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Suggestion of Risk Assessment Methodology for Decommissioning of Nuclear Power Plant

원자력발전소 해체 위험도 평가 방법론 개발

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The decommissioning of nuclear power plants should be prepared by quantitative and qualitative risk assessment. Radiological and non-radiological hazards arising during decommissioning activities must be assessed to ensure the safety of decommissioning workers and the public. Decommissioning experiences by U.S. operators have mainly focused on deterministic risk assessment, which is standardized by the U.S. Nuclear Regulatory commission (NRC) and focuses only on the consequences of risk. However, the International Atomic Energy Agency (IAEA) has suggested an alternative to the deterministic approach, called the risk matrix technique. The risk matrix technique considers both the consequence and likelihood of risk. In this study, decommissioning stages, processes, and activities are organized under a work breakdown structure. Potential accidents in the decommissioning process of NPPs are analyzed using the composite risk matrix to assess both radiological and non-radiological hazards. The levels of risk for all potential accidents considered by U.S. NPP operators who have performed decommissioning were estimated based on their consequences and likelihood of events.

Keywords: Decommissioning, Risk assessment, Risk matrix, Potential accidents, Hazards

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원전 해체를 준비함에 있어 정성적 또는 정량적 위험도 평가는 필수요소이다. 해체 공정간 발생하는 방사선학적 및 비방사선학적 위험요소는 해체 작업자 및 대중의 안전을 보장하기 위해 사전에 평가되어야 한다. 현재 해체 경험이 많은 미국의 기존 사업자들 및 NRC의 경우 위험의 중대성만 평가하는 결정론적 위험도 평가에 집중하고 있다. 하지만 최근 IAEA는 위험도 매트릭스를 활용한 위험도평가를 결정론적 위험도 평가의 대체안으로 제안하고 있다. 따라서 본 연구에서는 위험도평가에 앞서 해체 공정 별 해체 활동을 Risk Breakdown Structure에 맞추어 정리하였고, 미국 20여개 해체 원전에서 해체 공정별 위험도 평가 시행 중 선정한 해체 활동간 잠재적 사고를 해체 활동에 맞게 체계적으로 정리하였다. 그리고 복합 리스크 매트릭스를 개발 및 활용하여 해체 공정간 방사선학적 및 비방사선학적 위험요소의 위험도를 평가하여 정량적으로 수치화 하였다.

중심단어: 원전 해체, 위험도평가, 위험도 매트릭스, 잠재적 사고, 위험요소

1. Introduction

As of June 2017, Kori unit 1 nuclear power plant (NPP) permanently ceased its operation and phased into permanent shutdown. 11 other NPP units in Korea are also planned to be shut down by 2029 as they reach the end of their design lifetime. Once the NPPs permanently shut down, plants go through the decommissioning phase performing spent nuclear fuels removal, coolant water drain, decontamination, and dismantlement. The term 'decommissioning' can be defined as "administrative and technical actions taken to allow the removal of some or all of the regulatory controls from a facility" [1]. The decommissioning process is a long-term project that produces large amounts of various types of waste, and both radiological and non-radiological hazards are in place to endanger the decommissioning workers and publics. This has led regulators and operators to focus on the development of appropriate safety requirements and criteria for the decommissioning to ensure the safety of decommissioning process.

Korean regulation regarding decommissioning of nuclear facilities requires a decommissioning plan based on structural conditions with radiological characteristics of facilities and it should be supported by qualitative and quantitative safety assessment of the plan [2]. However, unlike the U.S. or other countries with decommissioning experience, Korea has never decommissioned commercial NPP unit and, thus, development of risk management of existing hazards through risk assessment on the decommissioning activities must be performed to accomplish safe decommissioning.

Risk assessment should include evaluation of the potential radiological and non-radiological consequences to the public and workers during the planned decommissioning activities and as a result of any credible accidents that might occur during such activities [3]. Risk assessment should allow the operator to identify, evaluate, and build control to minimize the potential risk during planned decommissioning activities and lead operators to succeed in safe decommissioning.

In this study, the risk factors in risk assessments for decommissioning of Nuclear Power Plant, which factors necessary to perform risk assessments, is identified and studied to suggest the risk assessment methodology. The study was based on risk assessment method of Korean Occupational Safety and Health Agency (KOSHA) using the data collected from U.S. Nuclear Regulatory Commission (NRC) and International Atomic Energy Agency (IAEA). The decommissioning activities organized in chronical order under stepwise process and related potential accidents of the activities are identified and classified. These potential accidents are estimated under developed composite risk matrix and result is analyzed.

2. Risk Assessment Methodology

IAEA describes risk assessment methodology with detail in safety report series (SRS) No. 77 [4]. This document focuses on deterministic approach, which mainly considers consequences of risk, to assess the risk in decommissioning activities. IAEA also suggests the risk matrix technique as the alternative to deterministic approach. U.S. department of Energy (DOE) also states deterministic approach as the standard method for risk assessment and if further evaluation is necessary, it can be performed on likelihood of event with method as risk matrix techniques [5].

