STUDY ON IVR STRATEGY FOR VVER1000/V320 REACTOR IN CASE OF SBO ACCIDENT

DOAN Manh Long
Nuclear Training Center, 140 Nguyen Tuan, Thanh Xuan, Ha Noi
Email: longdoanmanh28@gmail.com

Abstract: The In-vessel Melt Retention through Reactor Vessel External Cooling (so called IVR strategy) has been becoming an attractive accident management strategy for high power reactors such as AP1000, APR1400, VVER1000 … However, applicability of IVR for high power reactor has become more difficult than low power reactor like AP600 or VVER440 because of higher decay head and dependence of severe accident on each kind of scenario and individual reactor. Therefore, severe accident progression analysis for high power reactors is very important aspect for IVR study.

In the paper, the Station Blackout accident accompanied with IVR implementation has been analyzed for VVER1000/V320 reactor by using MELCOR code 1.8.6. Four SBO scenarios have been proposed for the study, among, two SBO scenarios with long-term high pressure showed that the failure of lower head occurred early due to high pressure and high temperature causing creep-rupture, and the external cooling only delayed the failure of lower vessel. Two others with short-term high pressure SBO by taking account the action of reactor vessel depressurization when temperature at core outlet (Tcoreoutlet) exceeds 650^{o}C (923^{o}K) as emergency operation procedures (EOPs) of VVER1000/V320 nuclear power plants, indicated that the failure of lower head lately occurred due to thermal load from hot debris according to creep-rupture mode, the external cooling only brought into play to delay the failure.

Key words: VVER1000, SBO, IVR strategy, severe accident, MELCOR.

1. INTRODUCTION

A molten pool established in lower plenum is a consequence of severe accident progression due to failure of reactor core cooling, and assuring the integrity of lower head vessel with the presence of a molten pool in lower plenum is the major concern in severe accident management strategy for power reactors in nuclear power plants. By flooding cavity before the core melt relocated into lower plenum, the integrity of lower head vessel could be secured. This novel measure known as In-Vessel Melt Retention through External Reactor Vessel Cooling (IVR/ERVC), firstly proposed by Prof.Theofanous **[1]**, have been successfully adapted for VVER440 in Loviisa nuclear power plant **[2]** and AP600 **[3]** as severe accident management strategy.

Currently, the ability of cooling molten debris in lower plenum by externally reactor vessel cooling has been becoming a crucial issue in severe accident management strategy for high power reactor. There are many efforts put in analytical studies and experimental facilities with the aim of proving that the application of IVR/ERVC for high power reactor such as AP1000 **[4]**, APR1400 **[5]** is feasible and reasonable. The efforts have been dedicated to determine the final bounding state of molten pool through various scenarios and to increase the critical heat flux of external reactor vessel cooling ambient. For second task, with optimization of cooling channel design and coating technique for external surface of reactor vessel have improved capability of cooling channel as results obtained ULPU-V facility **[4]** and GAMMA 3D facility **[5]**. The first task seems to be more difficult, unlikely as VVER440 and AP600, determining the final bounding state of molten pool based on conservative scenarios is not enough in high power reactor because of the complicate of severe accident progression which strongly depends on particular scenarios and individual nuclear reactor technology.

Recently, in-vessel melt retention analysis of a VVER1000 nuclear power plant have been presented in a technical report of Joint Research Centre, the European Commission’s in-house science service, as part of the outcome of the EU “Stress Tests” in 2012 **[6]**. In this study, the participating institute and utilities in Europe used various computer codes as MELCOR, SOCRAT, ASTEC, PROCOR, RELAP and CFD to perform IVR analysis for VVER1000. A Large – Break Loss of Coolant Accident (LBLOCA) due to double-ended break of the cold leg with equivalent diameter of 850 mm (the broken loop is assumed to be the loop with the pressurizer and the break is located near the reactor inlet) along with loss of offsite power was chosen as the most challenging for the IVR strategy. The primarily results presented in the report were different in different codes, and still had a lot of uncertainties. One of the conclusions of the study suggested that there needs more study focusing on accident analysis before a steady state molten pool established in lower plenum because some results showed that in transient period the thermal load from hot debris even was higher than from steady state molten pool.

With the hope of additional contribution on study of IVR application for VVER1000, the paper analyzed SBO accident accompanied with IVR implementation for VVER1000/V320 reactor by using MELCOR code. The SBO accident has been broken into four cases: in the two first cases without depressurizing for reactor vessel (long-term high pressure) and the two others with depressurizing for reactor vessel (short-term high pressure) when temperature at core outlet exceeded 650^{o}C by opening pressurizer safety valves following emergency operation procedures of VVER1000/V320 reactor.

2. BRIEF DESCRIPTION ON VVER1000/V320 REACTOR

A VVER1000/V320 is a version of VVER1000 reactor, which is a kind of Russian pressurized water reactors, produces 3000MW thermal power and generates 1000MW electrical power. The reactor core consists of 163 hexahedral fuel assemblies and 61 control rod clusters. Each assembly comprises 312 fuel rods and 18 control rod tubes.

