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Analysis of the Radioactive Release Characteristics for Development of the Containment Bypass Mitigation System

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1. Introduction

In recent years, a growing demand to strengthen safety regulations accompanied the increase in public distrust of the nuclear power plants (NPPs). To reduce off-site radiological consequences from NPP severe accidents, the nuclear industry and the regulatory agencies have begun to develop mitigative technologies and strategies that can limit the spread of radioactive material into the environment. One of the most popular such technology used in many NPPs is called containment filtered venting system (CFVS), which would vent the radioactive source term from the containment into the environment through filtration before the containment pressure buildup, thereby thus reducing chance of containment failure.

There has been many studies to reduce consequences of containment failure accidents, such as station blackout accident. However, according to the U.S. NRC State-of-the-Art Reactor Consequence Analysis (SOARCA), containment failure accident is not the only type of accident that poses significant risk to the environment and to the public. Though the containment bypass accidents are much less likely to occur compared with the containment failure accidents, consequences of the bypass accidents are more catastrophic [1]. Therefore, new technologies similar to CFVS should be developed to mitigate the consequences of the containment bypass accidents.

The purpose of this study is analyze the radioactive release characteristics from the containment bypass accidents. These characteristics may be required to develop environmental dispersion mitigation system for containment bypass accidents. The goal of developing this new technological system is to effectively capture and treat radioactive materials released from bypass severe accidents.

2. Required Radioactive Release Characteristics for Designing the Mitigation System

The bypass mitigation system will be comprised of two separate systems with different purposes. There should be a system to capture the released radioactive materials, and another system should filter and treat the captured radioactive materials. This concept is shown in Figure 1 for two bypass accidents: steam generator tube rupture (SGTR) and interfacing systems loss-of-coolant accidents (ISLOCA).

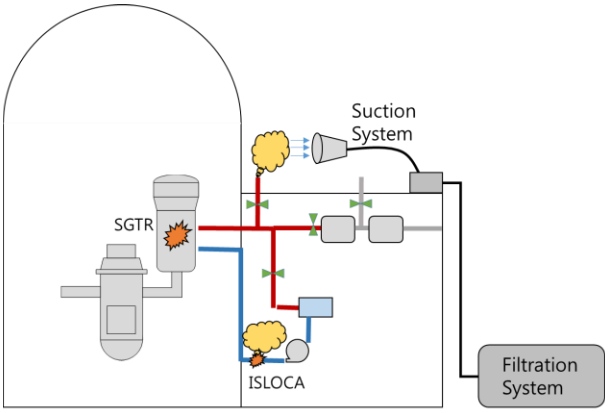

Figure 1. Concept of a Bypass Accident Mitigation System

To design such mitigation system, radioactive source term information is required. From previous studies of atmospheric releases (and accident experience), the most important radioactive materials that should be treated by the mitigation system are iodine and cesium, because these two contribute most of the environmental radioactivity. These may be released as some form of CsI, CsOH, Cs_2O, and I_2, in gas or aerosol form. Integrated severe accident codes such as MAAP and MELCOR can calculate the amount and the timing of the environmental releases during the severe accidents [2]. However, these severe accident codes cannot give other information required to design the mitigation system, because designing the suction system requires extra characteristics of the radioactive materials (including the released steam/water) *after* the environmental release occurs. This information should include the flashing of the released materials as well as the dispersion after the release. Figure 2 shows the required characteristics and the future research roadmap on how these characteristics may be found.

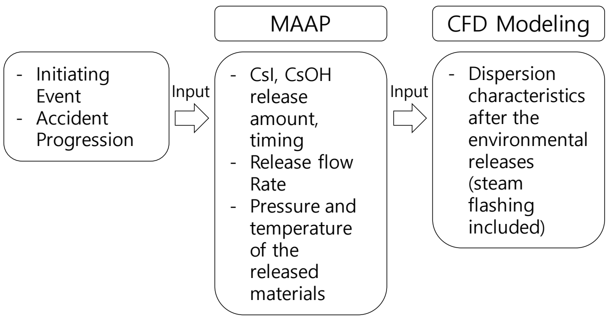

Figure 2. Required source term characteristics for analysis

3. Current Work using MAAP4

To effectively carryout the research roadmap on finding the required source term characteristics, it was required to define a representative bypass accident. The representative accident chosen for the beginning of this research is a thermally-induced steam generator tube rupture (TI-SGTR), based on the NRC SOARCA. The aforementioned accident was chosen, because it was a bypass accident that is a main contributor to mean early and latent cancer fatality risks for the public [1].

For the accident scenario, it was assumed that some significant external event took place that the loss of AC power occurred without recoverable offsite alternatives. It was also assumed that a loss of DC power occurred due to the external event (such as extremely powerful earthquake). Even turbine-driven auxiliary feedwater pump was unavailable. Due to loss of power, the main steam safety valve (MSSV) would be used to remove heat energy from the reactor. However, at 3 hours after repeated opening and closing of the MSSV, one with the lowest set opening pressure would get stuck open (i.e. MSSV failure). That would ultimately cause TI-SGTR, with loss of AC and DC power for mitigation, and the fission products from the primary side would be leaked to the secondary and ultimately into the environment through stuck-open MSSV.

The TI-SGTR results from the SOARCA assumed hot-leg creep failure about 45 minutes after the main steam safety valve being stuck open, which greatly reduced the amount of fission products released through containment bypass by releasing majority of the fission products into the containment through hot-leg failure. However, for the primary conservative calculation, it was assumed in current research that the hot-leg creep failure did not occur and that majority of the fission products were released through containment bypass. Otherwise, similar scenario as unmitigated TI-SGTR from the SOARCA was taken. An OPR-1000 plant was used to analyze the accident characteristics.

