A STUDY ON THE OFF-SITE CONSEQUENCE ANALYSIS FOR A VHTR

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Risk-Informed Design included in a VHTR design to ensure the level safety has been considered as an effective methodology. The use of risk information is essential to develop the next generation of NPPs. Therefore, a PSA for a VHTR was performed to assess the health effects on the public. However, there is no plant state of a VHTR comparable to the 'core damage frequency' and 'large early release frequency'. Thus, it is necessary to implement a new PRA procedure for a VHTR. This paper deals with the sequence and consequence levels of a PSA. In the sequence level, several scenarios where radioactive materials could be release are selected. In addition, a traditional event tree method is used to find a probability distribution of the release frequency. To calculate the system unavailability used for an event tree head, a couple of analysis methods are used. One of them is the survival reliability, which is used as a traditional method. For the other method, a structural reliability was used for a passive safety system such as a Reactor Cavity Cooling System (RCCS) because it has 100% reliability when calculated based on the survival reliability. After calculating the release frequency, the end states of an accident were defined according to the release categories of the radioactive materials. The consequence level of a PSA for off-site radioactive materials released during a severe accident was performed using the MACCS2 code. In addition, a risk profile was performed with a compensated cumulative distribution function (CCDF), which provides overall insights regarding the risk of VHTR accidents. In addition, the CCDF were compared to the CCDF of other plants in order to verify the safety of the VHTR. For further study, an assessment of the characteristics of a VHTR safety system and a precise quantification of its accident scenarios are expected to be conducted for a more certain consequence analysis. The methodology shown in this study might contribute to enhancing the safety of the VHTR design by utilizing the results having a far lower effect on the environment than the LWRs.

I. INTRODUCTION

Currently, fossil fuels are running out globally. If the current trends continue, crude oil will be depleted in 20 years and natural gas in 40 years. In addition, the use of fossil resources has increased the emissions of greenhouse gases such as carbon dioxide. Therefore, there has been a strong demand in recent years for producing large amounts of hydrogen as an alternative energy source (Ref. 1). To generate hydrogen energy, a very high temperature of more than 900℃ is required, but this level is not easy to reach. A Very High Temperature Reactor (VHTR), one of next generation reactors, is able to make reach this temperature, and is regarded as a solution to the above problem. In addition, a VHTR has excellent safety in comparison with existing and other next-generation reactors. In particular, a passive system, a Reactor Cavity Cooling System (RCCS), is adopted to remove radiant heat in the case of an accident. To achieve variety requirements of newly designed reactors, however, it is necessary to develop new methodologies and definitions that differ with the existing method. At the same time, the application of a probability safety assessment (PSA) has been proposed to ensure the safety of next generation NPPs. For this, risk-informed designs of the structures have to be developed and verified. In particular, the passive system needs to be evaluated for its reliability. The objective of this study is to improve the safety of a VHTR by conducting a risk profile.

II. PROBABILISTIC SAFETY ASSESSMENT FOR VHTR

II.A. Overview of VHTR PSA

II.A.1. Very High Temperature Reactor (VHTR)

There are two types of VHTR according to its fuels: a Pebble Bed Reactor (PBR) and a Prismatic Modular Reactor (PMR). The fuel block consists of carbon coated nuclear fuel particles, TRISO, and each particle minimizes the release of radioactive sources (Ref. 2). The coolant is helium, which has a very low absorption cross section. Therefore, several systems such as a heat exchange system are needed to cool down the helium. In Korea, researches are actively underway, focused on a prismatic block type VHTR with a high outlet temperature and a passive safety system. The main characteristics of a Korean VHTR are shown in table I.

TABLE I. Main Design Characteristics of Korean VHTR (Ref. 3)

Core Thermal Power	600 MW(t)
Helium Pressure	70 bar
Helium Flow Rate	250 kg/s
He In/Ex Temperature	490/950℃
Core bypass flow fraction	10%
Heat Removal by RCCS	Modeling
Reactor Cavity Relief Valve Opening Set-Point	1.7 bar

II.A.2. Probabilistic Safety Assessment for VHTR

Owing to the unique design characteristics of a VHTR, its PSA needs to apply a unique methodology. The main rule is not different from a conventional method, such as a fault tree analysis (FTA) / event tree analysis (ETA) and accident consequence analysis, but the inherent safety characteristics of a VHTR such as the core damage or nucleus safety features. Consequently, it is necessary to revise the existing PSA analysis system, as shown in Fig. 1. The figure describes three steps of a PSA changed into a couple of steps composed of a sequence level and a consequence level (Ref. 4). In the sequence level, initiating events are selected using a Master Logic Diagram (MLD) and accident scenarios are drawn through a system analysis. To quantify the frequencies of the accident scenarios, FTA and ETA were used to evaluate the accident sequences and derive the final states. In addition, in this level, the amount of radioactive materials and the occurrence frequency have to be measured. In this study, a sequence level PSA including level 1 and level 2 of the conventional PSA of an LWR was conducted. On the other hand, at the consequence level, the impact on the public and the environment after radionuclides are released following a nuclear accident is assessed. Its objective is the same as a quantifying level 3 PSA, and the influence of radioactive materials released into the atmosphere after an NPP accident is evaluated (Ref. 4).