The primary circuit consists of a reactor vessel which is internally divided into downcomer, lower plenum, reactor core, upper plenum and upper head area, and four loops connected with reactor vessel by hot leg and cold leg nozzles. Each loop has a main coolant pump, hot leg, cold leg and horizontal steam generator (SG). The pressurizer (PRZ) is connected to hot leg of 4th loop and the spray lines of pressurizer were connected to cold leg of 1st loop, and safety valves and relief valves are mounted on the top of PRZ.

The secondary circuit consists of four loops. Each loop has a horizontal steam generator and one steam line. On each steam line two SG safety valves, one main steam isolation valve (MSIV), one check valve and one atmospheric steam discharge valve (BRUA), are mounted.

The emergency core cooling system (ECCS) consists of high and low pressure injection systems along with hydro-accumulators (HAs).

3. DESCRIPTION OF VVER1000/V320 MODEL IN MELCOR INPUT DECK

MELCOR code **[7]**, a fully integrated system computer code developed by Sandia National Laboratories, composed of many packages, can simulate a broad spectrum of severe accident phenomena in light water nuclear power plants such as thermal-hydraulic response in the reactor coolant system, reactor cavity, containment and confinement buildings; core heat-up, degradation and relocation; core-concrete interaction; hydrogen production, transport, and combustion; fission product release and transport behavior. The version MELCOR 1.8.6 is a recent version of MELCOR code. It has some new features such as the improvement in molten pool model in lower plenum and new nodalization model for lower head vessel which improve the capability of MELCOR for calculating heat transfer between lower head vessel with internal or external ambient. A nuclear power plant is read in MELCOR code through input files in which it has been modeled and nodalized as control volumes and heat structures according to MELCOR guidelines.

The primary system was modeled into four loops representing the real four loops. Each loop was modeled by control volumes and flow paths representing for hot leg, cold leg, steam generator tubes, steam generator hot and cold collector. For more detail nodalization of primary system (Fig.1): hot leg was nodalized into two control volumes (HL1 and HL2); cold leg was nodalized into three control volumes (CL1, CL2 and CL3); steam generator tubes were divided into 5 levels, each level was divided into 2 parts as hot part (HP) and cold part (CP); hot collector and cold collector were modeled as individual control volume. Space inside reactor vessel were divided into 5 control volumes (Fig.1) representing for downcomer, lower plenum, core, upper plenum and upper head. The reactor core were divided into 12 levels and 6 rings including 5 core radial rings and 1 ring representing for downcomer (Fig.2).

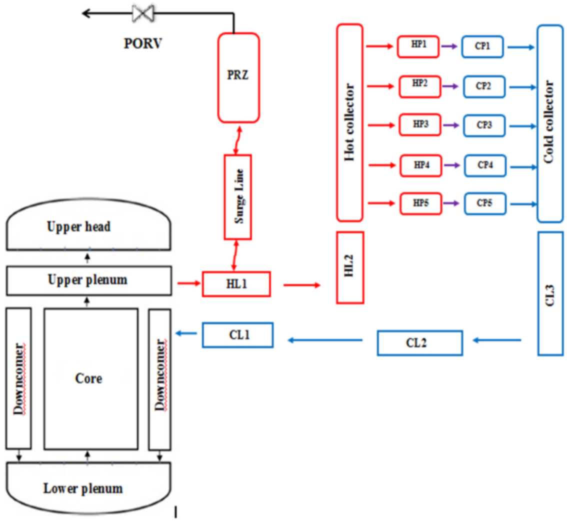

Fig 1: Nodalization scheme of primary loop

	Ring 1	2	3	4	5	6	
12	112	212	312	412	512	612	Upper core support plate
11	111	211	311	411	511	611	Core
10	110	210	310	410	510	610	Core
9	109	209	309	409	509	609	Core
8	108	208	308	408	508	608	Core
7	107	207	307	407	507	607	Core
6	106	206	306	406	506	606	Core
5	105	205	305	405	505	605	Lower core support plate
4	104	204	304	404	504	604	Lower plenum
3	103	203	303	403	503	603	Lower plenum
2	102	202	302	402	502	602	Lower plenum
Level 1	101	201	301	401	501	601	Lower plenum

