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Frequency Analysis of Containment Failure due to Internal Events According to the Design Characteristics of 4 Reactor Types

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1. Introduction

Korea is one of the countries that depend heavily on nuclear power for electricity generation. As of 2020, nuclear power generation share in Korea is about one quarter of the total power generation, the second highest after the thermal power generation. However, since the Fukushima nuclear accident, there has been an increase in public concern in Korea about the damage that maybe caused by nuclear power plant accidents.

When an accident occurs in a nuclear power plant and radioactive material leaks to the environment, it may cause fatality and economic loss. In order to prevent accidents from occurring at nuclear power plants, such safety features as containment building are equipped. There are several types of reactor types in Korea, where each reactor has different design characteristics. In this study WH600, WH900, OPR1000 and APR1400 reactors are used as reference reactors, and the conditional containment failure probability, which is a kind of level 2 probabilistic safety assessment (PSA) result, is analyzed and summarized.

2. Modelling

This chapter describes the level 2 PSA models for the four reactor types, WH600, WH900, OPR1000 and APR1400 and each step focusing on the comparison of the different characteristics. The quantification was performed by COFUN code.

2.1 Plant Damage State Logic Diagram

The core damage accident sequences derived from Level 1 PSA includes the systems that directly affect the core damage, but does not include the systems related to the severe accident scenarios and the performance of the containment building after the core damage has occurred.

Therefore, based on the information required for the classification of the plant damage state (PDS), the level 1 PSA event tree was modeled to plant damage state logic diagram (PDS LD) that was expanded to include all systems important for the analysis of the containment building accident progress. For this purpose, the headings and success conditions of the system and operator actions added to the event tree of the PDS are shown in Table 1. The SGTR-SCRUB title was added to consider the scrubbing effect in case of a steam generator tube rupture accident. In the case of the APR1400 reactor, it was considered by dividing it into two headings, SHR-FSG and CONSPRAY.

Table 1. PDS LD description for four reactors

Plant Damage State Heading	Reactor Type	Branch
Containment Bypass, CONBYPASS	WH600/900, OPR1000, APR1400	- NO BYPASS - Interfacing System Loss of Coolant Accident, ISLOCA - Steam Generator Tube Rupture, SGTR
Containment Isolation Status, CONISOLAT	WH600/900, OPR1000, APR1400	- ISOLATED - NOT ISOLATED - Rupture Before Core Melt, RBCM
Accident Type, TRANLOCA	WH600/900, OPR1000, APR1400	- Station Blackout, SBO - TRANSIENT - Large, Medium LOCA, L/MLOCA - Small LOCA, SLOCA
Power Recovery, POWRECOV	WH600/900, OPR1000, APR1400	- RAC-RVF - RAC-CF - RAC-NO
Status of In-vessel Injection, INVESSINJ	WH600/900, OPR1000, APR1400	- ON - DEADHEADED - FAILED
Steam and H2 Release Point, RELPOINT	APR1400	- INC - IRWST
Containment Spray Recirculation Cooling, CSRCOOL	WH600/900, OPR1000, APR1400	- YES - NO
Fan Cooler Cooling, FANCOOL	WH600/900	- YES - NO
Reactor Coolant System(RCS) Pressure during Core Damage, RCS-P	WH600/900, OPR1000, APR1400	- LOW - MEDIUM - HIGH
Secondary Heat Removal, SG	WH600/900, OPR1000, APR1400	- YES - NO
Status of Cavity, CAVCOND	WH600/900, OPR1000, APR1400	- FLOODED - NOT FLOODED

Water Injection for Scrub in Bypass Sequence, WATER-IN	WH600/900, OPR1000	- SCRUB-YES - SCRUB-NO

2.2 Containment Event Tree

The containment event tree (CET) is developed to analyze such behavioral characteristics of the containment building as phenomena that may occur during a severe accident, the containment building status, and the type of damage to the containment building. Therefore, the CET should be able to systematically model the progress of all possible accidents within the containment building. In addition, it should be written in detail so that it can reflect various severe accident phenomena, the process of severe accident, damage to the containment building, and operator actions important for the evaluation of radiation sources. The headings and branch contents of the CET are shown in Figure 1.

