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Severe Accident Analysis of an Asymmetric FWLB inside Containment for the CANDU Plant PSA Level 2

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1. Introduction

As in US SOARCA [1], K-SOARCA [2] is needed to focus on a detailed and realistic analysis of accident scenarios that have a relatively large impact on the atmospheric environment and inhabitants rather than a conservative and low level depth analysis due to a number of scenarios. To this end, it is necessary to review the major accident phenomena and scenarios of severe accidents such as the TMI and Fukushima accident, and to review the selection criteria of accident scenarios in the US SOARCA. For the US SOARCA project, accident sequences for which the core damage frequency is more than 1.0E-06/ry or the core damage frequency with early mass release of radionuclides is more than 1.0E-07/ry are selected as the core damage accident sequences.

In the case of Wolsong Unit 1, the same accident selection criteria as US SOARCA were applied. A total of 297 core damage events were analyzed through the analysis of event trees in the Wolsong Unit 1 Probabilistic Safety Assessment Report [3]. As a result of the review, total 6 accident sequences with core damage frequency higher than 1.0E-06/ry were selected and 3 of them are reactor building bypass accident sequences. The asymmetric feedwater line break between the control valve and the check valve inside containment is one of these accident sequences. This paper provides results of PSA Level 2 thermal hydrodynamic analysis, based on the MAAP- ISAAC[1] computer code.

2. Methods and Results

2.1 MAAP-ISAAC 4.03 Computer Code

The MAAP-ISAAC (Modular Accident Analysis Program - Integrated Severe Accident Analysis code for CANDU plants) computer code has been developed to simulate accident scenarios that could lead to a damaged core and eventually to containment failure at CANDU type plant. The MAAP4 computer code developed by EPRI for pressurized light water reactors was used as a reference code for development of MAAP-ISAAC [4]. As the Wolsong NPPs, which are CANDU 6 type reactors, differ from typical PWRs, the Wolsong-specific features are newly modelled and incorporated into the MAAP-ISAAC code. The code was used to assist in supporting PSA and estimating source terms by analyzing the accident progression beyond core damage resulting to the containment failure.

2.2 Accident Sequence

The 3rd event sequence in event trees of asymmetric feedwater line breaks between the control valve and the check valve inside containment is selected. At this event sequence, a reactor trips and secondary side overpressure protection function successfully operates but the event proceeds to core damage by failure of both auxiliary and emergency water supply to SGs. The event scenario is as follows:

IE-FWB2 = IE-FWB2 * /RS * /SGPRC * /FBIA * AFW * EWS

· IE-FWB2 : asymmetric feedwater line breaks between the control valve and the check valve inside containment
·/RS : Success in reactor shutdown
·/SGPRC : Success in SG overpressure protection
·/FBIA : Success in automatic isolation of feedwater line
·AFW : Fail to secondary aux. feedwater operation
·EWS : Fail to emergency water supply operation.

2.3 Assumptions

The common assumptions for the analysis regarding the availability of systems are as follows:

- Auxiliary Feedwater System (AFWS), Emergency Core Cooling System (ECCS), Emergency Water Supply (EWS) System, Moderator Cooling System (MCS), End Shield Cooling System (ESCS), Local Air Coolers (LACs), Containment Filtered Venting System (CFVS), Passive Autocatalytic Recombiners (PARs) are assumed to be not available during the transient after the reactor shutdown.
- The primary heat transfer (PHT) pump is assumed to be tripped by the phenomenon of a pump

[1] MAAP is an Electric Power Research Institute (EPRI) software program that performs severe accident analysis for nuclear power plants including assessments of core damage and radiological transport. A valid license to MAAP4 and/or MAAP5 from EPRI is required.

cavitation when the auto trip condition or void condition of 50% in the PHT system is met.

- Airlock seal failure at the reactor building occurs at 262 kPa (g) and the breakage area is assumed to be 0.027871 m^2.