Fig 2: Nodalization scheme of reactor core

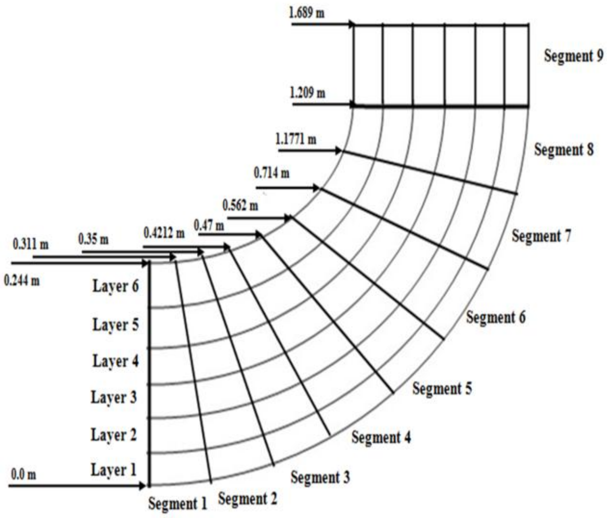

Fig 3: Nodalization scheme of lower head wall

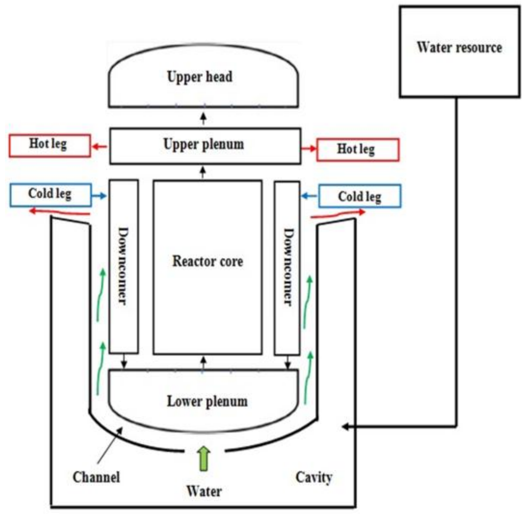

Fig 4: Nodalization scheme of cavity

Pressurizer and surge line were modeled as control volumes individually. The relief valves connected to pressurizer was modeled too (Fig.1).

For detail analysis heat transfer between lower head and hot debris (melt/particles), as well as water outside, the lower head wall was divided into 6 layers and 9 segments (Fig.3).

In order to deploy IVR strategy, the cavity and IVR cooling channel were also modeled. Water resource for IVR strategy was modeled as control volume which was assumed to be an unlimited water resource (Fig.4).

Four accumulators were modeled as control volumes (ACC1, ACC2, ACC3, ACC4). Two was connected to upper plenum and the two others were connected to down comer (Fig.5).

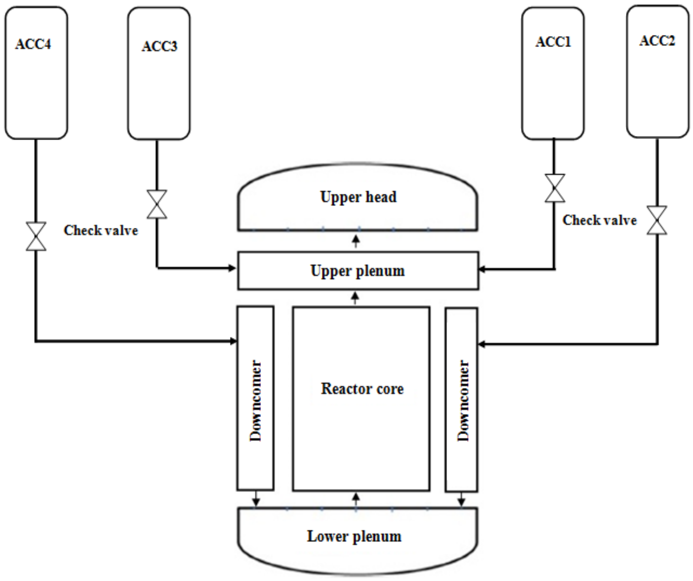

Fig5: Nodalization scheme of passive core cooling systems

In MELCOR code [7], the Larson-Miller parameter has been used to evaluate the failure of lower head vessel, based on the temperature profile through lower head to calculate the reactor vessel strain.

The Larson-Miller creep-rupture failure model gives the time to rupture, t_R, in seconds, as:

$$t_R = 10^{\left(\frac{P_{LM}}{T} - 7.042\right)} \tag{1}$$

where: T is the temperature of segment and P_{LM} is the Larson-Miller parameter and given by:

$$P_{LM} = 4.812 \times 10^4 - 4.725 \times 10^3 \log_{10} \sigma_e \tag{2}$$

where σ_e is the effective stress (Pa) and given following:

$$\sigma_e = \frac{(\Delta P + \rho_d g \Delta z_d) R_i^2}{R_o^2 - R_i^2} \tag{3}$$

with: ΔP is the pressure difference across the lower head; ρ_d and Δz_d are the density and the depth of the debris in lower plenum; R_i and R_o are the inner vessel radius and outer radius of load-bearing vessel.

The life-fraction rule gives the cumulative damage, expressed as plastic strain, $\varepsilon_{pl}(t)$, as:

$$\varepsilon_{pl}(t + \Delta t) = \varepsilon_{pl}(t) + 0.18 \frac{\Delta t}{t_R} \tag{4}$$

The failure occurs when the strain reaches to the value of 0.18.

4. INITIAL AND BOUNDARY CONDITIONS

The initial and boundary conditions of the important parameters are presented in the following tables. The comparison of plant initial parameters with MELCOR calculated stabilized values was presented in table 1, and the pressurizer safety valves set point for closing and opening was given in table 2.