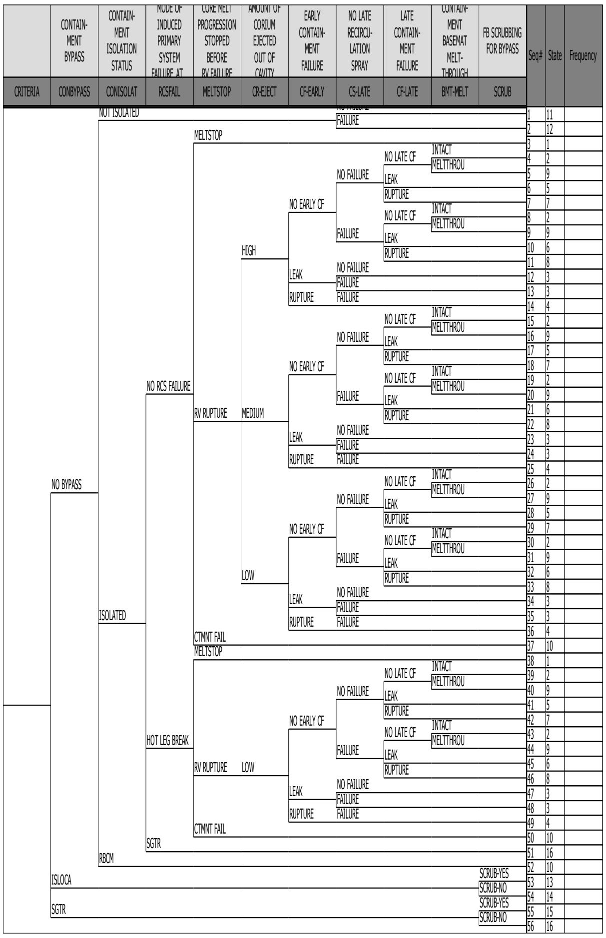

Fig. 1. Containment Event Tree of OPR1000 and APR1400

For each title of the CET, the sub-tree, that is the decomposition event tree (DET), was used to branch the PDS LD according to the logical rule, and the branching probability of each title was calculated. Each decomposition event tree was prepared for the phenomena that greatly affect the branching probability of the title. As an example of DET, 'BMT-MELT' DET is shown in Figure 2.

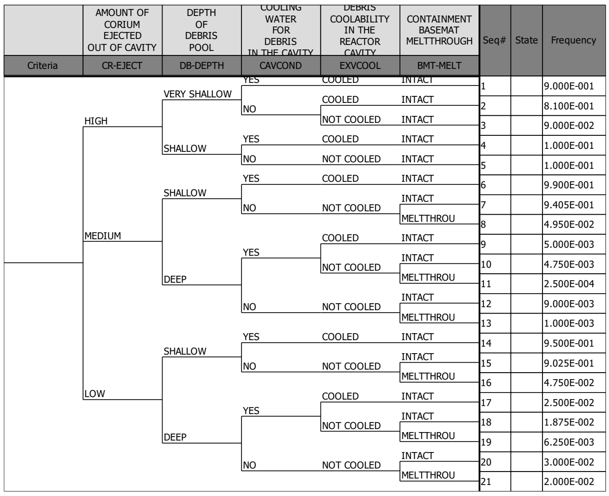

Fig. 2. BMT-MELT DET

The early containment failure (ECF) DET has the same title and branch for each reactor type. By using the calculation results from the MELCOR and TCE (Two Cell Equilibrium) codes, which are severe accident analysis codes developed by SNL (Sandia National Laboratories) for each reactor type, the peak pressure of the containment building is calculated for the direct containment heating (DCH) situation and the probability of containment failure was calculated. For the late containment failure (LCF) DET, the peak pressure of the containment building was calculated using the MELCOR code, and the failure probability of the containment building was calculated.