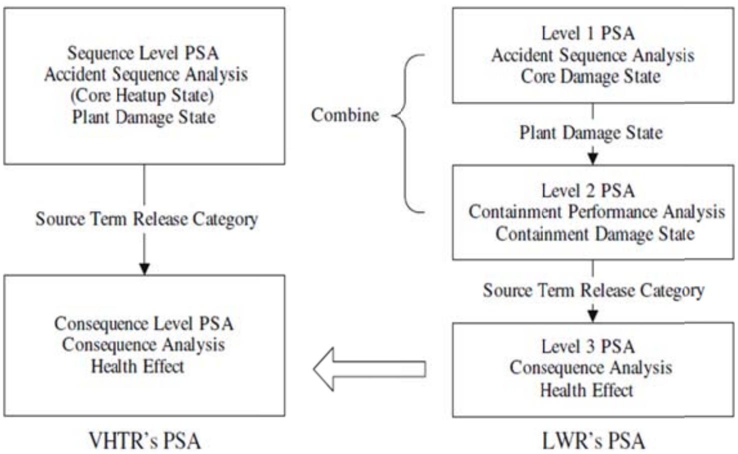

Fig. 1. Modified PSA process for VHTR (Ref. 4)

II.B. Accident Sequence Analysis of VHTR

II.B.1. Initiating Event Analysis

To select the initiating events, the MLD method was used. Because the MLD has no methodology format, it is easy to apply to a conceptual design. The selected initiating events ultimately result in a failure of the safety functions leading to leakage of radioactive materials. In other words, the events include failures of heat generation control, core heat removal, and chemical erosion control. The initial events causing a failure of these functions can be grouped by their similarity of accident sequences. The simplified MLD conducted in this study is shown in Fig. 2. Finally, the five types of initiating events are classified: Loss of Helium Pressure Boundary (LHPB), Loss of Secondary Heat Transport System (LSHTS), Water Ingress (WTIG), and Transient (TRN).

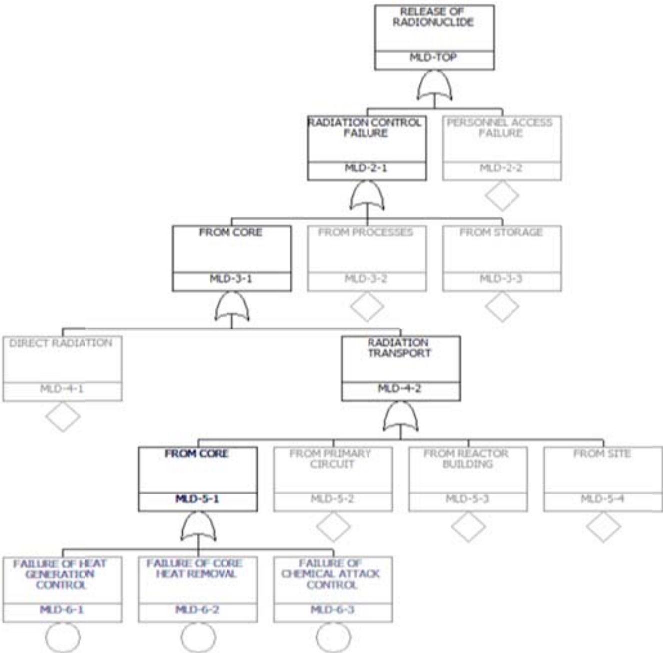

Fig.2. MLD for initiating event selection

First of all, LHPB is an event in which the helium pressure boundary is either ruptured or leaked and coolant releases from the primary loop. Because it causes a rise of pressure in the containment building, LHPB is able to threaten the integrity of the containment. The accident sequence of LHPB is similar with a small break loss of coolant accident (SLOCA) of an LWR. Therefore, the frequency and its error factor were presumed as that of an LWR SLOCA. Secondly, an LSHTS is a failure of heat removal function resulting from loss of coolant in a second system. The typical initiating events include failures of an intermediate or secondary loop. In this case, the helium boundary is intact but the pressure in the primary loop is able to rise owing to the cooling failure of the secondary loop. The frequency of this initiating event was calculated using an FTA, and the error factor was subjectively assumed. The reliability data used for an FTA are based on a U.S. Modular High Temperature Gas-cooled Reactor (MHTGR) and Korean LWR reliability data (Refs. 5, 6). Thirdly, WTIG is an event in which water from a steam generator in a secondary loop flows in a primary loop because of a rupture of both the steam generator and the heat exchanger in the immediate loop. This event may cause heat generation control failure, core heat removal failure, and chemical erosion control failure. The frequency of this event is from the reliability data of the steam generator and the heat exchanger of a U.S. MHTGR, and the error factor was subjectively assumed. Finally, a transient (TRN) consists of all events having similar accident sequences among other initiating events. In this case, the helium boundary is sound and the main safety functions work normally. Loss of offsite power is also comprised in this event to reflect the VHTR's low dependence of offsite power. Both the frequency and error factor are from data of an existing Korean LWR. A summary of the frequencies of initiating events is shown in table II.