Table 1: Comparison of initial parameters

Parameteres	MELCOR value	Design value
Core power (MW)	3100	3000+210
Primary pressure (MPa)	15.6	15.7±0.3
Maximum coolant temperature at reactor intlet (°K)	567	559.15±2.0

Average coolant temperature at reactor outlet (^oK)	596	593.15±3.5
Mass flow rate through reactor core (kg/s)	17650	17 611
Pressure in steam generator (MPa)	6.29	6.28±0.2
Steam mass flow rate at steam generator outlet (kg/s)	420	437

Table 2: Pressurizer safety valve characteristics

PRZ SV name	Characteristics	Design value	MELCOR value
SEMPELL1	Opening pressure (MPa)	18.11	18.11
	Closing pressure (MPa)	16.67	16.67
SEMPELL2,3	Opening pressure (MPa)	18.6	18.6
	Closing pressure (MPa)	17.07	17.07

5. DEFINITION OF SCENARIOS

The postulated accident scenarios used in this study was mainly SBO which was divided into four cases with the aim of study the necessary of reactor vessel depressurization as prerequisite requirement for IVR implementation and the effect of depressurizing action of operator following SAMGs to IVR strategy for VVER1000/V320. The main assumption of the SBO scenario and the four cases are given below.

Table 3: Additional assumption for four cases

Case	Additional assumption
1A	Main SBO accompanied by: - Without RV depressurization and; - Without IVR implementation.
1B	Main SBO accompanied by: - Without RV depressurization and; - With IVR implementation by flooding cavity when Tcoreoutlet > 650^{o}C
2A	Main SBO accompanied by: - With RV depressurization by opening PRZ safety valve when Tcoreoutlet > 650^{o}C and - Without IVR implementation.
2B	Main SBO accompanied by: - With RV depressurization by opening PRZ safety valve when Tcoreoutlet > 650^{o}C and; - With IVR implementation by flooding cavity when Tcoreoutlet > 650^{o}C

Definition of main SBO scenario:

- Loss of offsite and onsite power including diesel generator and batteries except batteries for BRU-A valves of SGs;
- Loss of all active safety system;
- Failure of Emergency Feed Water;
- Pressure of PRZ was controlled by safety valves with characteristics in table 2;
- There was not taken into account MCPs seal leakages;
- Pressure of SGs was control by BRU-A valves to maintain pressure in SGs below 6.7Mpa;
- Available of all hydro-accumulators.

The four cases were based on the main SBO scenario with additional assumption given in table 3.

6. RESULTS AND DISCUSSION

For the two firs cases, 1A and 1B, the paper studied and analyzed IVR strategy under long term high pressure accident without taking reactor vessel depressurizing action. The results showed that under the high pressure condition the failure of lower head vessel early happened at **15266s** and **20574s** respectively in case 1A and case 1B due to creep - rupture, even before the relocation of core melt to lower plenum. The case 1B, with IVR implementation, only means to delay the time of lower failure but prevents lower head vessel from failure.

In the case 2A and case 2B, the reactor vessel had been experienced high pressure condition for **8000s**, then it was depressurized when Tcoreoutlet > 650^{o}C following initial condition. In both cases 2A and 2B, the results showed that with depressurization the integrity of reactor vessel was kept longer than two first cases 1A and 1B, however, this action did not prevent lower head vessel from failure due to creep - rupture. The failure of lower head in case 2A and case 2B were at **51375s** and **63585s** respectively. The IVR implementation in case 2B only served to delay the failure of lower head vessel.

The main progression of accident in the cases was given below.

a) Two first cases (1A and 1B): without depressurizing for reactor vessel

Main events happened in case 1A and case 1B were given in Table 4, the results indicated IVR implementation did not affect the in-vessel accident progression when main events in case 1B happened at the same time as in case 1A, except the oxidation and the failure of lower head vessel.

Table 4: Main events in case 1A and case 1B

Main events / **Case**	**1A**	**1B**
SBO happened	0.0s	0.0s
Reactor tripped	1.0s	1.0s
Begin of core uncovery	2050s	2050s
Total core uncovery	3310s	3310s
Water in LP completely vaporized	3090s	3090s
Start to flood cavity	-	7957s
Start of oxidation	11800s	12050s
Start of cladding failure	-	-
Start of failure of supporting structures in LP	16750s	18250s
Start of core debris appearance in LP	-	-
Lower head vessel failure (CREEP-RUPTURE)	**15266s**	**20574s**

Without depressurizing, pressure in primary circuit in both cases was controlled and maintained by safety valves following the opening and closing set point (Fig.6) and the mechanical opening and closing of safety valves caused the difference in the trend of pressure in case 1A and case 1B as presented in Fig.6. However, pressure in both cases was still maintained at high value and secured the condition of scenario at high pressure.

Due to high pressure, all passive core cooling systems in the two cases was neutralized, therefore there was no water additionally injected into reactor vessel. The decay heat continued releasing in reactor core causing gradual decrease of water inventory in reactor vessel in the both cases 1A and 1B with the same rate as showed in Fig.7. Without in-vessel cooling, the temperature inside reactor vessel kept increasing, which led temperature of lower head vessel also increased (Fig.8).

