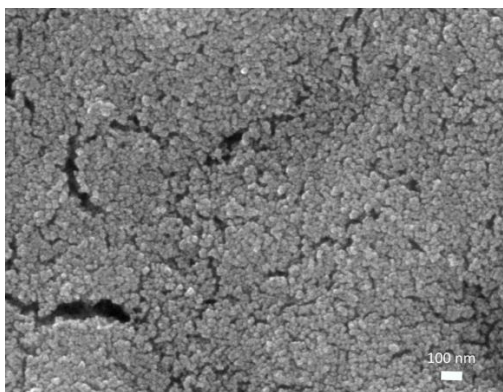


Fig. 1. SEM image of NiPB-GO on ITO electrode.

3.2 Electrochemical behaviors of NiPB and NIPBGO

The NiPB modified ITO electrode shows different redox peak potentials in various alkali metal electrolytes. However the intercalation onset potential of Cs^+ was very close to that of Na^+ . Fig. 2 shows the CVs of the NiPB-GO nanocomposite modified ITO electrode in 1 M CsNO_3 . The redox reaction peak potentials were well separated as observed with the NiPB modified electrode. The intercalation onset potential of Li^+ , Na^+ and Cs^+ in NiPB-GO nanocomposite appeared at 0.524, 0.611 and 0.711 V respectively. More positive potential of the intercalation onset of Cs^+ over the other alkali cations indicates that the NiPB-GO modified electrode improves the selectivity for Cs^+ compared to the pristine NiPB electrode [3].

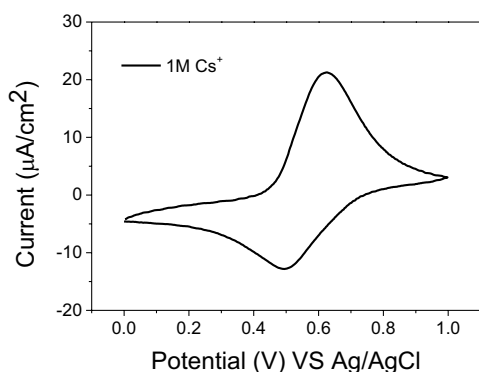


Fig. 2. Cyclic voltammograms of and NiPB-GO in 1 M CsNO_3 electrolyte solution at a scan rate of $5 \text{ mV} \cdot \text{s}^{-1}$.

4. Conclusion

The NiPB-GO nanocomposite has been prepared via a facile process in order to address current severe drawbacks of prussian blue and to efficiently separate the cesium by using electrochemical method.

The electrochemical properties of the NiPB-GO nanocomposite for Cs ion-exchange were evaluated in the present study. The NiPB-GO nanocomposite shows an excellent selectivity for Cs^+ over other alkali metals. Therefore, we expect that the NiPB-GO nanocomposite is effectively removal of Cs ion in aqueous environments.

REFERENCES

- [1] T. J. Yasunari, A. Stohl, R. S. Hayano, J. F. Burkhart, S. Eckhardt and T. Yasunari, Cesium-137 deposition and contamination of Japanese soils due to the Fukushima nuclear accident, Proc. Natl. Acad. Sci. USA. 108, 19530–19534 (2011).
- [2] G. Brumfiel, Fukushima set for epic clean-up, Nature 472, 146–147 (2011).
- [3] J. Yun, J. Pfisterer and A. S. Bandarenka, “How simple are the models of Na intercalation in aqueous media”, Energy & Environmental Science, 9, 955-961 (2016).

A Study on Chemical Separation of Pm-147 From RI Wastes

Jung Bo Yoo*, Sang Ho Lim, Junhyuck Kim, Hong Joo Ahn, and Kwang Yong Jee

Nuclear Chemistry Research Division, Korea Atomic Energy Research Institute, 111 Daedeok-daero-989,
Yuseong-gu, Daejeon, Republic of Korea

*yoojungbo@kaeri.re.kr

1. Introduction

Promethium-147 is a rare-earth group nuclide which is not found in nature. And all types of its isotopes have radioactive characteristics. Pm-147 has a 2.62 y half-life. Pm-147 has a great power density, low hazard and possesses several attractive properties [1].

A lot of methods of the separation skills of Pm-147 have been investigated in the past. The first quantitative analysis of Pm-147 was conducted at ORNL (Oak Ridge National Laboratory).

Separation process of radioactive wastes have been made by ion-exchange with lactate, citrate and α -HIBA (ammonium α -hydroxyisobutyrate) as eluents. Another research reviewed large number of eluting agents for the separation of lanthanides from each other. However, above methods have to use large amount of eluents which cost are very expensive. Furthermore, the chemical separation process is quite complicated [2].

The object of this study is to develop a chemical process for the separation of Pm-147 radionuclide from RI wastes.

2. Experimental Section

2.1 Reagents

All the chemicals were purchased from commercial suppliers with analytical reagent grade. The Neodymium standard solution with 10,000 mg/L was purchased from Accustandard, USA. Nitric acid (65%), ammonia solution (25%), ethanol (99%) and acetone (99%) were purchased from MERCK (Germany). The oxalic acid was purchased from Aldrich Chem. Co, USA.

2.2 Equipment

A radiation shielded ICP-AES system (Thermo Jarrell Ash, USA) composed of an ICP-AES and a stainless steel glove box was employed for the analysis of the radioactive waste samples. Doubly deionized water was prepared by passing demineralized water through a Milli-Q Plus Ultra Pure Water System (Milipore).

2.3 Chemical separation

2.0 mL of Nd standard solution was pipetted to a 50 mL plastic beaker, the mixture solution was evaporated to incipient dryness on a hot plate with IR lamp. The resulting residue was dissolved in 4 mL of 9 M HCl with hand-shake and transferred to a 50 mL centrifuge tube. Cool to room temperature and 15 mL of saturated oxalic acid were added with stirring for 1 h. After adjusting its pH to 3.5-4.0 with 10% NH_4OH , it was centrifuged at 5,000 rpm for 5 min.[3] The supernatant was discarded and sequentially washing the precipitate with oxalic acid-based cleaning solutions which were used 9 mL in 3 mL increments. The washed precipitate was transferred to a 15 mL of alumina crucible and evaporated on a hot plate with 70°C for 2 h. To acquire the crystalline Nd_2O_3 phase, the dried precipitate was annealed at 900°C for 1 h.

3. Results and Discussion

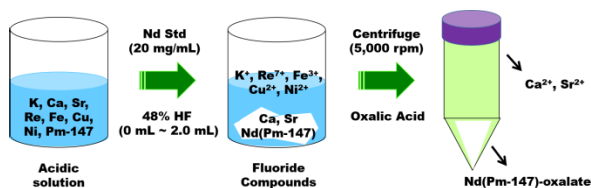


Fig. 1. Schematic illustration of Pm-147 chemical separation process.

A separation process of Pm-147 is shown in Fig. 1, and it is based on the wet chemistry. When the oxalic acid as a complexing reagent was added to the acidic solution, Nd-oxalate complex was spontaneously generated. Nd precipitates with oxalic acid at pH 3.5-4.0 in an oxalate medium and can be separated from some matrix elements that form insoluble metal-oxalate complex. After collecting the precipitates by using a centrifuge at 5,000 rpm for 5 min, thermal treatment was conducted for removal of moisture in the solidified products.

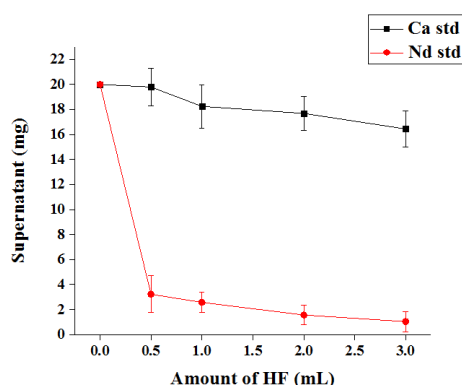


Fig. 2. Chemical behavior of Nd and Ca nuclides from radioactive waste sample solutions depending on the amount of HF (0 mL, 0.5 mL, 1.0 mL, 1.5 mL and 2.0 mL), which were measured with ICP-AES.

The relation between the precipitation behavior and the amount of HF is summarized in Fig. 2. When the HF was increased further from 0 mL to 2.0 mL, Nd nuclide was fully precipitated. On the other hand, Ca nuclide was not precipitated despite of its high K_{sp} values as well as high chemical affinity of Nd element with oxalic acid at pH 3.5-4.0. This result indicates that K_{sp} value of Nd was higher than Ca, which leads to precipitation of Nd as a fluoride compound.

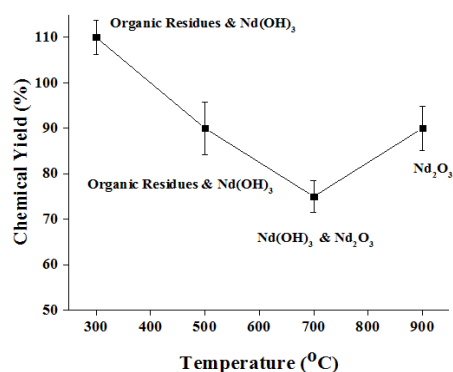


Fig. 3. Recovery rate for Pm-147 with an oxide compound (Nd_2O_3) at different temperatures, (a) at 300°C, (b) 500°C, (c) 700°C, (d) 900°C for 1h, respectively.

4. Conclusion

We developed the selective chemical separation process of Pm-147 from the radionuclides which include a variety of elements. As the chemical behaviors of Pm and Nd both are similar, Nd was used as a tracer of Pm. The formation of Pm-oxalate complex under acidic conditions by adding oxalic acid can be possible to separate unwanted radionuclides with low K_{sp} values. Based on the above results, the recovery yields of samples were measured with reliability.

REFERENCES

- [1] R. S. Pressly, C. L. Ottinger, P. B. Orr, E. E. Beauchamp, ORNL Report, 2928, 1960.
- [2] C-S. Lee, Y-M. Wang, W-L. Cheng and G. Ting, J. Radioanal. Nucl. Chem., 130, 21, 1989.
- [3] C. H. Lee, H. J. Ahn, J. Lee, Y-K. Ha, J-Y Kim, J. Radioanal. Nucl. Chem., 308, 809-816, 2016.

pH Dependence of Cs(I) and Sr(II) Coprecipitation With Fe(III) in Aqueous Solutions

TaeJun Kim^{1,2}, Minsik Kim^{1,2}, and Jei-Won Yeon^{1*}

¹ Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Rep. of Korea

² University of Science & Technology, 217, Gajeong-ro, Yuseong-gu, Daejeon, Rep. of Korea

*yeonysy@kaeri.re.kr

1. Introduction

A severe accident such as at Fukushima produces significant quantities of radioactive waste water. The water may contain non-volatile iodide (I^-) as a major fission product anion [1] and its oxidized form, molecular iodine (I_2), can be formed under irradiation. Although various techniques for the radioactive waste water treatment have been developed and used [2], a challenge still remains in keeping the volatility of radioactive iodine species low during the treatment. The water pH should be high enough for I_2 to decompose into non-volatile iodine species; such high pH conditions will also promote the formation of solid metal (hydr)oxides such as iron(III) (hydr)oxides. On the other hand, Jung et al. [3] demonstrated that one of water radiolysis products, hydrogen peroxide, can reduce I_2 to I^- at pH higher than 6. Thus, H_2O_2 addition to the waste water (pH > 6) may relieve the difficulty.

