Emergency condition analysis for MBLOCA along with SBO initiated severe accident using MELCOR

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Abstract. Much interest in severe accident research has been attracted since Fukushima accident, which indicates that severe accident could possibly happen although the probability is extremely low. Although many researches have been conducted in severe accident progression simulation and emergency condition analysis, it is still necessary to do some research on emergency conditions of severe accident that have not been sufficiently analysed. In this paper, a 250mm medium break loss of coolant accident (MBLOCA) in cold leg along with station blackout (SBO) initiated severe accident was simulated using MELCOR code. Before the calculation, steady state was simulated in order to check the model. Main parameters of the model were compared with operating parameters of nuclear power plant (NPP), and relative error was acceptable, which indicated that the model could be used to simulate the accident. Emergency condition progression of the accident was obtained, including start of core uncover, start of zirconium-water reaction, failure of fuel cladding and failure of the lower head. Thermal-hydraulics response of reactor was analysed. The calculation ended with failure of lower head, and the response of the containment was not studied. Results show that a 250mm break is relative a large one among MBLOCA, and the progression is similar with that of large break loss of coolant accident (LBLOCA). The accident progresses quite fast. The core starts to melt at 526s, and lower head fails at 2314s. Results of the paper could be used to provide basis for decision making in case of emergency and severe accident mitigation.

1. Introduction

Much interest in severe accident research has been attracted since Fukushima accident, which indicates that severe accident could possibly happen although the probability is extremely low. Many researches have been conducted in severe accident progression simulation and emergency condition analysis. Randall O Gauntt studied Fukushima accident using MELCOR. The paper provided some background into the development of accidents at the Fukushima Daiichi nuclear power station and their root causes, chief among them, the prolonged station blackout conditions that isolated the reactors from their ultimate heat sink — the ocean [1]. Mohsen Salehi and Gholamreza Jahanfarnia did research on small break loss of coolant accident (SBLOCA) without emergency core cooling systems (ECCS) using the RELAP5/SCDAP code in VVER-1000 reactor. In the study, a best estimate simulation event of a 25 mm break SBLOCA without ECCS was conducted until the reactor pressure vessel failed [2]. Longze Li et al. simulated and analysed a severe accident caused by SBO with failure of the steam generator (SG) safety relief valve (SRV) at Chinese pressurized reactor 1000-MW (CPR1000) power plant using MELCOR code [3]. Liang Hu et al. performed analysis for a Chinese three-loop Pressurized Water Reactor (PWR) severe accident induced by loss of coolant accident...
(LOCA) along with SBO using the MIDAC code [4]. Jun Wang et al. investigated core thermal hydraulic response for a hypothetical severe accident caused by SBO with failure of the steam generator safety relief valve at a Chinese pressurized reactor 1000-MW power plant using MELCOR [5]. Although the studies have been conducted, it is still necessary to do some research on severe accident emergency conditions that have not been sufficiently analyzed. In this paper, MBLOCA along with SBO initiated severe accident was simulated using MELCOR code. Emergency condition progression of the accident was obtained, and thermal-hydraulics response of reactor was analyzed.

2. Brief outline of MELCOR code and China three loop PWR

2.1. MELCOR code

The simulation is conducted using MELCOR 1.8.5. MELCOR is a fully integrated, engineering-level computer code that models the progression of severe accidents in light water reactor nuclear power plants. MELCOR is being developed at Sandia National Laboratories for the U.S. Nuclear Regulatory Commission as a second-generation plant risk assessment tool and the successor to the Source Term Code Package. A broad spectrum of severe accident phenomena in both boiling and pressurized water reactors is treated in MELCOR in a unified framework [6].

2.2. China three loop PWR

The China three loop PWR is a Generation II+ pressurized water reactor, which is designed based on French M310. It is known as CPR1000. Main parameters are shown in Table 1.

| Table 1. Main parameters of China three loop PWR. |
|-----------------------------------------------|
| Parameter                | Values and Units |
| Core power               | 2895MW           |
| Primary pressure         | 15.5MPa          |
| Core inlet temperature   | 565.55K          |
| Core outlet temperature  | 600.75K          |
| Primary flow rate        | 23790m³/h        |
| Secondary pressure       | 6.89MPa          |
| Secondary steam flow rate| 538.3kg/s        |

3. Method

3.1. Accident scenario

A 250mm break MBLOCA along with SBO is assumed to occur in cold leg in a China three loop PWR. The break is assumed to occur in the loop with the pressurizer. Medium break is the break that equivalent diameter is between 25mm~250mm [7]. In case of SBO, the larger the break is, the more severe the accident is. So, the largest medium break is chosen.

