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# Application of MELCOR to Westinghouse 4-loop PWR severe accident and evaluation of RPV lower head performance

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**Abstract:** In this study, MELCOR computer code is used to simulate the progression of a severe accident initiated from station blackout (SBO) accident for a Westinghouse 4-loop PWR. The hydraulic system is modeled using control volumes and flow paths. The reactor pressure vessel and internals, the primary loops with a pressurizer, steam generators, containment and accumulators are simulated for steady state in a good agreement with reference data.

The two scenarios concerning SBO are investigated. The first scenario simulates RCP seal leakage during SBO and the other is SBLOCA to highlight an effectiveness of accumulators as well as to compare with the first simulation. All active safety systems which depend on AC power are assumed to be unavailable in this analysis.

The main result of the study is an evaluation of RPV lower head integrity during severe accidents. This is preliminary work and expected to give the experience for further studies in the severe accident in nuclear power plants.

**Key Words:** *Severe Accident, Station Blackout (SBO), Core melt, RCP Seal leakage, RPV Lower head, Loss of coolant Accident, Accumulator.*

## I. INTRODUCTION

The research of accident phenomena and offsite consequences of severe accidents in nuclear power plants has been widely studied by the nuclear power industry, the international nuclear energy research community for many years. Since the crisis at the Fukushima nuclear power plant, the severe accident during a station blackout (SBO) in nuclear power plant has been recognized as a very important issue for severe accident analysis.

Station blackout accident involves a loss of offsite power, failure of AC power supply. All normal electrical equipment and most of the active safety systems are unavailable due to loss of all AC power. The loss of coolant may result in the fuel failure, core melting and RPV failure. Following the loss of all AC power the RCP seals will lose their cooling support systems and RCS water leakage occurs. For simplicity, the leakage of RCS water through the RCP seals is considered as small LOCA. The more severe scenario with SBLOCA which happens during SBO is also considered. The only safety injection tanks (accumulators) are assumed to be available and/or unavailable in this analysis.

In this study, MELCOR code is used to simulate the progression of a severe accident initiated from station blackout (SBO) accident for a Westinghouse 4-loop PWR. The hydraulic system is modeled using control volumes and flow paths. In order to confirm the results of the analysis, the steady state

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simulation has been performed and the comparison with reference data has been presented.

The main result of the study is an evaluation of RPV lower head integrity during severe accidents. The input deck has been developed and based on documents [1]. This is preliminary work concerning MELCOR and it is expected to give the experience for further studies in severe accident of NPPs.

## II. NODALIZATION

An Westinghouse 4-loop PWR is a four loop pressurized water reactor of 3250 MWth with 4 steam generators and 4 reactor coolant pumps (RCP). The hydraulic system is modeled using control volumes and flow paths. The reactor pressure vessel and internals, the primary loops with a pressurizer, steam generators, containment are modeled. Each RCS loop has 5 control volumes including hot leg, SG inlet plenum, tube side, shell side, outlet plenum, intermediate legs (RCP suctions) and cold leg. There are total 21 control volumes on the primary side of the RCS.

The secondary side of the steam generator has 3 volumes and the secondary side is connected to the turbine. The feedwater lines are simulated using control functions. The reactor pressure vessel (RPV) consists of 5 control volumes including the core channel, a bypass region, a lower plenum, an upper head and a down comer. Containment is simplified with 4 control volumes, reactor cavity (CV100) which is located below the RPV. The NPP nodalization schemes used in this study are presented in Figure 1 and 2.

In this study, the core and lower plenum are modeled as having 4 concentric radial rings and 12 axial nodes. Level 6 includes a core support plate, level 3 includes a bottom plate and level 1 includes an In-core Instrumentation (ICI) nozzle support plate. Levels from 7 to 12 are for fuel region.

Four control volumes, CV801, CV802, CV803 and CV804 are used to simulate four Safety Injection Tanks (SITs). The flow paths connecting these tanks with the cold legs are simulated as valves that will open and close depending on the pressure set points. The set point pressure of SITs to open is 4.24 MPa. In this model, safety injection pumps (SIPs) are not modeled as assumption of SBO.

The nodalization of containment and lower plenum shown in Figure 2, 3 are based on [1].The ICI penetrations are in the bottom of the RPV lower head. It is assumed that the RPV lower head will fail when the penetration temperature reaches the melting point.

The control function (CF) is used to simulate the main feed water lines as seen in Figure 1.