The deterministic approach is classical approach for safety assessment. The evaluation is performed intuitively using criteria. Once hazards are identified, evaluate the risk of hazards based on the criteria and if it is not qualified, suggest proper safety control to meet requirement for criteria. If deterministic approach is used for risk assessment, the risk criteria is determined by exposure dose of workers and publics. It is case for most of U.S. NPP operators, who have adopted deterministic approach in risk assessment for decommissioning of NPPs. IAEA also applied this approach to DeSa project, which is the project IAEA performed safety assessment on decommissioning of two systems in undisclosed boiling water reactor (BWR). When the deterministic approach is adopted, frequency or the likelihood of hazards is not considered even if risk is normally defined as multiplication of likelihood and consequences. IAEA states that using deterministic approach eliminates the need to perform analysis on likelihood of events and thus, it reasonably simplifies the risk assessment and save costs [4].

IAEA SRS No. 77 describes the deterministic approach in stepwise, however, the risk assessment related reports based on DOE standards generally does not states the detailed process of hazards and initial events identification, and how evaluated potential accidents were determined. Those documents rather focus on the evaluation of final potential accidents with high radiological risk.

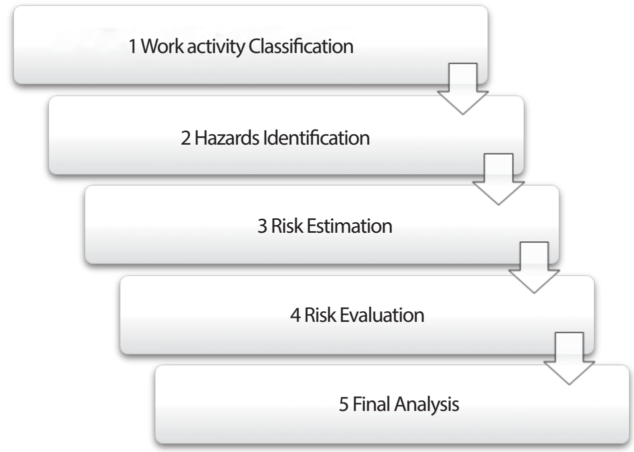

Fig. 1. Step risk factor analysis referenced from risk assessment methodology from KOSHA.

IAEA SRS No.77 also suggests alternative to the deterministic approach, risk assessment using the risk matrix. Unlike deterministic approach, the risk matrix evaluates the risk by considering both consequences and likelihood. The risk matrix technique is widely used on risk assessment in many different areas including nuclear industry. The risk matrix technique is considered as reasonably scientific since it is more simple than probabilistic approach, yet it also gives quantitative results. There are many documents describing the risk matrix technique conceptually for risk assessment of decommissioning like IAEA SRS No. 77, but there are not many documents explaining in detail how risk matrix should be adopted to the actual risk assessment for decommissioning of NPPs. In this study, we have developed the risk matrix based on risk assessment guidelines from KOSHA, using criteria acquired from IAEA, DOE, and KOSHA. Using this matrix risk assessment methodology for decommissioning of NPP is suggested.

3. Risk Assessment Methodology by KOSHA

Risk assessment guideline for general industries in Korea is provided by KOSHA. KOSHA defines Risk as a combination between probability of occurrence (likelihood) and magnitude of consequences for hazards or accident [6]. Risk assessment is defined as process to calculate the magnitude of potential consequences and the probability of these consequences to occur, and if the risk is higher than the regulatory standard, implement control to lower the risk. According to the KOSHA, risk assessment can be performed in 5 steps, which shows similarity to the IAEA safety assessment methodology given in safety standard series [1]. Based on these methodologies, risk assessment can be performed in 5 steps as shown in Fig. 1.

3.1 Decommissioning Activities, Processes & Stages

First step in assessment is to categorize the processes or activities of decommissioning. The activities are organized under processes it belongs in chronical order given by NRC [7]. Any activities that have chance to cause the accidents with consequences must be analyzed to ensure the safety of the decommissioning work.

Hazards and accidents occurring during the decommissioning process after the permanent shutdown are different from the hazards and accidents occurring during the normal operation. Thus, detailed identification of decommissioning process and activities must be performed to identify the hazards and accidents related to the decommissioning activities. NRC divides decommissioning processes into 4 stages for immediate dismantling strategy as shown in Fig. 2 [7].

Initial preparations for permanently ceasing plants operation is occurred at stage 1. This stage is primarily administrative. Main activities of this stages consists selection of decommissioning strategy and preparation in organizational structure (e.g. preparing workers specialized in decommissioning work). Since stage 1 activities are mostly planning, administrative, and organizational, there are only few hazards from these activities that require the risk assessment.

