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A Probability Risk Assessment for MACSTOR/KN-400 During An Air Inlet Blockage Accident Sequence

Jung Hyun Ryu and Moosung Jae *
Department of Nuclear Engineering, Hanyang University 17 Haengdang-Dong, Sungdong-Gu
Seoul, Korea

Chanwoo Jeong
Korea Institute of Nuclear Safety, P.O BOX 114, Yusong-Gu
Taejon, Korea

Abstract

The safety assessment framework for evaluating the spent fuel dry storage facility during the air inlet blockage accident composing of three phases has been established and applied to an interim storage system. They include the analysis of the failure probability of a basket and a cylinder, the accident modeling of spent fuel dry storage facility and the accident consequence assessments. The first phase of the analysis calculated the module failure probability by modeling of the basket and the cylinder, which is major element for containing radioactive substances. The second phase includes a modeling of spent fuel dry storage facility. At this phase, the probability that radioactive substances are released to outside when the initial event happens has been calculated by the construction of the event tree methods against a various elements which affects the air inlet blockage accident. At the third phase of releasing radioactive substances, the radiation damage to affect neighborhood and storage facility worker using MACCS2 code has been evaluated quantitatively.

Key Words: probabilistic risk assessment, MACSTOR/KN-400, basket failure rate, radiation material release, CCDF

I. Introduction

Since the commercial operation of Kori #1, twenty nuclear power plants in total are under operation in Korea. The capacity of the spent fuel storage pool, which each power plant possesses has been most tightly packed according to the present condition that spent fuels are produced through continued plant operation. Therefore, a long-term storage countermeasure is required. The interim spent fuel storage is considered as the facility to store safely and manage spent fuel recycling and eternal disposals. Currently, several types of interim spent fuel storage facilities are studied internationally for design, construction, and operation. Although the design concepts of these facilities are all different, the systems are designed to ensure the appropriate storage safety during design life. In general, spent fuels are transferred to the interim storage facility after initial keeping in the pool of the reactor building. Since the radionuclide radioactive decay heat is decreased gradually during this initial keeping, it may take the sufficient time reaching the management. Therefore, the treatment and storage facility involving spent fuels can preserve the stable safety without a complex system or initial automatic defensive system.

During design lifetime, issues which greatly affect safety of spent fuel storage include nuclear criticality, removal of radioactive decay heat, and shielding of radiation. The evaluation of these elements is major issue regarding a safety assessment of spent fuel storage facility. There are two way of safety assessment related nuclear facility. They are a deterministic and a probabilistic approach. In a deterministic one, the difference between calculated value against any parameter and limit value defined design basis is evaluated, while in a probabilistic methodology allowable risk ratio or frequency to estimate consequence caused the failure probability is evaluated [1]. Up to now most safety assessment of spent fuel storage facility has been used the deterministic methodology. Hence, in this study, the safety for the reference facility, MACSTOR/KN-400, which will be constructed at Wolsung NPPs has been analyzed quantitatively [2].

II. A Frameworks for Assessing Interim Spent Fuel Storage

The framework developed in this study for assessing the safety of spent fuel dry storage facility consist of three phases. In first phase, the failure probability of a basket and a cylinder, which is the major element of containing radioactive substances, is modeled. In second phase, the accidents of spent fuel dry storage facility are modeled. Finally, accident consequence analysis using MACCS2 code is performed. For the spent fuel dry storage facilities, the studies have included only the accident modeling of spent fuel dry storage facility and are performed with only deterministic analysis. PSA (Probability Safety Analysis) is a methodology that analyzes the risk of nuclear power plant by probability. In this study the safety analysis of spent fuel dry storage facility using simplified level 1, 2, and 3 PSA has been performed.