In case 1A, without external cooling, the temperature of lower head vessel increased with higher rate compared to that of case 1B. And under high pressure and high temperature, the failure of lower head vessel in case 1A was inevitable due to creep-rupture following Larson-Miller standard in MELCOR code at **15266s**. The time of failure of lower vessel predicted in 1A was close to result predicted by authors using ASTEC code in [8] at **15452s** with the same scenario.

However, in case 1B, thank to cavity flooding when Tcoreoutlet > 650°C at 7957s (Fig.8), external lower head vessel cooling brought into play to slow the increase of lower head temperature after 10000s that explained why the trend of temperature of lower head vessel in case 1B was below compared to that of case 1A as showed in Fig.9. Water level in cavity was remained at height of cold leg and heat transfer between reactor vessel wall and water in cooling caused vibration of water level as shown in Fig.8. However, the results in case 1B pointed out that the external cooling only brought into play to delay the failure of VVER1000/V320 lower head vessel under high pressure accident. The failure of lower head vessel 1B case happened at **20574s** due to creep rupture when its thermal strain reached to **0.18** according Larson-Miller standard in MELCOR code (Fig.10).

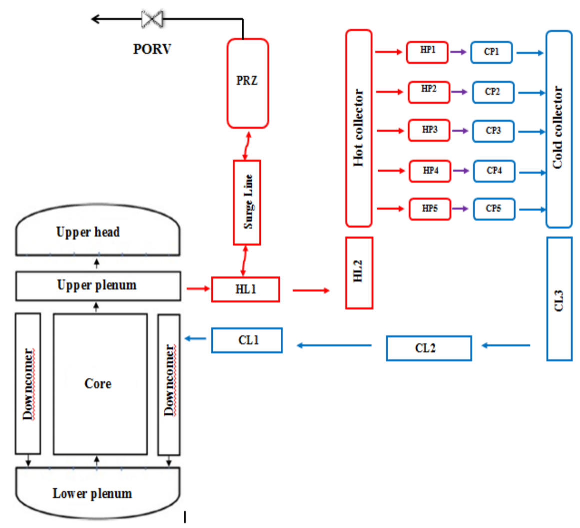

Fig 1: Nodalization scheme of primary loop

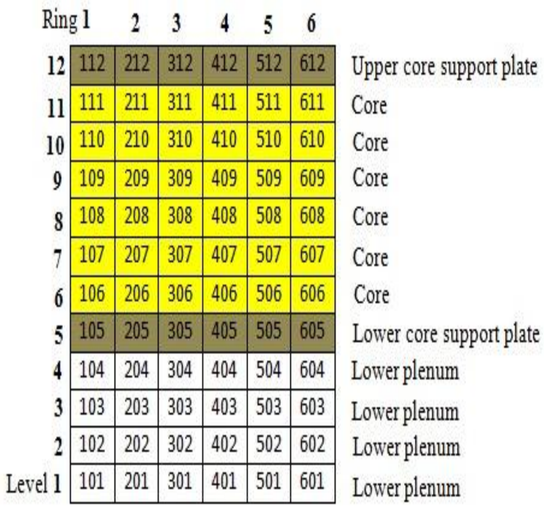

Fig 2: Nodalization scheme of reactor core

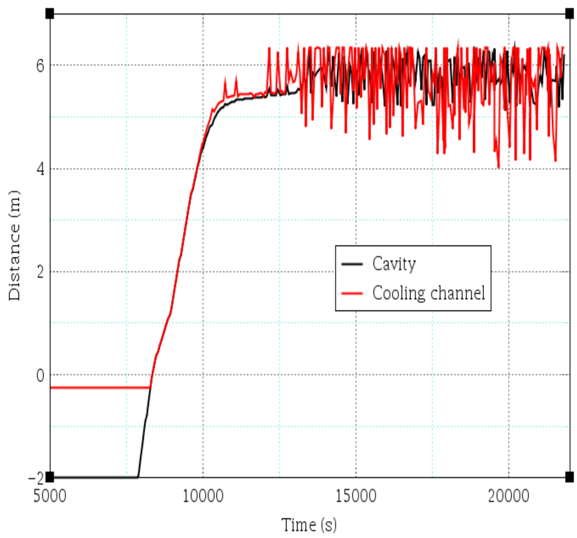

Fig.8: Water level in cavity (black) and cooling channel (red) in case 1B

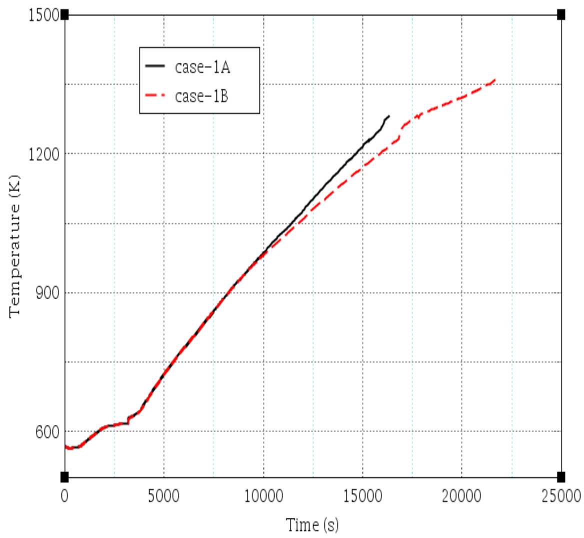

Fig.9: Temperature of lower head vessel in 1A (black) and 1B (red)