TABLE II. Frequencies and Error Factors of Initiating Event Categories

Initiating Event Category	Frequency (/RY)	Error Factor
Loss of Helium Pressure Boundary, LHPB	3.0×10-3	5.0
Loss of Secondary Heat Transport System, LSHTS	1.6×10-2	5.0
Water Ingress, WTIG	2.7×10-5	5.0
Transient, TRN	1.45	1.38

II.B.2. Accident Sequence Analysis

For the accident analysis, an event tree analysis (ETA) was used. First of all, an LHPB is an event in which the helium pressure boundary is either ruptured or leaked and coolant from the boundary is released from the primary loop. Because it causes a rise of pressure in the containment building, an LHPB is able to threaten the integrity of the plant. Once an LHPB occurs, the reactor will be automatically tripped followed by a decrease of pressure. As the reactor trip is failed, the pressure of the reactor vessel dwindles and helium release continuously increases. At this time, the shutdown cooling system (SCS) begins to operate because the cooling through a secondary loop does not work owing to the loss of the helium boundary. When all safety systems fail, the reactor cavity cooling system (RCCS) passively works to remove residual heat. Secondly, an LSHTS is caused by a loss of coolant in a second system and following a failure of the heat removal function. The typical initiating events include a failure of the intermediate or secondary loop. When an LSHTS occurs, the pressure in the primary loop increases and reactor shuts down. After the shutdown, SCS and RCCS remove the residual heat. In the case of an RCCS failure, the release of radioactive material occurs. Thirdly, WTIG is an event in which water from a steam generator in a secondary loop flows to a primary loop because of the leak of both the steam generator and the immediate heat exchanger. Flowing steam slows down the neutrons to increase the thermal output with the insertion of the reactivity. After high humidity or pressure stops the reactor from operating, the steam generator is isolated to restrict the inflow of steam. In addition, heat removal via SCS and RCCS is performed. Finally, the transient event includes all events having similar accident sequences. In the case of a transient, the helium pressure boundary and main safety systems are sound. After a reactor trip, generated heat is removed by the secondary loop heat exchanger. When heat removal through a secondary loop fails, the SCS or RCCS works to remove the residual heat. In the case of an RCCS failure, the release of radioactive material control fails. The event trees of all scenario sequences are shown in Fig. 3.