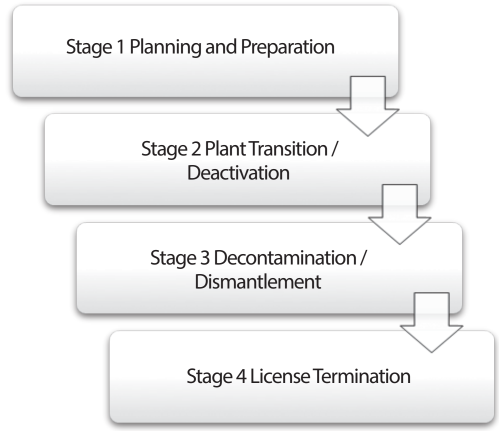

Fig. 2. Decommissioning stage by U.S. NRC [7].

Transfer of spent fuels (SFs) out of the reactor and into the spent fuel pool (SFP), isolation and stabilization of all unnecessary Structures, Systems, and Components (SSCs) are performed in Stage 2 as the transition period from the reactor operation to decommissioning begins. Installment of additional support systems needed for decommissioning activities can be done on this stage. The potential radiological and non-radiological hazards requiring risk assessment may increase depends on the optional activities performed on individual sites.

Main decontamination and dismantlement activities are performed in stage 3. Chemical decontamination of the primary system could possibly reduce radiation dose to workers by reducing level of contamination, but also it can raise the non-radiological hazards for workers due to the use of strong decontamination chemicals. Since every NPPs in US that have completed decommissioning or have started dismantling performed these activities in different ways and at different times in process, the activities varied depends on the strategy each site took to dismantle large component, for example, Rancho Seco had to perform segmentation to meet the land transportation requirement while Trojan had chosen to remove four steam generators, pressurizers, and reactor vessel as whole and varied at Hanford. Hazards for activities at this stage also varies since removing as whole reduces dose to workers significantly.

Table 1. The decommissioning Stages, Processes, and Activities [7]

Ⅰ. Stage	Ⅱ. Process	Ⅲ. Activities
1. Planning & Preparation	1.1 Procure Waste Containers & Special Equipment	1.1.1 Casks, Waste container, and other special equipment procurement
	1.2 Cure SFs storage option	1.2.1 Construction of ISFSI
2. Plant Transition / Deactivation	2.1 Transfer SFs to SFP	2.1.1 SF transfer
		2.1.2 SF cooling
	2.2 Drain & Flush Systems	2.2.1 Drain, Isolation and stabilization of unnecessary SSCs
	2.3 Move or install required support systems	2.3.1 Establishment of site construction power site
		2.3.2 Establishment of monitoring stations
		2.3.3 Establishment of radioanalytical facilities
		2.3.4 Design and fabrication of special shielding and contamination-control envelopes
		2.3.5 Establishment of radiological monitoring stations
		2.3.6 Installments of Radioactive waste Process utilities
3. Decontamination & Dismantlement	3.1 Empty SFP	3.1.1 Transfer SF to storage facilities
	3.2 Decontaminate Large components	3.2.1 Decontaminate reactor coolant system (RCS) and other larger-bore piping
		3.2.2 Decontaminate large component
	3.3 Remove NSSS and RPV Internals	3.3.1 Cut piping and instrumentation line
		3.3.2 Remove Large component intact or segmented
	3.4 Decontaminate External Surface and Structures	3.4.1 Decontaminate rest of components or Structure
	3.5 Dismantlement	3.5.1 Remove and package asbestos insulation
		3.5.2 Remove turbine control oil
		3.5.3 Remove nonradioactive materials, including fuel oil, lubricating oil, 1,1,1-trichlorethane, laboratory chemicals, lead, mercury, paints, and battery acid
		3.5.4 Concrete removal with Impact hammers, saw cutting, and diamond wire cutting
		3.5.5 HEPA filter Removal
	3.6 Minimize, Package, and Transport Decommissioning Waste	3.6.1 Process and ship radioactive materials
		3.6.2 Process and ship mixed wastes to approved disposal sites
4. License Termination	4.1 Soil Remediation	4.1.1 sit remediation and soil decontamination
	4.2 Final status Survey and License Termination	4.2.1 Final Radiation Survey