The results in both cases 1A and 1B indicated that in case of SBO accident without depressurizing action for reactor vessel. Even external cooling was deployed in case 1B could not prevent lower head vessel from failure due to creep-rupture under high pressure and high temperature. Therefore, the results confirmed again the need of depressurization for reactor vessel as one of the prerequisite requirements in order to secure the success of IVR strategy for a light water reactor.

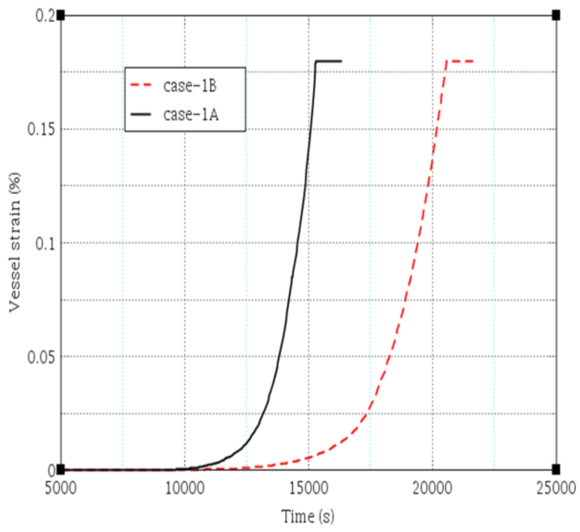

Fig.10: Thermal strain of lower head vessel in case 1A (solid line) and 1B dash line)

b) Two remaining cases: with depressurizing for reactor vessel (2A and 2B)

As severe accident management guides of VVER1000/V320, under high pressure accident when temperature at core outlet exceeds 650^{o}C, operators will fully open safety valves of pressurizer manually in order to depressurize for reactor vessel. This action will allow safety injection systems to inject water into reactor vessel with the aim of cooling reactor core. In the case 2A and case 2B, only passive systems (four accumulators) were available to supply water into reactor vessel when pressure in reactor vessel decrease below 5.8MPa. The main events happened in both cases 2A and 2B are presented in Table 5.

Table 5: Main events in case 2A and case 2B

Main events \ Case	2A	2B
SBO happened	0.0s	0.0s
Reactor tripped	2.0s	2.0s
Begin of core uncovery	2100s	2200
Total core uncovery	3400s	3400s
Water in LP completely vaporized	3300s	3300s
Time of primary depressurization	**8021s**	**8000s**
Start of flooding cavity	-	8000s
Begin of water injection from HAs	8562s	8566s
Stop of HAs injection	30000s	30000s
Start of oxidation	27800s	31000s
Failure of supporting system in lower plenum	39775s	44447s
Start of cladding failure	39800s	44500s
Start of core debris appearance in LP	40000s	45000s
Lower head vessel failure (CREEP-RUPTURE)	**51375s**	**63585s**

The difference in the second cases group (2A and 2B) compared to the first group cases (1A and 1B) is that **the action of depressurization for reactor vessel** was modeled, therefore, before the time of occurrence of depressurization the consequence of accident should be the same in two groups. However, there are differences in happening time of the events, the differences might be explained that MELCOR code is a very sensitive code, only changing or adding control volume or control function … also causes the vibration of time steps. In case 2A and 2B, with additional function for opening safety valves caused the differences in the time of events before occurrence of depressurization.

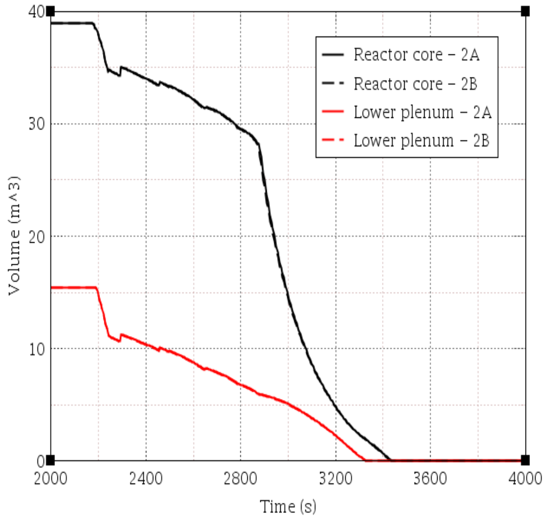

Fig.11: Water inventory in reactor core (black) and in lower plenum (red) in case 2A and 2B