At the first phase the destruction probability of a basket and a cylinder is calculated by estimating CDF(Core Damage Frequency) using one of level 1 PSA technique by modeling a basket and a cylinder containing radioactive substance. The second phase involves a modeling of spent fuel dry storage facility. In this phase, quantitative internal event and the external event that may occur in the dry storage facility is analyzed. And then, probability that each scenario can occur is calculated. In this phase, through the result of the first phase we can calculate more accurate probability of occurring accident. The third phase include an analysis for the consequence. Hence, we can quantitatively assess radiation damage affecting neighborhood and storage facility work from consideration of circumference area, population, and meteorological data. Figure. 1 shows the frameworks which are developed in this study [3].

Figure 1. A framework for assessing probabilistic safety of spent fuel dry storage facility

III. MODELING METHODS

III.A Modeling basket failure probability

The first phase in the framework focus on the quantification of the failure probabilities of both a basket and cylinder which contains radioactive substances. They are important data for assessing the interim storage facility. This method is called by stress-strength analysis model. The method calculates the failure probability that stress exceeds strength of the materials [5].

Stresses considered in the air inlet blockage accident are both load stress and thermal stress. Occurring at relatively lower temperature, it is assumed that total stress is linear combination of load stress and thermal stress. Hence, total stress are expressed as the following Eqs. (1).

$$d_{total} = d_{load} + d_{thermal} \qquad (1)$$

The load stress is described by the Eqs. (2) while thermal stress is the Eqs. (3) [6].

$$d_{load} = \frac{pR}{2t} \tag{2}$$

p : $pressure\ [MPa]$
R : $radius\ of\ basket\ [m]$
t : $thickness\ of\ basket\ [m]$

$$d_z = \frac{uEaq'''}{8(1-u)k}R^2 - Ea\left[\Delta T - \frac{q'''R^2}{4k}\right] \tag{3}$$

R : $radius\ of\ basket\ [m]$
u : $poisson\ ratio$
E : $modulus\ of\ elasticity\ [MPa]$
a : $coefficient\ of\ thermal\ expansion\ [mm/mK]$
q''' : $volumetric\ heat\ generation\ rate\ [W/m^3]$
k : $thermal\ conductivity\ [W/mK]$

Finally, total stress is obtained from the Eqs. (4)

$$d_{total} = \frac{pR}{2t} + \frac{uEaq'''}{8(1-u)k}R^2 - Ea\left[\Delta T - \frac{q'''R^2}{4k}\right] \tag{4}$$

Figure. 2 shows the simulation results. The distribution of total stress is assumed as a normal distribution with a mean of 50.49 MPa and a standard deviation of 7.97 MPa, respectively.