2.3 Source Term Category

The accident sequences up to the final point of the CET shows the paths from the initial event to the release of fission products. However, since there are many accident sequences similar to the release timing and quantity of fission products, it is unnecessary to analyze all accident pathways. Therefore, the final point of the CET was classified into a small number of source term categories (STCs) according to the behavioral characteristics of fission products emitted to the environment. Figure 3 shows the source term category logic diagram (STC LD) generally used in this study. In the case of the WH600 reactor, STC LD was used except for the CF-MODE heading in Figure 3 because the damage types of the containment building were not classified in detail as leak and rupture when evaluating the pressure capacity of the containment building.

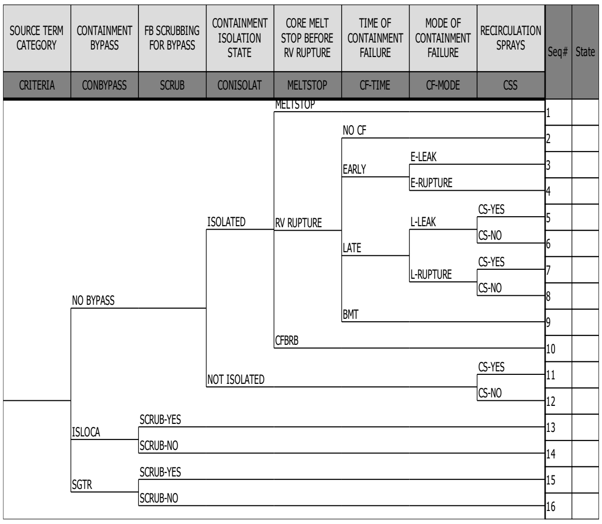

Fig. 3. Source Term Category Logic Diagram

3. Results and Discussion

According to the available time from the start of accident to the destruction of the containment building, it was classified as ECF and LCF, and it was considered that basemat melt-through (BMT) of the containment building could occur if LCF has not occurred. In addition, considering isolation failure or the leakage of radioactive materials to the outside of the containment building by bypassing the containment building, the case where the integrity of the containment building is maintained and the case where the containment building is damaged before the reactor vessel is damaged. The total of 7 cases are taken into account. Table 2 presents the containment building failure probability and quantification results related to the containment building failure types.

Table 2. The Results of the Containment Failure Frequency (continued)

	WH600		WH900	
Containment Failure Type	Frequency (/yr)	Percentage (%)	Frequency (/yr)	Percentage (%)
NO CF	5.53E-06	38.3	8.52E-06	68.9
ECF	4.21E-07	2.9	2.98E-08	0.2
LCF	7.00E-06	48.6	3.18E-07	2.6
BMT	2.31E-07	1.6	2.99E-07	2.4
CFBRB	2.31E-09	<0.1	2.22E-06	18.0
ISO. FL.	5.93E-07	4.1	2.96E-08	0.2
BYPASS	6.48E-07	4.5	9.54E-07	7.7
Total	1.44E-05	100.0	1.24E-05	100.0

Table 2. The Results of the Containment Failure Frequency

	OPR1000		APR1400	
Containment Failure Type	Frequency (/yr)	Percentage (%)	Frequency (/yr)	Percentage (%)
NO CF	3.42E-06	69.9	4.17E-06	80.6
ECF	2.04E-09	<0.1	9.16E-08	1.8
LCF	5.53E-07	11.3	1.09E-08	0.2
BMT	1.56E-08	0.3	2.81E-08	0.5
CFBRB	3.59E-07	7.3	2.72E-07	5.2
ISO. FL.	2.57E-09	0.1	4.44E-09	0.1
BYPASS	5.39E-07	11.0	5.98E-07	11.6
Total	4.89E-06	100.0	5.18E-06	100.0