In this context, this study was conducted to validate that the optimum pH for treatment of radionuclides in the water is about 7. Only the formation of I_2 and the precipitation of Fe(III), Cs(I), and Sr(II) at room temperature were considered here.

2. Materials and methods

2.1 Gamma irradiation to iodide solution

To confirm that the concentration of I_2 significantly decreases under irradiation as pH increases above 6, an experiment was conducted using a cobalt-60 gamma irradiator (MDS Nordion). The aqueous solutions of 5 mol dm^{-3} sodium iodide at various initial pH were irradiated (8 kGy); their pH

were adjusted using hydroiodic acid and sodium hydroxide (NaOH) solutions. The triiodide concentration, which is linearly proportional to the concentration of I_2 with excess I^- , was measured, using a UV-Vis spectrophotometer (Biochrom).

2.2 Metal coprecipitation

The mass ratios of Cs and Sr to Fe in the precipitates formed in the aqueous solutions (pH 7 to pH 10; NaOH for pH adjustment) of ferric chloride ($FeCl_3$), and caesium chloride ($CsCl$) or/and strontium chloride ($SrCl_2$) were determined using inductively coupled plasma spectrometers. The molarity of Fe(III), Cs(I), and Sr(II) were (36, 18, 9) mmol dm^{-3} , respectively. The spectrometry samples were the residues obtained by filtration of the solutions using Whatman Grade 541 filter paper, dried in a drying oven at 103°C for three hours, and then dissolved in 1 mol dm^{-3} hydrochloric acid.

Sodium iodide (99.5%), $CsCl$ (99.9%), $SrCl_2$ (99.9%), $FeCl_3$ (97%), NaOH (97%), hydroiodic acid (57% in mass fraction), and hydrochloric acid (35% in mass fraction) used for this study were all purchased from Sigma-Aldrich.

3. Results and discussion

3.1 pH dependence of volatile iodine formation

The concentration of triiodide in the gamma-irradiated samples at pH (2.8, 6.1, 6.4, 6.5, 7.0) was (0.42, 0.37, 0.09, 0.02, 0.00) mmol dm^{-3} . The results confirm that the concentration of I_2 is drastically reduced under irradiation as the solution pH increases above 6.

As Bull et al. [4] mentioned, linear relationships between the Sr–Fe mass ratio and the initial pH are found with coefficients of determination greater than 98% (see Fig. 1). The Cs–Fe mass ratios were only a few percent at most and were not linearly dependent on the initial pH. With Sr(II), Cs(I) precipitated less than without Sr(II). The smallest Sr–Fe mass ratio was found at the solution pH about 7, the lowest experimental pH.

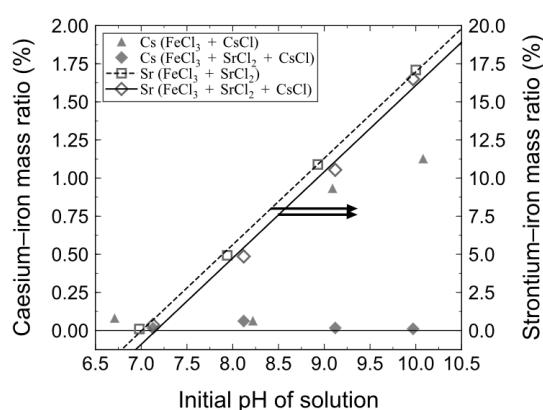


Fig. 1. Metal coprecipitation experiment results.

4. Conclusion

It appeared valid at least for the experimental conditions that the optimum pH for treatment of radionuclides in the waste water is about 7. Experiments on the aqueous solutions more alkaline and those containing other chemical species should be conducted to determine the relationship between the solution pH and the coprecipitation of Cs(I) and Sr(II) with Fe(III) under the conditions closer to the waste water.

ACKNOWLEDGEMENT

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) (No. 2017M2A8A4015281).

REFERENCES

- [1] C.-C. Lin, and J.-H. Chao, “Reassessment of Reactor Coolant and Iodine Chemistry under Accident Conditions,” *Journal of Nuclear Science and Technology*, 46, 1023–1031 (2009).
- [2] P. Sylvester, T. Milner, and J. Jensen, “Radioactive Liquid Waste Treatment at Fukushima Daiichi,” *Journal of Chemical Technology and Biotechnology*, 88, 1592–1596 (2013).
- [3] S.-H. Jung, J.-W. Yeon, S. Y. Hong, Y. Kang, and K. Song, “The Oxidation Behavior of Iodide Ion Under Gamma Irradiation Conditions,” *Nuclear Science and Engineering*, 181, 191–203 (2015).
- [4] P. S. Bull, J. V. Evans, and R. J. Knight, “Removal of Radioactive Strontium from Water by Coagulation–Flocculation with Ferric Hydroxide,” *Journal of Applied Chemistry and Biotechnology*, 25, 801–807 (2007).

7분과

방사화학 (Poster)



Physicochemical Properties of Trivalent Rare Earth Elements-Doped UO_2

Jandee Kim ^a, Seohyeon Park ^{a,b}, Jeongmi Park ^{a,b}, Jeongmook Lee ^a, Young-Sang Youn ^a,
Jong-Goo Kim ^a, Jong-Yun Kim ^{a,c}, and Sang Ho Lim ^{a,c*}

^aNuclear Chemistry Research Division, Korea Atomic Energy Research Institute: 111, Daedeok-daero 989,
Daejeon, Yuseong-gu, 34057, Republic of Korea

^bDepartment of Chemistry, University of Chungnam National University, Daejeon, Yuseong-gu, 34134,
Republic of Korea

^cRadiochemistry & Nuclear Nonproliferation, University of Science and Technology, Daejeon, Yuseong-gu,
34113, Republic of Korea

*Corresponding author: slim@kaeri.re.kr

1. Introduction

Spent nuclear fuel is consisted with not only stoichiometric UO_2 , but also parts of hypo-, and hyper-stoichiometric $\text{UO}_{2\pm x}$ matrix. Because the non-stoichiometry of the $\text{UO}_{2\pm x}$ matrix can influence on the corrosion/dissolution behavior of the spent fuel by itself, result in, radionuclides in matrix can be released. Therefore, it is important to understand the chemical and physical properties of $\text{UO}_{2\pm x}$ matrix, which determine the reactivity of $\text{UO}_{2\pm x}$ matrix [1-2]. And also, the influence of fission products including RE (Rare earth) element on the oxidation of UO_2 has been extensively investigated [3-4]. RE-doping enhances the stability of the cubic fluorite structures of U_4O_9 -type to oxidize to U_3O_8 depending on RE contents as compared with un-doped UO_2 . Oxidation of un-doped UO_2 occurs by simple two-step reaction to U_3O_8 via U_4O_9 , however, RE-doped UO_2 represents the resistance to change from cubic U_4O_9 to U_3O_8 . RE-dopants stabilized urania matrix by forming RE-oxygen vacancy clusters as increase the content of dopants, and resulted in decrease the interstitial sites could be occupied with O^{2-} during oxidation.

This work presents structural and electrochemical properties depending on stoichiometry of UO_2 matrix and Gd-doping level. Stoichiometry of all pellets was determined by non-destructive analysis of XRD, and distinguished into hypo-, hyper-, and stoichiometric $\text{U}_{1-y}\text{Gd}_y\text{O}_2$ series, respectively. Also, grain structure and electrical conductive property are estimated in terms of lattice structural changes depending on stoichiometry and Gd-doping. The influences of non-stoichiometric urania matrix and Gd-dopants on

dissolution behaviors of $\text{UO}_{2\pm x}$ matrix were examined by electrochemical study.

2. Experimental

2.1 Pellet preparation

Green pellets prepared by using unirradiated UO_2 and/or mixed with Gd_2O_3 powder were pressed, and sintered in a linear type of furnace at 1700°C in a reducing mood for 18 h. And then, stoichiometry of all pellets are adjusted by followed mild oxidation controlling the oxygen potential with CO/CO_2 mixed gas ratio, and temperature according to the Ellingham diagram which shows thermodynamic conditions of solid solution forms of Gd-doped $\text{UO}_{2\pm x}$. Prepared pellets have ~ 8 mm dia. and less than 1 mm thickness. Electrodes were polished with 3000 grit SiC paper prior to an experiment and then cathodically cleaned at a potential of -1.2 V for 5 minutes before an experiment.

3. Results

3.1 Lattice parameter

The variation of lattice parameter as a function of the Gd-doping level for $\text{U}_{1-y}\text{Gd}_y\text{O}_{2\pm x}$ specimens shows the 3 types of linear slopes. The slope of stoichiometric $\text{U}_{1-y}\text{Gd}_y\text{O}_2$ is -0.0173 , a reasonable contraction factor in agreement with published literature. This contraction could be caused by the possible charge compensation mechanisms as the Gd content increases; (i) the creation of the smaller U_V

cation; and (ii) the formation of O_{vacancy} . For the hyperstoichiometric $U_{1-y}Gd_yO_{2+x}$ the contraction factor is -0.0233 , considerably higher. By contrast, the lattice parameter for hypostoichiometric $U_{1-y}Gd_yO_{2-x}$ changes more gradually with Gd content (slope = -0.00995).

3.2 Grain structure and growth rates

Scanning electron microscopic results, using 20 keV electron acceleration voltage with 10 mm of working distance, revealed morphological evolutions and grain features of freshly prepared pellet depending on Gd-doped level and stoichiometry. The grain structures of $U_{1-y}Gd_yO_{2\pm x}$ changed drastically depend on Gd-doped level as shown in Fig. 1. In case of un-doped hypo-stoichiometric UO_{2-x} , it has polygonal structures consisted of wrinkled and flattened domains, are observed ~ 5 to ~ 10 μm sized, and the domain size decreases rapidly with increase of Gd doping level.

On the other hand, stoichiometric and hyper-stoichiometric pellets prepared by through the CO/CO_2 oxidation of hypo-stoichiometric pellets, have more big domains of over 20 μm sized and smooth textured surface. The interstitial oxygen activates internal strain of distorted cubic lattice, followed by enhanced uranium diffusion results in crystalline growths. Therefore, it can be explained that the distribution of excess O^{2-} makes to develop domain growth affects to domain structures of various size, texture and figurations. And as Gd-doping level increase, grain features got small in non-stoichiometric and stoichiometric pellets. However, hyper-stoichiometric pellets showed tolerance toward size-change to small. Probably, Gd plays a role to prevent inflowing of extrinsic oxygen into hyper- UO_{2+x} matrix. In fact, oxidation process of un-doped UO_{2-x} to UO_{2+x} needed more high oxygen potential than 10 mol% Gd-doped UO_{2-x} .

3.3 Electrochemical behaviors of oxidative dissolution

The influences of non-stoichiometric urania matrix and Gd dopant on dissolution behaviors of $UO_{2\pm x}$ matrix were investigated by electrochemical study. Cyclic voltammetry (CV) was recorded in carbonate solution with rotating electrode in scanning potential range from -3 V to 4 V, which can show catalytic behavior of water decomposition according to Gd-

doping. Distinguished two oxidation peaks, indicating two steps of oxidation mechanisms, defined as surface oxidation and dissolutive oxidation, respectively. Because dissolution process includes these whole oxidation mechanisms, oxidative behaviors of each steps has been investigated to understand the dissolution of $U_{1-y}Gd_yO_{2\pm x}$.