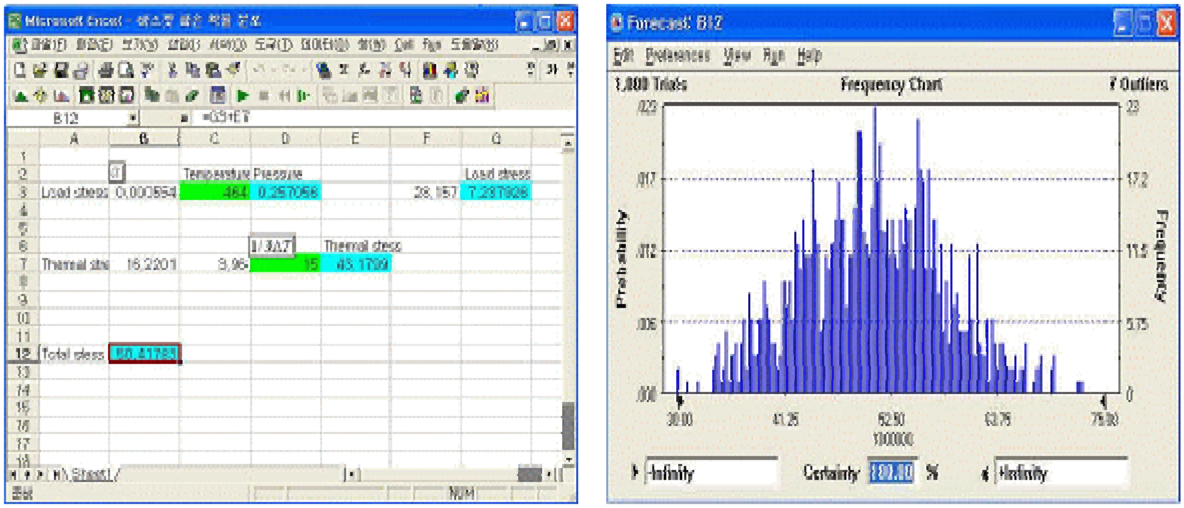


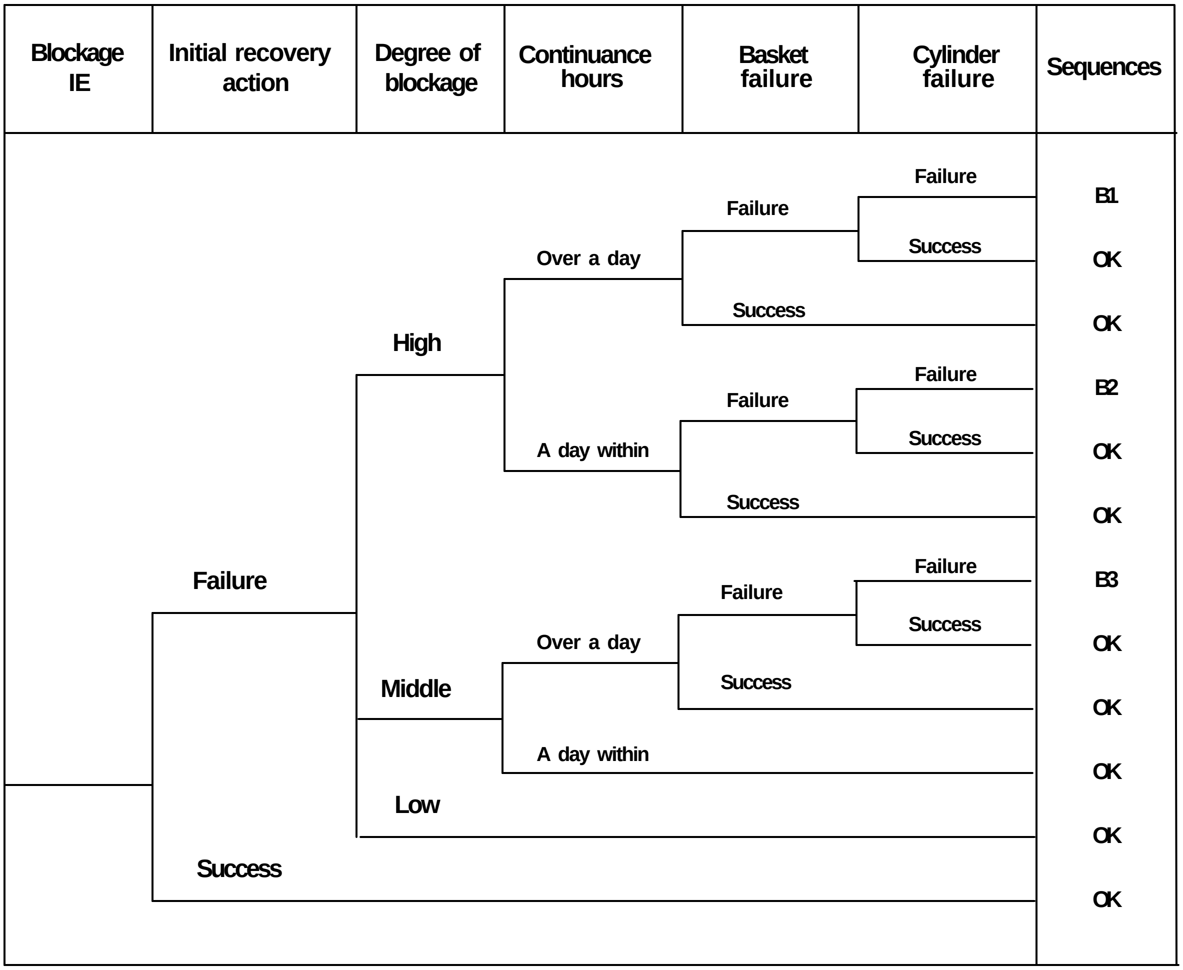
Figure 2. Uncertainty calculation process for total stress

