Sensitivity Analysis of Improved Reactor Building Parameters Using MAAP-ISAAC 4.03 Code

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1. Introduction

The reactor building is primarily designed to prevent or mitigate the uncontrolled release of radioactive materials to the environment in operational states and in accident conditions. The main purpose of this paper is to evaluate the effect of the improved variables related to reactor building on behavior of containment pressure and radiological transport during severe accident initiated by the station blackout (SBO) and the small loss of coolant accident (SBLOCA) in the Wolsong plants which are a typical CANDU-6 type using MAAP-ISAAC 4.03 (Modular Accident Analysis Program - Integrated Severe Accident Analysis Code for CANDU Plant)[1] computer program [1].

2. Scope of the Modeling

The reactor building is modeled by compartments (nodes), junctions (flow paths), heat sinks (distributed and lumped), and initial conditions, etc. in the MAAP-ISAAC code. As shown in Figure 1, the compartments and the junctions consist of 13 nodes and 26 flow paths. These nodes are connected by flow junctions. The heat sinks consist of 34 distributed heat sinks and 12 lumped heat sinks as shown in Figure 2 [2].

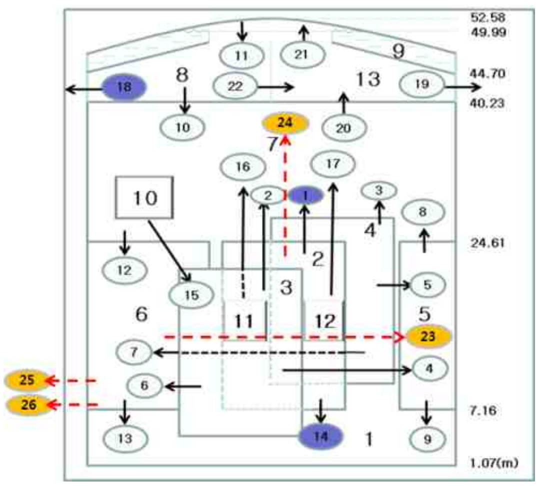

Figure 1. MAAP-ISAAC Modeling of the Reactor Building [2].

The junctions 23, 24, 25 and 26 are not modeled in the existing parameter file. The junction #23 is connecting the moderator room and the access area, the failure junction #24 is the rupture disk installed to protect the overpressure of the calandria vault, the junction #25 is the containment filtered vent system (CFVS) for the controlled depressurization of containment atmosphere under severe accident condition, and the failure junction #26 is modeled as airlock seal failure. In the case of heat sinks, 175 variables were improved including the existing distributed heat sink #18 divided in detail as five distributed heat sinks (#18, #31, #32, #33 and #34) [3].

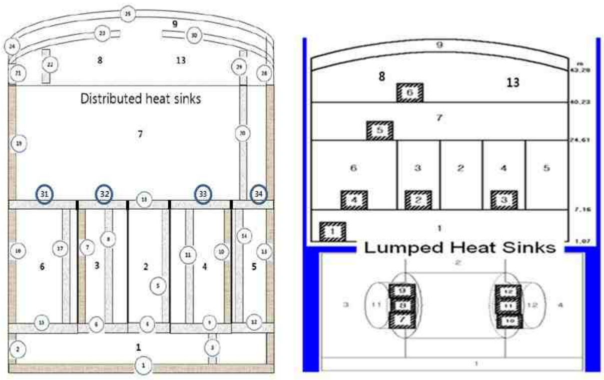

Figure 2. Distributed & Lumped Heat Sinks in the Reactor Building [2].

In the MAAP-ISAAC code, there are a total of 1414 variables related to reactor building. Among them, 402 variables modified based on design data, operation data, and procedures for the reactor building system of the Wolsong plants were shown in table 1 [4].

3. Method and Results

3.1 Description of Analyzed Cases and Assumptions

Selected cases are the station blackout (SBO) [5] which is representative pressurization accident and the small loss of coolant accident (SBLOCA) [6] which is representative de-pressurization accident. The common assumptions regarding the availability of systems are as follows:

[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.

- Auxiliary Feed Water System (AFWS), Emergency Core Cooling System (ECCS), Emergency Water Supply (EWS) System, Moderator Cooling System (MCS), End Shield Cooling System (ESCS), Local Air Cooler (LAC), Containment Filtered Venting System (CFVS), Dousing System (DS), all Passive Autocatalytic Recombiners (PARs) are assumed to be not available during the transient.
- It is assumed that there are no reactor building failure and airlock seal failure.

The analysis results during SBO and SBLOCA are compared for those by using existing variables. Since the number of MAAP-ISAAC input variables related to the reactor building is large, four types are divided as followings: 1) Compartment, 2) Junction, 3) Initial condition, 4) Heat sink.