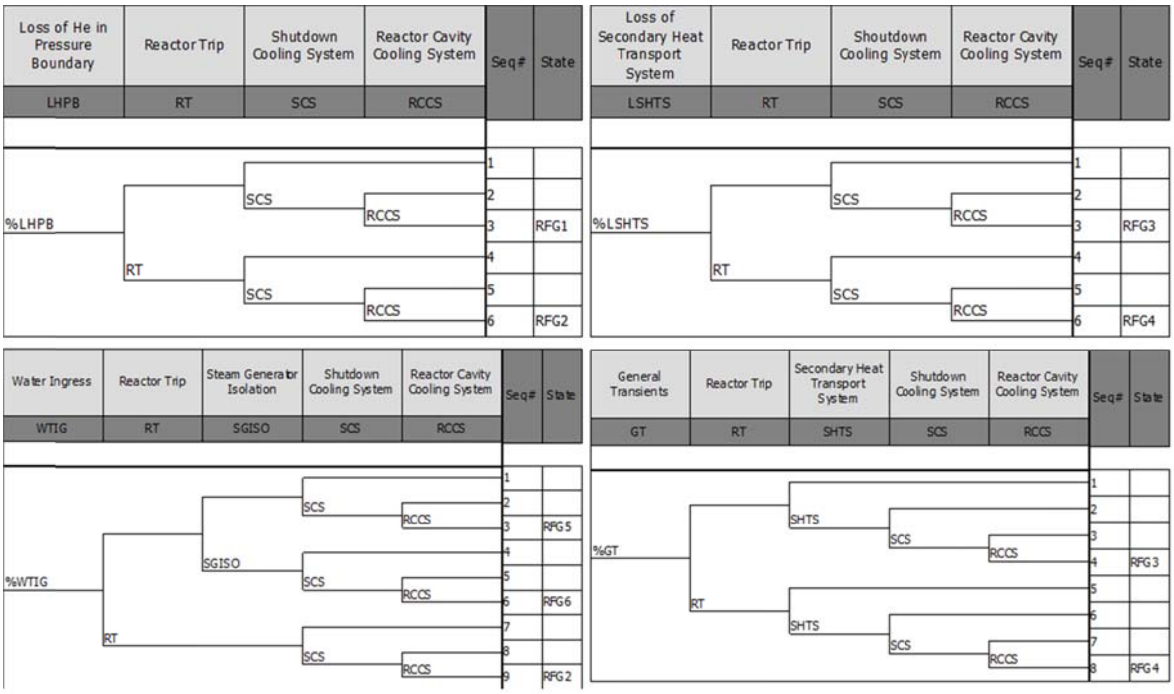

Fig. 3. Event trees of each scenario.

II.B.3. System Analysis

The system failure rates after initiating events were evaluated by a Fault Tree Analysis (FTA). The data used for the FTA were mainly from the U.S. MHTGR reliability data. When the data were unavailable, the reliability data of a Korean LWR were used. Failure probabilities were assumed to have log-normal distributions. Fig. 4 shows the fault trees of the SCS and SGISO system. Meanwhile, the passive safety system, RCCS, is unable to be measured by the reliability data. Because it is a passive system, it has 100% reliability when calculated using the survival reliability method. Therefore, in this study, a load-capacity model, which is a structural method, was used to calculate the failure probability of an RCCS (Ref. 7).

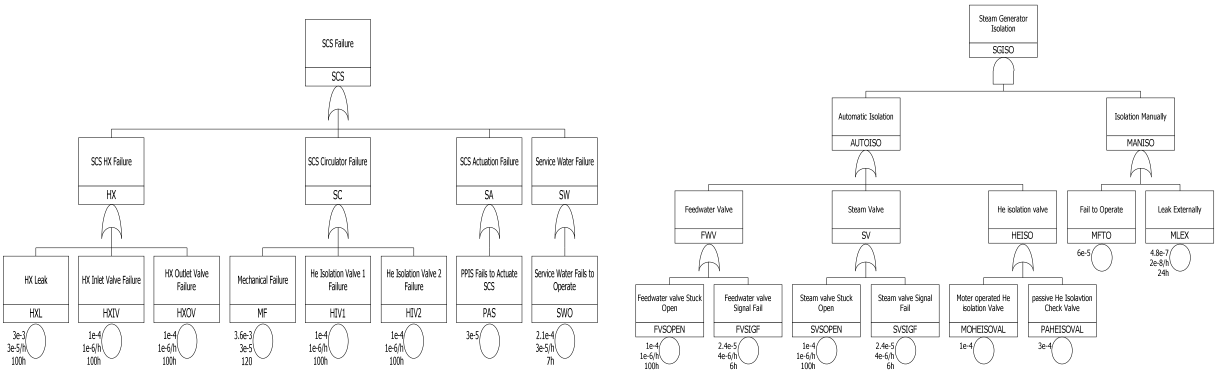

Fig. 4. Fault tree of SCS (left) and SGISO (right)

TABLE III. System Failure Probabilities and Error Factors

Safety System	Failure Probability	Error Factor
Reactor Trip, RT	5.1×10^{-6}	5.0
Loss of Secondary Heat Transport System: LSHTS	1.6×10^{-2}	5.0
Shutdown Cooling System: SCS	2.7×10^{-5}	5.0
Steam Generator Isolation: SGISO	3.9×10^{-8}	5.0
Reactor Cavity Cooling System: RCCS	1.1×10^{-4}	1.2

II.B.4. Accident Sequence Quantification

The accident sequence quantification was performed with all accident scenarios followed by four initiating events. The scenarios obtained are classified according to whether or not the safety system is successful. Then, the end states are categorized based on a sequence in which an accident proceeds with identical system failures. In this study, those end state groups are named the ‘Release Fraction Group’ because it assumed that all sequences included in the same RFG have the same release fraction of radioactive materials.

The quantification is calculated with initiating event frequencies and failure rates of the safety systems. To calculate, Monte Carlo sampling was used to obtain the distributions of the event tree terminated states and RFGs, and the number of samples was 100,000 per state. By combining the sequences of the same initiating event type, the final distributions of the accident frequencies were obtained. The results are shown in table IV. The sum of all releases was $3.416\text{x}10^{-8}$/RY.

The contributions of each RFG are shown in Fig 5. RFG-1 occupies most of the frequency and RGF-1 has the second largest contribution. Furthermore, the contribution of the initiating events is also shown in Fig. 5. The transient has largest contribution, and water ingress has smallest contribution.