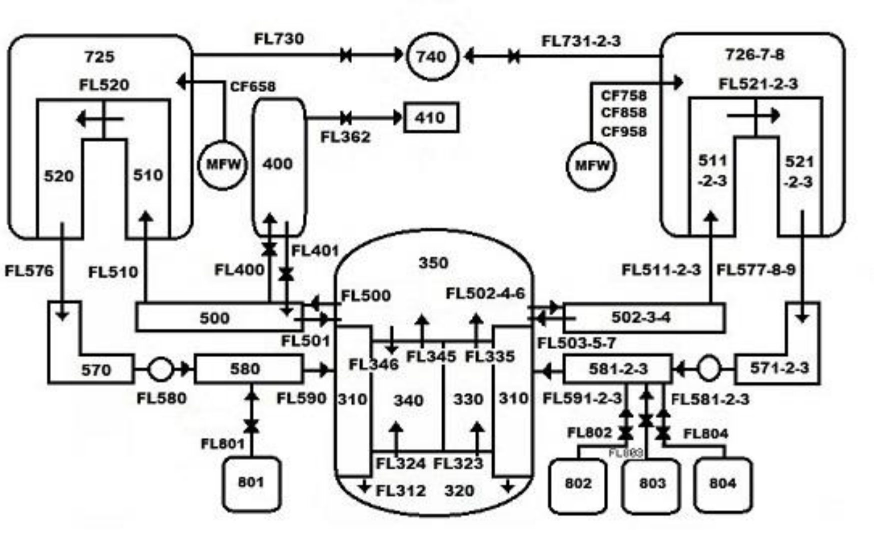


**Fig.1.** Westinghouse 4 loop RCS nodalization scheme

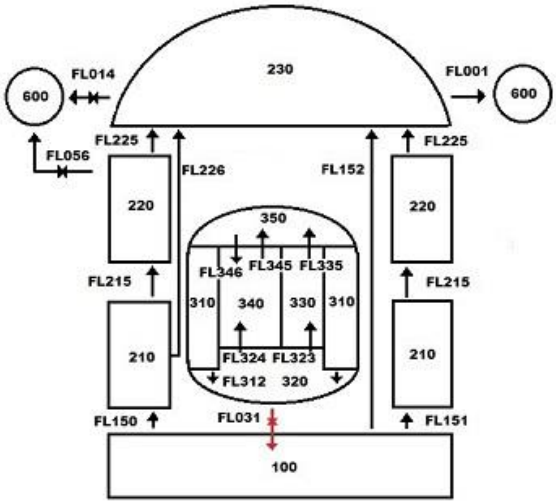


**Fig.2**. NPP nodalization scheme for reactor building compartements.

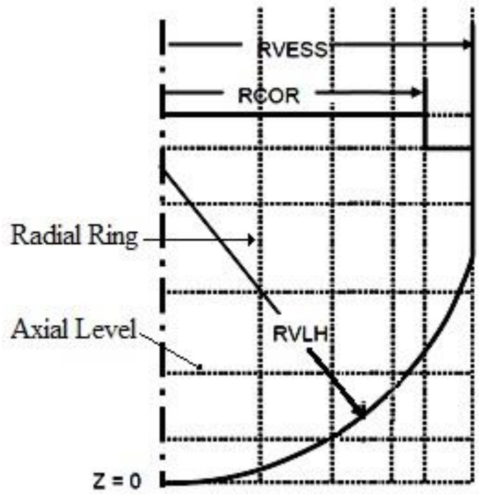


**Fig.3**. Core and lower plenum nodalization.

## III. STATION BLACKOUT (SBO) ACCIDENT ANALYSIS

### A. Initial Conditions and Assumptions

By using system nodalization and taking into account the reference data for Westinghouse 4 loop PWR [3], [4] and running for steady state, the calculation results are in good agreement with reference data as presented in Table 1. Water in the RCS is discharged through the leak seals of RCP, which results in a decrease in the RCS water inventory. The reactor core is uncovered and eventually damaged if the safety injection system (SIS) cannot be recovered in time. In this calculation, only the SITs are assumed to be operable.

In this study, a station blackout accident, which is initiated by a loss of off-site power and a failure of on-site power, is assumed to occur at the time of 0 s. Reactor trip, reactor coolant pump (RCP) trip and main feed water (MFW) trip occur following the SBO. The auxiliary feed water system (AFWS) is assumed not to work so that steam generator secondary side will dry out. The heat transfer between primary and secondary sides of steam generators cannot be maintained after a steam generator secondary side dry out. Water in the reactor pressure vessel will boil-off because of the high temperature of the RCS.

