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Creep Rupture Analysis of Reactor Coolant System Boundary during a Total Loss of Feed Water Transient

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1. Introduction

The failure of a reactor coolant system (RCS) boundary, such as hot leg, pressurizer surge line, steam generator U tubes, affects severe accident progression before a failure of the reactor pressure vessel during a severe accident. In particular, the possibility of such a failure is very important in the consideration of direct containment heating (DCH), because a large enough failure may allow the RCS to depressurize sufficiently by the time the vessel fails to preclude early containment failure by DCH. The failure of the steam generator tubes may also provide a path for a containment bypass, in that fission products could be transported through the ruptured steam generator tubes into the secondary side of the steam generators and from there, through the safety relief valves to the atmosphere. For this reason, an evaluation on the creep rupture of the RCS boundary is very important during a severe accident of a high RCS pressure sequence. A total loss of feed water (TLOFW) of the APR1400 has been evaluated from an initiating event to a creep failure of the RCS boundary by using the SCDAP/RELAP5 computer code[1].

2. SCDAP/RELAP5 Input Model

The input model for the SCDAP/RELAP5 calculation of the APR1400 was a combination of the RELAP5, SCDAP, and COUPLE input models. Heat structures for the fuel rods and the lower part of the reactor vessel in the RELAP5 input model were replaced by the SCDAP and COUPLE input models, respectively. In the RELAP5 model, the reactor core was simulated as 3 channels to evaluate the thermal-hydraulic behavior in detail, and each channel was composed of 10 axial volumes, as shown in Fig. 1. A surge line and a pressurizer were attached to one of the hot legs in the primary coolant loop. The power operated safety relief valves (POSRVs) which control the RCS pressure are attached to the top of the pressurizer. The secondary side pressure is controlled by safety relief valves (SRVs) in the main steam line.

In the SCDAP input model of this study, the component numbers for the fuel and the control rods were 3 and 3. The axial node number of the fuel and control rods was 10 each, and the radial node numbers for the fuel and the control rods were 6 and 2, respectively. In the COUPLE input, the lower part of the reactor vessel was divided into 234 nodes and 204 elements.

A model based on creep rupture theory is used to calculate the damage and nearness to rupture of structural components. Two different theories are applied: (1) Larson-Miller[2], and (2) Manson-Haferd[3]. The particular theory to be applied is dependent on the material composition and stress. For 316 stainless steel and Inconel materials, the Larson-Miller theory is used. For A-508 Class 2 carbon steel, the Manson-Haferd theory is applied for the lower range of stress and the Larson-Miller theory for the higher range of stress. In this study, the steam generator U tubes of Inconel, the pressurizer surge line, the hot leg, and the reactor vessel are modeled as a creep input. The steam generator U tubes are modeled by Inconel, and the others are modeled by carbon steel.

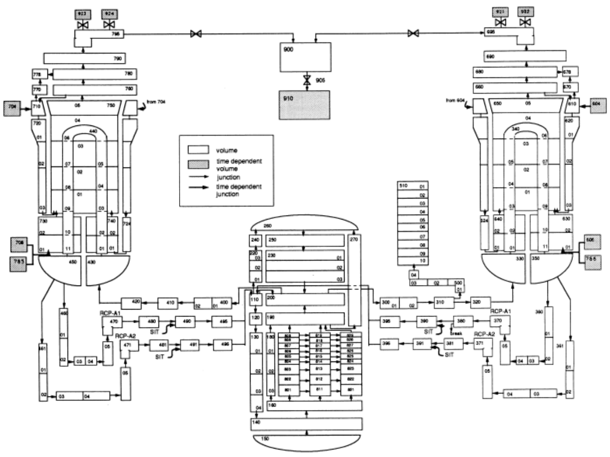

Fig.1. SCDAP/RELAP5 input model for the APR1400.

3. Results and Discussion

The TLOFW transient is initiated when the main and auxiliary feedwater are lost, and is assumed to occur at 0 second. The steam generator secondary side water level decreases due to the steam generation by boiling and finally the steam generators cannot act as effective heat sinks when the steam generator secondary sides dry up. The reactor and the RCP are tripped due to the low steam generator level and coolant sub-cooling margin, respectively. Since the core decay heat is not completely eliminated by the steam generators due to the decreased steam generator secondary side's water level, the pressure and temperature of the RCS increases. The SRVs of the steam generator regulates the pressure of the secondary side. The RCS pressure increases up to 17.4 MPa at 2,720 seconds, which is the opening pressure of the pressurizer SRV.