Based on the accident scenario, some assumptions are made. The reactor is assumed to run at 100% full power before the accident. The break and SBO happen at the same time, which is treated as initial time of the simulation. Reactor is assumed to trip in 0.1s after the accident happens, because primary pressure would drop to saturated pressure of the coolant in 0.1s once the break happens, and void fraction of the core increases, which introduces enough negative reactivity to scram the reactor. Main pumps run down for 30 seconds. Leakage rate of the containment is 0.1% full volume per day. High pressure safety injection (HPSI) and low pressure safety injection (LPSI) are lost because of loss of on-site and off-site power, and simultaneously main feed water and auxiliary feed water of the steam generators are lost for the same reason. The calculation ended with failure of lower head, and the response of the containment was not studied.
3.2. Model

The model is made up of primary system, secondary system, accumulator and containment. Some of key parameters of the China three loop NPP are shown in Table 2.

The nodalization of the NPP model is shown in Figure 1, which consists of 46 control volumes (CV) and 52 flow paths (FL). Containment is not included in Figure 1. As shown in Figure 1, all of the three loops are modelled. RPV consists of upper plenum, upper head, active core region, core bypass, lower plenum, lower head and downcomer. Each loop consists of cold leg, hot leg, loop seal, rising of the U-tube and down leg of the U-tube. Three loops share one surge line and pressurizer. Main pumps are modelled using control function and table function in MELCOR code rather than using control volume and flow path. Table 3 shows the meaning each control volume represents for.

Figure 2 shows nodalization of the lower plenum and core. They are divided into 14 axial levels and 4 radial rings. Axial level 1 represents for secondary support plate and barrel base plate, level 2 for flow diffuser plate, level 3 for core lower plate, level 4-13 for active core, and level 14 for core upper plate.

| Parameter                     | Values and Units |
|-------------------------------|------------------|
| Core power                    | 2895 MW          |
| Primary pressure              | 15.5MPa          |
| Core inlet temperature        | 565.55K          |
| Core outlet temperature       | 600.75K          |
| Coolant flow rate in cold leg | 23790m³/h        |
| Outer diameter of fuel rod    | 9.5mm            |
| Secondary pressure            | 6.89MPa          |
| Secondary steam flow rate     | 538.3kg/s        |
| Feed water temperature        | 226°C            |
| Failure temperature           | 1000°C           |

Table 3. Control volumes of NPP model.

| Control volume | Name                             |
|----------------|----------------------------------|
| CV110          | Downcomer                        |
| CV120          | Lower plenum and lower head      |
| CV130          | Active core region               |
| CV140          | Upper plenum and upper head      |
| CV126          | Core bypass                      |
| CV210          | Hot leg                          |
| CV220          | Rising of the U-tube             |
| CV230          | Down leg of the U-tube           |
| CV240          | Loop seal                        |
| CV250          | Cold leg                         |
| CV270          | Surge line                       |
| CV280          | Pressurizer                      |
| CV290          | Relief tank of Pressurizer       |
| CV620          | Safety injection tank            |
| CV700          | Upper shell of SG                |
| CV710          | Letdown of SG                    |
| CV720          | Secondary U-tube area            |
| CV740          | Outlet of steam                  |
| CV750          | Main steam line                  |
| CV780          | Feedwater tank                   |
Before the calculation, steady state was simulated in order to check the model. Main parameters of the model were compared with operating parameters of NPP, as shown in Table 4. Relative error is less than 1%, which indicates that the model can be used to simulate the accident.

| Parameter                  | Model          | Operating in NPP | Relative error |
|---------------------------|----------------|------------------|----------------|
| Core power                | 2898MW         | 2895MW           | 0.104%         |
| Primary pressure          | 15.47MPa       | 15.5MPa          | 0.194%         |
| Core inlet temperature    | 601.4K         | 600.75K          | 0.042%         |
| Core outlet temperature   | 566.6K         | 565.55K          | 0.186%         |
| Primary flow rate         | 23652m³/h      | 23790m³/h        | 0.580%         |
| Secondary pressure        | 6.87MPa        | 6.89MPa          | 0.290%         |
| Secondary steam flow rate | 535.6kg/s      | 538.3kg/s        | 0.502%         |

![Figure 1](image1.png)  
**Figure 1.** Nodalization of NPP.