4. Conclusion

Stoichiometric and non-stoichiometric Gd-doped $U_{1-y}Gd_yO_{2\pm x}$ pellets are prepared by mechanical blending method adjusting the oxygen potential. Stoichiometry of all pellets was confirmed by non-destructive X-ray analysis.

The interstitial oxygen in hyper-stoichiometric $U_{1-y}Gd_yO_{2+x}$ improves the uranium diffusion in solid solution to allow grain size to be large. And oxidative dissolution of $U_{1-y}Gd_yO_{2\pm x}$ can be described with surface oxidation and followed dissolutive oxidation mechanisms. Gd dopants influence on decrease the surface oxidation rate which can control the whole oxidative dissolution reaction. Gd doping in hypo-stoichiometric pellets induces more oxygen deficiency ($x = y/2$) by forming Gd-oxygen deficiency clusters which makes cubic structure unstable, however, the dissolutive behaviors show that the surface oxidation decrease as increase the Gd doping level.

REFERENCES

- [1] J. C. Wren, D. W. Shoesmith and S. Sunder, "Corrosion behavior of uranium dioxide in alpha radiolytically-decomposed water", J. Electrochem. Soc. 152 (2005).
- [2] R. J. Mc Eachern and P. Taylor, "A review of the oxidation of uranium dioxide at temperatures below 400°C ", J. Nucl. Mater. 254, 87 (1998).
- [3] R. J. Mc Eachern, D. C. Doern and D. D. Wood, "The effect of rare-earth fission products on the rate of U_3O_8 formation on UO_2 ", J. Nucl. Mater. 252, 145 (1998).
- [4] L. E. Thomas, R. E. Einziger and H. C. Buchanan, "Effect of fission products on air-oxidation of LWR spent fuel", J. Nucl. Mater. 201, 310 (1993).

Separation of Niobium From Nickel-Chromium-Iron Containing Nickel-Based Alloys

Kyungwon Suh* and Youngkyong Ha

Nuclear Chemistry Research Division, KAERI, 111, Daedeok-daero 989beon-gil, Yuseong-gu,
Daejeon, Republic of Korea

*kyungwon0707@kaeri.re.kr

1. Introduction

Nickel-based alloys have been extensively used and been one of the important materials in nuclear facilities because of their high resistant to aqueous corrosion in a variety of environment and their resistant to high temperature. These alloys can be applied to a nuclear structure tube such as steam generator tube for PWRs and for some CANDU reactor. They can also be used as fuel element cladding and subassembly wrappers due to the high resistance to radiation-induced void swelling. Recently, there is a revival of interest in these alloys for Generation IV reactor and molten salt reactor, as well as in consideration of their use for the pressure vessel of the Prometheus space reactor [1].

Niobium is one of the alloying elements in nickel-based alloy. With the presence of niobium, it improve the intensity at high temperature, the temper brittleness and sensitivity to superheat even though less than 0.1% of niobium. In other words, the addition of niobium to nickel-based alloy provides solid solution strengthening, strengthening through carbide formation and precipitation hardening as well [2].

In addition, niobium is present as impurity in structural component of nuclear reactor pressure vessel and the cladding of nuclear fuel. By neutron activation of the stable niobium, the long-lived radioactive niobium-94 was generated. The radioactive concentration of niobium-94 was too low to direct gamma-measurement. Therefore, it is important to separate niobium from other elements and to quantify the amount of niobium present in nickel-based alloy for modification of thermal properties in nickel-based ally.

In this paper, the procedure for separation of niobium from nickel, chromium, and iron containing nickel-based alloy was demonstrated using following method: precipitation, solvent extraction, and anion exchange chromatography.

2. Experimental

2.1 Reagents

All reagents were commercially available and used without further purification. The niobium standard solution was from Accustandard. The nickel, chromium, and iron solution were prepared with following reagent.

Table 1. Each element composition

Element	composition
Ni	Ni powder and HNO ₃
Cr	Cr(NO ₃) ₃ and HNO ₃
Fe	Fe(NO ₃) ₃ and HNO ₃

MIBK (methyl isobutyl ketone) was purchased from SHOWA. Anionic exchange resin (AG 1-X8) was from BioRad.

2.2 Method

Each metal solution was mixed to prepare the simulation sample, which contained 150 mg nickel, 30 mg chromium, 20 mg iron, and 0.2 mg niobium.

For a niobium separation, main elements like nickel, chromium, and iron should be selectively isolated. First, precipitation was applied to lower the nickel content by adding 10% NH₄OH.

Second, iron was isolated by MIBK extraction from the precipitate containing nickel, chromium, iron, and niobium generated at precipitation step.

Last, niobium was separated from chromium and nickel using anion exchange chromatography.

Each element was determined by measurement of ICP-AES (Horiba Jobin Yvon, Activa M) in each step.

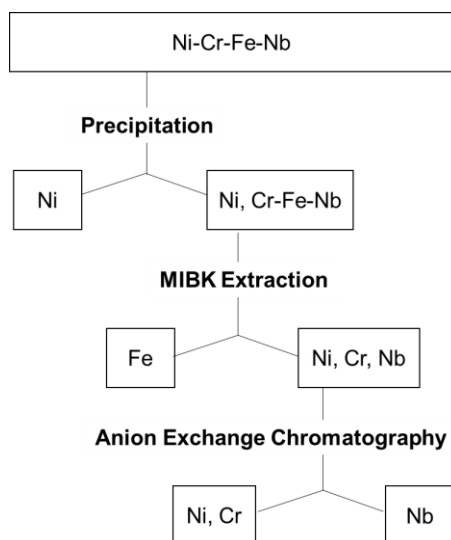


Fig. 1. Scheme for Separation.

3. Results

3.1 The precipitation with 10% NH_4OH

A different volume of 10% NH_4OH was added to the simulated solution for an isolation of the nickel. Although chromium, iron and niobium were precipitated regardless of the volume of added NH_4OH , the nickel content was dependent on the volume of NH_4OH . In Fig. 2, the nickel content was the highest in a solution at 0.8 mL of 10% NH_4OH , which meant that was the lowest in a precipitate.

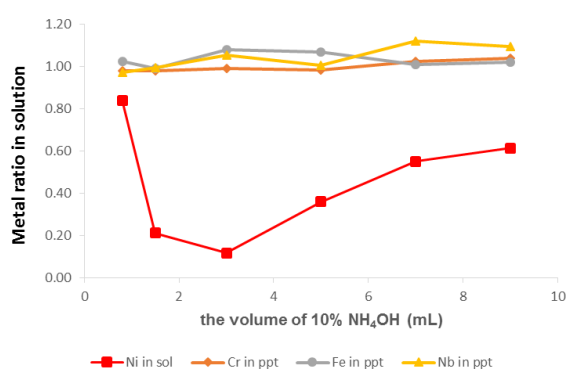


Fig. 2. Metal contents in precipitation step.

3.2 Solvent extraction with MIBK

There were nickel, chromium, iron, and niobium in a precipitate after the precipitation. Iron was extracted at the combination of MIBK and HCl [3].

For solvent extraction, the precipitate was dissolved in 4 M HCl and dissolved solution was mixed with MIBK. Iron was isolated from nickel and chromium, however, niobium recovery was 60 ~ 70%.

3.3 Anion exchange chromatography

Anion exchange chromatography was applied to separate niobium from nickel and chromium after MIBK extraction. A solution containing nickel, chromium, and niobium was loaded to anion resin (AG 1-X8) pretreated with 8 M HCl , the column was washed with 8 M HCl , and niobium was stripped 14% NH_4Cl and 4% HF mixture.

4. Conclusion

In here, the procedure for separation of niobium from nickel-based alloy containing nickel, chromium, and iron was demonstrated. First, precipitation was applied to lower the nickel content following MIBK extraction to isolate iron. Last, anion exchange chromatography was utilized to separate niobium from nickel and chromium. Although the final recovery of niobium was moderate, niobium was separated from nickel, chromium, and iron. This procedure can be applied to separate niobium in a radioactive metallic waste.

REFERENCES

- [1] "Comprehensive Nuclear Materails", 1st Ed, R. Kongins, R. Stoller, T. Allen, and S. Yamanaka, Eds., Elsevier, 2012.
- [2] G. D. Smith and S. J. Patel, "The role of niobium in wrought precipitation-hardened nickel-base alloys", International Symposium on Superalloys 718, 625, 706, and Derivatives, TMS, 2005.
- [3] J. Pinajian, "The methyl isobutyl ketone extraction of trivalent iron from hydrochloric acid solution and the effect of an inert dilution on the distribution coefficient" (1966).

Development of an In-Hotcell Bromine Method for the Determination of Metallic Contents of Electrolytic Reduction Products in the Pyroprocessing of Spent Nuclear Fuel

Young-Hwan Cho^{*}, Ji-Hye Kim, and Tae-Hong Park

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

^{*}kaerisan@gmail.com

1. Introduction

One of the key steps of pyroprocessing is electrolytic reduction of metal oxide fuel into metallic form. In this step, an appropriate analytical method is required to measure the metal conversion yield of the electrolytic reduction process product. One needs to develop a reliable analytical method to determine the metallic constituents in the metal and oxide mixtures. And system establishment for the implementation of chemical analysis of highly radioactive pyroprocessing materials has become an imminent task.

2. Chemical hot cell and shielded ICP-AES system

A shielded ICP-AES system, chemical hot cell will be used to carry out bromine method and provide chemical analysis data of highly active pyroprocessing materials. Fig. 1 shows the chemical hot cell and shielded ICP-AES system to be used for highly radioactive samples.

3. Considerations in performing bromine method in hot cell conditions

We have carried out numerous bromine methods to quantify the metallic contents from the methodology development stage to the real application to PRIDE and ACPF(cold test) samples. The samples tested comprise different physicochemical properties produced under various process operation conditions.

A standard bromine method has been established to meet the process materials for Korean concept of pyro electrolytic metal reduction process. The experience obtained from those test works enables the bromine method to be applied under the hot cell conditions in more practical ways.

3.1 Reaction time

In general, the bromine reaction with metallic component is fast and exothermic. In case of powder or porous sample, the reaction time is shorter than bulk sample. The higher the density (e.g. reduced ingot after CP, cathode process), the more time is required for the completion of bromine reaction. From our experiences 2 hr reaction time will be appropriate. During the course of bromination reaction the sample tumbles in the reaction vessel by thermal motion due to exothermic nature of the reaction. This facilitates the reaction progress. Therefore, magnetic stirring can be replaced by gentle shaking the reaction bottle by manipulator. This may help protect the electronic part of the stirring device and save the space in the hot cell. Comparative study between using magnetic stirring and gentle shaking method showed the same results.

3.2 Solid-liquid phase separation

The metallic components selectively dissolved into ethyl acetate-bromine organic liquid phase are separated from undissolved oxide solid part. A normal way of phase separation method is

centrifugation. In case of hot cell with limited space, one needs to skip the installation of centrifuge device into highly radioactive hot cell condition. In that case a filtration method is a good alternative method for phase separation.

