Analysis of the Interfacing System LOCA Source Term for a PHWR

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1. Introduction

The reactor building bypass sequences are distinctly different from the non-bypass sequences in that there exists a direct flow path from the primary system to outside the reactor building boundary which bypasses the main reactor building gas volume. Hence the gravity fallout and engineered mitigation features inside the reactor building become ineffective for reducing a fission product release into the environment. Consequently, bypass sequences can result in relatively large source term releases soon after the onset of a core damage. The reactor building bypass sequences can be further subdivided into an interfacing system LOCA (ISLOCA or Event V) group and steam generator tube rupture (SGTR) group, because the radionuclide release pathways are different. Both of these have been defined by a plant damage state attribute "Reactor building Bypass" [1]. The purpose of this paper is to analyze the interfacing system LOCA source term for the Wolsong plants.

2. Methodology of Source Term Evaluation

A particular release category consists of a group of Containment Event Tree end points which have similar source term release characteristics. Once the release categories are set up, various accident sequences are allocated to that category.

To select the representative sequence for a specific release category, the following processes are used:

1) Select the plant damage state (PDS) with the largest contribution to the release category's total frequency
2) Among the accident sequences corresponding to the PDS, choose the dominant sequence for the release category. This defines the initiating event and the status of the various plant systems
3) The CET and plant accident sequence definitions are retrieved to determine if any special phenomenological conditions have to be specified
4) A reactor building failure pressure, failure time and failure condition are specified based on the release category definition

In this study, deterministic analyses of representative sequences from each significant release category were performed with an accident progression source term assessment code. The ISAAC (Integrated Severe Accident Analysis Code for CANDU Plant) computer code [2], which is an accident progression source term assessment code, is used for this purpose.

3. Accident Analyses

For Event V, the most likely failure point occurs in the auxiliary building because of a check valve failure between the PHTS and the emergency core cooling system (ECCS), placed outside the reactor building. Hence the release pathway becomes the PHTS, ECCS piping and the auxiliary building. The only factor that affects the source term assessment is whether or not the release point in the auxiliary building is submerged at the time fission product releases are occurring. If the break point in the auxiliary building is located under water, a scrubbing effect through the water pool can reduce the amount of soluble fission product. However, these decontamination effects are not considered here. No account of any other auxiliary building release reduction mechanisms is taken.

To simulate the Event V sequence, a eighteen inch diameter emergency core cooling pump line is assumed to be broken in the auxiliary building. The open area between the auxiliary building and the environmental atmosphere is also assumed to be 9.29 m^2 ($100ft^2$). This study assumes that the safety systems such as the Emergency Core Cooling System, Shield Cooling System and Moderator Cooling System are unavailable. Dousing spray system is automatically initiated when the containment pressure exceeds 2 psig and stops when the pressure decreases below 1 psig, resulting in a cycling operation. And 72 hours of a mission time is assumed.

Figure 1 shows the mass fraction of CsI in the reactor building and the environment for the Event V sequence. All the mass fractions of the fission products released to the environment for the Event V sequence are summarized in Table 1. The fraction of the initial inventory released from the fuel travels from the core through the safety injection piping to the auxiliary building with a certain fraction being deposited in the calandria and piping on the way to the break, while a part of the fission products escapes to the reactor building. An initial CsI inventory of 0% remains in the calandria, 9% of the CsI remains in the Loop 1 PHTS, 29% of the CsI remains in the Loop 2

PHTS, 50% is transported to the reactor building or auxiliary building and 12% is released to the environment (Figure 1, Table 1) within seventy-two hours after the accident is initiated.

Figure 2 shows the mass fraction of the noble gases, CsI and CsOH released into the environment for the Event V sequence. About 96%, 13.2% and 12% of the initial inventories of the noble gases, CsOH and CsI are released to the environment respectively.

4. Conclusion

The Interfacing system LOCA source term for the Wolsong plants is analyzed. An initial CsI inventory of 50% is transported to the reactor building or auxiliary building, 29% of the CsI remains in the Loop 2 PHTS, 12% is released to the environment, 9% of the CsI remains in the Loop 1 PHTS. About 96%, 13.2% and 12% of the initial inventories of the noble gases, CsOH and CsI are released to the environment respectively.

Table 1 Fractional Source Term Release to the Environment for V Sequence

STC	V sequence
NOBLE GAS	9.6E-01
CsI	1.2E-01
TEO2	2.3E-02
SRO	1.8E-04
MOO2	4.5E-05
CsOH	1.3E-01
BAO	1.6E-04
LA2O3	2.5E-06
CEO2	1.2E-05
SB	3.8E-03
TE2	2.2E-06
UO2	3.3E-08

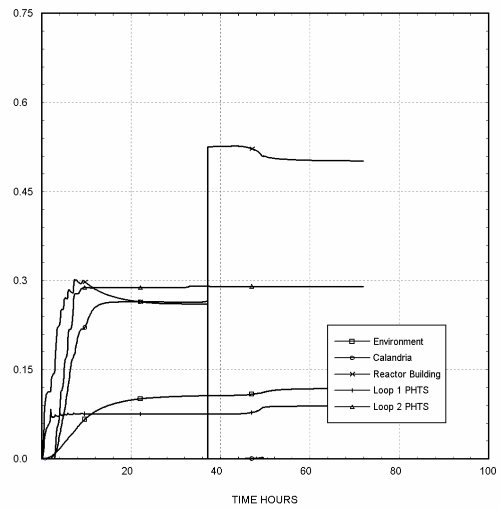

Figure 1 The Mass Fraction of CsI in the Reactor Building and the Environment for the Event V sequence

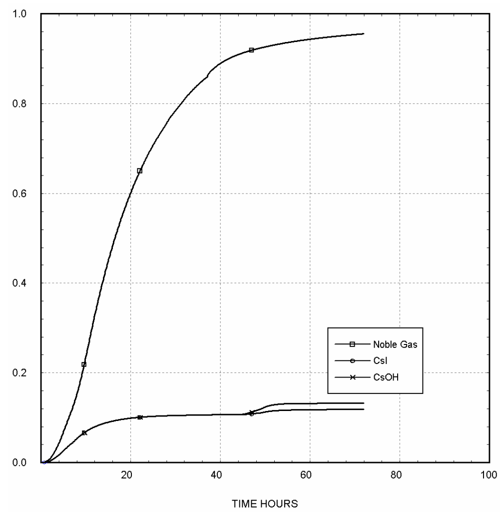

Figure 2 The Mass Fraction of the Noble Gases, CsI and CsOH Released into the Environment for the Event V sequence

Acknowledgement

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REFERENCES

[1] KHNP. "Probabilistic Safety Assessment for Wolsong Unit 1", December, 2003.
[2] Dong Ha Kim, "Development of Computer Code for Level 2 PSA of CANDU Plant", KAERI/RR-1573/95, December 1995.