TABLE IV. Frequencies of Release Fraction Groups

RFGs	Frequency (/RY)	Error Factor	90% Confidence Range
RFG-1	2.402×10^{-9}	5.1	$[2.89\times10^{-10},\ 7.50\times10^{-9}]$
RFG-2	1.239×10^{-14}	10.5	$[4.75\times10^{-16},\ 4.66\times10^{-14}]$
RFG-3	3.173×10^{-8}	3.5	$[7.07\times10^{-9},\ 8.29\times10^{-8}]$
RFG-4	1.620×10^{-13}	6.0	$[1.61\times10^{-14},\ 5.23\times10^{-13}]$
RFG-5	2.164×10^{-11}	9.8	$[8.47\times10^{-13},\ 8.09\times10^{-11}]$
RFG-6	8.481×10^{-19}	15.4	$[1.20\times10^{-20},\ 3.28\times10^{-18}]$
Sum	3.416×10^{-8}	3.27	$[8.59\times10^{-9},\ 8.58\times10^{-8}]$

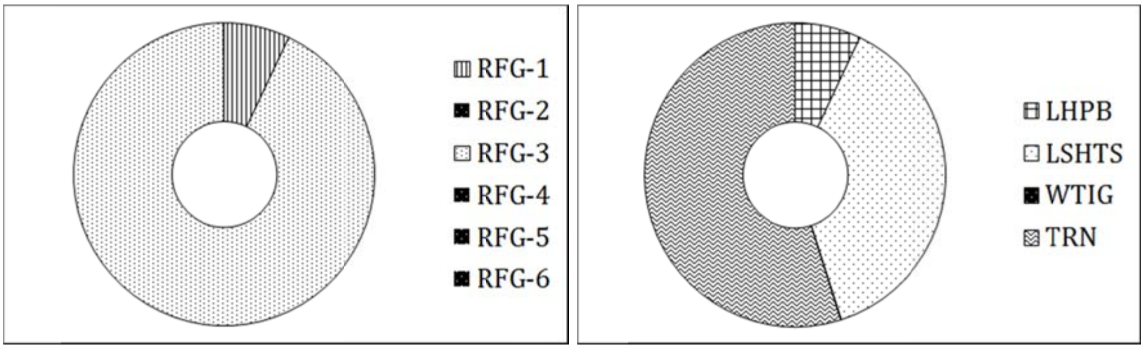

Fig. 5. Contributions of each RFG (left) and initiating event (right)

II.C. Accident Consequence Analysis of VHTR

II.C.1. Initial Inventory and Release Fraction Groups

Because the Korean VHTR is in the design phase, the specific design for the implementation of a PSA has not been confirmed yet. In this study, therefore, the confirmed design details of the Korean VHTR up to date were compared with those of foreign VHTRs. The key features of the Korean VHTR include a TRISO, helium coolant, designed power, and intermediate heat exchanger between the reactor and hydrogen production facility. Among the foreign VHTRs, NGNP VHTR in U.S. has the most similar features with the Korean VHTR features. In particular, both VHTRs have the same heat output, 600 MW(t). Because the initial inventory is able to be tracked with the heat output, it was assumed that the Korean VHTR and NGNP VHTR have the same initial inventory.

The release fractions of each scenario were taken from the release fraction of scenarios of the NGNP and U.S. MHTGR, which have the same sequences. These release fractions are from the NGNP release fraction analysis report by Idaho National Laboratory (INL) (Ref. 8) and the PSA report of the U.S. MHTGR by the U.S. Department of Energy (Ref. 5). Table V shows the release fractions of the RFGs used in this study.

TABLE V. Release Fractions for each Scenario

Radionuclide Groups	Release Fraction Groups (RFGs)					
	RFG-1	RFG-2	RFG3	RFG-4	RFG-5	RFG-6
NG	4.23×10^{-6}	3.32×10^{-4}	3.45×10^{-6}	8.47×10^{-5}	3.92×10^{-6}	8.06×10^{-8}
I	1.99×10^{-7}	8.04×10^{-4}	8.47×10^{-8}	2.01×10^{-4}	2.73×10^{-7}	2.07×10^{-7}
Cs	3.29×10^{-7}	2.65×10^{-5}	3.47×10^{-9}	6.65×10^{-6}	1.12×10^{-7}	6.67×10^{-7}
Te	1.40×10^{-7}	1.03×10^{-4}	8.68×10^{-8}	2.60×10^{-5}	1.73×10^{-7}	2.54×10^{-8}
Sr	4.06×10^{-8}	2.87×10^{-10}	5.38×10^{-13}	7.17×10^{-11}	1.84×10^{-8}	4.04×10^{-7}
La	8.25×10^{-9}	-	-	-	9.77×10^{-9}	1.73×10^{-9}
Ru	4.39×10^{-8}	1.68×10^{-9}	1.68×10^{-9}	2.44×10^{-6}	3.12×10^{-8}	-

II.C.2. Accident Consequence Analysis

The MACCS2 code was used to calculate the accident consequence. MACCS2 was developed to assess the public and environmental impact resulting from radionuclides released after a nuclear accident. Because its object is to quantify a level 3 PSA, it presents the influence of radioactive materials in the atmosphere. The site was assumed to be located at Gyeong-ju where the second Korea Atomic Energy Research Institute will be built. The area chosen has a 30 km radius distance from the VHTR site. This area includes Gyeong-ju, Ulsan, and Pohang, where about eight hundred thousand people live. With this area set, the land fraction and population data were obtained. The emergency response activities were assumed to be a 95% evacuation and 5% sheltering. The factors used in the calculation, such as the shielding factor and inhalation factor, were recommendation values of the U.S. NRC.