### B. SBO with RCP seal leakage

The results reported in WASH-1400 indicated that breaks of an equivalent diameter in the range of 0.5 to 2 inches in the RCS pressure boundary are an important event which may lead to core-melt. The overall probability of core-melt due to SBLOCA could be dominated by events such as RCP seal failures was also interested [7].

The water mass in the reactor core and lower plenum decreases and then recovered by water injection when RCS pressure reaches the setpoint of accumulators. At about 8h after reactor trip, the core is uncovered again and collapse in fuel ring 1 occurred. The core center (ring 1) is totally failed at 9.7h. The sequences of core degradation are presented in Figure 4.

The RCP seals may subsequently fail due to high temperatures that causes a leak increase. However, as commented above, the seal leakage in each RCP is assumed unchange and the leakage flow is about 1.21 kg/s. The results are compared with SNL report [6] which used the increased leakage flow and presented in Table 2.

The RCS pressure is decreased following reactor trip, and then increased due to increasing of decay heat and steam. The increase of the decay heat may be explained by the SBO that causes the unavailability of residual heat removal system (RHRS). When the pressure reaches the openning setpoint of accumulator (4.24 MPa), water is injected to the cold legs and the water level in the core is recovered as indicated in figure 5. From the Figure 5, it is also seen that the core is totally uncovered at 8.3 h and there is no water in RPV at 14.4 h. The water level in reactor core decreases and recovered by ACC injection at 4.3h as seen in Figure 6 and then RCS pressure is also decreased after increase due to water loss (Figure 6).

**Table I**. Steady state simulation for WH 4-LOOP PWR

| Parameters | WH data ref.[3,4] | SCDAP/RELAP [ 3 ] | MELCOR |
|---|---|---|---|
| Reactor power (MWth) | 3250 | 3250 | 3250 |
| PZR Pressure (MPa) | 15.51 | 15.509 | 15.55 |
| PZR water/steam vol. (%) | 60/40 | 61.2/38.8 | 50.5/49.5 |
| RCS Coolant Flow Rate (kg/s) | 17010 | 17010 | 17087 |
| Cold Leg Temperature (K) | 565.5 | 549.9 | 565.6 |
| Hot Leg Temperature (K) | 598.5 | 585.5 | 598.9 |
| SG Secondary Pressure (MPa) | 4.964 | 4.892 | 4.969 |
| Feedwater Temperature (K) | 493.5 | 493.48 | 493.5 |
| Steam Flow Rate per SG (kg/s) | 440.9 | 439.9 | 435,9 |
| Liquid Volume per SG (m3) | 52.05 | 52.66 | 52.29 |

In MELCOR the fuel rods are assumed to have their integrity until cladding and fuel temperature reaches their melting points. The particulate debris is formed due to cladding failure. With the core support plate failure, hot corium is relocated to the lower plenum and heats up the ICI penetration. ICI penetrations heat up and failures which results in the RPV lower head failure. The chronology of events is described in Table 2. The major important results of events are shown in Figures 4 through10.

Core uncovery will occur as the RPV water level decreases to the top of the active fuel (TAF). As a result, the fuel and structure start to heat up. Due to the cladding temperature increase, oxidation of the cladding occurs. Failures of cladding in the top of active fuel (TAF) and bottom of active fuel (BAF), which correspond with axial levels 12 and 7 respectively, are shown in Figure 7.