Table 2. The potential accidents, the risk level, and the risk class

Ⅲ. Activities	Ⅳ. Potential Accidents	Risk Level	Risk Class
1.1.1 Casks, Waste container, and other special equipment procurement	1.1.1.1 Heavy load drop (equipment and casks) (Fort. St. Vrain)	3	Ⅳ
1.2.1 Construction of ISFSI	1.2.1.1 Heavy load drop (equipment and casks) (Fort. St. Vrain)	3	Ⅳ
2.1.1 SF transfer	2.1.1.1 SF handling accident (Trojan, San Onofre 1, Rancho Seco, Humboldt Bay 3, Yankee Rowe)	3	Ⅳ
	2.1.1.2 SF handling accident in SFP (Yankee Rowe, Main Yankee)	3	Ⅳ
	2.1.1.3 SF drop (Haddam Neck)	2	Ⅳ
	2.1.1.4 SF cask drop (San Onofre 1)	4	Ⅳ
	2.1.1.5 SF cask drop in SFP (Haddam Neck, Maine Yankee)	4	Ⅳ
	2.1.1.6 Heavy load drop into SFP (Big Rock Point, Indian Point 1, Humboldt Bay 3, La Crosse)	4	Ⅳ
	2.1.1.7 Fuel failure (Indian Point 1, Shoreham, Dresden Unit 1)	3	Ⅳ
2.1.2 SF cooling	2.1.2.1 Loss of SFP cooling by loss of offsite power (Big Rock Point, Rancho Seco, San Onofre 1)	4	Ⅳ
	2.1.2.2 Loss of SFP cooling (Indian Point 1, La Crosse)	4	Ⅳ
	2.1.2.3 Loss of SFP water by loss of offsite power (La Crosse, Big Rock Point)	4	Ⅳ
	2.1.2.4 Loss of SFP water (Yankee Rowe, La Crosse, Big Rock point, Indian Point 1, Yankee Rowe, Trojan)	4	Ⅳ
	2.1.2.5 Loss of SFP water by earthquake beyond design basis (Haddam Neck)	4	Ⅳ
	2.1.2.6 Loss of SFP decay heat-removal capability (Main Yankee)	4	Ⅳ
	2.1.2.7 Loss of SFP water from pool rupture of unknown origin (Humboldt Bay 3)	4	Ⅳ
	2.1.2.8 Loss of prestressed concrete reactor vessel shielding water (Fort St. Vrain)	4	Ⅳ
	2.1.2.9 Failure of auxiliary electrical systems related to SFP cooling (Dresden 1)	4	Ⅳ
	2.1.2.10 Non-mechanistic loss of cooling and airborne release (Humboldt Bay 3)	4	Ⅳ
	2.1.2.11 SFP drain-down (Dresden 1)	4	Ⅳ
	2.1.2.12 SFP system pipe break (La Crosse)	4	Ⅳ
	2.1.2.13 Inadvertent criticality by misplaced SF in SFP (Maine Yankee)	4	Ⅳ
	2.1.2.14 Criticality by SF rearranged from seismic or other events (Humboldt Bay 3)	4	Ⅳ
2.2.1 Drain, Isolation and stabilization of unnecessary SSCs	2.2.1.1 Leaks and failures in radioactive liquid waste systems (LWS) (Maine Yankee)	8	Ⅲ
	2.2.1.2 Liquid waste tank rupture (Fermi 1, Three Mile Island (TMI) 2, Saxton, Trojan, Humboldt Bay)	12	Ⅱ
	2.2.1.3 Liquid waste discharge pumped to river without sampling (La Crosse)	6	Ⅲ
	2.2.1.4 Condensate storage tank contents pumped into ground during in-service leak test (Dresden 1)	6	Ⅲ
2.3.1 Establishment of site construction power site	2.3.1.1 Nonradioactive materials handling events (Maine Yankee)	9	Ⅲ
2.3.2 Establishment of monitoring stations	2.3.2.1 Nonradioactive materials handling events (Maine Yankee)	9	Ⅲ
2.3.3 Establishment of radioanalytical facilities	2.3.3.1 Nonradioactive materials handling events (Maine Yankee)	9	Ⅲ
2.3.4 Design and fabrication of special shielding and contamination-control envelopes	2.3.4.1 Nonradioactive materials handling events (Maine Yankee)	9	Ⅲ
2.3.5 Establishment of radiological monitoring stations	2.3.5.1 Nonradioactive materials handling events (Maine Yankee)	9	Ⅲ
2.3.6 Installment of Radioactive waste Process utilities	2.3.6.1 Nonradioactive materials handling events (Maine Yankee)	9	Ⅲ
3.1.1 Transfer SF to storage facilities	3.1.1.1 SF handling accident (Trojan, San Onofre 1, Rancho Seco, Humboldt Bay 3, Yankee Rowe)	3	Ⅳ
	3.1.1.2 SF handling accident in SFP (Yankee Rowe, Main Yankee)	3	Ⅳ
	3.1.1.3 SF drop (Haddam Neck)	2	Ⅳ
	3.1.1.4 SF cask drop (San Onofre 1)	4	Ⅳ
	3.1.1.5 SF cask drop in SFP (Haddam Neck, Maine Yankee)	4	Ⅳ

