NUMERICAL STUDY OF IN-VESSEL CORIUM RETENTION IN BWR REACTOR

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Abstract

Recently, the strategy of melt retention inside the Reactor Pressure Vessel (RPV) in case of severe accidents in the nuclear reactors is getting more and more attention worldwide. The paper presents a numerical study of severe accident in the nuclear reactor and the analysis of corium coolability inside the reactor pressure vessel. The RELAP/SCDAPSIM 3.4 code and the integrated module COUPLE were used to analyze this problem. A full plant model of a ~2000 MW thermal power BWR reactor was used and a large LOCA with total failure of cooling water injection was assumed. The paper presents full accident sequence from normal operation conditions to core heat-up, melting and relocation into lower head. The temperature of RPV wall was used as a criterion for the integrity of the vessel. The results showed that if the reactor cavity is filled (to the level above the lower head) before meltdown of the core and slumping of the material to the lower head, the RPV could be cooled so that corium should stay inside the vessel.

1. INTRODUCTION

Nuclear energy is currently counting more than half of a century of its existence. That being said, the technology still keeps advancing and evolving, eliminating more and more safety issues in the Nuclear Power Plants (NPP). The safety issues of course include severe accidents as the most dangerous scenarios for the NPPs. Recently the strategy of melt retention inside the Reactor Pressure Vessel (RPV) in case of severe accidents in the nuclear reactors is getting more and more attention worldwide. This paper presents a numerical study of severe accident in the nuclear reactor and the analysis of corium coolability inside the reactor pressure vessel. The RELAP/SCDAPSIM 3.4 code and the integrated module COUPLE were used to analyse this problem. For the analysis of the IVR strategy in BWR, the General Electric BWR-5 reactor installed in Mark-II containment was chosen. The schematic view of a typical BWR-5 with Mark II containment and are shown in FIG. 2 and FIG. 1.
2. ACCIDENT SCENARIO DESCRIPTION

In this study the worst case scenario, i.e. a guillotine break of external circulation pump suction pipe - large Loss of Coolant Accident (LOCA) with total failure of cooling water injection due to station blackout and the failure of the Emergency of Core Cooling Systems (ECCS) is considered. The main assumptions for this scenario are:

- Reactor SCRAM at time moment \( t = 0 \) s;
- Main coolant pumps are tripped at time moment \( t = 0 \) s;
- Main safety valves closes at time moment \( t = 0 \) s;
- No feed water and emergency core cooling system water supply.

This is a hypothetical scenario in order to cause the core melt rapidly and observe the slumping of corium to the lower head, which then leads to heating up of the bottom of the RPV. The primary containment is simulated and external cooling of the reactor vessel is considered. The technical measures for supplying the water into the reactor cavity are not analysed in this study. It was assumed that the cavity is filled before the meltdown and slumping of the core material into the lower head.

After the initiating large LOCA with total failure of cooling water injection accident, there were two sub-scenarios considering different conditions for lower head cooling were analysed:

1. The water is continuously supplied to the reactor pit so that the water level does not decrease;
2. The water fills the reactor cavity and then stops, which leads to gradual boil-off of the water and uncover of the lower head.

3. MODEL OF THE REACTOR

The RELAP/SCDAPSIM computer code was used to simulate the severe accident [3], [4]. RELAP/SCDAPSIM is designed to describe the overall reactor coolant system (RCS) thermal hydraulic response and core behaviour under normal operating conditions or under design basis or severe accident conditions. The SCDAP models calculate the behaviour of the core and vessel structures under normal and accident conditions.

![FIG. 3. Scheme of the reactor ex-vessel model.](image-url)
COUPLE model takes into account the decay heat and initial internal energy of slumped debris and then calculates the transport by conduction of this heat in the radial and axial directions to the wall structures and water surrounding the debris.

In the model the convective boundary is connected to the external surface of the RPV lower head, and the heat from COUPLE finite-element mesh is transferred to hydrodynamic volumes of RELAP5 code. The model for simulating the ex-vessel cooling is presented in FIG. 3. The ex-vessel model consist of 2 time dependent volumes, 2 single volumes, one vertical pipe component and time dependent junction. Time dependent volumes used in order to give the boundary conditions to the developed model. 2 single volumes are used as the buffering volumes in the model. Vertical pipe component represents the available volume around ex-vessel, according the available Mark II containment data (see FIG. 2). Pipe component have 10 internal nodes. Each node are connected to the RPV vessel modelled using COUPLE. In order to have more precise calculation results flow area and height is varying according to the nodes developed for the COUPLE model. The height of the first nodes is small and increasing at the top of the lower head. In order to simulate the ex-vessel cooling conditions the time dependent junction was used.