![Figure 2](image2.png)  
**Figure 2.** Nodalization of the lower plenum and core.

![Figure 3](image3.png)  
**Figure 3.** Primary pressure.
4. Results and analysis

Simulation of MBLOCA along with SBO initiated severe accident was conducted using the MELCOR model. Emergency condition progression of the accident and thermal-hydraulics response of reactor were obtained.

4.1. Accident emergency condition progression

A 250mm break is relatively a large one among MBLOCA, and the progression is similar with that of LBLOCA. Comparison of MBLOCA and LBLOCA emergency condition progression is shown in Table 5[8]. The accident progresses quite fast, and at 526s the core starts to melt.

| Events                              | MBLOCA | LBLOCA |
|-------------------------------------|--------|--------|
| Accident occurs                     | 0      | 0      |
| Pressurizer exhausted               | 15     | 10     |
| Start of accumulator injection      | 53     | 18     |
| Core complete uncover               | 75     | 28     |
| Clad bursting                       | 108    | 31     |
| Accumulator inventory exhausted     | 163    | 110    |
| Start of rapid Zr-H₂O reaction      | 373    | 377    |
| Core starts to melt                  | 526    | 522    |
| Lower head failure                  | 2314   | 1973   |

4.2. Thermal-hydraulics response

Once the break happens, primary pressure would drop to its saturation pressure in a second, and then drops very quickly until it equals containment pressure, as we can see in Figure 3. When primary pressure is under accumulator tank pressure(4.2MPa), the accumulators are activated and start to inject emergency cooling water into the core, as shown in Figure 4. It is the only coolant source during the accident, because HPSI and LPSI are lost. Figure 5 shows the leakage flow rate through the break, and the leakage flow is two phase flow. In the first few seconds, leakage flow rate is critical flow because primary pressure is much larger than containment pressure. It lasts for just a few seconds because of quick depressurizing in primary loop. When primary pressure equals containment pressure, leakage flow rate becomes very small, but it does not equal zero because coolant in the core is heated into steam by decay heat and escapes through the break. Core water level is shown in Figure 6. Leakage flow rate is so heavy that accumulator can not fill up the core and water level decreases. When accumulators start to inject coolant into the core, water level increases. After a while, accumulators exhaust and water level drops once again. An increase of water level between 500s and 1000s is caused by core melt. At 526s the core start to melt, and relocates to lower plenum.

Figure 7 shows the max temperature of cladding, which can reflect the temperature trend of all cladding. While core water level is low and core is uncovered by coolant, max temperature of cladding grows and vice versa. Figure 8~ Figure 11 show cladding temperature of radial ring 1~4. As introduced in section 3, cell 304 is the bottommost cell of radial ring 3 in active core, and cell 313 is the topmost one. Radial ring 3 is the first one that starts to melt, while radial ring 1 is intact throughout the accident. Figure 12 shows total hydrogen production during the accident. In the early stage of the accident, zirconium-water reaction is not rapid and there is not much hydrogen. In the late stage of the accident, zirconium-water reaction is very rapid and hydrogen production rate is much more quick.
5. Conclusions
A 250mm break MBLOCA in cold leg along with SBO initiated severe accident was simulated using MELCOR code in this paper. Emergency condition progression of the accident was obtained, including start of core uncover, start of zirconium-water reaction, failure of fuel cladding and failure of the lower head. Thermal-hydraulics response of reactor was analyzed, including primary pressure, core water level, leakage rate and clad temperature. Results show that a 250mm break is relative a large one among MBLOCA, and the progression is similar with that of LBLOCA. The accident progresses quite fast, and at 526s the core starts to melt.
Figure 10. Cladding temperature of radial ring 2. Figure 11. Cladding temperature of radial ring 1.

Figure 12. Total hydrogen production.

Results of the paper could be used to provide basis for decision making in case of emergency and severe accident mitigation. Other accidents could be simulated in future work to investigate their emergency condition, such as SBLOCA and SGTR.

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