Ⅲ. Activities	Ⅳ. Potential Accidents	Risk Level	Risk Class
3.1.1 Transfer SF to storage facilities	3.1.1.6 Heavy load drop into SFP (Big Rock Point, Indian Point 1, Humboldt Bay 3, La Crosse)	4	Ⅳ
	3.1.1.7 Fuel failure (Indian Point 1, Shoreham, Dresden Unit 1)	3	Ⅳ
	3.1.1.8 Transportation accident (TMI 2, Shoreham, Yankee Rowe)	2	Ⅳ
3.2.1 Decontaminate reactor coolant system (RCS) and other larger-bore piping	3.2.1.1 Decontamination events (Yankee Rowe)	6	Ⅲ
	3.2.1.2 Gross leak or accident during in situ decontamination (Trojan, Saxton)	6	Ⅲ
3.2.2 Decontaminate large component	3.2.2.1 Gross leak or accident during in situ decontamination (Trojan, Saxton)	6	Ⅲ
	3.2.2.2 Decontamination events (Yankee Rowe)	6	Ⅲ
	3.2.2.3 Accidental spraying of concentrated contamination with high-pressure spray (TMI 2)	3	Ⅳ
	3.2.2.4 Concentrated contamination spray (TMI 2)	3	Ⅳ
3.3.1 Cut piping and instrumentation line	3.3.1.1 Contamination release by accidental cutting or breaking of contaminated piping (TMI 2)	8	Ⅲ
3.3.2 Remove Large component intact or segmented	3.3.2.1 Loss of engineering controls during dismantlement of reactor cavity (Big Rock Point)	8	Ⅲ
	3.3.2.2 Contamination release during dismantlement of main coolant system loop (Yankee)	8	Ⅲ
	3.3.2.3 Dismantlement of RCS and safety injecting piping without or with loss of local engineering controls (Saxton)	8	Ⅲ
	3.3.2.4 Materials handling events (Yankee Rowe)	8	Ⅲ
	3.3.2.5 Steam Generator load drop inside or outside of containment (Fort St. Vrain, Trojan)	1	Ⅳ
	3.3.2.6 Dropping the reactor pressure vessel (pathfinder)	1	Ⅳ
3.4.1 Decontaminate rest of components or Structure	3.4.1.1 Gross leak or accident during in situ decontamination (Trojan, Saxton)	6	Ⅲ
	3.4.1.2 Decontamination events (Yankee Rowe)	6	Ⅲ
	3.4.1.3 Accidental spraying of concentrated contamination with high-pressure spray (TMI 2)	3	Ⅳ
	3.4.1.4 Concentrated contamination spray (TMI 2)	3	Ⅳ
	3.4.1.5 Spent resin handling accident (Haddam Neck, Saxton, Maine Yankee, TMI 2, Trojan)	15	Ⅱ
3.5.1 Remove and package asbestos insulation	3.5.1.1 Nonradioactive materials handling events (Yankee Rowe)	2	Ⅳ
	3.5.1.2 Packaging events (Yankee Rowe)	2	Ⅳ
3.5.2 Remove turbine control oil	3.5.2.1 Explosion of large fuel-oil storage tanks (Humboldt Bay 3, Trojan)	4	Ⅳ
3.5.3 Remove nonradioactive materials, including fuel oil, lubricating oil, 1,1,1-tricholorethane, laboratory chemicals, lead, mercury, paints, and battery acid	3.5.3.1 Nonradioactive materials handling events (Yankee Rowe)	20	Ⅰ
	3.5.3.2 Packaging events (Yankee Rowe)	12	Ⅱ
3.5.4 Concrete removal with Impact hammers, saw cutting, and diamond wire cutting	3.5.4.1 Absence of blasting mat during removal of activated concrete (Trojan)	3	Ⅳ
	3.5.4.2 Dropping of concrete rubble (Fort St. Vrain, Trojan)	12	Ⅱ
3.5.5 HEPA filter Removal	3.5.5.1 Rupture of contamination-control envelope (Shoreham)	4	Ⅳ
	3.5.5.2 HEPA filter failure (TMI 2)	4	Ⅳ
	3.5.5.3 Loss of integrity of portable filtered ventilation enclosure (Trojan)	12	Ⅱ
	3.5.5.4 Pressure-surge damage to filters during blasting of activated concrete bio-shield (Trojan)	4	Ⅳ
	3.5.5.5 Temporary loss of local airborne contamination control during blasting or scarfing of contaminated concrete surfaces with jackhammer (Trojan)	8	Ⅲ
	3.5.5.6 Loss of contamination-control envelope during oxyacetylene cutting of the reactor-vessel shell (Trojan)	8	Ⅲ
3.6.1 Process and ship radioactive materials	3.6.1.1 Materials handling events (Yankee Rowe)	4	Ⅳ
	3.6.1.2 Packaging events (Yankee Rowe)	12	Ⅱ
	3.6.1.3 Transportation accident (TMI 2, Shoreham, Yankee Rowe)	3	Ⅳ
3.6.2 Process and ship mixed wastes to approved disposal sites	3.6.2.1 Nonradioactive materials handling events (Yankee Rowe)	4	Ⅳ
	3.6.2.2 Packaging events (Yankee Rowe)	4	Ⅳ
	3.6.2.3 Transportation accident (TMI 2, Shoreham, Yankee Rowe)	3	Ⅳ
4.1.1 Soil Remediation	4.1.1.1 Decontamination Events (Yankee Rowe)	6	Ⅲ
4.2.1 Final status Survey and License Termination	4.2.1.1 Materials handling events (Yankee Rowe)	1	Ⅳ