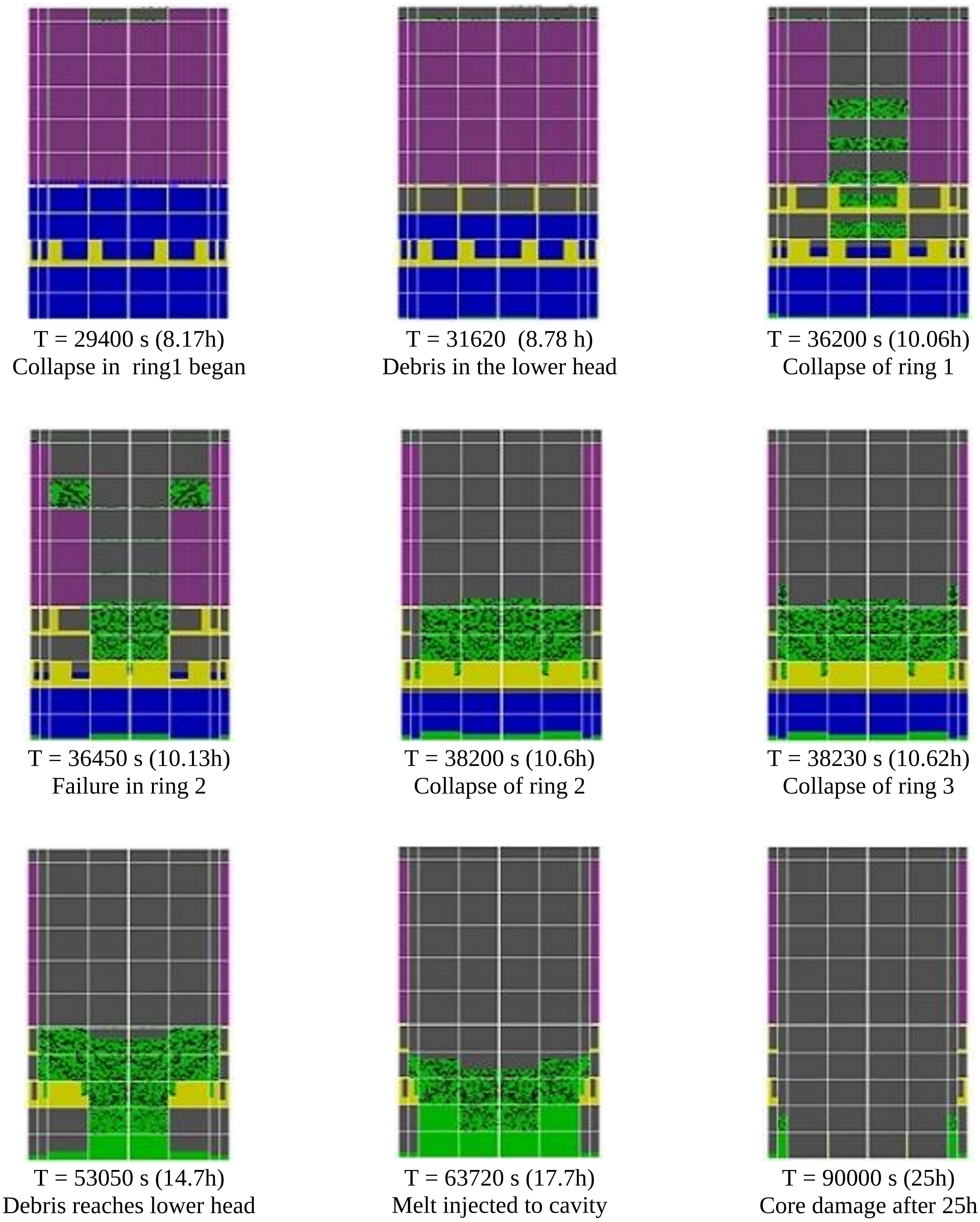


**Fig.4:** Accident sequences in reactor core and lower head.

The cladding temperature heat-up and exceeds the melting temperature, the cladding failure starts to occur from top of ring 1 at 8.16 hours after that it spreads to other areas as shown in Figure 4 and 7. At that time, the fuel cladding oxidation is also accelerated rapidly leading to the peak of pressure in the reactor core around time of 9 to 10h (Figure 6) and hydrogen content increases in core as shown in Figure 10.

Due to having depressurize from the RCS to the containment through leakage flow, so the containment pressure increases as seen in Figure 9 and the pressure in RCS (Figure 6) decreases to get the pressure balance between RCS and containment. However, the containment failure may occur when the containment pressure exceeds 0.441MPa. This issue is of great importance. And this is also well simulated by MELCOR. To examine and understand all physical phenomena inside the containment, the further studies are under consideration and it will be performed in the next work.