Table 1: List of Major Improvements

Classification	Variable		Number (#)
Compartment	VOLRB(1)~(7),(9),(11),(12), AGKEY(1)~(13), XRBRRB(1)~(13), XHBRRB(2),(8)~(10),(13), XRBLK(2,2),(3,2),(4,2),(5,2),(2,8),(2,10),(2,13), VRBLK(4,1),(2,2),(3,2),(4,2),(5,2),(2,3),(2,4),(2,5),(2,6),(2,7),(4,9),(5,9),(2,11),(2,12), ASEDRB(1)~(7),(10)~(12), ACMPLB(1)~(2)		74
Junction	Failure #1, #16~#18, #24, #26, Junction #2~#10, #12~#13, #20, #23, #25		114
Initial condition	TCN0, ZWCVMU, TWMUCV, TGRB0(1)~(13), PRB0(1)~(10),(13), FRHB0(9),(10), XWRB0(2),(10), TWRB0(2),(9)~(12), MSPRB0, WWSTHX, WWESHX		39
Heat sink	Distributed heat sink	HSTYP(31)~(34), NIWALL(6),(9),(12),(15),(18),(31)~(34), AHSRB(1)~(21),(24)~(28),(31)~(34), XTHSRB(1)~(21),(24),(28),(31)~(34), XHSRB(5),(21),(22),(25),(28),(29),(31)~(34), ZHSRB(31)~(34), N1HSRB(5),(25),(31)~(34), N2HSRB(5),(6),(9),(12),(15),(18),(25),(31)~(34), XLN1(4),(5),(31)~(34), XLN2(31)~(34), RG1(31)~(34), RG2(31)~(34), WWSP1(31)~(34), WWSP2(31)~(34), TWSP1(31)~(34), TWSP2(31)~(34), XPERHS(1)~(34)	175
	Lumped heat sink	ZEQRB(7)~(12)	

3.2 Results and Discussion

The analysis results of the pressure in the reactor building are shown in Fig. 3 and Fig. 4. As shown in the figures, in the analysis case of using only modified variables related to compartment (Mod-Compartments), it can be confirmed that the time of corium release from the calandria tank to the calandria vault, in other words, the time of the failure of the calandria tank was delayed and the reactor building pressure behaviors were shifted slightly to the right than cases of using existing variables (Reference). This comes from that the input variable for total free volume of each compartment is increased compared to the existing input variable. In the case of changing input parameters related to the junctions (Mod-Junctions), there is no significant difference from the reference case (Reference) except that the reactor building pressure behavior was moved slightly to the right than the reference case. In the analysis case of using only modified variables related to initial conditions (Mod-Initial conditions), the results were similar to 'Mod-compartments' cases because the initial water level and the cooling water mass in the calandria vault are increased. In the case of the modified input parameters related to the heat sinks (Mod-Heat sinks), it has the greatest influence on improved parameters relevant to reactor building and the pressure in reactor building has been significantly reduced than the reference case. This is due to the increased area of the distributed heat sink. The locations of the lumped heat sink have no effect on the analysis results.

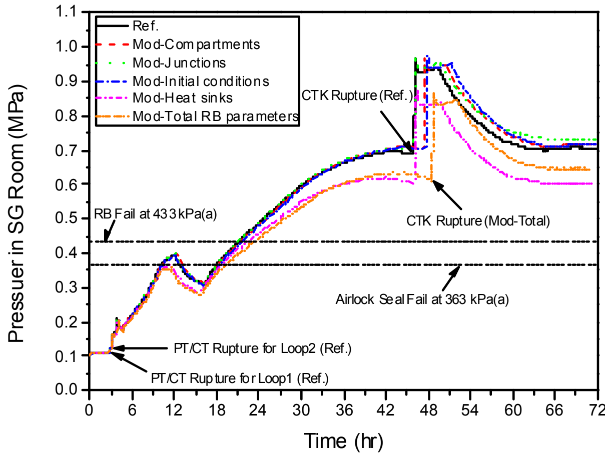

Figure 3. Pressure Transient in the Reactor Building during SBO.

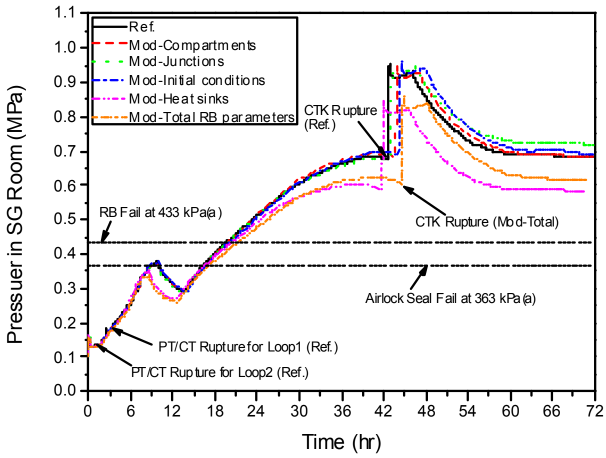

Figure 4. Pressure Transient in the Reactor Building during SBLOCA.