Table 3. The level of radiological consequences

Radiological Consequences	Level of Exposure	Consequence Level
Insignificant	< 0.1 mSv onsite < 0.01 mSv offsite	1
Minor Exposure	0.1 - 1 mSv onsite 0.01 - 0.1 mSv offsite	2
Moderate Exposure Under Dose Limit	1 - 20 mSv onsite 0.1 - 1 mSv offsite	3
Major Exposure Above Dose Limit	20 - 50 mSv onsite 1 - 5 mSv offsite	4
Critical Exposure	> 50 mSv onsite > 5 mSv offsite	5

Table 4. The level of non-radiological consequences

Non-radiological Consequences	Duration of Treatment	Consequence Level
Insignificant Injury (no treatment required)	No treatment	1
Minor Injury	< 1 month	2
Moderate Injury	1 – 6 months	3
Serious (Major) Injury	6 months – 1 year	4
Fatality (Long-term illness or Death)	> 1 year or Fatal	5

Table 5. The level of likelihood

Likelihood	Level of Likelihood	Likelihood Level
Highly Unlikely	< 20%	1
Reasonably Likely	20 – 40%	2
Even Chance	40 – 60%	3
Highly Likely	60 – 80%	4
Almost Certain	> 80%	5

Stage 4 is License termination. Decommissioning activities at this stage are final site characterization, final radiation survey, submission of final license termination plan, and possibly site remediation and soil decontamination. Mainly administrative activities are performed other than site remediation and soil decontamination, thus, the risk for workers and publics are relatively low.

NRC also provides a detailed decommissioning activities list, which is obtained from sites that are recently completed the decommissioning activities or began the decommissioning activities. The list is categorized by construction, decontamination, contamination control, dismantlement, removal of the reactor vessel & internals, other large components and systems, radioactive waste management, and spent fuel etc. [7]. This list was recategorized using risk breakdown structure (RBS) method under each process in chronical order, shown in Table 1. Level Ⅰ is the decommissioning stage in four different steps and level Ⅱ is the decommissioning process under each stage. Lastly, level Ⅲ is the decommissioning activities that performed in each process.

3.2 Potential Accidents

The identification of hazards for individual activities is performed in this step. However, in this study, the potential accidents that were considered by twenty of U.S. decommissioned plants were identified as hazards. These accidents are possible NRC has utilized their research efforts, industry-related documents, and licensing-basis documents such as post-shutdown decommissioning activity reports (PSDARs), final safety analysis reports (FSARs), environmental assessments (EAs), or environmental impact statements (EISs) to obtain a list of potential accidents and their consequences from twenty of U.S. plants that are decommissioned or in decommissioning. NRC has categorized these accidents into 5 headings, which are fuel related accidents, accidents involving radioactive materials (nonfuel related), accidents initiated in external events, offsite transportation related accidents, and hazardous non-radiological chemical events [7].

NRC has listed all accidents that licensees have included even if licensees did not evaluate in detail of its consequences. Most licensees did not describe in detail for potential

Likehood	Level	Level					
Radiological Consequences			Insignificant	Minor Exposure	Moderate Exposure Under Dose Limit	Major Exposure Above Dose Limit	Critical Exposure
Non-Radiological Consequences			Insignificant Injury (no treatment required)	Minor Injury	Moderate Injury	Serious (Major) Injury	Fatality (Long-term illness or Death)
			< 0.1 mSv onsite < 0.01 mSv offsite	0.1 -1 mSv onsite 0.01 -0.1 mSv offsite	1 -20 mSv onsite 0.1-1 mSv offsite	20 -50 mSv onsite 1-5 mSv offsite	> 50 mSv onsite > 5 mSv offsite
			No treatment	<1 month	1 month -6 month	6 month-1 year	> 1 year or Fatal
		Level	1	2	3	4	5
Highly Unlikely	< 10%	1	1	2	3	4	5
Reasonably Likely	10% - 25%	2	2	4	6	8	10
Even Chance	25% - 50%	3	3	6	9	12	15
Highly Likely	50% - 75%	4	4	8	12	16	20
Almost Certain	> 75%	5	5	10	15	20	25

Fig. 3. Risk matrix of radiological and non-radiological hazards for decommissioning activities.

accident, for example, most document discussed the analysis of release of liquid radioactive waste did not indicate the cause of accident. Not to mention, many of these potential accidents could fall under one or more different categories [7]. However, many of the potential accidents that licensees considered were fell under the low risk criteria due to result in insignificant consequences.