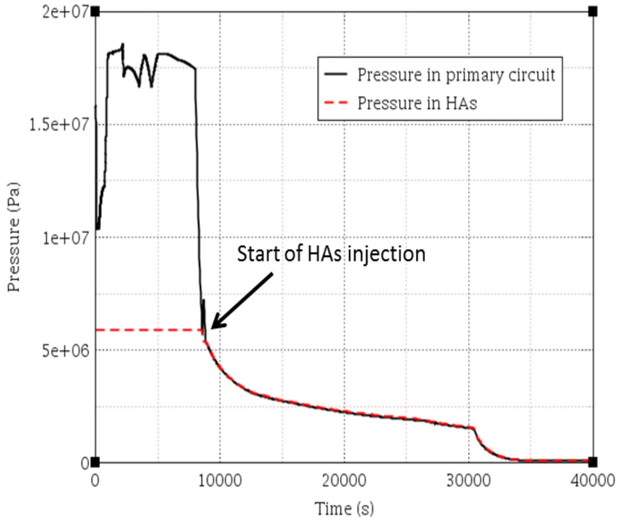

Fig.12: Pressure in primary circuit and HAs

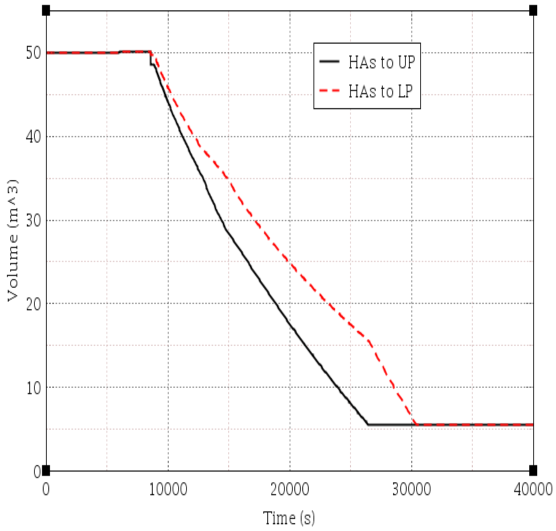

Fig.13: Volume of water in HAs: HAs connected to upper plenum (black line) and connected to lower plenum (red line) in both cases 2A and2B

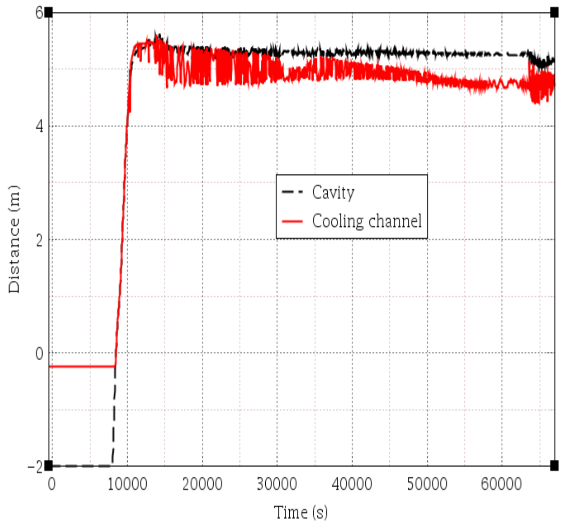

Fig.14: Water level in cavity (black line) and cooling channel (red line) in case 2A

Come back to case 2A and case 2B, as having the same main scenarios and only difference in implementing IVR strategy in case 2A, most main events in both cases 2A and 2B happened at the same time with slight difference, except from start of oxidation event to the failure of lower head vessel. Before the depressurization happened at 8021s in case 2A and at 8000s in case 2B, water in reactor vessel in the both cases completely evaporated (Fig.11). When pressure in reactor vessel decreased to below 5.8MPa at 8562s in case 2A and 8566s in case 2B, water from hydro-accumulators (HAs) was injected to reactor vessel (Fig.12), and the injection was stopped when volume of water in HAs decreased to 5 m^3 (Fig.13) to prevent inert gas from going to reactor vessel.

The IVR strategy in case 2B was initiated at the same time of depressurization which presented water level in cavity and cooling channel and the vibration of water level in channel demonstrated heat transfer between lower head wall and water in cooling channel as demonstrated in Fig14. The external reactor vessel cooling in case 2B slowed the increase of lower head temperature compared to that of case 2A as showed in Fig15, the average temperature of lower wall at segment 5th case 2B advanced with lower rate compared to that of 2A. With the cooler reactor vessel, the ambient inside reactor vessel was also cooler that explained why the oxidation and failure of fuel cladding in case 2B happened late at 31000s and 44500s compared to 27800s and 39800s in case 2A respectively. Without any water was injected into reactor vessel in both cases after HAs stopped supplying and the failure of fuel cladding in case 2A was earlier than that of 2B, therefore, the relocation of core melt also happened earlier in case 2A compared to 2B, and core debris appeared in lower plenum in both cases 2A and 2B were at 40000s and 45000s respectively. Total mass of debris in lower plenum in 2A and 2B was graphically given in Fig16 and Fig17 respectively and numerically in Table 6.