**Table II.** Time events for SBO with RCP seal leakage

| Event | MELCOR | SNL Report [6] |
|---|---|---|
| SBO, s | 0 | 0 |
| RCP seal leakage, s | 0 | 0 |
| Beginning of core uncovering, h | 4.3 | 6.5 |
| Start of fuel cladding failure, h | 8.16 | 8.0 |
| Start of core support plate failure, h | 10.13 | 11.3 |
| Debris relocation to lower head, h | 16.6 | 14.3 |
| Vessel rupture, h | 18.85 | 17.9 |

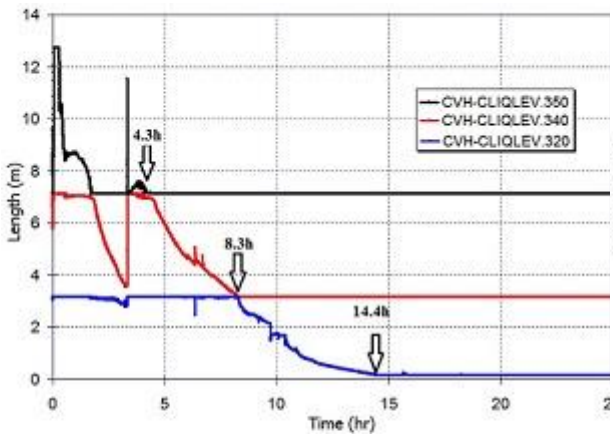


**Fig.5**. The water level in the RPV

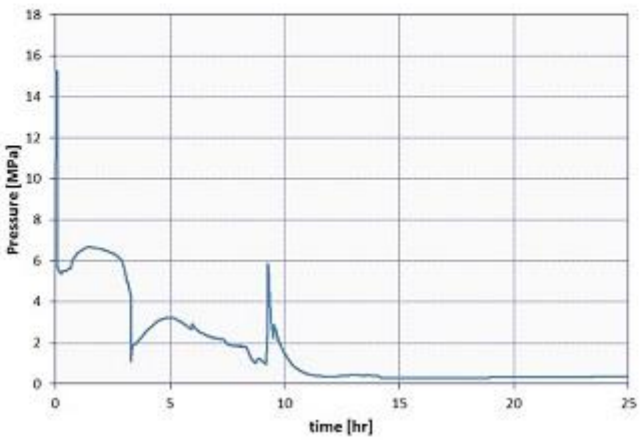


**Fig. 6.** The changes in RCS pressure

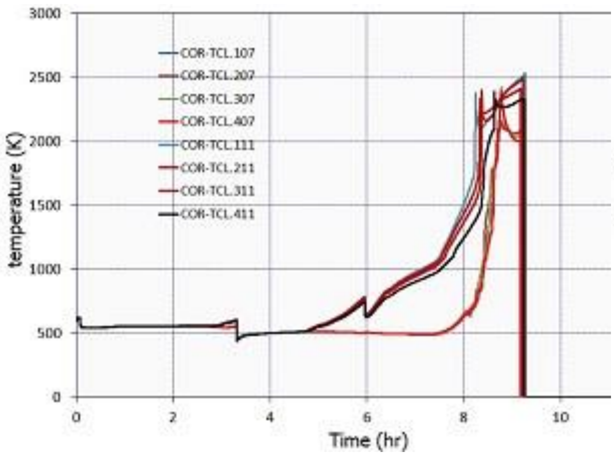


**Fig. 7.** Clading temperature at the BAF(x07) and TAF(x12).

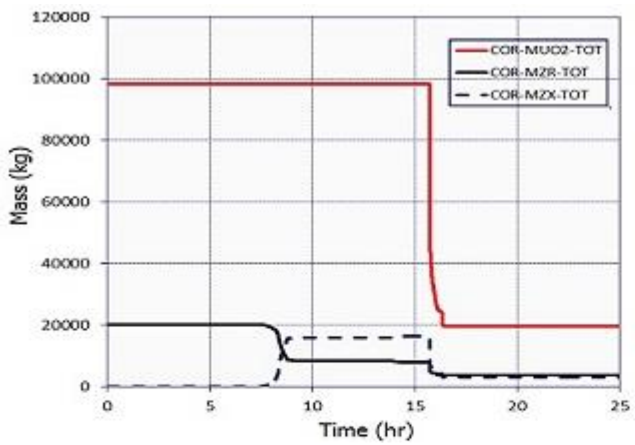


**Fig. 8.** Total mass of UO2 , zircaloy (MZR) and $ZrO_2$ (MZX) in the core.

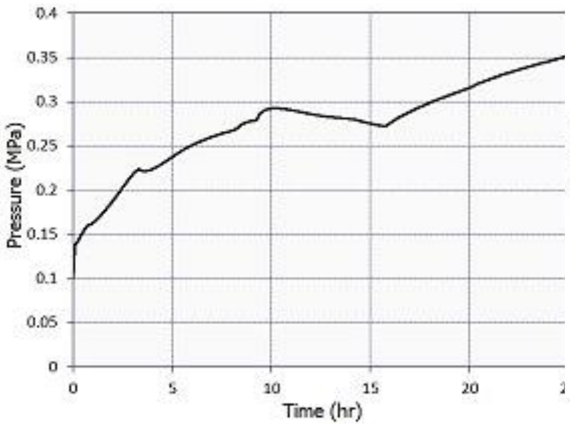


**Fig. 9**. Pressure in the containment.

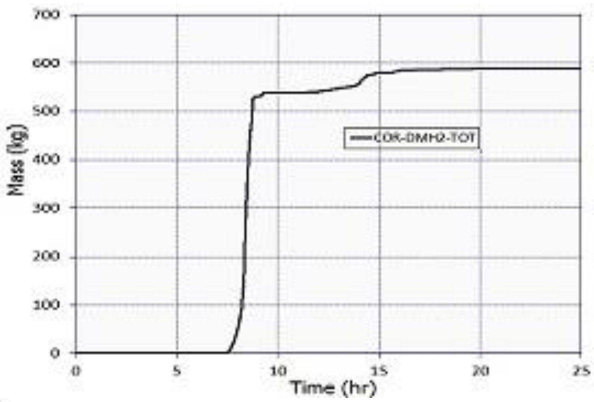


**Fig. 10**. Hydrogen release.

### C. SBO with SBLOCA

The accident event is initiated by opening the valve simulation the 2.54cm (1") break in the cold leg. The reactor trip further caused the turbine trip, main feed water trip and coast-down of the four reactor coolant pumps (RCPs). Due to SBO, the high-pressure and low-pressure safety injection systems are unavailable. Two scenarios are assumed for calculations with and without the accumulators (ACC).

