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The Plant-specific Impact of Hot-leg and Cold-leg Break LOCAs on the Evolution of a Severe Core Degradation

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1. Introduction

The main objective of this paper is to provide the MELCOR 1.8.5 analysis results on the plant-specific impact of RCS hot- and cold-leg breaks on the evolution of a severe core degradation and the insights from the analyses. The APR 1400 has been chosen as a referenced plant for this purpose. The analysis results show that the RCS hot-leg break LOCA leads to a much faster and severer degradation of the reactor core when compared with the cold-leg break of the same size. This seems different from the existing viewpoint that the cold-leg break leads to a severer result than the hot-leg break of the same size and thus the existing severe accident management (SAM) has mainly focused on the cold-leg break. An in-depth study is required to clear up this issue which may be a unique feature of plant systems.

2. Methods and Results

The APR (Advanced Power Reactor) 1400 [1] is a 2-loop pressurized water reactor (PWR) with a large dry containment, whose electric power has been designed to be 1400 MWe. Unlike most other reactor systems in which the emergency core cooling (ECC) water is injected into the cold-legs, the APR 1400 employs a concept of a direct vessel injection (DVI) to reduce the bypass effect of the ECC water via a break during a design basis LOCA. The DVI takes a suction of water from an in-containment refueling storage tank (IRWST) which was designed so that all the RCS and Spray water released into the containment flow into it.

2.1 Accident Conditions

The present LOCA sequence is assumed to occur with a size of 0.5 ft^2 (9.56 inches in diameter) for the break in the cold-leg of the primary loop connected with a pressurizer surge line in one case and a hot-leg break of the same size in the other case. This size of the RCS break corresponds to a boundary between a medium and large break LOCA. A previous analysis for the APR 1400 [2] shows that if at least one of the 4 safety injection pumps (SIPs) is available during the LOCA sequence, the accident is no longer progressed to a severe core degradation phase and an in-vessel core cooling is eventually maintained. By this reason, the 4 SIPs are assumed to fail. Whereas, the 4 SITs are assumed to be available since they are passive systems. The initial water inventory of the SIT and its injection pressure are set as 1,858 ft^3 (52 m^3, whose net free volume is 68 m^3) and 4.24 MPa (600 psig), respectively. The other engineered safety systems (e.g., containment spray) are also assumed to fail. This is one of the representative accident sequences that has been regarded as an important contributor to a severe accident risk of the existing PWRs.

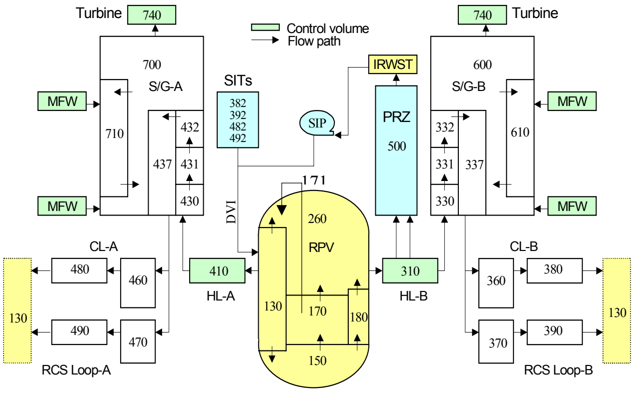

Fig. 1 MELCOR Model of the APR 1400 RCS and the S/G

2.2 MELCOR Plant Models

The MELCOR input for the APR 1400 is modeled with 41 control volumes (29 in primary and secondary systems, 12 in containment), 71 flow paths (41 in primary and secondary systems, and 30 in containment), and 144 heat structures (77 in primary and secondary systems, 67 in containment). The reactor core has been taken as a separate model from the core control volume for a hydrodynamic calculation, i.e., 39 core cells divided into 13 axial segments and 3 radial rings. Axial levels 4 through 13 are comprised of the active core region, and levels 1 through 3 corresponds to the lower plenum. The lower core support plate is in level 3. Figure 1 illustrates the MELCOR 1.8.5 configuration of the primary and secondary systems used for the present study.

2.3 Plant-specific Results

As a result of the MELCOR 1.8.5 simulation for the forgoing plant model, Figures 2 through 4 show the plant-specific impacts of the two aforementioned break locations on the key RCS thermal-hydraulic responses which can play an essential role in determining the subsequent core degradation and the evolution of the core materials in the RPV.