2.4 Modification of the MAAP-ISAAC Parameter File

A total of 2013 variables have been revised or added based on design data, operation data, and plant procedures of the Wolsong plants. Among them, 1047 variables are related to the auxiliary building. The input for auxiliary building is newly developed for all severe accidents leading to the failure of reactor building airlock seal [5].

The rest are variables reflecting the specific data of Wolsong plants, excluding the two variables reflecting the consultation results [6, 7]. And there are some of modified input variables reflecting the initial inventory of radionuclide and decay heat fraction using the latest ORIGEN code.

2.5 Analysis Results

The reactor was assumed to trip at 10 seconds due to SDS1 steam generator feedline low pressure which was the result from FSAR of Wolsong plants, and the operator recovery measures were not considered thereafter. The calculation of this event was carried out up to 72 hours after the start of the event requested in Level 2 PSA analysis, and the main event occurrence is summarized in Table I.

Table I: Problem Description.

Time	Description
10 sec	Reactor Trip due to SG Feedwater Low Pressure (FSAR)
18 sec	Dousing start
0.86 hr	LRV open
0.88 hr	4 SGs Dryout
1.15 hr	PHT Pump trip (50% void condition)
1.48 hr	Core Starts Uncover
1.72 hr	Loop 1 Fuel Channel Rupture due to Creep
2.96 hr	SAMG Condition
21.28 hr	Containment Airlock Seal Failure
50.14 hr	Calandria Failure
72.0 hr	Calculation Ends

After the reactor tripped, the main feedwater pumps are stopped immediately and the auxiliary feedwater pump is not available. However, it is assumed conservatively that the water through the break area is discharged for about 46 seconds. The pressure of the PHT system is gradually increased after the reactor trip as shown in Figure 1, and the LRVs are opened at about 0.86 hours, and then the pressure drops abruptly by the pressure tube rupture due to creep at about 1.72 hours. Figure 2 shows water level in the steam generator. The steam generator pressure increases at the beginning of the accident, and the MSSVs are opened at about 13.23 seconds. After that, the steam generator inventory starts to decrease. Since the loss of the PHT system inventory due to the PT/CT rupture reduces the heat transfer to the secondary side through the steam generator, the pressure in the steam generator starts to decrease and maintains the low pressure condition during the accident transient.

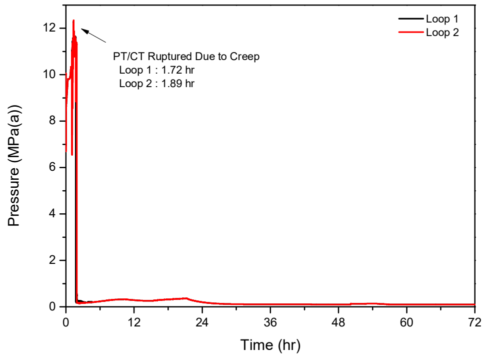

Fig. 1. PHTS Pressure for Case FWB2.

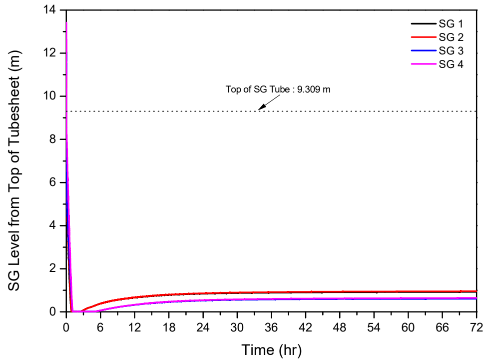

Fig. 2. SGs Level for Case FWB2.

The pressure in the containment is shown in Figure 3. A pressure in the containment begins to rise due to the water being discharged into the containment, reaching dousing spray initiation setpoint at about 18.33 seconds. The PHT pressure is reduced due to the PT/CT rupture occurring in about 1.72 hours. The pressure rise in the containment due to the coolant discharged through this part is stopped while the DCRV(Degasser Condenser Relief Valves) is closed. The rise of pressure in the containment continues due to the release of the moderator into the containment caused by the breakage of the calandria tank overpressure protection rupture disk. The airlock seal of the containment is failed in about 21.28 hours. Subsequent pressure in the containment is continuously reduced to the atmospheric pressure level until the time of the calandria tank failure occurring at about 50.14 hours. When the calandria tank fails, the pressure in the containment rapidly rises again after the sudden vaporization of the core melt moved into the reactor vessel, and then decreases to the atmospheric pressure level.