The event sequences for the scenarios are indicated in Table 3. With the operation of accumulators the time of failures is longer than unavailable accumulators about 10 to 12 h. The calculated results are conformed to the sequence of a SB-LOCA with typical phases, i.e. blow down, natural circulation, loop seal clearance, boil off and core recovery phases.

The sequence of events is described by the primary pressure, the break flow rate, and other parameters as shown in the Figures 11 to 16.

It can be seen that, when the accumulators are available, water is injected to re-flood the core, core melting is later, and the vessel failure is also later about 10 hours in comparison with the unavailable accumulator (the vessel failure is at 24.6 hours with the available accumulator and at 14.19 hours with the unavailable accumulator). It is clear that the accumulator system plays an important role in the reduction of core degradation in the basic design accidents as well as beyond design basic accidents.

By choosing 2.54cm (1") break SBLOCA at the cold leg, it should be noted that this SBLOCA case gives the results of consequences basically same as RCP seal leakage. The time of lower head failure, mass of hydrogen generation and so on can be compared between the two scenarios. The peak of the RCS pressure in the case of ACC active is lower than inactive ACC case due to the water injection from ACC as seen in Figures 11 and 12.

**Table III**. Event sequences for 2.54cm (1'') SBLOCA

| Event | Accumulator on | Accumulator off |
|---|---|---|
| Accumulator injection begins, h | 6.76 | Off |
| Core plates failure, h | 21.3 | 10.92 |
| Lower head failure, h | 24.6 | 14.19 |