Suppose the allowable stress is assumed to be a value of 0.5 as a yield stress based on ASME code while for stainless steel 304 the distribution is defined as a normal distribution with a mean of 85 MPa and a standard deviation of 8.5 MPa, respectively [7].

III.B System modeling

The elements of probable accident scenarios include a cask drop, a flood, a fire and an explosion, a lightning, an earthquake, a loss of shielding, an adiabatic heat-up, a tornado and a missile attack. In this study it is assumed that the air inlet blockage accident is one of important accident scenarios. Table 1 is an event tree describing the air inlet blockage accident scenario. Possible methods are surveyed to analyze this problem. Since each method can have some merit or weak points, is concluded that use of ETA and FTA is appropriate because the dry storage facility is simple and its accident occurs in order. Therefore, the ETA is used for modeling the air inlet blockage accident sequence to result in obtaining the probability of radiation detection failure.

Table 1. Event tree of air inlet blockage accident



The initial events of this accident may occur by an earthquake, a flood, a tornado or a factitious behavior. In this case the probability of initial recovery failure, degree of blockage, duration time, failure probability of a basket and a cylinder need to be quantified. The data used for quantification has been cited from an existing PSA report and IEEE data. The failure probability for the basket and the cylinder is derived from the results of the first phase called reliability physics.

Table 2. Data of air inlet blockage accident

Heading	Mean	Standard deviation
Initial accident frequency	1 time/year	-
Initial recovery failure	3.0×10^{-1}	1.0×10^{-2}
Degree of blockage	2.0×10^{-1}	10×10^{-2}
	3.0×10^{-1}	
	5.0×10^{-1}	
Duration time	8.0×10^{-1}	1.0×10^{-2}

III.C Consequence modeling

One of PRA (Probability Risk Analysis) levels evaluate the consequence effects when radioactive substances are released is performed. Although radioactive substances are released by design character for the dry storage facility, inside radioactive substances are not leaked out suddenly into outside because the inside pressure of a basket and a cylinder is not high. However, consequence analysis is need because radioactive substances can be leaked. In this study, MACCS 2 code for consequence analysis is used. MACCS2 (MELCOR Accident Consequence Code System) is an accident consequence analysis code that evaluates the effect caused by radioisotope released. MACCS2 code is developed by SNL (Sandia National Laboratory) to analyze severe accident risk of NRC (Nuclear Regulatory Commission). This code is used all over the world, benchmark study is being progressed to solve problem that model is complicate and input data is huge, and international user group are composed. A major input data of MACCS2 code are circumference area data, meteorological data, population data of reference system, and release fraction, inventory of radioisotope inside a basket, etc. In this study CCDF, average individual risk and whole body dose are calculated. Figure. 3 shows an environment effect assessment system.

