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ISLOCA Analysis for a Typical 1000MW PWR: Accident Progression, FP Distribution, and Accident Management

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1. Introduction

An interfacing system loss of coolant accident (ISLOCA) is a loss of primary coolant outside the containment via a system that interfaces with the re-actor coolant system (RCS) and for which the pressure boundary is outside the containment. An ISLOCA is presumed to result from exposing low pressure piping of the interfacing system to the full pressure of the primary system due to failures of multiple pressure barriers.

An ISLOCA will provide a direct path, which bypasses the containment, for the release of radioactive materials into the environment. The fission product behaviors with a modeling of an auxiliary building, which have not been evaluated for domestic nuclear plants in Korea, are very different from the behaviors without the modeling of an auxiliary building, and are scenario-specific. Accordingly, a scenario-specific analysis of the fission product source terms during a severe accident might be noteworthy. This paper is intended to evaluate the characteristics of a fission product source term release or distribution during an ISLOCA with the modeling of an auxiliary building in a typical 1000 MW PWR. In addition, an evaluation is included for the effectiveness of external cooling water injection strategies using fire trucks during a potential ISLOCA accident.

2. Methodologies

2.1 ISLOCA in Reference Plant

A brief outline of a typical 1000MWe PWR design with special reference to the mitigation capability during an ISLOCA is provided in this section.

The reactor uses pressurized water with a core thermal output of 2815 MWth. In the highly unlikely event of a loss-of-coolant accident, the safety injection system (SIS), including high-pressure and low-pressure safety injection pumps, a refueling water tank, and safety injection tanks, inject borated water into the reactor coolant system. This provides cooling to limit the core damage and fission product re-lease and ensures an adequate shutdown margin. The SIS can also provide continuous long-term, post-accident cooling of the core by the recirculation of borated water from the containment sump. However, the recirculation mode through the containment sump is not available for the ISLOCA sequences.

Potential paths for ISLOCA are identified from [1]. After the screening process for the potential paths is performed, the remaining two are considered as ISLOCA sources in this analysis. The two potential paths are located at the chemical volume and control system letdown line and shutdown cooling system suction line. Figures 1 and 2 show simplified diagrams of the paths for the chemical volume and control system (CVCS) letdown line and shutdown cooling system (SCS) suction line. In the ISLOCA sequences, the tentative pathways are (1) RCS, chemical volume and control system piping, and the auxiliary building, and (2) RCS, shutdown cooling system piping, and the auxiliary building. 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.

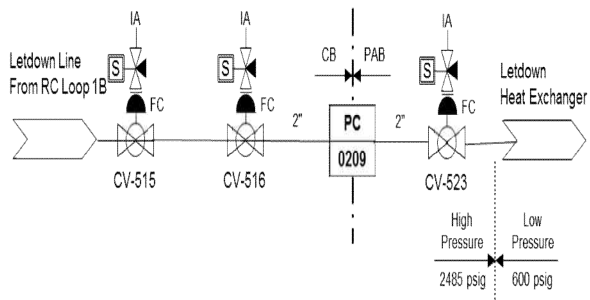

Figure 1. Simplified diagram of CVCS letdown line

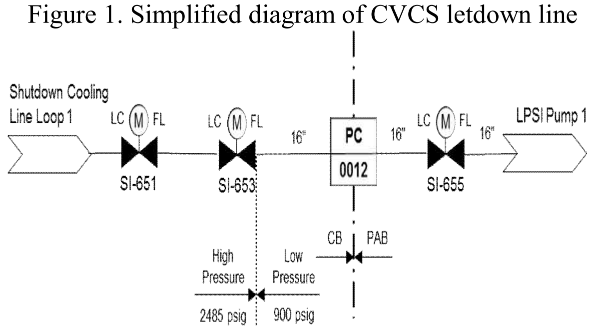

Figure 2. Simplified diagram of SCS suction line

2.2 Analyzed ISLOCA Cases

Three representative sequences are selected: (1) CVCS letdown line break sequence (Case 1), (2) SCS suction line break without water pool scrubbing effect

(Case 2), and (3) SCS suction line break with water pool scrubbing effect (Case 3). To simulate a Case 1 sequence, a 2 inch diameter CVCS line (be-tween PC0209 and CV523 in Figure 1) is assumed to be broken in the auxiliary building, and a 16 inch diameter SCS line is assumed to be broken in the auxiliary building for Case 2 (between PC0012 and SI-655 in Figure 2) and Case 3 (between SI-655 and LPSI pump in Figure 2).