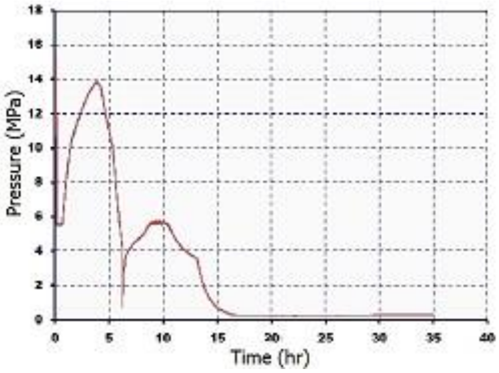


**Fig. 11**: The primary pressure with ACC on.

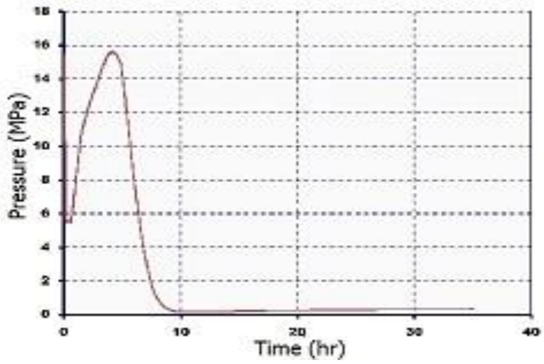


**Fig. 12:** The primary pressure with ACC off.

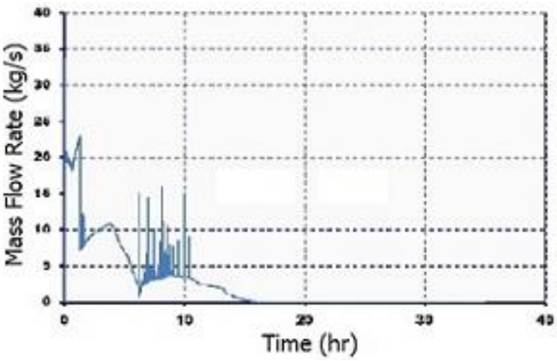


**Fig. 13:** The leakage mass flow rate with ACC on.

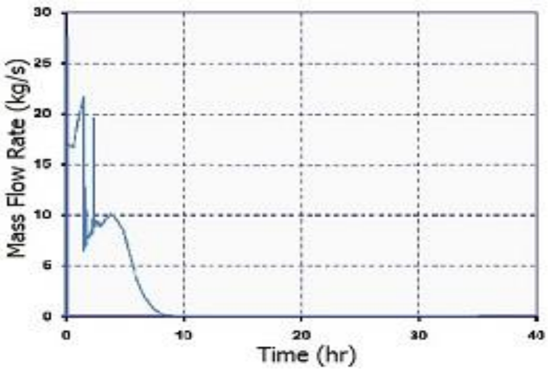


**Fig. 14:** The leakage mass flow rate with ACC off.

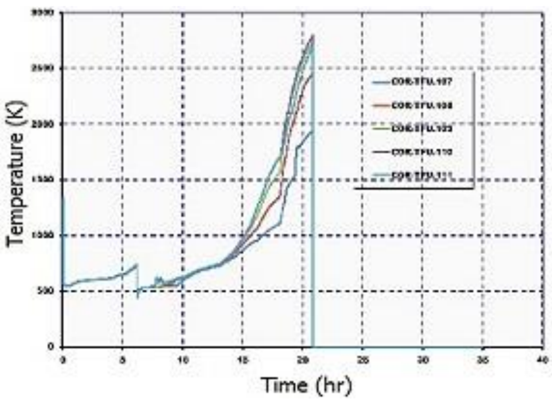


**Fig. 15**: The fuel temperature with ACC on.

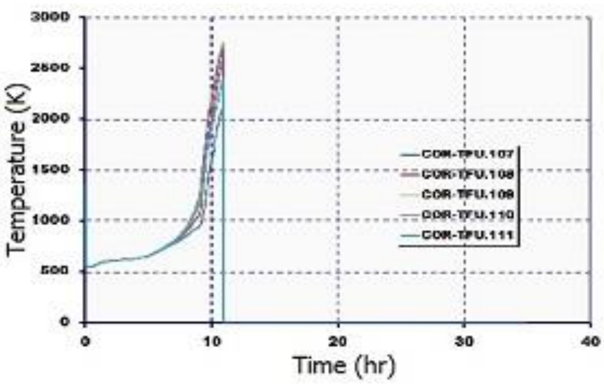


**Fig. 16:** The fuel temperature with ACC off.

Due to water injection from accumulators, the leakage flow rate is increasingly fluctuated as seen in Figure 13, but not eliminated as Figure 14 for case of accumulators unavailable. This also makes the failure of fuel rods more longer as shown in Figures 15 and 16.

### D. Comparison between the calculated result and reference data

Westinghouse 4-loop PWR has been investigated for many years. Catawba Units 1 and 2 and McGuire Units 1 and 2 are Westinghouse 4-loop PWRs is taken as reference data [6] as MELCOR was also used. Each of the four plants is similar in design, and a single representative MELCOR model was developed for their study. As the nodalization and design data are different, it is also supposed that DC power only failure after 3 hours and RCP leakage rate was not constant as our assumption The total mass of $UO_2$ fuel and thermal power are little different. There are also many uncertainties not only in design data, but also in other parameters needed to be discussed. As shown in Table 2, the progression of core damage is same and they are quite comparable.

As shown on Figures 17 and 18, the hydrogen genration is about over 500 kg, it is similar with the calculated result by SNL report. The water levels in RPV are presented in Figures 19, 20 respectively. Although the water level in our simulation decreases and the core is uncovered earlier but core support plate failure, debris relocation in lower head and RPV rupture occur later than ones simulated in [6].