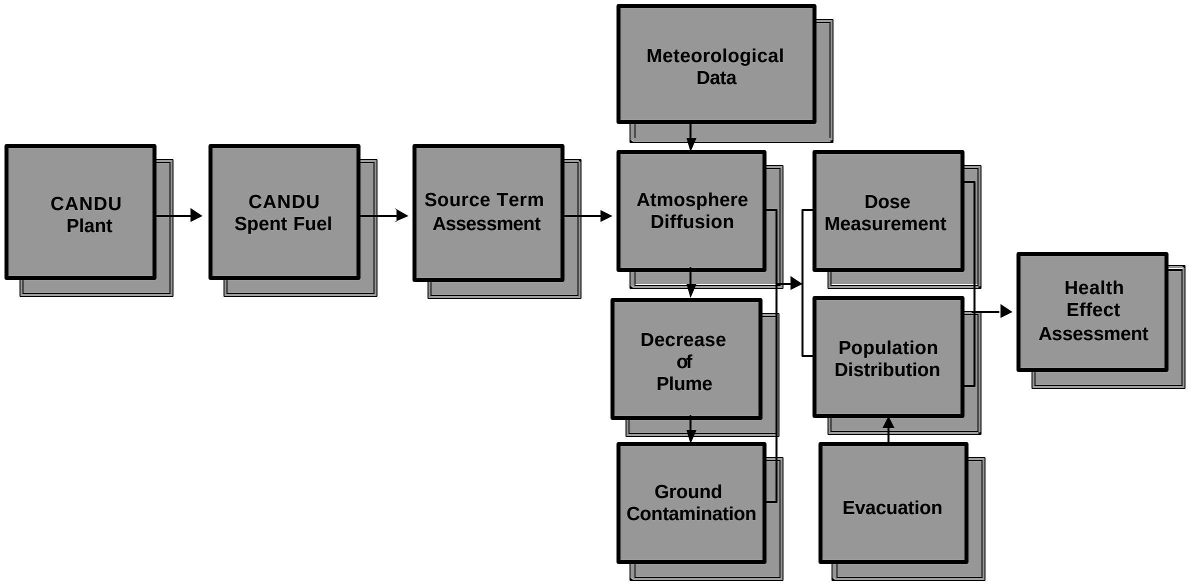


Figure 3. Module of accident consequence assessment

Inventory of radioisotope inside a basket which is most important input data in this study is defined as spent fuel cooled in the pool for six years, its burnup is 7,800 MWD/MTU that is design basis fuel. Source strength of MACSTOR/KN-400 is calculated by ORIGEN-2 code. Table 3 is an inventory of radionuclide included in spent fuel that is a radiation source of containment.

Table 3. Source term inventory

Nuclide	Amount of nuclide (Bq/bundle)	Total amount
H-3	6.96×10^{10}	1.67×10^{15}
Kr-85	1.10×10^{12}	2.64×10^{16}
Sr-90	3.45×10^{1}	8.28×10^{5}
Y-91	1.06×10^{13}	2.54×10^{17}
Zr-95	2.79×10^{3}	6.70×10^{7}
Nb-95	3.62×10^{4}	8.69×10^{8}
Ru-103	8.03×10^{4}	1.99×10^{9}
Ru-106	1.23×10^{-2}	2.95×10^{6}
Ag-110m	2.31×10^{12}	5.54×10^{16}
Sb-124	7.03×10^{-1}	1.69×10^{6}
Sb-125	4.85×10^{11}	1.16×10^{16}
I-129	5.51×10^{6}	1.32×10^{11}
Cs-134	1.39×10^{12}	3.34×10^{16}

Cs-137	1.58×10^{13}	3.79×10^{17}
Ce-141	3.85×10^{6}	9.24×10^{10}
Ce-144	1.70×10^{12}	4.08×10^{16}
Pm-147	1.06×10^{13}	2.54×10^{17}
Eu-152	2.32×10^{8}	5.57×10^{12}
Eu-154	4.81×10^{11}	1.15×10^{16}
U-234	1.88×10^{8}	4.51×10^{12}
U-235	3.07×10^{6}	7.37×10^{10}
U-238	2.31×10^{8}	5.54×10^{12}
Np-237	1.86×10^{7}	4.46×10^{11}
Np-239	3.69×10^{8}	8.86×10^{12}
Pu-238	6.25×10^{10}	1.50×10^{15}
Pu-239	1.18×10^{11}	2.83×10^{15}
Pu-240	1.64×10^{11}	3.94×10^{15}
Pu-241	1.22×10^{13}	2.93×10^{17}
Pu-242	1.67×10^{8}	4.01×10^{12}
Pu-244	7.40×10^{0}	1.78×10^{5}
Am-241	1.41×10^{11}	3.38×10^{15}
Am-243	3.69×10^{8}	8.86×10^{12}
Cm-242	2.98×10^{18}	7.15×10^{22}
Cm-244	9.95×10^{9}	2.39×10^{14}