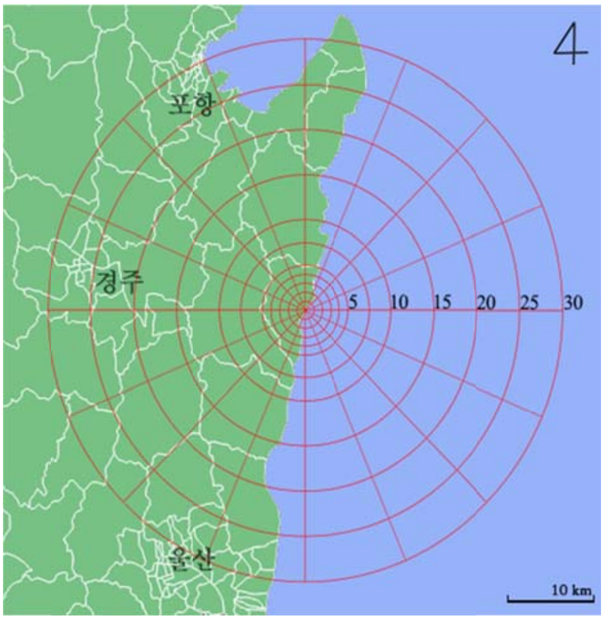

Fig. 6. Area set for land fraction and demographic input data

II.C.3. Dose and Cancer Fatality Results

An accident consequence evaluation was conducted. In addition, the results of the dose and fatality as the effects of an accident were obtained. First, the whole body doses at each distance are shown in Table VI. The Early Fatality results came out as 0% for all of the considered area set for all scenarios. This means that nobody immediately dies when the VHTR has an accident. The cancer fatality results are shown in tables VII through VIII and are much lower than the safety standard.

TABLE VI. Whole Body Dose at Area Sets

Distance (km)	Whole Body Dose					
	RFG-1	RFG-2	RFG-3	RFG-4	RFG-5	RFG-6
1.0-2.0	36.20	5220	1.38	1560	14.5	73.6
2.0-3.0	18.60	3040	0.74	809	7.49	37.9
3.0-4.0	11.70	2000	0.47	513	4.73	23.8
4.0-5.0	8.18	1060	0.34	358	3.30	16.6
5.0-7.5	4.90	733	0.21	217	1.99	9.91
7.5-10.0	2.78	335	0.12	123	1.13	5.62
10.0-15.0	1.43	212	0.06	63.7	0.58	2.88
15.0-20.0	0.69	105	0.03	30.6	0.28	1.38
20.0-25.0	0.36	53.2	0.02	16.2	0.15	0.73
25.0-30.0	0.20	31.5	0.01	9.09	0.08	0.41

TABLE VII. Mean individual Cancer Fatality

Distance(km)		RFG-1	RFG-2	RFG-3	RFG-4	RFG-5	RFG-6
Mean	7.5-10.0	7.38×10^{-8}	2.44×10^{-6}	1.92×10^{-9}	2.40×10^{-6}	2.81×10^{-8}	1.47×10^{-7}
	10.0-15.0	3.78×10^{-8}	1.23×10^{-6}	9.85×10^{-10}	1.23×10^{-6}	1.44×10^{-8}	7.52×10^{-8}
	15.0-20.0	1.81×10^{-8}	5.81×10^{-7}	4.71×10^{-10}	5.85×10^{-7}	6.87×10^{-9}	3.59×10^{-8}
	20.0-25.0	9.53×10^{-9}	3.03×10^{-7}	2.50×10^{-10}	3.08×10^{-7}	3.62×10^{-9}	1.89×10^{-8}
	25.0-30.0	5.33×10^{-9}	1.68×10^{-7}	1.41×10^{-10}	1.72×10^{-7}	2.03×10^{-9}	1.06×10^{-8}