3.3 Removing U using UTEVA column for precise FP measurement

Uranium is the predominant constituent of nuclear fuel. In case of quantification of major fission product components such as TRU and lanthanides, the predominant U signal interfere significantly with weak minor element signals. Therefore for precise measurement of TRU and REEs, one needs to remove U selectively from all other chemical components. For this extraction chromatography using UTEVA column is a good method. By using ICP-MS, we measured the concentration down to 1 ppb level. For accurate reduction yield determination of TRU and REEs, removing of U is a must, not an option.

3.4 Choice of plastic lab ware materials

In the chemical operation in hot cell conditions, disposable plastic lab wares are highly recommended.

Bromine is a very reactive chemical and ethyl acetate solvent may give damage to certain plastic materials. Typical plastic lab wares are severely damaged by bromine and ethyl acetate. From our experience, for chemical operations in bromine method PP (polypropylene) based plastic wares are suitable.

3.5 Validation of the test procedure

One need to check and validate the whole process of bromine test and ensure the reliability of the data. Simple way of doing this is back calculating the U recovery. Checking major element ratio such as La/Ce, La/Nd may help to diagnose the test results.

4. Conclusion

Understanding the basic underlying principles behind the bromine method and the accumulated test experiences enables the operation in hot cell conditions in effective ways.



Fig. 1. Chemical hot cells (upper) and a shielded ACP-AES system (bottom).

ACKNOWLEDGMENTS

This work was supported by a National Research Foundation of Korea (NRF) grant, funded by Korean government.

REFERENCES

- [1] Young Hwan CHO, et al “Determination of lanthanide metallic contents by selective dissolution in bromine-ethyl acetate solution”, KAERI/TR-5299/2013 (2014).

Electrochemical Dissolution of Platinum in a Chloride-Based Molten Salts

Suhee Choi*, Sang-Eun Bae, and Tae-Hong Park

Korea Atomic Energy Research Institute, 111 Daedeock-daero, Yuseong-gu, Daejeon, Korea

*shchoi926@kaeri.re.kr

1. Introduction

Pt is precious transition metal that has outstanding catalytic and electrical properties. Pt has been widely used in many applications, such as catalysts for chemical reaction and fuel cells and electrode material in chemical processes.

Pt is widely used as an electrode material in the electrochemical system because it is one of the most inert materials. These days, stability of Pt is one of key issues in pyrochemical process. Pt is used as a counter electrode in the electro-reduction process where the Pt is easily dissolved during the operation of the process [2]. In order to prevent the dissolution of Pt in the high temperature molten salt, it is needed to acquire the electrochemical properties of Pt in the molten salt.

In the present work, we investigated electrochemical dissolution of Pt in the various chloride-based molten salts. by using linear sweep voltammetry (LSV) and chrono-amperometry (CA).

2. Experimental

All the experiments were carried out in a glove box under Ar. The O₂ and H₂O level were maintained to be less than 1 ppm. Lithium chloride (LiCl)/potassium chloride (KCl) eutectic salts (anhydrous beads), lithium chloride (LiCl), calcium chloride (CaCl₂) and silver chloride (AgCl) were obtained from Sigma Aldrich.

Electrochemical studies were performed using a quartz tube (20 mm in outer diameter and 2 mm in wall thickness). Pt wire (DSM, dia. 1 mm) and glassy carbon (Alfa Aesar, dia. 2 mm) electrodes were used as the working and counter electrodes, respectively. The reference electrode was Ag wire includes 1 mol% AgCl in the LiCl-KCl eutectic melt.

The electrochemical experiments were performed using a Gamry Reference 3000.

3. Results and Discussion

It is well-known that the Pt dissolution reaction is impeded due to the formation of a passivation layer on the electrode surface where the passivation layer is composed of K₂PtCl₆ [3]. Fig. 1 shows the linear sweep voltammograms (LSVs) of Pt in various chloride-based molten salts. The anodic current began to increase at 0.69 V in LiCl-KCl (Fig. 1a) and then to be suddenly dropped. This reaction is attributable to the formation of the passivation layer.

In case of LiCl melt, the increment of current was observed from 0.683 V to 1.2 V. After passing the 1.2 V, the current was slightly decreasing. In this case, the decrease of the current resulted from the decrease of the electrode surface area.

We also measured the LSVs in the LiCl melt after adding KCl to the LiCl melt. The result showed that the passivation layer started to form after adding 10 wt% KCl (Fig. 1c).

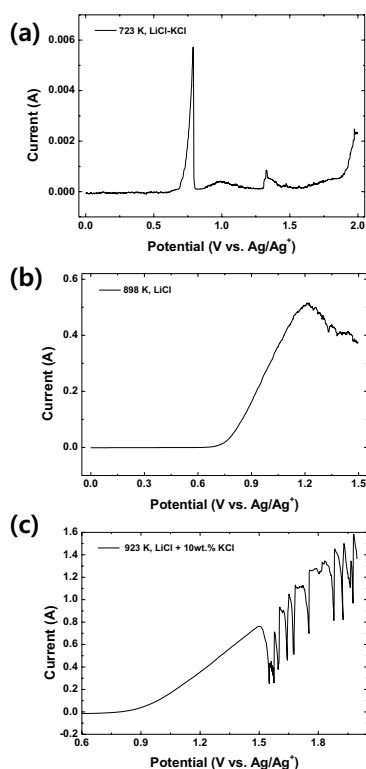


Fig. 1. LSV obtained dissolution of Pt in the (a) LiCl-KCl and (b) LiCl. (c) CV obtained dissolution of Pt in the LiCl + 10wt% KCl.

By using the LSVs, we also built Tafel graphs for the Pt dissolution reactions and estimated Tafel slope, exchange current, etc.

4. Conclusion

Here, we studied the dissolution of Pt in the various chloride-based molten salts. In the LiCl-KCl molten salt, the formation of the passivation film was observed during the dissolution of Pt while the dissolution of Pt continuously occurred in LiCl. The adding 10 wt% KCl to the LiCl melt induced the formation of the passivation layer, indicating that the addition of the KCl can reduce the loss of the Pt during the electro-reduction process.

REFERENCES

- [1] C. Deferm, J. Hulsege, C. Moller and B. Thijs, "Erratum to: Electrochemical dissolution of metallic platinum in ionic liquids", *Journal of Applied electrochemistry*, 43, 797 (2013).
- [2] S. M. Jeong, H.-S. Shin, S.-H. Cho, J. -M, Hur, H. S. Lee, "Electrochemical behavior of a platinum anode for reduction of uranium oxide in a LiCl molten salt", *Electrochimica Acta*, 54, 6335 (2009).
- [3] T. Takenaka, K. Iwasa, Y. Hashigaya , and M. Izaki, "Anodic Dissolution and Chemical Reduction of Noble Metals in LiCl-KCl Melt", *Electrochemistry*, 77, 633 (2009).

Chemical Properties of Hypo-Stoichiometric $U_{1-y}Nd_yO_{2-x}$ Produced by Neodymium Doping Into UO_2

Jeongmi Park^{1,2}, Young-Sang Youn¹, Jeongmook Lee¹, Jandee Kim¹, Seohyeon Park^{1,2},
Choong Kyun Rhee², and Sang Ho Lim^{1*}

¹Nuclear Chemistry Research Division, Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon, 34057, Korea

²Department of Chemistry, Chungnam National University, Deahak-ro 99, Yuseong-gu, Daejeon, Korea

*slim@kaeri.re.kr

1. Introduction

Over the past decades, research on UO_2 fuel has been actively performed because it is nuclear fuel to be used in light water reactors (LWRs) [1-5]. The fission products generated from UO_2 under irradiation can affect the chemical properties of UO_2 [2-4]. In particular, the fission products such as Nd, Ce, La, and Pr were easily dissolved into UO_2 matrix at high temperature, to form solid solution, which influenced the fuel thermal conductivity and fuel performance [5]. Among these fission products, Nd was used as a representative trivalent element to study the effect of Nd on UO_2 . In this study, the chemical properties of the hypostoichiometric Nd-doped UO_{2-x} pellets with various mol% Nd were investigated using scanning electron microscopy (SEM), X-ray diffraction (XRD), and raman spectroscopy.

2. Experimental

Nd-doped UO_2 pellets with 0, 5.9, and 9.5 mol% Nd were prepared using UO_2 and Nd_2O_3 powders. The fabricated pellets with a diameter of 6.35 mm were sintered in an alumina tube furnace at 1700 °C for 18 h under H_2 atmosphere to produce hypostoichiometric

$U_{1-y}Nd_yO_{2-x}$ pellets. SEM experiments were performed using a JEOL JSM-6610LV with an Oxford Instruments EDS. XRD data were obtained by Bruker-AXS D8 Advance system in the 2θ range of 20° to 120° with a scanning step of $0.02^\circ/0.1s$. $Cu K_\alpha$ radiation was used at beam current of 40 mA and beam generation power of 40 kV. The lattice parameter was calculated from the refinements of diffraction patterns based on a Pawley method using Bruker TOPAS program. ANDOR Shamrock SR500i Raman spectrometer with a He-Ne laser of 632.8 nm wavelength were used for Raman spectroscopic studies.

3. Results

Fig. 1 shows SEM images of Nd-doped UO_{2-x} pellets with 0, 5.9, and 9.5 mol% Nd. As shown in Fig. 1, the grain size of pellets decreased with higher Nd contents, which could be attributed to the lattice contraction of UO_2 by introducing Nd.

Raman spectrum of UO_{2-x} in Fig. 2a displayed two peaks at 445 and 1150 cm^{-1} , which corresponds to a cubic fluorite structure of uranium dioxide [6]. The band at 445 cm^{-1} was associated with the triple-degenerate Raman activity (T_{2g}) mode of U-O symmetric stretching mode in UO_2 . In addition, the

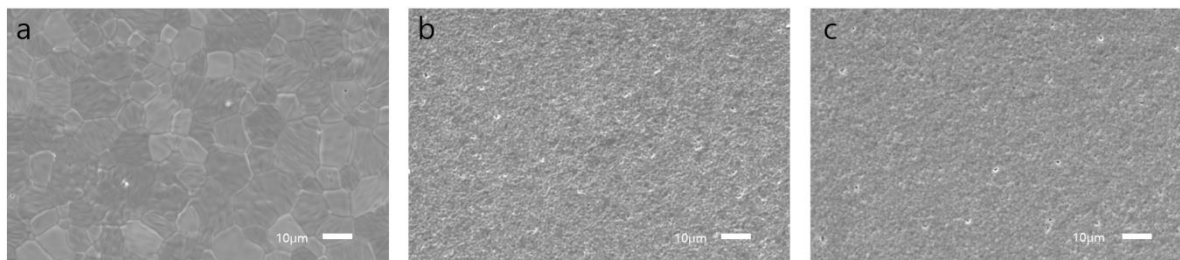


Fig. 1. SEM micrograph of the $U_{1-y}Nd_yO_{2-x}$ pellet with (a) $y = 0$, (b) $y = 0.059$, and (c) $y = 0.095$.