It is expected that temperature of fuel during the period of storage and accident is lower than 200°C. If it is assumed that radioactive substances are released by destruction of a basket and a cylinder, radionuclide will almost not be released. Only Kr-85 and H-3 expected to have gas phase in a basket will be released. Therefore, release fraction used major input data in MACCS2 code is about zero except inert gas group. However, in this study we used release fraction of SLOCA accident corresponding STC 3, which is used in LEVEL 3 of nuclear power plant to apply more conservatively. Table 4 shows their release fractions [8].

Table 4. Release fraction of a sequence, Small LOCA

	Xe/Kr	I	Cs	Te	Sr	Ru	La	Ce	Ba
Release Frequency	1.0	7.7×10^{-3}	8.5×10^{-3}	5.2×10^{-3}	2.6×10^{-4}	1.2×10^{-6}	4.7×10^{-6}	2.5×10^{-8}	1.3×10^{-4}

In this study it is assumed that duration time is 0 sec, release time is 86,400 seconds, release height is ground release which is a conservative value. It used meteorological data arranged from the weather tower of nuclear power plant neighborhood during 1 year against wind velocity, wind direction, rainfall, atmosphere stability of 8,760 things, To calculate dispersion in atmosphere it is divided 10 lattice up to radius 80 km with 16 azimuth in the center of nuclear power plant site, and considered population in each lattice. Table 5 is population at Wolsung neighborhood.

Table 5. Population distribution of Wolsung site

radius (km)	0-1.6	1.6-3.2	3.2-4.8	4.8-6.4	8-16	16-32	32-48	48-64	64-80	Total
Wolsung	848	2,762	2,449	2,338	59,288	1,090499	558,491	639,866	4,492820	6,852,099

IV. RESULTS

IV.A Basket failure probability modeling

The failure probability of a basket caused by two probability distributions can be obtained from two distribution intersections. In this case distribution of stress and distribution of yield stress which are assumed to be normal distributions are used as competing random variables. In other words, a variable, X is a random variable representing stress while another variable Y is a random variable representing material strength. The random variable Z is then defined as below Eqs. (5).

$$Z = X - Y \qquad (5)$$

The mean of new random variable Z is obtained from the following Eqs. (6).

$$E[Z] = \mu_x - \mu_y \qquad (6)$$

And standard deviation is Eqs. (7)

$$d_z = \sqrt{d_x^{\ 2} + d_y^{\ 2}} \qquad (7)$$

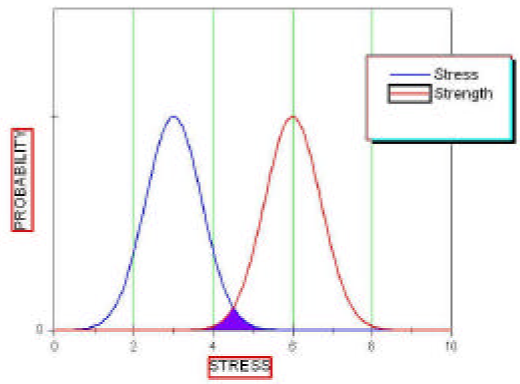


Figure 4. Reliability physic model

Hence, total intersection area can be calculated using the following Eqs. (8).

$$R = \Pr(Z > 0) = \int_0^{\infty} \frac{1}{d_h\sqrt{2p}} \exp[\frac{-(x - E(z))^2}{2d_x^{\ 2}}]dx = \Phi(\frac{m_x - m_y}{\sqrt{d_x^{\ 2} + d_y^{\ 2}}}) \qquad (8)$$

Thus, the probability of basket failure results in a value of 1.12×10^{-3}.

