IFCI Simulation of Steam Explosion Loads for Reactor Cases

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1. Introduction

SERENA(***S***team ***E***xplosion ***RE***solution for ***N***uclear ***A***pplication) program runned by OECD is an international concerted program on Fuel Coolant Interaction. To resolve the still unsolved issue of calculating the steam explosion load is one purpose of the program. For this each participant to the program is supposed to calculate the in-and ex-vessel steam explosion loads for the assumed reactor conditions. The present paper summarizes the simulation results obtained using IFCI (Integrated Fuel Coolant Interaction) code [1].

2. Simulation Methods and Results

In this section the IFCI code used in the simulation, the in-and ex-vessel reactor conditions to simulate, and the results are described in a summarized way .

2.1 Code and Assumed Reactor Conditions

IFCI cide is a best-estimate computer code for analysis of phenomena related to mixing of molten nuclear reactor core material with reactor coolant. The code can address all aspects of FCI phenomena, including coarse fragmentation and mixing of molten material with water, triggering, propagation and fine fragmentaion, and expansion of the melt-water system. IFCI provides a two-dimensional, r-z geometry, four-field hydrodynamics model, whose fields consist of vapor, water, solid fuel, and melt. The code was validated against a lot of small and intermediate scale experimental data. The lack of validation against large scale experiments requires us to be careful in interpreting the calculation results but this lack of validation seems to be a general defect for all FCI codes.

The reactor situations to be simulated are shown in the figure 1. The left figure is a condition for an ex-vessel case. 100 tons of melt are supposed to pour into water-filled cavity as a single central jet through a 0.5m diameter in the vessel. The melt is composed of 80 w% UO_2 and 20 w% ZrO_2. The right figure shows the condition for an in-vessel case. The melt mass and composition are the same as those of ex-vessel case, but the melt slumps down into the lower head of reactor vessel through 25 multi-jets of diameter 0.08m. The aim of the simulation is to calculate the loads of steam explosion occurring for these seemingly conservative reactor situation.

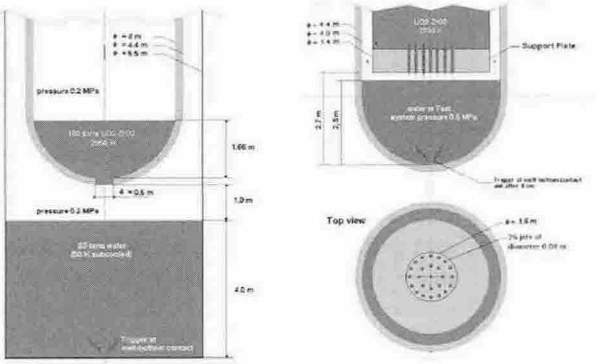

Figure 1. Left figure shows the ex-vessel steam explosion situation. Right one shows the in-vessel steam explosion situation.

2.2 Steam Explosion Load of Ex-Vessel Case

The steam explosion load for ex-vessel case of figure 1 is calculated using 21 x 12 (axial x radial) IFCI nodes. The system pressure is 0.2MPa, water temperature is 393K, jet free fall in the air is 1.0m, water pool height is 4.0m and the melt temperature is 2950K. The loads obtained are shown in figure2. The straight line is the pressure development at the bottom of the cavity center. It reaches the maximum pressure of ~ 80MPa at 1.6 ms after the triggering. The pressure at the cavity wall is shown by the dotted line in the figure 2. The maximum pressure at the wall is ~5 MPa at 3.6 ms which means that the pressure peak has moved from center of the cavity to the cavity wall at the velocity of ~2.0E3 m/s .
From the figure 2, we can see clearly that the very high pressure peak of 80MPa has been diminished to ~5MPa at the wall. A rough estimation of a dynamic impulse at the wall seems to be around 3 kPa-s. Considering that the MELCOR[2] analysis of the corium mass at the time of vessel breach done for operating plant or APR1400 suggests the generally lower corium mass than 100 tons [3,4], the assumed condition of 100 tons of corium to participate in the FCI seems to be highly conservative. Also it is not clear whether the whole mass of 100 tons could really go through fine fragmentation process and thus contribute to the explosion in a real case. Thus the authors have the impression that this peak pressure of 80MPa is a sufficiently high pressure as can be obtained by steam explosion. This result tells us that reliable prediction of how much a peak would diminish propagating through the melt-water mixture is very important to estimate the cavity integrity.