Boiling starts to occur in the reactor core, which causes core uncovery at the top of the fuel rods at 4,977 seconds. Ballooning of the fuel rod cladding continues, and finally the cladding fails by overstrain at 5,833 seconds. Oxidation of the fuel cladding began when the cladding surface temperature reached approximately 1,000 K and produced oxidation heat. When the fuel rod temperature reaches approximately 1800 K, the fuel rod cladding oxidation occurs vigorously and the cladding temperature increases rapidly due to oxidation heat generation in the core. ZrO_2 is ruptured and relocated to the lower part of the core when the fuel rod cladding temperature reaches 2,500 K at 6,345 second. The melted core material initially relocated to the lower plenum of the reactor vessel at 7,596 seconds. Most of the molten pool slumps to the lower plenum of the reactor vessel when the crust fails. Finally, the reactor lower head vessel failure occurs by creep. However, the pressurizer surge line failed by creep at 8,305 seconds before the reactor vessel failure in the TLOFW transient of the APR1400. The steam generator U tubes did not fail.

Figure 2 shows the SCDAP/RELAP5 results for the pressurizer and secondary pressure in the TLOFW. When the TLOFW transient occur at 0 second, the main feedwater is not supplied to the steam generator and the MSIV is closed. Hence the steam generator secondary side pressure increases up to the set pressure (8.75 MPa) of the SRV. The pressurizer pressure increases up to the opening set pressure of the pressurizer SRV (17.4 MPa), and then the RCS inventory is lost through the opened POSRV. When the pressurizer pressure decreases to the closing set pressure of the POSRV (15.1 MPa), the POSRV closes and the pressure builds up again. The pressurizer pressure fluctuates between the opening and closing setting pressure of the POSRV. When the RCS boundary fails, the pressurizer pressure maintains a high value between 15.1 MPa and 17.4 MPa.

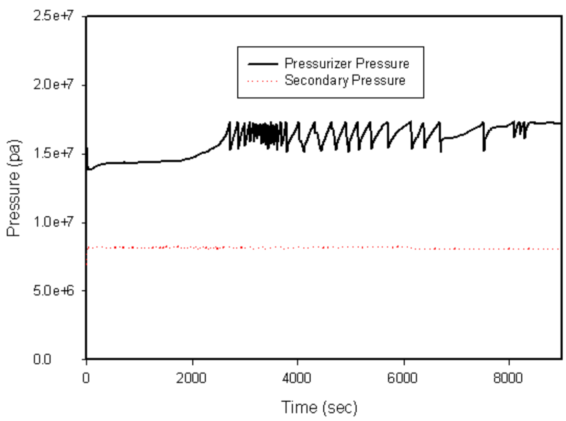

Fig. 2. SCDAP/RELAP5 results on the primary and secondary pressure.

Figure 3 shows the SCDAP/RELAP5 results for the average temperature of the RCS boundary. The path of the hot steam from the core is the hot leg, pressurizer surge line, pressurizer, POSRV, and a containment atmosphere. The inner surface temperature of the hot leg is very similar to the surge line. However, the average temperature of the hot leg is lower than that of the pressurizer surge line, because the thickness is bigger. For this reason, the average temperature of the pressurizer surge line is higher than that of the other RCS boundaries.

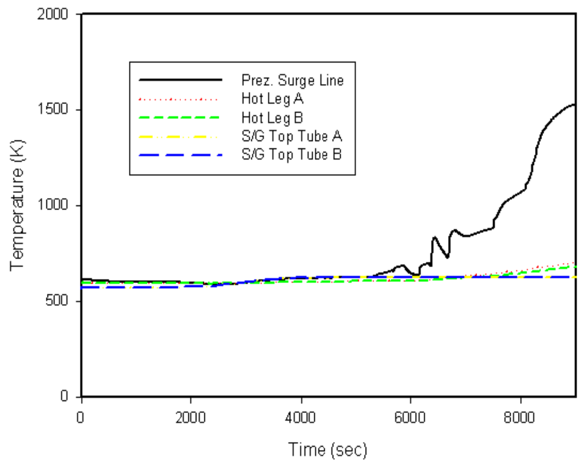

Fig. 3. SCDAP/RELAP5 results on the average temperature of the RCS boundary.

4. Conclusion

A TLOFW of the APR1400 has been evaluated from an initiating event to a creep failure of the RCS boundary by using the SCDAP/RELAP5 computer code. The SCDAPRELAP5 results have shown that the pressurizer surge line failed before the reactor vessel failure, but the steam generator U tubes did not fail.

ACKNOWLEDGMENTS

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