$$P(Y > X) = P(Z > \frac{m_x - m_y}{\sqrt{d_x^{\ 2} + d_y^{\ 2}}}) = 1.12\times10^{-3} \qquad (9)$$

IV.B System modeling

Inlet blockage accident can have many different initial events because it may occur with earthquakes, floods, tornados and missile attacks, factitious behaviors, etc. The radioactive substances release accident by inlet blockage is an event that radioactive substances leak out by containment failure of both a basket and a cylinder. Although this accident can occur for MACSTOR/KN-400, which is a reference system, the result release outside may not be severe because pressure in a basket is low. Along with RISKMAN code, it uses 1,500 data samples in Monte Carlo sampling from the given average and standard variation, and Figure. 5 show the processes.

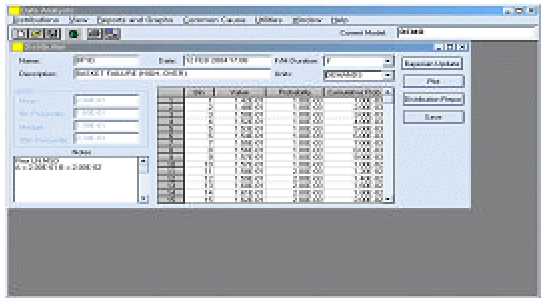


Figure 5. Input data inserting sequence

The elements affecting the air inlet blockage accident are initial recovery behavior, degree of blockage, duration time, etc. It uses a different probability because the probability occurring by each case affects each other. Using RISKMAN code through this consideration, we obtain the FTA such as Figure. 6. The shape of this FTA is designed based on a FTA against the air inlet blockage described above.

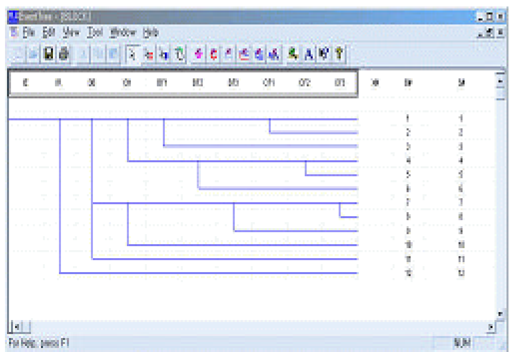


Figure 6. Event tree of air inlet blockage accident.

It is assumed that it uses large data, conservatively. However, it is obtained from the consequence that probability of the radioactive substances release against the air inlet blockage accident is very small and safe because probability of the air inlet blockage accident is also very small. We conclude that the MACSTOR/KN-400 is a safe facility during the design operation period. The result of radiation material release frequency results in a value of 1.46×10^{-7}.

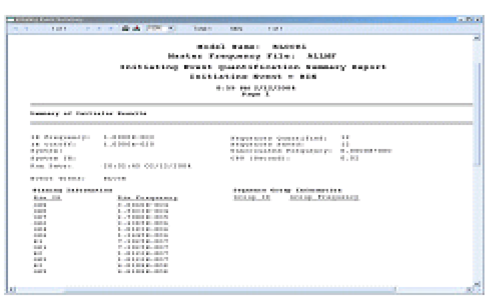


Figure 7. Results of air inlet blockage accident

IV.C Consequence modeling

The calculation procedures using MACCS2 code in this study are as follows.

1. The CCDF of early rising and cancer death based upon evacuation Scenarios (NUREG-1150 Base Case)
2. Average Individual Risk of cancer death
3. Whole body Dose with respect to the distance from the interim storage facility

The input preparation for MACCS2 code and calculation has been performed for the reference facility of MACSTOR/KN-400 to be installed around the area of Wolsung #1 unit. Figure. 8 shows the results of NUREG-1150 base case showing the result of CCDF, being 95% and 5% of evacuation and non-evacuation, respectively. The early occurring cancer death rate was almost zero for the entire category as already expected by insights. For the latent cancer death rate the result of CCDF showed that the MACSTOR/KN-400's risk was much lower than that of Wolsung's. The Figure. 9 shows average individual risks of cancer deaths. As shown in the figure, about double digit difference is shown compared with Wolsung's nuclear power plant. The Figure 10 shows the whole body dose, which is also generally lower than that of Wolsung's power plant.