In the case of nuclear containment building isolation failure, WH600 and WH900 reactors have values of 4.1% and 0.2%, respectively, and the OPR1000 and APR1400 reactors have 0.1% and 0.1%, respectively. The chemical volume control system (CVCS) letdown line has a valve that blocks the connection with the outside of the containment building. The Westinghouse reactor is driven only by a motor operated valve (MOV), so power is essential, whereas for OPR1000 reactor and APR1400 reactor, CVCS letdown line driven by air operated valve (AOV) or MOV, the valve can be closed through AOV even if there is no power. So there is a difference between the Westinghouse reactor and others in the rate of isolation failure of the containment building.

Even though it is the same Westinghouse model, the CFBRB of WH600 and WH900 have a value of less than 0.1% and 18.0%, respectively. CFBRB is mostly determined from level 1 event trees. This is an accident where the coolant in the core is released to the containment building for some reason and the integrity of the core is secured by the successful operation of the safe injection system, but the heat of the coolant discharged to the containment building cannot be removed and the containment building cannot maintain the integrity. #LOCCW-003, which is initiating event is loss of component coolant water (LOCCW) and core and containment building heat removal fails, accounts for more than 99% of the total CFBRB of WH900 reactor. This is the accident process in which the primary system cooling water loss accident occurs, the reactor coolant system seal is broken, the coolant is released into the containment building, and the operation of the safety injection system succeeds, but heat removal of the containment building fails. In case of WH900 reactor, cooling of the charging pump is lost due to the loss of cooling water in the primary system, so cooling of the charging pump was considered in consideration of supplying raw water as an alternative coolant. It was modeled that the safety injection system operation was impossible due to the loss of cooling. Therefore, WH600 reactor had low CFBRB occurrence percentage because it is impossible to operate the safety injection system in the case of a loss of primary coolant accident.

The LCF fraction of the WH600 reactor is a value of 48.6%, which is high compared to those reactors. The containment building pressure capacity of WH600 reactor is lower than that of other reactors, and as described above, unlike WH900 reactor, WH600 reactor loses the cooling function of the pump in the event of a primary equipment loss of cooling water, so that the safety injection system and the heat removal system of the containment building work and account for more than 50% of the LCF fraction. In WH600 reactor, if room cooling fails, safety injection is not performed due to the failure of the safety injection pump, and the heat removal of the containment building fails due to the failure of the containment building fan cooler and the containment building injection pump. However, in the case of WH900 reactor, when the room cooling fails, safety injection is not performed due to the failure of the charging pump, the fan cooler of the containment building and the water spray pump of the containment building. Since heat can be removed, the integrity of the containment building can be maintained. The result of the uncertainty analysis of large early release frequency (LERF) is shown in Table 3.

Table 3. Uncertainty Analysis for LERF

Reactor Type	Mean	5th	50th	95th
WH600	1.70E-6	5.19E-7	1.25E-6	4.10E-6
WH900	9.82E-7	2.97E-7	7.56E-7	2.38E-6
OPR1000	5.45E-7	1.40E-7	3.30E-7	1.46E-6
APR1400	7.04E-7	2.33E-7	5.12E-7	1.67E-6

4. Conclusions

In this study WH600, WH900, OPR1000 and APR1400 reactor were used as reference reactors and the conditional containment failure probability according to the design characteristics, which is a kind of the level 2 probabilistic safety assessment (PSA) result, was analyzed and summarized.

ECF was evaluated as low as the result of TCE calculation, and LCF was affected by the MELCOR results used for the containment heat removal system and the LCF DET. BMT was rated as low as a value of 2.0% or less for all reactors. the floor area per heat output meets the EPRI guidelines for all reactors. The containment building failure before reactor vessel failure, containment isolation failure, and containment bypass are mainly affected by the Level 1 PSA results for all reactors.

Acknowledgments

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