TABLE VIII. 95 Percentile Individual Cancer Fatality

Distance (km)		RFG-1	RFG-2	RFG-3	RFG-4	RFG-5	RFG-6
95 Per-centile	7.5-10.0	7.61×10^{-8}	2.50×10^{-6}	1.94×10^{-9}	2.47×10^{-6}	2.87×10^{-8}	1.50×10^{-7}
	10.0-15.0	3.91×10^{-8}	1.27×10^{-6}	1.02×10^{-9}	1.27×10^{-6}	1.48×10^{-8}	7.81×10^{-8}
	15.0-20.0	1.85×10^{-8}	6.03×10^{-7}	4.84×10^{-10}	6.08×10^{-7}	7.17×10^{-9}	3.73×10^{-8}
	20.0-25.0	9.89×10^{-9}	3.17×10^{-7}	2.59×10^{-10}	3.22×10^{-7}	3.77×10^{-9}	1.95×10^{-8}
	25.0-30.0	5.58×10^{-9}	1.74×10^{-7}	1.46×10^{-10}	1.77×10^{-7}	2.13×10^{-9}	1.11×10^{-8}

II.C.4. Risk Profile

For PRA applications, the radiological consequences are presented in the form of a complementary cumulative distribution function (CCDF). It shows the frequency in which a consequence will exceed a given magnitude. The cancer fatalities are shown in Fig. 7 for each accident scenario.

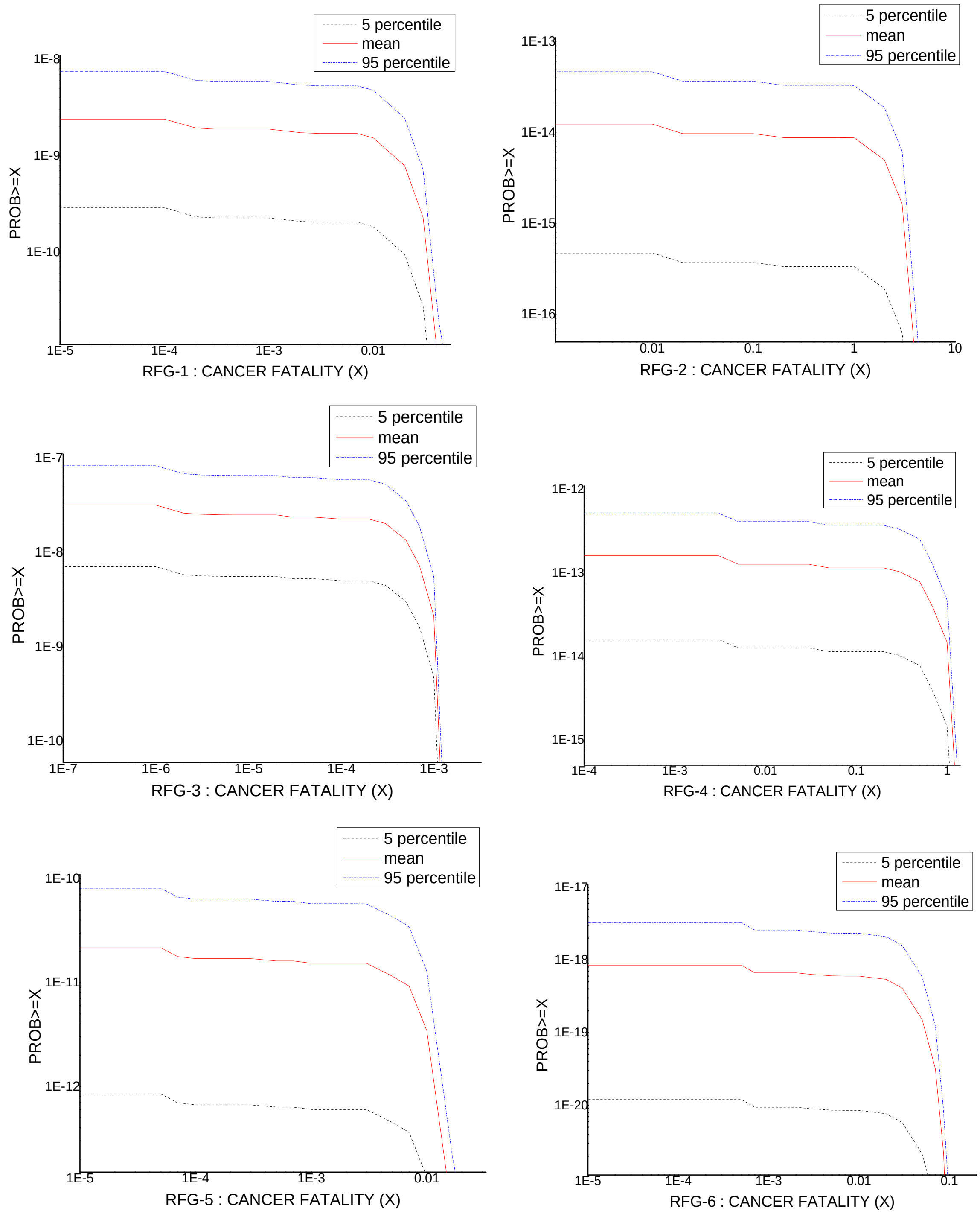

Fig. 7. CCDF Results of Each Accident Scenario

By summing the CCDF results for each accident scenario, the total CCDF of a VHTR was obtained as shown in Fig. 8. The RFG-1 and RFG-3 have the largest effect on the total CCDF. This is because these two accidents have a relatively high frequency or large release amount of radioactive materials.