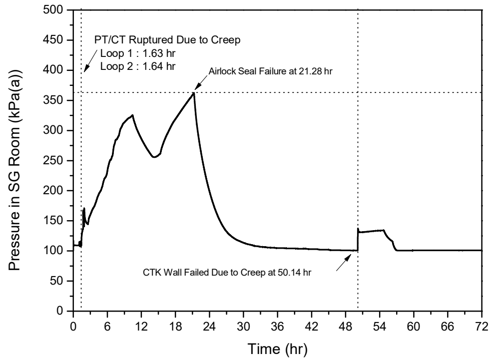

Fig. 3. SG Room Pressure for Case FWB2.

The core uncovery due to reduced coolant inventory starts from about 1.49 hours, and then fission products in the fuel cladding gap are released to the coolant in about 1.72 hours due to the damage by rise of the fuel cladding temperature. Inert gas Xe and Kr nuclei are released into the environmental atmosphere through normal containment leakage (0.5%/day). The remaining radionuclides begin to release after about 2.78 hours and rapidly increase at about 3.61 hours when the corium present in the core begins to move into the calandria tank, as shown in Figure 4. The radionuclide released into the atmosphere by the failure of the airlock seal of the containment which occurred in about 21.28 hours suddenly increases again, but it is a small amount for a certain period. However, when the calandria tank is broken, the amount of radionuclides emitted to the atmosphere was slightly increased by MCCI. When the debris gets expelled from the calandria tank to the reactor vault, MCCI could occur. As shown in Fig. 5, it can be checked by mass of hydrogen from the concrete whether the MCCI is occurred or not. After the calandria tank failed at 50.14 hrs, the hydrogen in the containment is equal to the mass of hydrogen from concrete. The subsequent emission of additional radionuclides is insignificant.

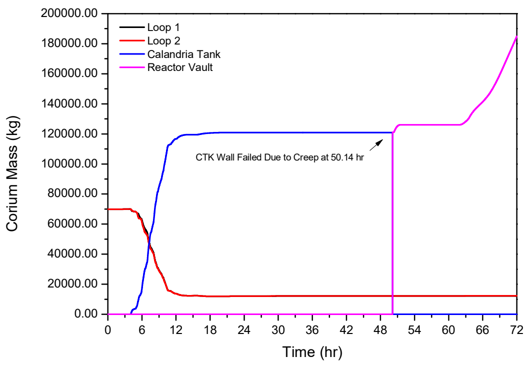

Fig. 4. Corium Masses in Both Loops and Calandria for Case FWB2.

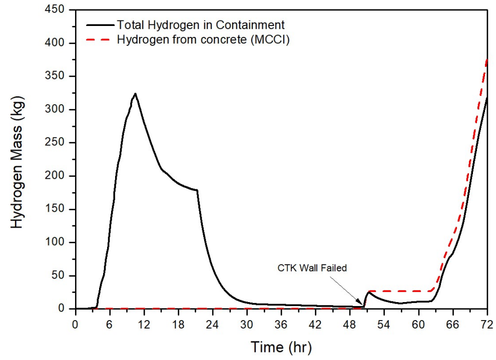

Fig. 5. Mass of Hydrogen for Case FWB2.

Figure 6 shows the Cs mass at each position. The Cs nuclides present in the core can be seen to migrate into the containment as the fuel melts and the core melt moves into the calandria tank. Figure 7 shows the form of Cs in the containment, most of Cs had deposited in the containment before the airlock seal failed. As shown in Figure 6, the mass of Cs emitted from the containment to the atmosphere is 0.07 kg, which is small compared with the mass of 25.05 kg of Cs deposited inside the containment.