Using MAAP4 code, some of the results were found below. The time is zero at the accident initiation. Figures 3 and 4 can be used for verification purposes with the TI-SGTR data from the SOARCA (Figures 5-61 and 5-62 in the SOARCA report), which used Surry as a reference plant [1].

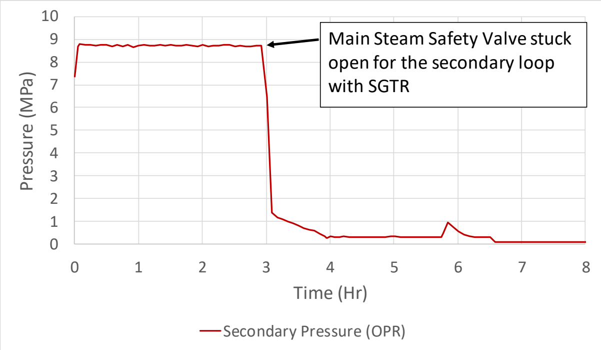

Figure 3. Unmitigated TI-SGTR secondary pressure

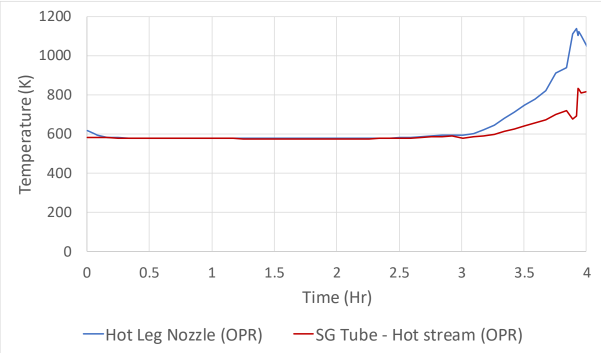

Figure 4. Unmitigated TI-SGTR hot leg nozzle and steam generator tube-hot stream interface temperature

As shown in Figure 3, first three hours maintain near the lowest pressure limit for the main steam safety valve (MSSV), as the MSSV opens and closes rapidly. However, at 3 hours, the MSSV fails and gets stuck open, which results in rapid depressurization of the secondary side with the SGTR. Within one hour, the pressure drops rapidly from around 8.7 MPa, and afterword the pressure stays at near atmospheric pressure. Figure 4 shows the hot leg temperature as it goes into the steam generator.

The flow rate through MSSV into the environment is shown in Figure 5. Before about 3 hours, the MSSV opens and closes very rapidly, so the actual flow rate is below the design flow rate of the MSSV at the design pressure (122.6 kg/s at 8.72 MPa). When the MSSV fails and gets stuck open after about 3 hours, the flow rate peaks at around 36 kg/s, which then stays above 3 kg/s for few hours and then drops to less than 0.5 kg/s afterword.

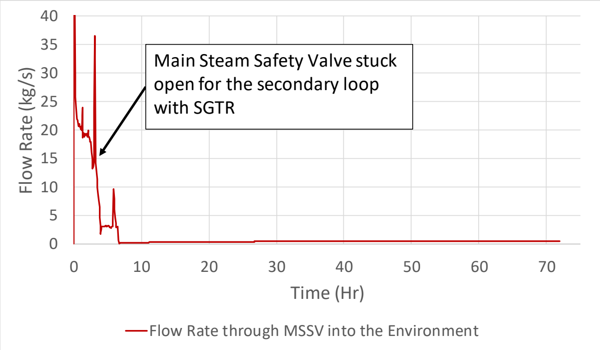

Figure 5. Environmental flow rate through MSSV after the SGTR

In the SOARCA, hot-leg creep rupture occurred within an hour of MSSV being stuck open, thereby releasing majority of the radioactive materials into the containment. However, the conservative assumption used in this research for strictly having environmental releases through containment bypass (i.e. no hot-leg creep rupture) resulted in very high environmental mass releases. Example of that is shown in Figure 6 for CsI, where mass fraction of CsI in the reactor coolant system, fraction in the steam generator, and fraction released into the environment are compared.

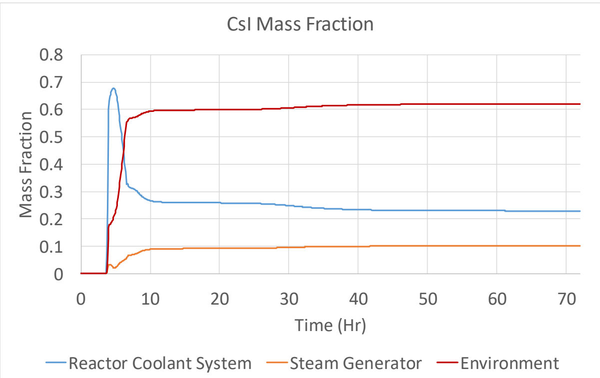

Figure 6. CsI mass fraction information in the TI-SGTR accident with no hot-leg creep rupture

Unlike the results from SOARCA where the majority of the fission product from the reactor coolant system gets transferred into the containment through hot-leg creep rupture, majority of the fission products in the simulation were released into the environment through containment bypass (assuming no hot-leg creep rupture). Furthermore, majority of the environmental releases occurred within 10 hours of accident initiation. Therefore, in case of such accident, the mitigation system should be installed within 3~4 hours after the initiating event, should loss of AC and DC occur at the same time due to some kind of catastrophic event.

4. Future Work

To design the containment bypass mitigation system, information from integrated severe accident code such as MAAP is not enough; the dispersion characteristics of the environmentally released radioactive material must be known. To do so, the future plan of this research is to model using CFD the behavior of the released materials. The goal within next few months is to use output from the integrated severe accident code as input for the CFD modeling to predict the radioactive release behavior after being released through the MSSV.

Acknowledgment

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