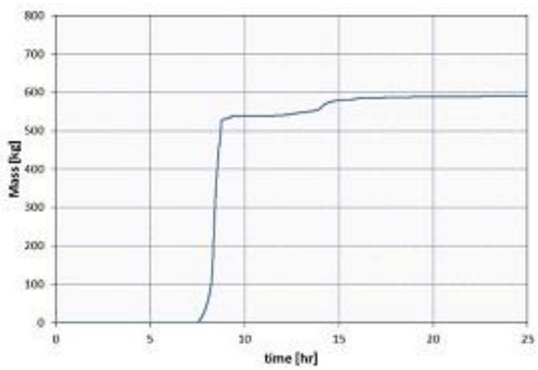


**Fig. 17:** Hydrogen release in-vessel.

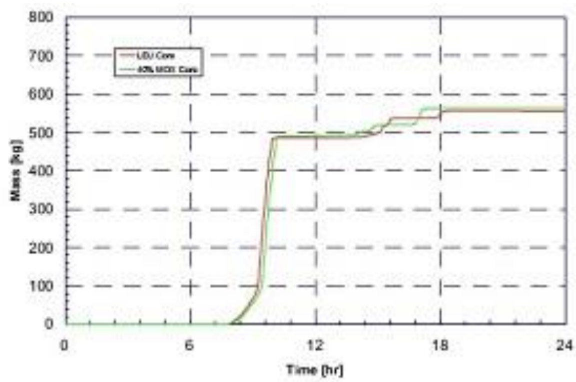


**Fig. 18:** Hydrogen release in-vessel [6].

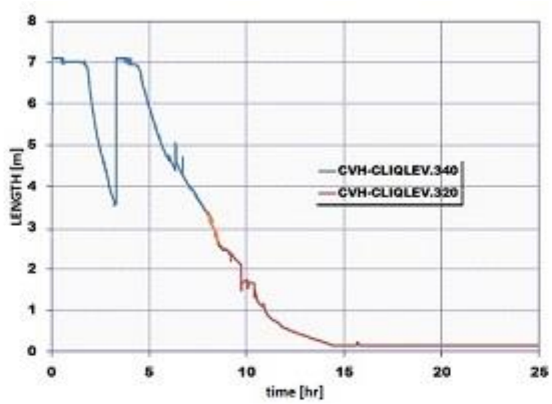


**Fig. 19:** The change of water collapsed level in RPV.

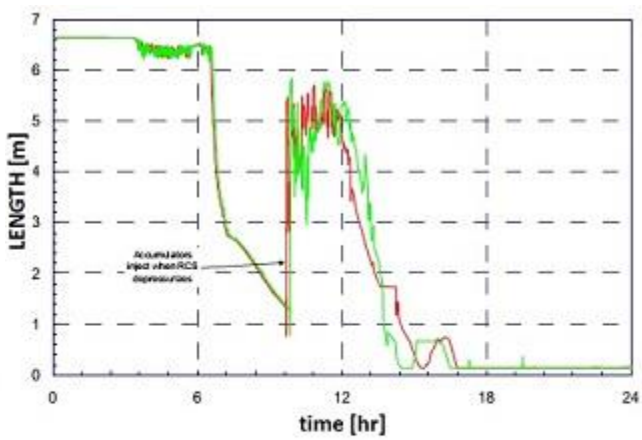


**Fig. 20:** The change of water collapsed level in RPV [6].

## IV. CONCLUSIONS

Westinghouse 4-loop PWR is modeled using MELCOR computer code. Two cases are considered for SBO accident: SBO with RCP seal leak actuation and SBO with LOCA actuation. In case of SBO with RCP seal leak actuation, the RPV lower head will fail after 18 hours since SBO occurs. In case of SBO with SBLOCA actuation and the accumulators are available, the RPV lower head may failure after 24.6 hours and in case of unavailability of the accumulators, the RPV lower head failure after 14.19 hours since the accident occurs. The calculated results show that the time of failure of RPV lower head depends on the break size (or the leak rate), the plant response, and the availability of safety systems.

The packages CVH, COR, HS and others have been intensively studied during using of MELCOR. The high temperature causes the failure of reactor lower head in the simulation. The material relocation models as well as the partitioning of fission products between metallic and oxidic phases can affect heat generation. The natural convection in core debris retained within the RPV lower plenum and other phenomena occurred during core melting should be in detail investigated.

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