The potential accidents listed in NRC documents are reorganized under the decommissioning activities, shown in Table 2. The potential accidents are categorized under Level Ⅳ of RBS.

3.3 Development of Risk Matrix

Risk is defined as a measure of the probability for an accident to happen and of the potential severity of the consequences. Both probability and potential severity can be measured in risk matrix, which is widely used as in the industry. IAEA also suggests the risk matrix as the measure for risk assessment in both quantitative and qualitative measure [1].

First step in development of risk matrix was to determine the criteria for consequences. Decommissioning activities inherently own both radiological and non-radiological hazards, two different criteria for each consequence are must be applied to the risk matrix. Both offsite (publics) exposure and onsite (workers) exposure are considered for the level of radiological consequence. 5 different ranges of exposure criteria are acquired through modifying offsite criteria from IAEA Safety Series No. 77 Annex 1 Part A [8], and onsite criteria from U.S. DOE order [9], shown in Table 3. On the other hands, the duration of treatment for the injury is a commonly used criteria for non-radiological consequences as explained in KOSHA guideline [10], shown in Table 4. The criteria for likelihood is simply implemented from KOSHA guideline, which separates the level of likelihood in percentage from 0 to 100%, shown in Table 5. The risk matrix for both radiological and non-radiological hazards is developed combining above all the criteria, shown in Fig. 3.

The risk levels are quantitatively measured by the multiplication of the level of consequence and level of likelihood. Then, the risk levels are classified into four different class from risk class Ⅰ to risk class Ⅳ, shown in Table 6. The risk classification is modified from classification in IAEA SRS No. 77 [5] and in KOSHA risk assessment guideline [10]. KOSHA classifies risks into 3 classes while IAEA classifies into 4 different classes. Since the IAEA classification has considered radiological risks, it was considered as more suitable classification for this project. IAEA

Table 6. Risk classification

Risk class	Criteria	Remarks
Risk class Ⅵ (Blue)	Risk level 1 – 5	Do not require any safety measure
Risk class Ⅲ (Yellow)	Risk level 6 – 10	Safety management program by operator required
Risk class Ⅱ (Orange)	Risk level 12 – 16	1 independent safety measure required
Risk class Ⅰ (Red)	Risk level 20 – 25	2 independent safety measure required

classification was based on 4×4 risk matrix, so it has been modified for 5×5 risk matrix level. The four classes are colored in four different color; risk class Ⅳ as blue (risk level 1-5) , risk class Ⅲ as yellow (risk level 6-10), risk class Ⅱ as orange (risk level 12-16), and risk class Ⅰ as red (risk level 20-25). This classification is adopted from Korean national emergency response system. Each of four risk classes are given with different requirement to mitigate the risk. The risk class Ⅰ and Ⅱ requires independent safety measure to mitigate the risk under operational level and independent complete safety measure is defined as the safety measure independent from the potential accidents and other complete safety measure, which detects the accident and suitably mitigate the accident under the risk class Ⅱ criteria.

3.4 Risk Estimation

Once Risk Matrix is ready, risk level of all potential accidents is estimated using reference data. The consequence levels for the radiological potential accidents are acquired through NRC research data and U.S. licensee documents (PSDARs, EAs, ERs, or EISs). The highest offsite doses calculated for potential accidents were given to determine the consequence level for individual potential accidents [11, 12]. The likelihood level of radiological potential accidents were not quantitatively given in database. However, the documents mention likelihood level in qualitatively (often, unlikely, highly unlikely etc.). These can be matched into risk matrix likelihood level and becomes quantitative through brainstorming method. The consequence levels for the non-radiological potential accidents are acquired through KOSHA. KOSHA has database for risk assessment on common industrial accidents. It gives the consequence level and likelihood level for individual non-radiological potential accident. It is stated by IAEA that non-radiological hazards for decommissioning of NPPs are similar to hazards found in decommissioning of other chemical factories or industrial buildings.

The level of likelihood for all the accidents is performed through qualitative analysis. Many accident analyses done by NRC performed the qualitative analysis on the likelihood of accident, for example most of the spent fuel related accidents were analyzed to have very low likelihood of occurrence [11] while the decontamination related accidents were analyzed to have moderate likelihood of occurrence [13]. The levels of consequence and likelihood are multiplied to acquire the risk level of individual potential accidents, shown in Table 2.