As shown in Figures 2 through 4, the RCS thermal-hydraulic behavior after the SIT injection is somewhat different when compared with that before the SIT injection. Fig. 2 shows that during the SIT injection phase the hot-leg break consistently maintains a lower pressure than the cold-leg break of the same size, and in turn a greater difference between the RPV and the SIT pressures causes a faster and enhanced injection of the SIT water into the RPV. Whereas, both the cold-leg break flow and the SIT injection are much slower when compared with the hot-leg break of the same size (see Figures 3 and 4), thus leading to a greater chance that the injected SIT water could contribute to a core cooling and thus it could be vaporized to steam in the RPV. As a result, more steam is released via the cold-leg break and the SG inlet plenum when compared with the corresponding hot-leg break of the same size. Therefore, this leads to a lower RPV temperature in the case of the hot-leg break.

In summary, both the break flow and the SIT injection are accelerated more in the case of the hot-leg break than the cold-leg break, and as a result the hot-leg break lessens the chances that the injected water could contribute to a core cooling. The greater accelerated rate of injection of the SIT water in the case of the hot-leg break than the cold-leg break leads to a faster depletion of the SIT inventory, and in turn this leads to a much longer injection of the SIT water into the RPV in the case of the cold-leg break. The aforementioned RCS thermal-hydraulic conditions after the SIT injection lead to a much faster degradation of the core and a much faster evolution of the core materials in the case of the hot-leg break, when compared with the cold-leg break of the same size. The MAAP4 application to the foregoing two cases [4] also shows a very similar trend.

3. Concluding Remarks

The present MELCOR 1.8.5 analysis on the APR 1400-specific impact of the RCS hot- and cold-leg breaks on the evolution of a severe core degradation has provided a result different from the existing viewpoint that the cold-leg break leads to a severer core degradation than a hot-leg break of the same size. The reason why the two different types of reactors (i.e., the conventional PWRs and the APR 1400) leads to such a contrary trend for a different break location of the same size results in is not clear yet. A unique feature of plant systems leading to a different physics for a severe accident may be a possibility. An in-depth study is required to clear up the issue.

ACKNOWLEDGEMENTS

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REFERENCES

[1] KEPCO, Korean Next Generation Reactor Standard Safety Analysis Report (Rev.1), 2001.

[2]. K.I. Ahn, S.Y. Park, D.H. Kim, MELCOR 1.8.4 Sensitivity Analysis of Severe Accident Progressions for LOCA Scenarios in APR 1400, International Conference on Nuclear Engineering (ICONE-11), Tokyo, Japan, April 2003.

[3] R.O. Gauntt et al., MELCOR Computer Code Manual: Primer and Users' Guide Version 1.8.5, Prepared for U.S.NRC, NUREG/CR-6119, Vol.1, Rev.2, Sandia National Laboratories, May 2000.

[4] S.Y. Park and K.I. Ahn, Technical Communication on MAAP4 Analysis Results of the APR1400 LOCA, KAERI, Daejeon, Korea, August, 2005.

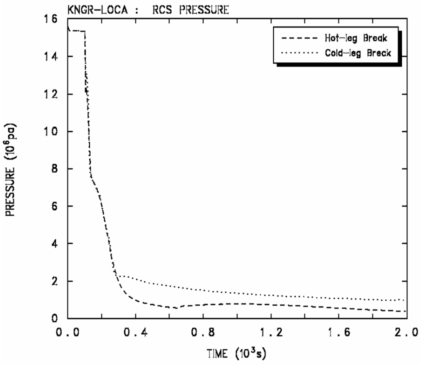

Fig.2 RPV pressures for both breaks

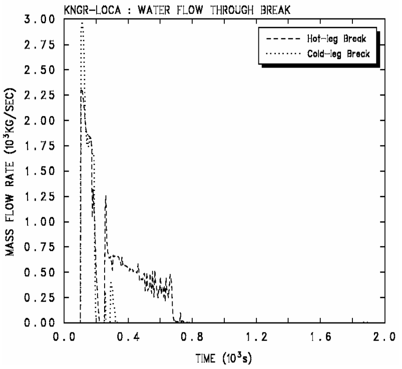

Fig.3 Blow-down rates via the breaks

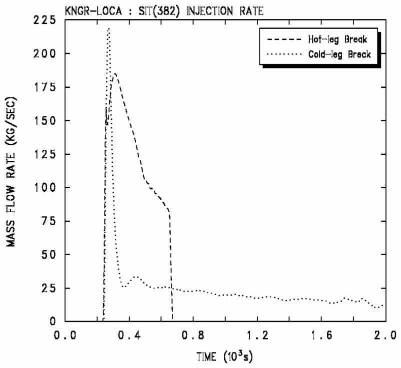

Fig.4 SIT Injection rates into the RPV