2.3 Analysis Program

Thermal hydraulic, severe accident phenomenological, or radiological analyses for an evaluation have been conducted using the PWR version of MAAP (Modular Accident Analysis Program) version 5 [2]. The code is a system level computer code capable of performing integrated analyses of severe accident progression, supporting level 2 probabilistic safety assessment studies or accident management strategy developments.

3. Accident Progression and Radiological Analyses

Fission product elements are grouped into 18 fission product species in the MAAP code. Among these fission product groups, noble gas and cesium iodine groups are considered as more volatile. Although noble gases carry significant amounts of activity during a severe accident, they commonly have only a short-term impact as they have mostly a very short half-life and quickly disperse into the atmosphere without causing land contamination. A noble gas, which is a chemically inert material, poses considerably less danger to human health. Instead, iodine is one of the elements vital to the proper functioning of the human body. Iodine exits from the damaged fuel rods predominantly as cesium iodine (CsI) rather than as molecular iodine (I2). Therefore, the CsI group is selected as a representative fission product. 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 vessel and piping on the way to the break, while a part of the fission products escapes into the containment building.

The following shows the calculation results of three representative sequences of the interface system LOCA.

3.1 Case 1

The result of the MAAP calculation shows that the core uncovery begins from 9.1 hours, and the reactor vessel fails at 12.3 hours after the accident initiation. The CsI release into the environment was initiated when the fuel damage occurred at about 10.3 hours. An initial CsI inventory of 100% is released in the reactor vessel, 69.4% of CsI remains in the reactor coolant system, 29.8% is transported to the containment or auxiliary building, and 0.7% is released into the environment by 72 hours after the accident is initiated (Figure 3).

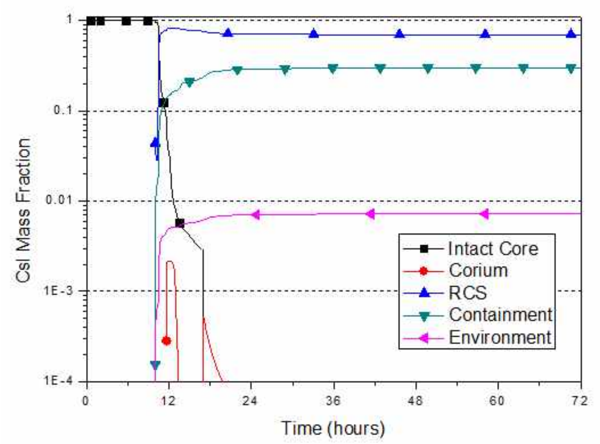

Figure 3. Distribution of CsI Group for Case 1.

3.2 Case 2

The core uncovery begins from 3.7 hours and the reactor vessel fails at 6.2 hours after the accident initiation. The CsI release into the environment was initiated when the fuel damage occurred at about 4.4 hours. An initial CsI inventory of 100% is released in the reactor vessel, 8.2% of CsI remains in the reactor coolant system, 78.8% is transported to the containment or auxiliary building, and 13.0% is released into the environment by 72 hours after the accident is initiated (Figure 4).

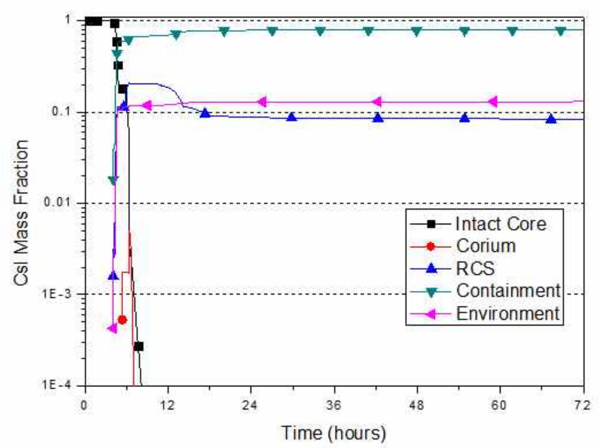

Figure 4. Distribution of CsI Group for Case 2.

3.3 Case 3

The core uncovery begins from 3.8 hours and the reactor vessel fails at 6.4 hours after the accident initiation. The CsI release into the environment was initiated when the fuel damage occurred at about 4.4 hours. An initial CsI inventory of 100% is released in the reactor vessel, 9.4% of CsI remains in the reactor coolant system, 83.0% is transported to the containment or auxiliary building, and 7.6% is released into the environment by 72 hours after the accident is initiated. In this case, because the break point in the auxiliary building is located under water at the initial stage, a scrubbing effect through the water pool can significantly reduce the amount of soluble fission products; however, the water pool is dried out at about 30 hours, and the release increases owing to the re-vaporization (Figure 5).