The risk levels for all potential accidents are also analyzed to study distribution of the risk level for the potential accidents in decommissioning, shown in Fig. 4. Overall, 80 potential accidents in the decommissioning activities were analyzed and it shows that 62.5% of the potential accidents were classified as the risk class Ⅳ, while 28.75% as the risk class Ⅲ, 7.5% as the risk class Ⅱ, and 1.25 (1 case) as the risk class Ⅰ. The potential accident with the highest risk level was the accident while removing nonradioactive materials, including fuel oil, lubricating oil, 1,1,1-trichlorethane, laboratory chemicals, lead, mercury, paints, and battery acid. This accident was classified as risk class Ⅰ with the risk level of 20, 5 for level of consequence and 4 for level of likelihood. The result showed that accidents with high risk is more likely to be related to non-radiological hazards than to be related to radiological hazards. The distribution of the result was categorized in each of decommissioning process (RBS level Ⅱ) and decommissioning stage (RBS level Ⅰ). As shown in Fig. 5, the decommissioning process 2.1 (transfer of SF to SFP) had highest number of accidents in overall with all of them in risk class Ⅳ, due to many accident U.S. operators considered were SF related, yet, they have concluded that the SF related accident will highly unlikely to occur during the decommissioning of NPPs. Except decommissioning process 2.1, the decommissioning process 3.5 (dismantlement) clearly has the highest risk because not only it has the accident ranked in risk class Ⅰ, but also has the most numbers of accidents in the risk class Ⅱ. Again, most of the activities performed in the decommissioning process 3.5 (dismantlement) process is related to non-radiological hazards, which shows the significant of the non-radiological hazards in the decommissioning process. Furthermore, if this distribution data is categorized in terms of decommissioning stage as shown in Fig. 6, It is certain that which decommissioning stage will have the most risk to workers or publics. As it was mentioned above in section 2, the stage 1 and 4 is the phase where most administrative work with low risk are performed, while stage 2 and 3 is phase where more active decommissioning activities are performed.

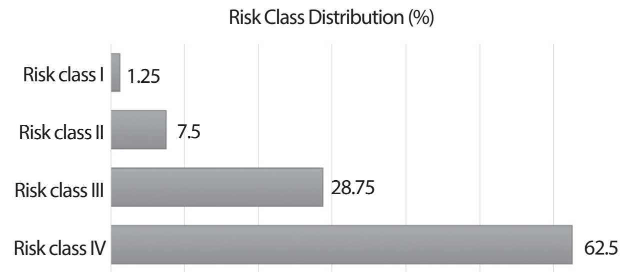

Fig. 4. Potential accident risk class distribution.

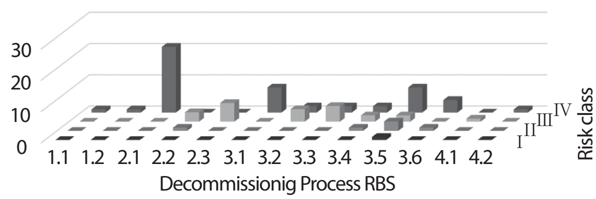

Fig. 5. Distribution of the risk in the decommissioning process.

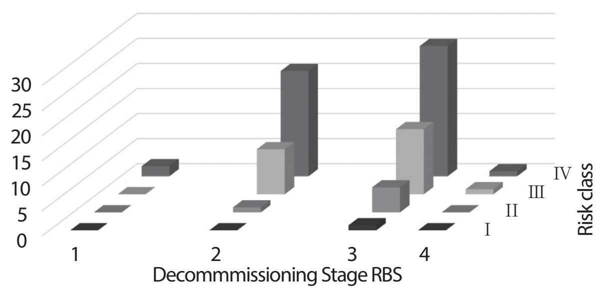

Fig. 6. Distribution of the risk in the decommissioning stage.

4. Conclusion

The risk assessment was performed on the potential accidents considered by U.S decommissioned NPP operators, on basis of the developed risk matrix and KOSHA risk assessment methodology. All of the decommissioning activities were recategorized under each decommissioning process and related potential accidents were reorganized under each decommissioning activities. The criteria for radiological hazard and non-radiological hazards were studied through existing criteria from IAEA, DOE, and KOSHA. Then, new criteria for both radiological and non-radiological hazards consist of five rank were developed. The composite risk matrix was developed using developed criteria and level of likelihood used in KOSHA. All risks of the potential accidents already categorized were estimated using risk matrix all offsite dose exposure calculation result were acquired through NUREG Guide, PSDAR, EIS of decommissioned NPPs. The risk estimation showed that most of the risk of potential accidents were classified under risk class Ⅳ and smaller as it goes to risk class Ⅰ. the risk distribution can be explained that some of the radionuclide inventory significantly decrease shortly following shutdown, and then continues to decrease at entire decommissioning period, which the radiological hazard would be much lower than the operational period. The U.S. NRC standardized the deterministic risk assessment, which some of the potential accidents U.S. operators considered did not include the initiating event or how they have picked the potential accidents. Yet, U.S. NRC states that these potential accidents are the worst potential accidents, which means they did not consider any accident that has higher level of consequences than these accidents have. Therefore, the result of the potential accidents risk estimation gives conservative values for the level of risk in decommissioning process of NPPs.

As the result shows, the potential accidents related to non-radiological hazards have higher level of risk than the potential accidents related to radiological hazards. Afterward, risk assessment in decommissioning of NPPs can be focused more on to non-radiological hazards, to develop more detailed assessment.

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