For the case of all parameters related reactor building modified (Mod-Total RB parameters), the times of airlock seal failure, reactor building failure, and calandria tank failure were delayed from 10.1 hours to

19.09 hours, from 21.06 hours to 23.28 hours, from 45.97 hours to 49.35 hours, respectively compared to the reference case during SBO. For SBLOCA case, the times of airlock seal failure, reactor building failure, and calandria tank failure were delayed from 8.42 hours to 16.94 hours, from 19.58 hours to 21.16 hours, from 42.55 hours to 45.74 hours, respectively compared to the reference case. And these results have the effect of reducing the mass release fraction of CsI from the reactor building to the external environment. For the airlock seal failure case during SBO and SBLOCA, the mass release fraction of CsI are sharply reduced from 0.1868 to 0.0351 and from 0.1746 to 0.0308, respectively because the deposition effect in the reactor building is increased as the time of airlock seal failure is delayed as shown in Fig. 5 and 6. But as the first pressure peaks in Fig. 3 and 4 in modified cases are so close to the airlock seal failure pressure, the results of failure time delay could change easily considering uncertainties in them.

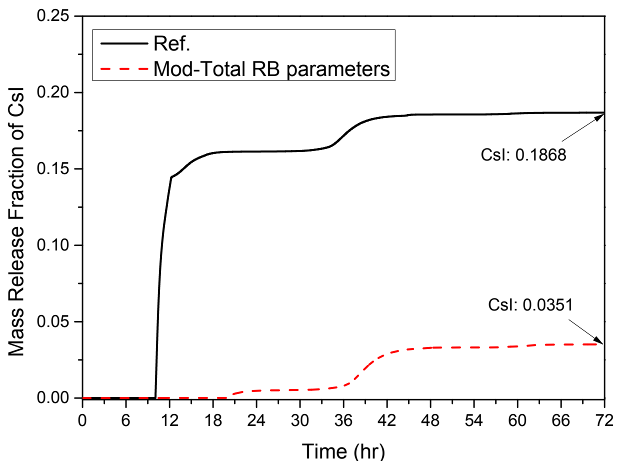

Figure 5. Total Mass Fraction of CsI Released to the Environment for the Airlock Seal Failure Case during SBO.

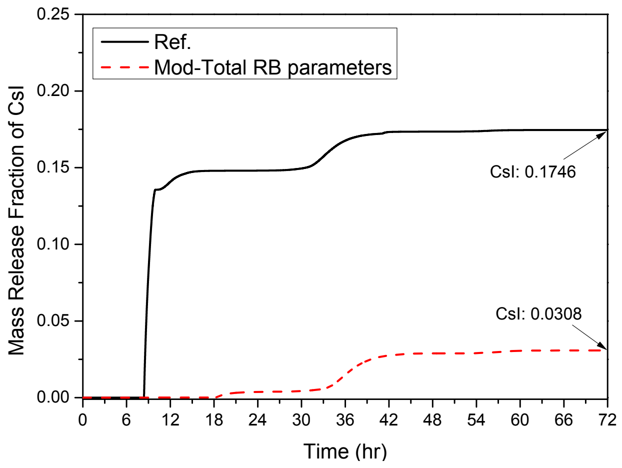

Figure 6. Total Mass Fraction of CsI Released to the Environment for the Airlock Seal Failure Case during SBLOCA.

4. Conclusions

The parameters relevant to reactor building of MAAP-ISAAC 4.03 which is the severe accident analysis code were improved and their effects were evaluated. As a representative accident, the station blackout (SBO) and the small loss of coolant accident (SBLOCA) in the Wolsong plant were selected.

As a result of the analysis, it was confirmed that distributed heat sinks were the greatest influence of the input variables related to reactor building. In the analysis case of using modified parameter file (Mod-Total RB parameters), the progress of the severe accident was delayed and mitigated. In other words, the times of airlock seal failure, reactor building failure, and calandria tank failure were delayed and the mass release fractions of CsI were sharply reduced than cases of using existing parameters (Reference).

Therefore, it may be necessary to improve the parameters relevant to reactor building. The analysis result that reflects the modification of the input parameters can be applied to the improvements such as the Level 2 PSA or the severe accident management guidance (SAMG) which use the result of the severe accident analysis.

REFERENCES

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