4. RESULTS AND DISCUSSION

The objective of the analysis was to investigate the effectiveness of ex-vessel cooling, so that the RPV would stay intact. The main parameter for such evaluation was the wall temperature of the RPV wall. It was assumed in our analysis, that if temperature of the wall reaches the melting temperature of carbon steel (1723 K) the geometry is assumed as melted.

The LOCA occurs at the time = 0 s, and at the same moment the feed water stops. Reactor scram occurs immediately. After LOCA is initiated, the top of active fuel is uncovered in 32 seconds after the break, and fully uncovers in 50 seconds. The remaining water in the lower head fully evaporates in 3400 s after the accident, as it is heated by slumped debris. The more rapid decrease of the water level (at 2600 s after the accident) is due to starting of UO$_2$ slopping into lower head.

After the fuel is uncovered, the heat-up of the fuel rods takes place. The first rupture due to ballooning occurs in 421 s after the accident. In case of no ECCS in operation as it was assumed in our work, the heat up of the reactor core progresses and ends up with melting of the core materials and relocation of the debris into the lower head.

The melting and relocation of the core begins with the lighter metals, and the earliest slumps of debris contain Fe, Zr, ZrO$_2$ and B$_4$C absorber. The first slump is observed at 847 s after the accident. The slumping of debris to the lower head by material type is shown in FIG. 4.

![FIG. 4. Mass of different materials in the lower head.](image)

In our case the reactor cavity is filled with air during steady state, and then is filled with water before the debris starts slumping into the lower head. It was assumed in the model that the injection of water in the ex-vessel cooling system starts when the peak fuel cladding temperature exceeds the melting temperature of carbon
steel, before the beginning of corium slumping into the lower head. It was assumed that the water supply stops after filling the reactor cavity just a bit above the lower head.

The temperature of the external RPV wall is shown in FIG. 5. The reference nodes correspond to the model shown in FIG. 3, where the 1st node is at the bottom, and are placed along the external wall up to 11th. As we can see from the results, the external surface temperature is maintained around 430 K and the critical heat flux is not reached. As long as the wall is submerged in the water, it is cooled successfully. When the water level decreases and the wall dries out, the overheating and melting of the RPV wall is inevitable. Therefore, the main factor for the given reactor is to maintain the water level in the reactor pit.

![FIG. 5. Temperature and void fraction at the external wall surface of the lower head.](image)

The wall thickness of the lower head is 190 mm, and due to high power in debris the steel wall starts melting from the inside. RELAP/SCDAPSIM does not take into account melting of the wall, but the remaining wall thickness was calculated by the temperature profile.

![FIG. 6. Temperature and void fraction at the external wall surface of the lower head.](image)

The results of this first calculation show that if the lower head is submerged into the water, the heat is successfully removed and no CHF is expected to occur. For this reason, additional calculations were performed to determine the minimum water supply to the reactor pit, which is needed to avoid the dry-out. The analysis showed that if the water supply of 15 kg/s to the reactor pit is maintained, the in-vessel retention is achieved successfully. The results of this calculation are presented in FIG. 6. As we can see from this figure, the external wall temperature remains stable at around 430 K, which means it’s cooled and no CHF has been achieved.

5. CONCLUSIONS

For the analysis the most conservative initiating event – guillotine break of suction pipe in recirculation loop with total failure of cooling water injection was selected. This scenario leads to the very fast overheating of the core and the start of fuel melting at high decay heat level (~600 seconds after the beginning of the accident).
The dry-out of the lower head occurs approximately 1 hour after beginning of the accident. It was assumed in the modelling that the reactor cavity (pit) around the reactor vessel is flooded by water in advance. For this initiating event two cases were analysed:

1) without additional supply of water in the ex-vessel compartment (insufficient cooling, which leads to the reactor vessel failure).

2) with constant cold water flow (15 kg/s) around the lower head which allows the sufficient lower head cooling through the wall of bottom part of reactor vessel; the heat transfer coefficient from external surface of reactor vessel varies in interval 5000 – 7000 W/(m² K);

The modelling results show that critical heat flux is not reached in the analysed cases. Only the dry-out of water due to evaporation and decreasing water level can cause the failure of the RPV. Therefore, for given cases, if the reactor cavity is flooded, the ex-vessel cooling is sufficient and failure of the lower head does not occur. Although it must be mentioned that the analysis did not take into account the failure of the guide tubes, penetrating the lower head of RPV. In case of uncover of the external surface of the lower head (below the level of the debris bed) the RPV failure occurs almost instantly.

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6. REFERENCES