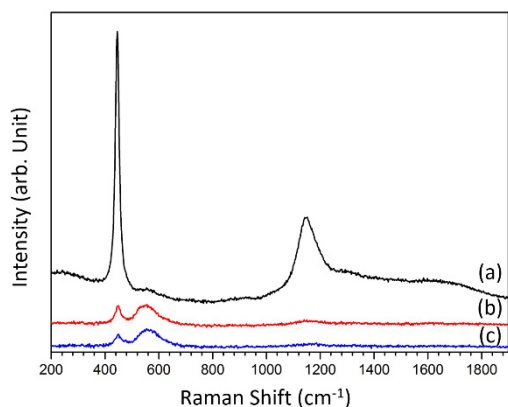


Fig. 2. Raman spectra of (a) UO_{2-x} and $\text{U}_{1-y}\text{Nd}_y\text{O}_{2-x}$ solid solutions with (b) $y = 0.059$ and (c) $y = 0.095$.

broad peak at 1150 cm^{-1} was assigned to the overtones (2L-O) of the primary L-O phonon (575 cm^{-1}). However, a new broad band appeared at 550 cm^{-1} as shown in Fig. 2b and 2c. As the concentration of Nd element in $\text{U}_{1-y}\text{Nd}_y\text{O}_{2-x}$ increased, the characteristic peaks of uranium dioxide at 445 and 1150 cm^{-1} decreased. On the other hand, the new signal at 550 cm^{-1} increased. These changes might be caused by the distortion of UO_2 crystal structure due to the introduction of Nd into the UO_2 matrix.

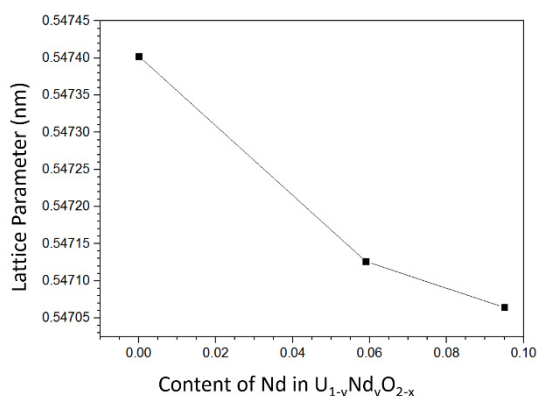


Fig. 3. Lattice parameters as a function of Nd content in $\text{U}_{1-y}\text{Nd}_y\text{O}_{2-x}$.

The lattice parameters of the $\text{U}_{1-y}\text{Nd}_y\text{O}_{2-x}$ were shown in Fig. 3. With the increase of Nd concentration in $\text{U}_{1-y}\text{Nd}_y\text{O}_{2-x}$ samples, their lattice parameters almost linearly decreased. In other words, when Nd was doped into UO_2 , the U^{4+} around Nd^{3+} atoms can be oxidized to U^{5+} , leading to the reduction in the lattice parameter.

4. Conclusions

We have investigated the chemical properties of $\text{U}_{1-y}\text{Nd}_y\text{O}_{2-x}$ using SEM, Raman spectroscopy, and XRD. Their grain sizes decreased with increasing Nd doping rates. Raman spectroscopy results showed the distortion of UO_2 lattice structure due to Nd element. The lattice parameters obtained from the refinements of obtained diffraction patterns show the lattice contraction with increasing Nd doping rates, which were consistent with the results of Raman data.

ACKNOWLEDGEMENT

This research was supported by the National Research Foundation of Korea (NRF) grant funded by the Korean government (MSIP) (No. 2017M2A8A5014754)

REFERENCES

- [1] P.G. Lucuta, R.A. Verrall, H.J. Matzke, B.J. Palmer, "Microstructural features of SIMFUEL – Simulated high-burnup UO_2 -based nuclear fuel", J. Nucl. Mater, 178, 48-60 (1991).
- [2] H. KLEYKAMP, "The Chemical State of the Fission Products in Oxide Fuels", J. Nucl. Mater, 131, 221-246 (1985).
- [3] R.C. Ewing, "Long-term storage of spent nuclear fuel", Nat. Mater, 14, 252-257 (2015).
- [4] D.A. Macinnes, P.W. Winter, "The effect of the chemically inert fission products on the chemistry of irradiated UO_2 ", J. Phys. Chem. Solide, 143-150 (1988).
- [5] S. Ishimoto, M. Hirai, K. Ito, Y. Korei, "Effects of Soluble Fission Products on Thermal Conductivities of Nuclear Fuel Pellets", J. Nucl. Science and Technology, 796-802 (1994).
- [6] M. Razdan, D.W. Shoesmith, "Influence of Trivalent-Dopants on the Structural and Electrochemical Properties of Uranium Dioxide (UO_2)", J. Electrochem. Soc, H105-H113 (2014).

Structural and Electrochemical Properties of Non-Stoichiometric UO_{2-x} Doped With Trivalent Rare Earth

Seohyeon Park^{1,2}, Jandee Kim¹, Jeongmook Lee¹, Jeong-mi Park^{1,2}, Young-Sang Youn¹,
Choong Kyun Rhee², and Sang Ho Lim^{1*}

¹Nuclear Chemistry Research Division, Korea Atomic Energy Research Institute, 989-111 Daedeok-daero,
Yuseong-gu, Daejeon, 34057, Republic of Korea

²Department of Chemistry, Chungnam National University 99-Daehak-ro, Yuseong-gu, Daejeon, 34134,
Republic of Korea
shpark16@kaeri.re.kr

1. Introduction

Currently, geological disposal is globally considered as a way to dispose of spent nuclear fuels (SNFs) [1]. In order to apply the geological disposal method, the reactivity of SNF and external environments such as unpredictable natural disasters and geochemical changes should be considered. Factors influence on the external of SNF including the reactivity of SNF by fission products (FPs), the reactivity of the UO_2 matrix itself, and disposal environment factors such as groundwater inflow [2]. Among the FPs, many studies on trivalent-rare earth have been performed [3-4]. In particular, Nd is one of the major FPs of spent fuels, and is a representative rare earth trivalent element used for estimating burn-up level using Nd isotopes [5]. In addition, since the stoichiometry of the UO_2 matrix affects to the reactivity of SNF, the physical and chemical properties of SNF according to stoichiometry need to be investigated.

In this study, we prepared simulated spent nuclear fuel doped with various molar concentrations of Nd(III) in hypostoichiometric UO_{2-x} and investigated the effects of carbonates in electrolyte aqueous solution on the electrochemical oxidation behaviors of SNF.

2. Experimental

The UO_2 powder and Nd_2O_3 powder were mixed with different molar concentrations of Nd(III) percentages (0, 2, 10, 18 mol%). After thoroughly grinding, resulting mixed powders were pelletized at 300 MPa pressure. And then, it was sintered under H_2 atmosphere at 1700 °C for 18 h to form a UO_{2-x} or $\text{U}_{1-y}\text{Nd}_y\text{O}_{2-x}$.

For observing the surface morphologies and estimating grain size of non-stoichiometric $\text{U}_{1-y}\text{Nd}_y\text{O}_{2-x}$ prepared in this study, Scanning electron microscopy (JSM-6610LV, JEOL, Japan) was used under specific(optimal) operation condition (25-30 keV accelerating voltage and 8 mm of working distance).

Electrical conductivities of each pellet were determined by measuring the specific resistivity at three positions on the surface of the pellet using a 4-point probe technique (HM21-Jandel Co., UK). As a working electrode in three electrode system, each pellet was placed on a stainless steel support, followed by molding with epoxy resin. After then, the working electrode was polished with a sand paper (3000 grids), and the oxide layer on the surface was removed by cathodic cleaning process (by applying 1.5 V for 5 minutes) prior to use. Pt mesh and SCE (Saturated Calomel Electrode) were used as a counter electrode and reference electrode, respectively. 0.1 M NaClO_4 and/or 0.01 M carbonate ion in 0.1 M NaClO_4 aqueous solutions were used as electrolytes. All electrochemical analyses were performed using potentiostat (CHI-600D, USA).

3. Result

Fig. 1 shows the SEM image of $\text{U}_{1-y}\text{Nd}_y\text{O}_{2-x}$ pellet depending on Nd(III) doping level. Initially, the surface of UO_{2-x} , which does not contain Nd, represented wrinkled morphology with more than 30 μm of a grain size having polygonal structure as shown in Fig. 1 (a). $\text{U}_{1-y}\text{Nd}_y\text{O}_{2-x}$ ($y = 0.02$, Fig. 1 (b)) indicated with a smaller grain size (less than 5 μm), and still retains its polygonal structure.

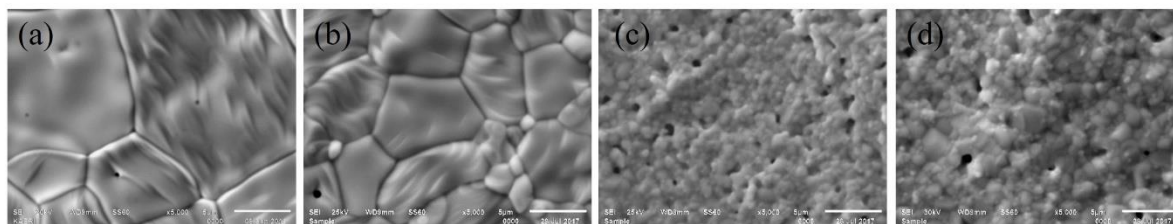


Fig. 1. SEM images of $U_{1-y}Nd_yO_{2-x}$ surface according to the Nd-doping level on the UO_{2-x} matrix. (a) $y=0$, (b) $y=0.02$, (c) $y=0.10$ and (d) $y=0.18$.

However, the grain size was significantly reduced and the polygonal structure disappeared at 10 mol% of Nd doping level. Moreover, pores started to appear on the surface (Fig. 1 (c)). At 18 mol% of Nd doping level, the numbers and sizes of pores were increased, which might cause the durability of pellet to deteriorate.

The electrical conductivities were also investigated depending on Nd(III) doping levels. The conductivity of $U_{1-y}Nd_yO_{2-x}$ pellet increased with increasing of doping level, but decreased sharply at the 18 mol% level.

In cyclic voltammetry (CV) analysis of Nd(III) doped UO_{2-x} electrode in 0.1 M $NaClO_4$ and/or 0.01 M carbonate ion in 0.1 M $NaClO_4$ as shown in Fig. 2, the surface oxidation occurred in the range of about -0.2 to 0.2 V, and dissolution started at ~ 0.2 V. The surface oxidation increased as the amount of Nd(III) increased. Comparing the CV results obtained in the two electrolyte system, it was confirmed that carbonate ions accelerated surface oxidation and dissolution of $U_{1-y}Nd_yO_{2-x}$ pellet. In addition, at a doping level of more than 10 mol%, the oxidation reaction peak by Nd itself might be observed in the -1.0 to 0 V region, so it seems that the solid solution did not form well.

4. Conclusion

The structure and electrochemical properties of $U_{1-y}Nd_yO_{2-x}$ were investigated. As the Nd doping level increased, the size of grain domain decreased and the size and the number of pores increased. Electrical conductivity also increased up to 10 mol% doping level, but decreased at 18 mol% level.

Through CV analysis, it was confirmed that the electrochemical behavior of $U_{1-y}Nd_yO_{2-x}$ pellet proceeded in the order of dissolution after the surface oxidation. Surface oxidation increased depending on the amount of Nd(III). The presence of carbonate ions is likely to have an effect on accelerating surface oxidation and dissolution as well.