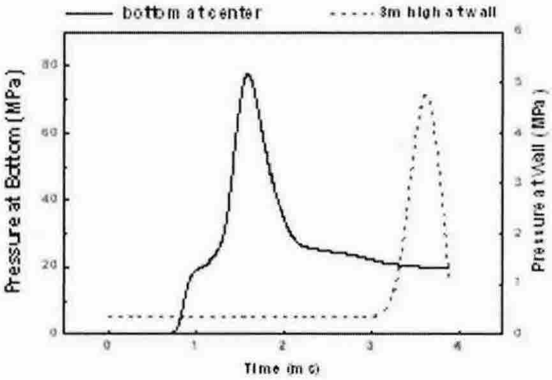

Figure 2. Pressure development at the center and at the wall of reactor vessel.

2.3 In-Vessel Reactor Case

The load for in-vessel steam explosion was simulated using the 12□21 (axial□radial) IFCI nodes. The system pressure in the vessel is 0.5 Mpa, the melt mass and temperature are 100 tons and 2950 K respectively. Figure 3 shows the pressure development at the bottom of the reactor vessel.

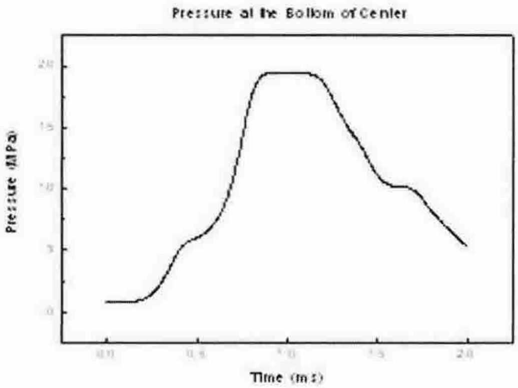

Figure 3. Pressure development at the bottom of the reactor vessel due to an in-vessel steam explosion.

The maximum pressure reached was ~20Mpa and the estimated dynamic impulse is ~ 15kPa-s. For the reactors having ICI penetration at the bottom of the vessel, this order of dynamic impulse seems to be high enough to intimidate the vessel integrity.

3. Conclusion

Simulation was performed to estimate the loads caused by the in-vessel and ex-vessel steam explosion assuming a seemingly real reactor condition as part of an concerted OECD research program. Because there still remain much uncertainties in the prediction of FCI codes ,it is premature to conclude anything from the present results. More reliable code models need to be developed in predicting fine fragmentation process, wave propagation through the corium-water mixture and the scaling effects.

Even then, the present research sheds some light in understanding where the major safety threats could be. The peak of ex-vessel FCI load is very high but in propagating through the melt-water mixture, the peak seems to diminish to much lower magnitude. On the other case of in-vessel steam explosion, although the pressure peak itself is smaller than the ex-vessel case, the dynamic impulse can affect the ICI penetration parts directly without diminution, thus threatening the vessel integrity.

REFERENCES

[1] F. J. Davis, M. F. Young, Integrated Fuel-Coolant Interaction (IFCI 6.0), NUREG/CR-6211, 1994.

[2] R.O.Gauntt et al. , MELCOR Computer Code Manuals, NUREG/CR-6119, 2000.

[3]Byung-Chul Lee et al. Development of the Methodology for Assessment of Kori Unit 1 Severe Accident Management Strategy Entry Points, KINS/HR-582, 2004.

[4] J. F. Ziegler, J. P. Biersack, "SRIM-2000, 40: The Stopping and Range of Ions in Matter", IBM-Research, Yorktown, NY 2000.