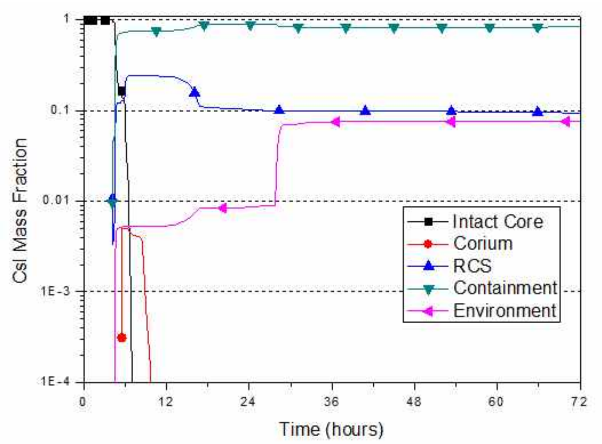

Figure 5. Distribution of CsI Group for Case 3.

4. Accident Management

The initial plant response to an ISLOCA is the same as the response to an equivalent sized loss of coolant accident (LOCA) inside the containment. However, the RCS inventory and consequential makeup inventory are discharged outside the containment, and are not returned to the containment sump. When the refueling water tank (RWT) inventory for the RCS makeup is depleted, the safety injection (SI) pumps will have no water source to provide a makeup to the RCS. The core will be uncovered and a core melt will occur.

One of the measures to increase the mitigation capability during extreme ISLOCA accidents is to provide emergency cooling water of external sources using fire engines into the reactor cooling system (RCS). An emergency cooling water system consists of a fixed pipe connected from the RCS to the outside of the containment. A standby valve is installed on the pipe. Following the occurrence of an ISLOCA, movable equipment (for example, a fire truck hose) can be connected to the pipe hole at the opening of the isolation valve. Nevertheless, the emergency cooling water injection strategy was not taken into account for the ISLOCA accident management strategy of the reference plant. Therefore, it is necessary to develop some guidelines or strategies to cope with extreme ISLOCA scenarios using the newly installed injection flow paths and fire engines.

The USNRC performed a state-of-the-art reactor consequence analysis (SOARCA) to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents. The availability of the external cooling water injection time was assessed to occur at 3.5 hours [3], which include the following: (1) initial plant status assessment by operators, (2) attempt to start an emergency diesel generator (EDG) manually, (3) manning and operation of the onsite technical support center (TSC) and offsite emergency operations facility (EOF), (4) decision-making of TSC and EOF for the recommendation of operator actions, and (5) operator's assessment and implementation of recovery actions.

Once the RWT inventory for the RCS makeup is depleted during an ISLOCA accident, an external cooling water injection strategy is the only mitigation measure. If external water injection is employed as an accident management strategy, the available time for the operators to respond might be a key feature for a successful strategy implementation.

Based on the MAAP calculation results described in the previous section, the core uncovery of Case 2 begins from 3.7 hours where the HPSI pump flow rate is not controlled. The HPSI pump flowrate can be controlled if an adequate water level can be maintained to minimize the spill rate into the auxiliary building. Figure 6 shows the difference in the core uncovery time between the cases with (Case 2) or without (Case 4) the HPSI pump flow rate control. The core uncovery time of Case 4 is delayed to 92.4 hours.

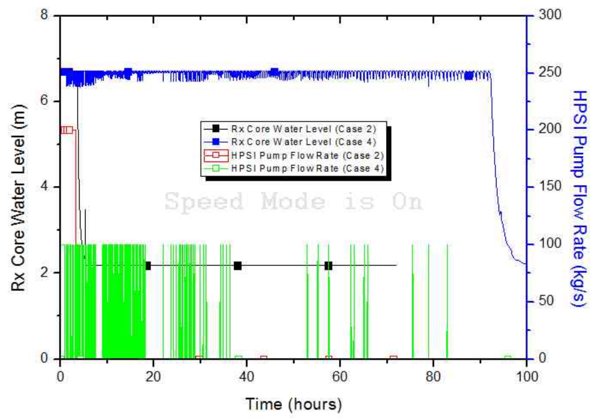

Figure 6. HPSI Pump Flow Rate and Rx Core Water Level for Case 2 and Case 4

The analysis results led to the summary that, with regard to the external cooling water injection strategies using fire engines, an HPSI operation with flow control is the key mitigation feature for a successful strategy implementation.

ACKNOWLEDGEMENT

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