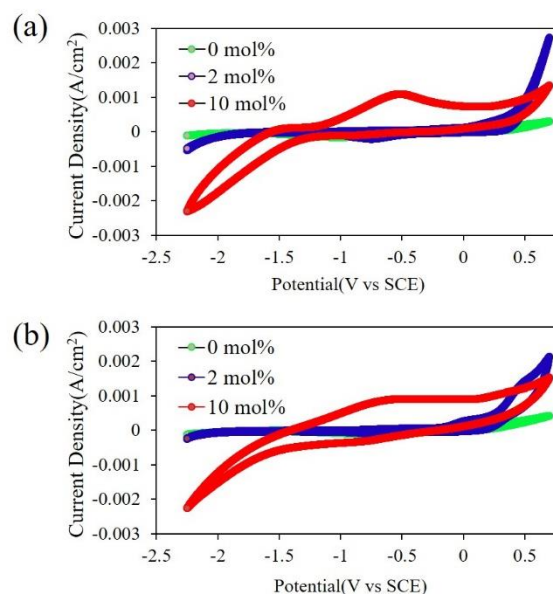


Fig. 2. CVs of $U_{1-y}Nd_yO_{2-x}$ depending on the Nd(III) doped amounts in the (a) 0.1 M $NaClO_4$, and (b) 0.01 M HCO_3^-/CO_3^{2-} contained 0.1 M $NaClO_4$ electrolyte, respectively.

REFERENCES

- [1] R.C. Ewing, "Long-term storage of spent nuclear fuel", *Nature Materials*, 14, 252–257 (2015).
- [2] H. Kleykamp, "The chemical state of the fission products in oxide fuels", *Journal of Nuclear Materials*, 131, 221–246 (1985).
- [3] J.M. Solomon, A. Navrotsky, M. Asta, "Energetics and defect clustering trends for trivalent rare earth cation substituted in UO_2 ", *Journal of Nuclear Materials*, 457, 252–255 (2015).
- [4] T. Ohmichi, S. Fukushima, A. Maeda and H. Watanabe, "On the relation between lattice parameter and O/M ratio for uranium dioxide-trivalent rare earth oxide solid solution", *Journal of Nuclear Materials*, 102, 40–46 (1981).
- [5] J.P. Ackerman, "Chemical basis for pyrochemical reprocessing of nuclear fuel", *Industrial & Engineering Chemistry Research*, 30(1), 141–145.

Study on Dissolution of Glass for Measurement of Activity of α , β , and γ Emitting Nuclides in Vitrification Radioactive Waste Sample

Kwang-Soon Choi*, Jaesik Hwang, Simon Oh, Young-Bok Kim, and Hong Joo Ahn

Nuclear Chemistry Research Division, Korea Atomic Energy Research Institute, 111 Daedeok-daero-989,
Yuseong-gu, Daejeon, KOREA

*nkschoi@kaeri.re.kr

1. Introduction

Nuclear power plants (NPPs) have been commercially operated in Korea since 1978, 24 NPP units being in operation and which accounting for 32% of Korean electricity generation. As a glass has advantages such as low leachability of hazardous constituents, vitrification as stabilizing of radioactive waste has been studied in the UK in the 1950s [1]. Vitrification of radioactive waste sample in Korea was tested in Hanwool NPP. As such, drums of low and intermediate level radioactive waste which include vitrification, are managed temporarily in the NPP sites. To transport radioactive waste from the temporary disposal site to the repository, > 95% of the total activity in a drum should be accounted for based on the radioactive waste acceptance criteria specified by nuclear safety and security commission (NSSC) notice No. 2015-4. Thus, we herein report the development of a dissolution procedure for radioactivity measurements of α , β , and γ ray-emitting nuclides in radioactive vitrification waste samples.

2. Experiments

2.1 Grinding of glass using a ball mill

To increase the surface area of vitrification waste sample, we made bulk sample small with a hammer and grinded the glass into small pieces using a tungsten carbide ball mill.

2.2 Dissolution of glass using a hot plate

The powdered inactive glass sample (0.5 g) was dissolved in the mixture of aqua regia (28 mL) with and without HF (1 mL) using a hot plate at 250°C. Following dissolution of the fresh glass sample, subsequent insoluble residue was centrifuged and filtered, respectively, using a centrifuge and a Watman No 41 filter paper. The obtained solutions were transferred to a 100 mL volumetric polypropylene (PP) flask and the volume made up to 100 mL using distilled and demineralized water (DDW). The filter paper was transferred to a Pt crucible, and then the crucible was ignited at 1000°C using an electric muffle furnace. After cooling in the desiccator, weight the crucible (labelled weight A). To measure the content of silicone dioxide by gravimetric analysis, 1:1 H₂SO₄ (3 mL) and HF (4 mL) were added to the Pt crucible. The solution in the crucible was evaporated to dryness on a hot plate at 250°C. And then the crucible was ignited at 1000°C using an electric muffle furnace. After cooling in the desiccator, weight the crucible (labelled weight B). Subsequent insoluble residue in the crucible was dissolved in 1:1 HCl (20 mL) and HF (0.1 mL). The obtained solution was transferred to a 50 mL volumetric PP flask. Following the dissolution of inactive glass sample, the contents of constituent elements were measured by inductively coupled plasma atomic emission spectroscopy (ICP-AES).

2.3 Dissolution of glass using an HP-MDS

The powdered inactive glass sample (0.5 g) was dissolved in the mixture of HCl (13.6 mL), HNO₃ (3.4 mL), and HF (0.6 mL) using a high-performance microwave digestion system (HP-MDS) [2]. After centrifugation, the obtained solution was transferred to a 100 mL volumetric PP flask and the volume made up to 100 mL using DDW. Subsequent insoluble residue was dissolved using an HP-MDS in the same conditions mentioned above. Following the dissolution of inactive glass sample, the contents of constituent elements were measured by ICP-AES.

3. Results & discussion

The powdered fresh glass sample was not dissolved in the mixture of HCl, HNO₃, and HF using an HP-MDS. However, the undissolved residue was completely dissolved using the second operation of an HP-MDS in the same mixture of acids. Although the sample was completely dissolved, the analytical results of SiO₂ determined by ICP-AES was lower than those of certified value, owing to the volatile loss of SiF₄ despite the closed microwave digestion system.

On the other hand, the weight difference between A and B corresponds to the content of silicone dioxide. According to preliminary tests, the amount of dissolved Al, Ca, K, Na, and Si in the sample decreased as the amount of HF increased. Especially, 50% Na was not dissolved in the excess presence of HF. Although the loss of SiO₂ as SiF₄ happened using a hot plate, we selected a hot plate for dissolution of radioactive waste samples due to handling a lot of samples.

the measurement of radioactivity in vitrification radioactive waste samples. The ICP-AES analytical results showed that principal constituent elements except SiO₂ were quantitatively recovered in dissolving sample with a mixture of aqua regia and HF on a hot plate. This procedure will be applied to vitrification radio-active waste samples for the measurement of α , β , and γ ray-emitting nuclides.

REFERENCES

- [1] Mike T.H., "Vitrification of High Level Waste in the UK", *Procedia Mater Sci*⁺, 7 10-15 (2014).
- [2] Fiona L., Lian Z., "Microwave digestion for the quantification of inorganic elements in coal and coal ash using ICP-OES", *Talanta*, 101 346-352 (2012).

4. CONCLUSIONS

We developed a sample dissolution procedure for

Electrical Conductivities of UCl_3 - LiCl - KCl Molten Salts Measured by Using a Short Time Interval and Multiple-Potential Step Technique

Sang-Heon Lee, Jong-Yun Kim*, Sang-Eun Bae, and Tae-Hong Park
Nuclear Chemistry Research Division, Korea Atomic Energy Research Institute
111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, 34057, Korea
*k jy@kaeri.re.kr

1. INTRODUCTION

Many physical properties of high-temperature molten salt such as electrical conductivity, viscosity, phase transition and temperature are very critical in designing and operating pyroprocess. Such properties also provide important information with regard to the process operation and nuclear material accountancy. Among the many valuable physical properties, the electrical conductivity is an essential physical property providing information on monovalent and trivalent ions in molten salts. However, almost of the commercially available measurement systems are not adequate for measuring the electrical conductivity of high-temperature corrosive molten salts [1]. Recently, we have developed a new electrical conductivity measurement system installed inside a glove box, which is controlled by a custom-made software based on the LabView program [2].

Uranium trichloride (UCl_3) is one of the most important components in the pyrochemical process because the nuclear spent fuel comprises mainly uranium. The molten salt in the process usually contains up to ~9wt% of UCl_3 so that the physical properties of the molten salt containing a large quantities of UCl_3 are very important in the process. In this work, we have measured and reported the electrical conductivities of the high temperature molten salts containing UCl_3 and lanthanide trichlorides.

2. METHOD

The electrical conductivity measurements of LiCl - KCl molten salts with various compositions of UCl_3 , LaCl_3 , CeCl_3 , and NdCl_3 were performed in a glove box equipped with a high-temperature furnace under an argon atmosphere. In-house designed capillary cell and measurement system were used for the determination of the electrical conductivities of molten salts. In our present study, a chronoamperometry technique featured with short time interval and multiple-potential step was used to minimize measurement errors resulted from polarization. In this technique, the potential is applied for a very short period of time (a few microseconds) to minimize the influence of the polarization of the electrode. With a series of standard solutions with known conductivities, the cell constant was determined from the slope of the peak current at each time interval versus the potential. Subsequently, the same procedure and the same capillary cell were used for the determination of the electrical conductivities with a test sample solution containing 0 to 9 mol% UCl_3 at various temperatures (300 to 630 °C).

3. RESULTS

The measurements of the electrical conductivity revealed one or two phase transition temperatures for the multi-component LiCl - KCl molten salt systems

comprising uranium and lanthanides when the temperature was ramping down from 630 to 300 °C.

4. CONCLUSIONS

In the present work, the electrical conductivity measurement technique not only determine the electrical conductivities of LiCl-KCl molten salts containing uranium and lanthanide elements but also show a potential to measure phase transition temperatures of the multi-component molten salts.

Physical property measurement techniques for electrical conductivity and viscosity are under development and will be very useful for safe and efficient process operation. In addition, such fundamental physical properties provide the basis for understanding and predicting the macroscopic behavior of the actinides and lanthanides in the molten salt of the pyroprocess.

ACKNOWLEDGEMENTS

This work was supported by a National Research Foundation of Korea (NRF) Grant, funded by the Korean government.

REFERENCES

- [1] S.F. Katyshev, Yu.F. Chervinskii, and V.N. Desyatnik, *At. Energ.* Vol. 53, pp. 108, 1982.
- [2] J.Y. Kim et al., *Asian J. Chem.* Vol. 25(12), pp. 7028, 2013.

Burnup Determination of Irradiated Sintered Solid and Annular Pellets by Neodymium Monitor Methods Based on Isotope Dilution Mass Spectrometric Measurements

Jung Suk Kim*, Byungman Kang, Yang-Soon Park, Jai Il Park, and Yeong Keong Ha

Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon, Korea

*njskim1@kaeri.re.kr

1. Introduction

One of the important parameters required for the study of a nuclear fuel is its burnup, which is the number of fission per 100 heavy nuclide atoms (mass ≥ 232) initially present in the fuel. Destructive methods, which is based on the determination of specific nuclides by a chemical analysis after appropriate separation of the heavy elements and a monitoring fission product, are widely used as a reference method to measure the burnup of irradiated fuel[1]. The isotope ^{148}Nd was selected mainly because its fission yield is independent of the fissioning actinide, and because of its low thermal neutron capture cross-section. However, a serious drawback of ^{148}Nd burnup monitor is its reported susceptibility to a ^{147}Nd neutron capture effect, that is, a large thermal neutron capture cross-section for ^{147}Nd which would cause increasing amounts of ^{148}Nd with increasing neutron flux. The isotope ^{142}Nd , meanwhile, is used to monitor and correct possible contamination of the fission product with natural Nd during chemical separation. Another approach is to use a different monitor such as $^{145}\text{Nd} + ^{146}\text{Nd}$ because the sum of ^{145}Nd and ^{146}Nd appears to be invariant with neutron flux and fluence[2]. In thermal reactor spectra, considerable burnout of ^{143}Nd and ^{145}Nd occurs to ^{144}Nd and ^{146}Nd respectively. Therefore, their fission yields and their abundances in the fission product neodymium have to be added.