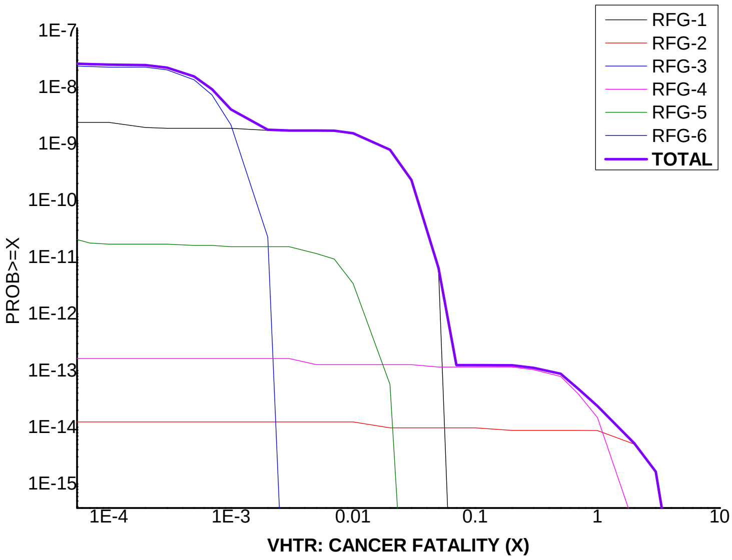

Fig. 8. Total CCDF of VHTR

Furthermore, to compare the results of a VHTR accident consequence with the existing LWR and advanced reactor, an accident consequence assessment on the Korea Standard Nuclear Power Plant (KSNP), OPR1000, and APR1400 was conducted. The risk of every scenario is evaluated for the VHTR, OPR1000, and APR1400. A comparison graph of the results is shown in Fig. 9. As the figure shows, the VHTR has a lower cancer fatality risk compared with the existing LWR.

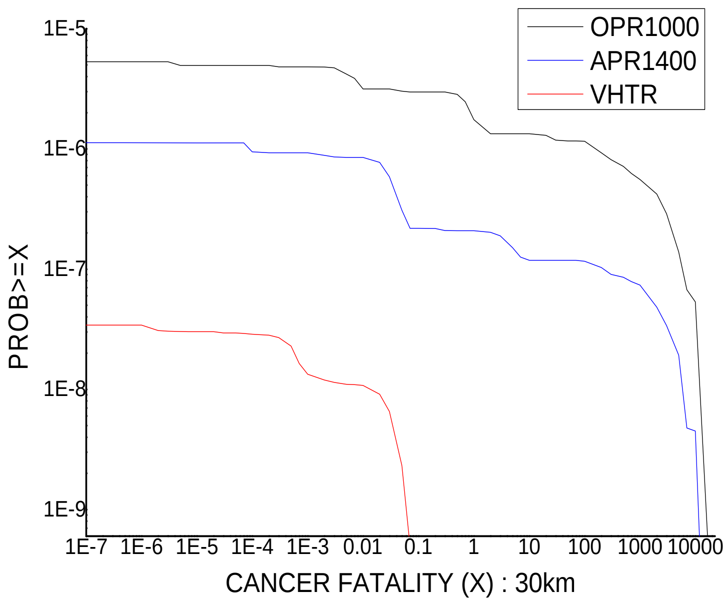

Fig. 9. Comparison of CCDF Result

III. CONCLUSIONS

An offsite consequence analysis for a VHTR using the MACCS code has been performed. Because a passive system, the RCCS (Reactor Cavity Cooling System), is equipped, the frequency of occurrence of accidents has been evaluated to be very low (Ref. 5). For further study, an assessment of the characteristics of a VHTR safety system and a precise quantification of its accident scenarios are expected to be conducted for a more certain consequence analysis. The methodology shown in this study might contribute to enhancing the safety of a VHTR design by utilizing the results of having a far lower effect on the environment than the LWRs.

ACKNOWLEDGMENTS

This work was supported by a National Research Foundation of Korea (NRF) grant funded by Ministry of Science, ICT & Future Planning and partly supported Nuclear R&D Programs of Ministry of Science, ICT and Future Planning of Korea and partly supported by the Nuclear Safety Research Program of the Korea Foundation Of Nuclear Safety (KOFONS), granted financial resources from the Korean Government's Nuclear Safety and Security Commission (NSSC) (No. 1305008).

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