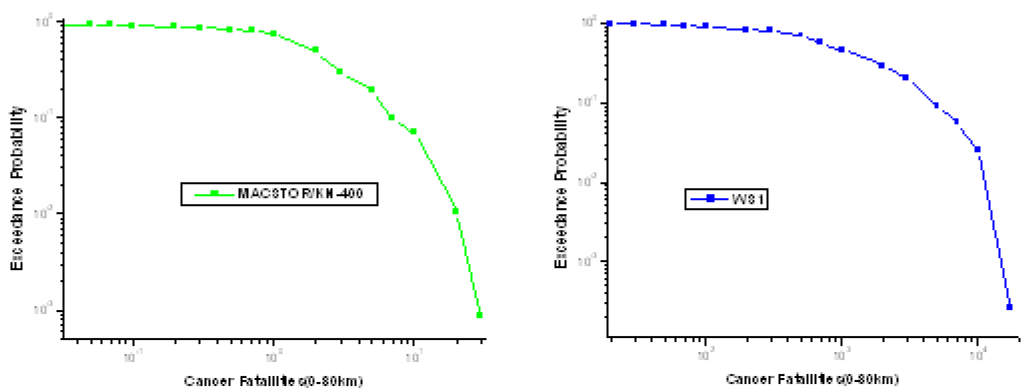


Figure 8. CCDF of MACSTOR/KN-400 and Wolsung plant (0~80km)

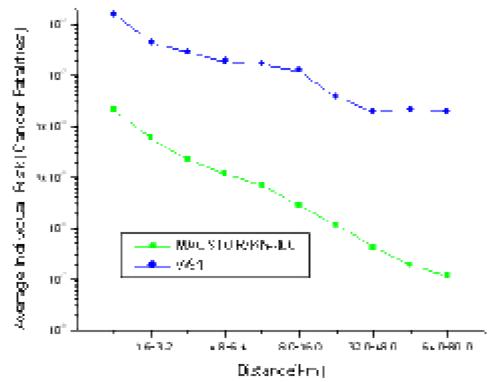


Figure 9. Average individual risk (0~80km)

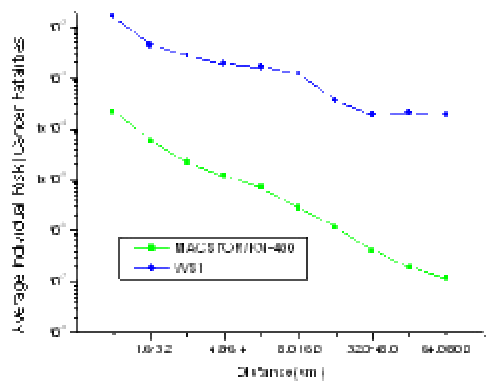


Figure 10. Whole body dose (0~80km)

V. Conclusions

A framework for evaluating the spent fuel dry storage facility during the air inlet blockage accident composing of three phases has been established and applied to an interim storage system. They include the analysis of the failure probability of a basket and a cylinder, the accident modeling of spent fuel dry storage facility and the accident consequence assessments. The first phase of the analysis calculated the module failure probability by modeling of the basket and the cylinder, which is major element for containing radioactive substances. The second phase includes a modeling of spent fuel dry storage facility. At this phase, the probability that radioactive substances are released to outside when the initial event happens has been calculated by the construction of the event tree methods against a various elements which affects the air inlet blockage accident. At the third phase of releasing radioactive substances, the radiation damage to affect neighborhood and storage facility worker using MACCS2 code has been evaluated quantitatively. Using this framework, a risk of the interim storage facility has been quantified by the measure of early and latent cancer fatality.

Acknowledgement

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