The aim of the present work is to determine the total burnup by using various neodymium isotope monitors for the same samples from the sintered solid and dual-cooled annular pellets irradiated in the Hanaro reactor at KAERI, and is to compare the results, so as to determine the respective validity of the methods.

2. Experiments

2.1 Chemicals

Certified ^{233}U (99.470 atom%) and ^{150}Nd (96.13 atom%) spikes were obtained from Oak Ridge National Laboratory (ORNL). Certified ^{242}Pu spike solution (99.9033 atom%, IRMM-044) was obtained from the Institute for Reference Materials and Measurements.

2.2 Sample Preparation and Isotopic Measurement

The irradiated pellet sample having been precisely weighed was placed into a 100 mL dissolution flask of the dissolution apparatus. 30~40 mL of 8 M HNO_3 for the samples of S1, S5, S6 and S7, and 30~40 mL of 8 M HNO_3 and 3 mL of c-HCl for the samples of S2, S3 and S4, which contain some additives such as Al, Mn and Cr, was then added, while applying a water cooling. All samples was refluxed for more than 12 hours. Chemical separation was carried out for both the unspiked and the spiked sample solutions in the same experimental conditions in a glove box. The isotopic compositions of U, Pu and Nd in the unspiked and spiked samples were determined using a thermal ionization mass spectrometer (TIMS, Finnigan TRITON).

3. Results & Discussion

3.1 Determination of the Isotopic Composition

Table 1 shows the isotopic compositions of neodymium in the irradiated pellet samples measured by the TIMS. In this work, all the measured average ratios of neodymium were corrected for a mass discrimination and the contribution of a natural

contamination, so as to achieve a high accuracy for the burnup measurement.

3.2 Determination of the Total burnup and Burnup Parameters

Table 2 gives the total burnup in GWd/MtU determined by the neodymium isotope monitors, that is, ^{148}Nd , $^{143}\text{Nd}+^{144}\text{Nd}$, $^{145}\text{Nd}+^{146}\text{Nd}$, and the total of the Nd isotopes for the irradiated pellet samples. The number of fissions by the ^{148}Nd monitor was calculated with a correction for the excess ^{148}Nd produced from the capture on ^{147}Nd [1]. The data obtained by using other neodymium isotope monitors are in agreement, within a deviation of 5%, with those by the ^{148}Nd monitor. Table 3 shows the F_5 (^{235}U fractional burnup) and D_5 (^{235}U depletion) values, $^{148}\text{Nd}/\text{U}$ and Nd/U mass ratios. The F_5 is calculated by assuming no loss of ^{238}U or ^{236}U during an irradiation. The D_5 (the difference between the initial and final ^{235}U content) is expressed by $W_5^0/(W_5^0-W_5)$ [2].

4. Conclusion

The contents of U, Pu, Nd and their isotopes in irradiated sintered solid and annular pellet samples and the total burnup by using various neodymium isotope monitors can be determined simultaneously by the isotope dilution mass spectrometric techniques. The neodymium isotope patterns provide information on the real irradiation characteristics which are necessary for evaluating a fuel's performance in a reactor. A comparison between independently determined burnup values provides a check on the validity of the results

Table 1. Isotopic Compositions of the Neodymium Separated from the Irradiated Pellet Samples

	Atom%*					
	^{143}Nd	^{144}Nd	^{145}Nd	^{146}Nd	^{148}Nd	^{150}Nd
S1	10.35	42.31	16.21	17.74	9.31	4.08
S2	10.98	41.72	16.35	17.60	9.28	4.07
S3	11.09	41.72	16.36	17.57	9.23	4.03
S4	10.69	42.04	16.27	17.68	9.27	4.04
S5	10.20	42.58	16.10	17.75	9.29	4.08
S6	9.84	42.73	15.99	17.92	9.39	4.14
S7	9.87	42.70	16.00	17.86	9.40	4.16

* Corrected for contribution due to natural contamination and mass discrimination.

Table 2. Total Burnup Determined by the Neodymium Isotope Monitor Methods for the Irradiated Pellet Samples

	GWd/MtU			
	^{148}Nd	$^{143}\text{Nd}+^{144}\text{Nd}$	$^{145}\text{Nd}+^{146}\text{Nd}$	Nd-total
S1	33.28 ±1.04	34.21 ±1.07	34.32 ±1.07	33.53 ±1.05
S2	33.22 ±1.04	33.47 ±1.05	33.90 ±1.06	33.18 ±1.04
S3	33.20 ±1.04	33.31 ±1.04	33.80 ±1.06	33.05 ±1.03
S4	34.00 ±1.06	34.34 ±1.07	34.78 ±1.09	33.95 ±1.06
S5	35.45 ±1.11	35.81 ±1.12	36.23 ±1.13	35.40 ±1.11
S6	36.43 ±1.14	36.84 ±1.15	37.31 ±1.17	36.39 ±1.14
S7	38.83 ±1.21	36.87 ±1.15	37.28 ±1.17	37.18 ±1.16

Table 3. Determination of Some Burnup Parameters for the Irradiated Pellet Samples

	Atom%*				
	$^{148}\text{Nd}/\text{U}$	Nd/U	D_5	F_5	F_5/F_T
S1	4.02E-4	4.23E-3	1.033	2.244	0.647
S2	3.94E-4	4.16E-3	1.039	2.223	0.642
S3	3.89E-4	4.13E-3	1.039	2.226	0.644
S4	4.02E-4	4.25E-3	1.035	2.234	0.631
S5	4.19E-4	4.42E-3	1.030	2.244	0.608
S6	4.35E-4	4.54E-3	1.025	2.258	0.595
S7	4.36E-4	4.54E-3	1.025	2.257	0.558

D_5 : ^{235}U depletion in weight, F_5 : ^{235}U fractional burnup in atom% fission, F_T : total burnup in atom% fission

REFERENCES

- [1] ASTM, "Standard Test Method for Atom Percent Fission in Uranium and Plutonium Fuel (Neodymium-148 Method)", Annual Book of ASTM Standards 12.02, E321-96 (reapproved 2012), (2012).
- [2] J. S. Kim, Y. S. Jeon, S. D. Park, S. H. Han, and J. G. Kim, "Burnup Determination of High Burnup and Dry Processed Fuels Based on Isotope Dilution Mass Spectrometric Measurements", J. Nucl. Sci. & Technol., 44(7) 1015-1023 (2007).

Determination of ^{129}I in Simulated Radwastes Using Anion-Exchange Resin

Hwa Sung Yoo*, Ke Chon Choi, Keon Ho Kim, and Hong Joo Ahn

Korea Atomic Energy Research Institute, 111, Daedeok-daero 989beon-gil, Yuseong-gu, Daejeon, Korea

*yumars@kaeri.re.kr

1. Introduction

There is a growing consensus that the regulation of nuclear waste management of alpha or beta decay nuclides such as ^{129}I has to be emphasized, and that a selection basis has to be made in terms of a disposal program of nuclear waste generated from a nuclear power plant and other nuclear facilities.

Generally, concentration of difficult-to-measure (DTM) radionuclides, such as ^{129}I , contained in nuclear wastes generated from nuclear power plants can be predicted from the scaled factor ($^{129}\text{I}/^{137}\text{Cs}$, $^{129}\text{I}/^{60}\text{Co}$) by measuring the radionuclide instead, such as ^{60}Co , ^{137}Cs which are relatively easier to measure. Therefore, the activity of ^{129}I has to be measured accurately in order to calculate the scaling factor (radioactivity ratio of ^{129}I and ^{137}Cs) in relation to the activity of the standard nuclide. The latter will then be used as described above. However, the radioactivity ratio of ^{129}I to ^{137}Cs varies widely, owing to many different classes of radioactive waste forms generated from nuclear power plants. Consequently, it can be concluded that a trustworthy chemical analysis is required to determine the characteristic scaling factors for each plant [1].

Xiaolin Hou et al. washed the samples with a KHSO_3 solution of 0.5 M and distilled water to measure the iodine content in natural water and seawater. I_2 and I^- were then separated from the other interfering ions using an anion exchange resin column (AG1-X4, Cl^- form) [2].

2. Experimental

5 g of a simulated radwaste, 5 mL of a phosphate buffer solution (pH 7) and 20 mL of distilled water were added into a plastic beaker containing 5.0 g of the anion exchange resins (AG 1-X4, 50-100 mesh Cl^- form). The resin solution was prepared and allowed to stand for 12 hours to wait for complete absorption into the resin. Next, 100 mg of KI were added to the solution, and the resin was transferred into a plastic column. 8 mL of a 7 % NaClO solution (oxidant) passed through the column, and the same procedure was repeated 2 times. The eluent was then

collected in a vial, 1.25 mL of 65% HNO_3 was added, and finally, the reducing agent (1000 mg, $\text{NH}_2\text{OH}\cdot\text{HCl}$) was added into the eluent solution. After the solvent extraction procedure was conducted with separatory funnel in which 35 mL of CHCl_3 contained was shaken for 3 minutes, the organic phase was separated and collected in a vial. After adding 5 mL of the 0.2 M NaHSO_3 solution into the separated organic phase, the solution was back-extracted to the aqueous phase. A precipitate formed by 0.2 M AgNO_3 in an aqueous phase went through an aging process for 5 minutes. After the centrifuging process, the final sample was now transferred to a planchet, and dried below 70°C . Its activity is measured for 10,000 seconds with a low energy photon spectrometry system.

3. Results and discussion

3.1 Optimum chemical condition for the separation and recovery of non-radioactive iodine

The recovery rate of iodine as a function of acidity (on the variation of pH in the eluent) was measured to find the optimum chemical conditions for the separation of non-radioactive iodine. The recovery rate of iodine was measured, while the acidity of the sample was controlled in the range between 0.0 and 1.6 M.

Fig. 1 shows the results of the optimum concentration of the acid for the extraction process with an organic solvent. The concentration range between 0.4 and 1.2 M provides the highest recovery rate, and the NaClO solution was used as an eluent to extract the iodine absorbed on the anionic resin. This may be due to the oxidation of iodide ions, under the high acidity condition, to non-volatile oxyiodate such as IO_3^- or IO_4^- , which are not extracted into the organic layer.

Fig. 2 shows the analytical results of the effect of the eluent concentration in the range of 2.0 ~ 12.0 % on the iodide recovery, using a column elution method. As shown, the concentration range of 8~10.0 % for NaClO was found to be suitable for the best elution of iodide from the anionic column.