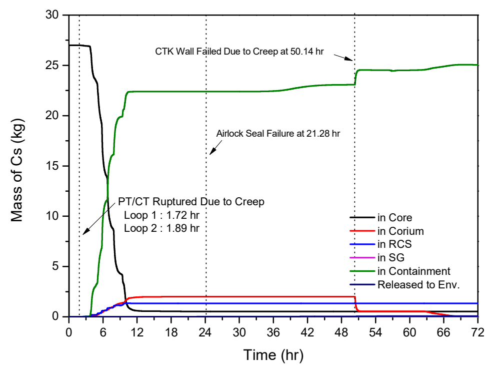

Fig. 6. Mass of Cs in Loops, Calandria and Containment for Case FWB2.

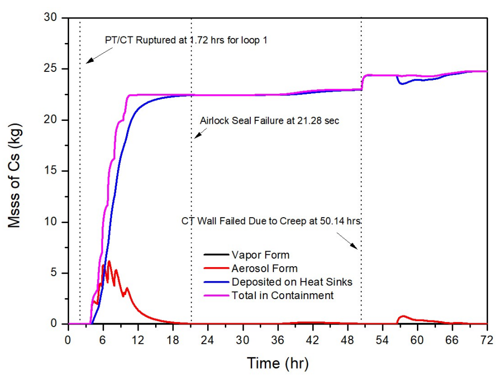

Fig. 7. Cs Mass Form in Containment for Case FWB2.

Hydrogen in the containment is appropriately removed by the PAR until containment airlock seal fails. After the airlock seal fails, the amount of hydrogen present inside the containment decreases sharply. After the collapse of the calandria tank, the amount of hydrogen in the containment increases slightly due to the MCCI phenomenon between the core melt and the concrete. This is because hydrogen is not removed by the PAR as the oxygen amount in the containment is exhausted. The amount of hydrogen at 72 hours of the accident is 237 kg, and the high vapor fraction effect in the containment does not cause overpressure in the containment due to the hydrogen explosion for 72 hours in the transient.

3. Conclusions

In this paper, the level 2 PSA thermal hydraulic analysis of asymmetric feedwater line breaks between the control valve and the check valve inside containment have been carried out. While the relevant accident sequence was covered in the FSAR, it has not been treated as a severe accident up to now. When operator actions are not taken, the analysis has shown that the release of Cs to the environment can exceed 100 TBq. However, if operation action was taken or mitigation measures such as LACs are carried out, it is expected that the release of Cs to the environment will be much lower compared to the present analysis result.

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REFERENCES

[1] USNRC, State-of-the-Art Reactor Consequence Analyses Project Volume 2: Surry Integrated Analysis, DC, NUREG/CR-7110 Vol.2, 2013.
[2] Kwang-Il Ahn, Seok-Won Hwang, The KHNP SOARCA Project: Development Strategy, KHNP Severe Accident and SOARCA Workshop, Sandia National Laboratories, Albuquerque, NM, April 17, 2017.
[3] KHNP, At-Power Internal Events Level 1 PSA for Wolsong Unit 1 Vol. 7. Quantification, 2015.
[4] KAERI, MAAP-ISAAC Computer Code User's Manual, KAERI/TR-3645, 2008.
[5] Bong-Jin Ko et. al., Wolsong Auxiliary Building Modeling for KHNP-SOARCA Project, Transactions of the KNS Spring Meeting, Jeju, Korea, May 17-18, 2018.
[6] Chul-Kyu Lim et. al., An Evaluation of Improved Distributed Heat Sink Models in MAAP-ISAAC 4.03 Code, Transactions of the KNS Autumn Meeting, Gyeongju, Korea, October 26-27, 2017.
[7] Chul-Kyu Lim et. al., Sensitivity Analysis of Improved Reactor Building Parameters Using MAAP-ISAAC 4.03 Code, Transactions of the KNS Spring Meeting, Jeju, Korea, May 17-18, 2018.