3.2 Recovery rate of iodide by stage using ^{125}I tracer

Several stages of the separation process including oxidation, adsorption, elution, reduction, and extraction, were required to analyze quantitatively the iodine present in a concentrated liquid waste sample. Some contents of the iodine can be lost through various separation processes of physical or chemical reactions owing to its high volatile property. In this study, the recovery rate of iodine for each separation process was measured using a radioactive iodine isotope tracer added into the starting sample at the initial step to compensate for the calculation of the total recovery rate of iodine.

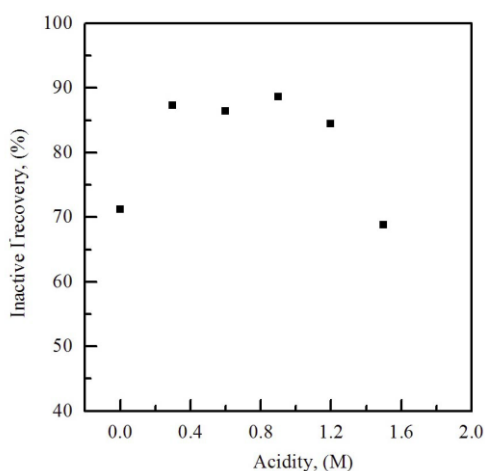


Fig. 1. Effect of Acidity on the Recovery of Iodide.

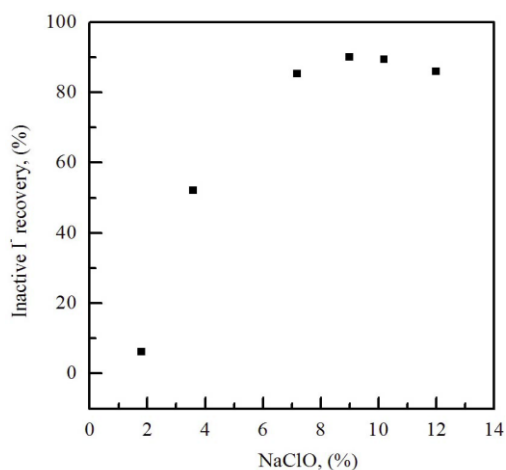


Fig. 2. Effect of Concentration of Eluent on the Iodide by Column Elution Method.

4. Conclusions

An analytical procedure and a chemical separation technique were studied for the quantitative determination of ^{129}I , which is classified as a

difficult-to-measure (DTM) nuclide. The recovery rate was measured by adding ^{125}I as an iodine tracer to the simulation sample to measure the activity concentration of ^{129}I . The optimum condition for the maximum recovery yield of iodine on the anion exchange resins (AG1-X4, 50-100 mesh, Cl^- form) was found to be pH 7.

REFERENCES

- [1] K. C. Choi, Y. J. Park, K. Song, "Determination of the ^{129}I in primary coolant of PWR", Nuclear engineering and thchnology, 45, 61-66 (2013).
- [2] X. Hou, H. Dahlgard, B. Rietz, U. Jacobasen, S. P. Nielsen and A. Aarkrog, "Determination of chemical species of iodine in seawater by radiochemical neutron activation analysis combined with ion-exchange presepation", *Anal.Chem.*, 71, 2745-2750 (1999).

Comparison of ^{55}Fe Recovery Results by Gravimetric and Spectrometric Analysis

Hyeon Jeong Oh*, Jung Bo Yoo, Hong Joo Ahn, and Kwang Yong Jee

Nuclear Chemistry Research Division, Korea Atomic Energy Research Institute, 111 Daedeok-daero-989,
Yuseong-gu, Daejeon, Republic of Korea

*ohj@kaeri.re.kr

1. Introduction

According to the Notice No. 2015-4 from the Nuclear Safety and Security Commission (NSSC), it should be known to clarify radioactivities of gross α as well as over 10 radionuclides such as ^3H , ^{14}C , ^{129}I , ^{60}Co , ^{55}Fe , $^{59,63}\text{Ni}$, ^{90}Sr , ^{94}Nb , ^{99}Tc and ^{137}Cs . Among them, ^{55}Fe , $^{59,63}\text{Ni}$, ^{94}Nb , ^{99}Tc nuclides have to be solely separated because of their low energy beta- and X-ray emitting radiochemical characteristics. What if it does not separate those nuclides individually, detecting signals show the overlapping circumstance. For that reason, development of chemical separation process is an important issue. In fact, KAERI (Korea Atomic Energy Research Institute) has investigated the analytical method regarding sequential chemical separation procedure for those radionuclides from radioactive waste samples [1].

To acquire the accurate results, it is essential to manage the determination techniques of individual radionuclides. There are two representative ways to determine recovery results. The first one is a gravimetric analysis. Gravimetric analysis is the process of producing and weighing a compound or element in as pure form as possible after some form of chemical treatment has been carried out on the substances to be examined. It is one of the most accurate and precise analytical skills of macro quantitative method. Another one is ICP-AES (Inductively Coupled Plasma-Atomic Emission Spectrometry) analysis. It is very rapid (within 1 ~ 2 min) simultaneous multi-element analysis. ICP-AES has good precision (0.5~1%) and accuracy as well as good reliability and inherent safety of modern instrumentation.

Herein, we have focused on comparing the recovery results of ^{55}Fe by gravimetric method and spectrometric method. To compare the recovery results based on the measured techniques, we

separated the ^{55}Fe nuclide from the synthetic waste samples. The gravimetric analytical results are in good agreement with ICP-AES analysis.

2. Experimental section

2.1 Reagents

All the chemicals were purchased from commercial suppliers with analytical reagent grade. The Fe standard solution with 10,000 mg/L was purchased from Accustandard, USA. HF (48%), ammonia solution (25%) were purchased from MERCK (Germany).

2.2 Separation process

Radioactive waste sample solutions were prepared according to the previously procedure. In brief, from 1.0 mL to 5.0 mL of Fe carrier solutions (10 mg/mL) was added into a 50 mL plastic beaker. To evaporate the acidic solutions, the beaker was placed on the hot plate for 3 h. And the dried product was dissolved in 2 mL of 1 M HNO_3 . After adjusting its pH to 4.0 ~ 5.0 with 10% NH_4OH and diluting this solution by adding deionized water. The diluting solution were passed through the anion exchange resin column. 100 mL of 0.1 M NH_4 -oxalate solution (pH 4.0 ~ 5.0) and 25 mL of 4 M HF solution were sequentially poured into the column. After that, Fe nuclide was collected in a 200 mL plastic beaker with 65 mL of saturated boric acid by adding the 60 mL of 4 M HF solution [2]. This solution with Fe nuclide not only was measured with an ICP-AES but also a gravimetric method. To gain a precipitate, 25% NH_4OH solution was added into the 200 mL beaker. The brown precipitate was made as an amorphous

$\text{Fe}(\text{OH})_3$ phase. It was centrifuged at 4,000 rpm for 5 min. The precipitate was transferred to a 15 mL of alumina crucible and evaporated on a hot plate with 100°C for 3 h. The dried precipitate was annealed at 800°C for 30 min to get a crystalline Fe_2O_3 [3].

3. Results and discussion

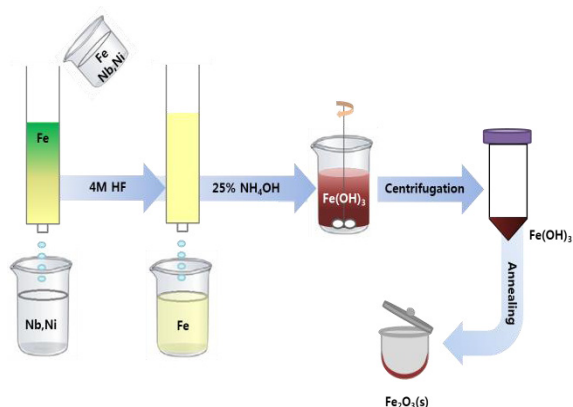


Fig. 1. Schematic illustration of ^{55}Fe chemical separation process. Bio Rad AG 1×8 (100~200 mesh) anion exchange resin was packed in the Econo-Pac Disposable Chromatography Column (20 mL, Bio Rad, U.S.A.).

A separation process of ^{55}Fe is shown in Fig. 1. When the solution with those nuclides were passed through the anionic exchange resin column, two nuclides (Fe, Nb) were placed in the column with green color which is correspond to Fe element. To collect the Fe nuclide, 4 M HF was put into the column. Fe-oxalate anion bonds were break by adding HF, the effluent was collected in a 200 mL beaker. The amount of Fe standard solutions were changed from 10 mg to 20 mg, 30 mg, 40 mg and 50 mg, respectively. In the case of gravimetric analysis, the chemical yields were $95.7 \pm 3\%$, $93.2 \pm 1\%$, $89.7 \pm 6\%$, $87.2 \pm 1\%$ and $84.8 \pm 8\%$. When the ICP-AES was used as a tool, the chemical yields were $92.6 \pm 1\%$, $94.7 \pm 3\%$, $88.0 \pm 3\%$, $83.9 \pm 6\%$ and $81.5 \pm 2\%$. When the amount of Fe carrier was increased, the chemical yields were decreased. This results indicate that the resin has its own capacity. In this case, when the amount of anion exchange resin was 10 mL, the maximum ability to capture the Fe moiety is about 20 mg.

Table 1. Calculated chemical Yields for ^{55}Fe samples measured by gravimetric and ICP-AES analysis. The amount of Fe standard solutions were changed from 10 mg to 20 mg, 30 mg, 40 mg and 50 mg, respectively

	Gravimetric Method	ICP-AES Analysis
10 mg	$95.7 \pm 3\%$	$92.6 \pm 1\%$
20 mg	$93.2 \pm 1\%$	$94.7 \pm 3\%$
30 mg	$89.7 \pm 6\%$	$88.0 \pm 3\%$
40 mg	$87.2 \pm 1\%$	$83.9 \pm 6\%$
50 mg	$84.8 \pm 8\%$	$81.5 \pm 2\%$

4. Conclusion

We have carried out comparison of the recovery results of ^{55}Fe by gravimetric method spectrometric (ICP-AES) analysis. To define the recovery result above two techniques, we separated the ^{55}Fe nuclide from the synthetic waste samples by an extraction chromatography. Based on the above results, the gravimetric analytical results are well matched with spectrometric results. To verify those recovery results, further study is still required.

REFERENCES

- [1] C. H. Lee, M. Y. Suh, K. Y. Jee, W. H. Kim, J. Radioanal. Nucl. Chem., 272, 187, 2007.
- [2] C. H. Lee, M. H. Lee, S. H. Han, Y. K. Ha, K-Song, J. Radioanal. Nucl. Chem., 288, 319, 2011.
- [3] C. H. Lee, H. J. Ahn, J. Lee, Y-K. Ha, J-Y Kim, J. Radioanal. Nucl. Chem., 308, 809, 2016.

2017 추계학술논문요약집

발행인 송기찬

발행처 한국방사성폐기물학회

주 소 대전광역시 유성구 대덕대로 989번길 111
한국원자력연구원 원자력성과확산관(E7) 202호

연락처 Tel +82-42-861-5851, +82-42-866-4205
Fax +82-42-861-5852 E-mail krs@krs.or.kr

인쇄처 신진기획인쇄사 Tel +82-42-638-7887

Abstracts of Proceedings of the
Korean Radioactive Waste Society AUTUMN 2017

www.krs.or.kr

KOREAN RADIOACTIVE WASTE SOCIETY



사단 한국방사성폐기물학회
법인 Korean Radioactive Waste Society