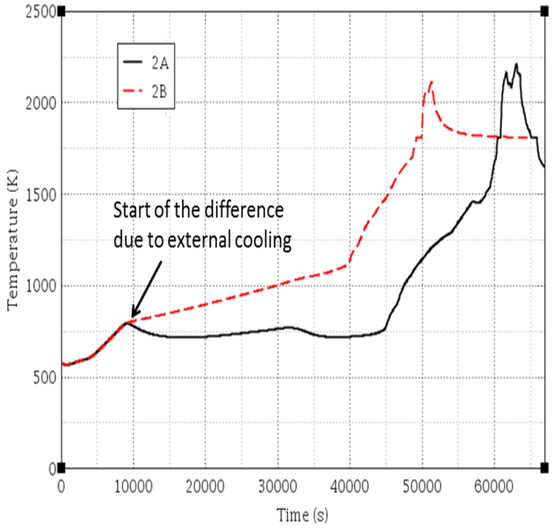

Fig.15: The average temperature of lower head wall at segment 5 in both cases 2A and 2B

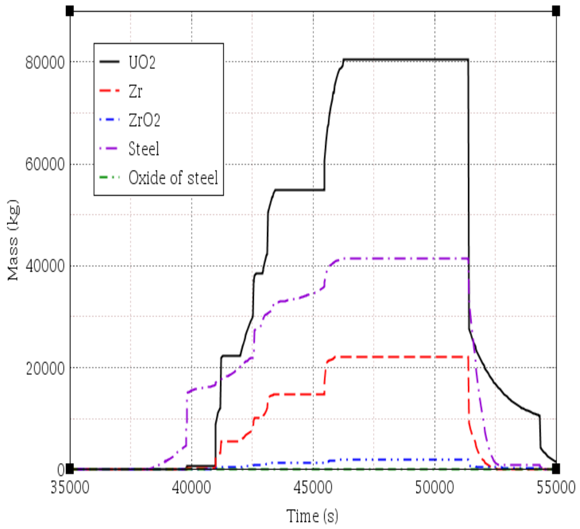

Fig.16: Mass of elements debris in lower plenum in case 2A

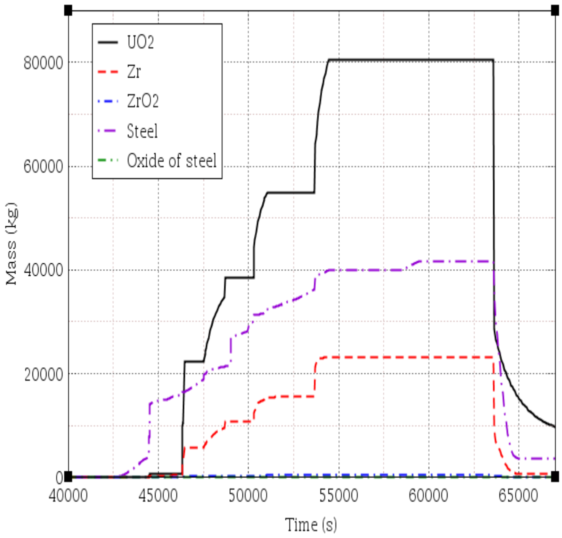

Fig.17: Mass of elements debris located at lower plenum in case 2B

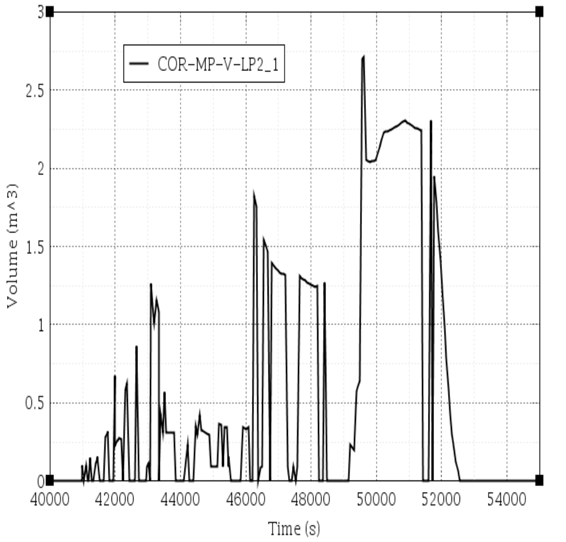

Fig.18: Volume of steel molten pool in lower plenum in case 2A

Discussion on results of case 2A

In case 2A, the failure of supporting systems (columns) in lower plenum established steel debris (pink line) in there at 39800s as showed in Fig.16. The hot core debris relocated to lower plenum at 40000s and directly contacted to lower wall, which caused the increase of lower head temperature at 40000s as shown in Fig.15 (red line). There was no water in lower plenum, therefore, debris started to melt due to residual decay heat in core debris. The elements have lowest melting point like steel which would be melt first. At first, the melting of steel happened locally which formed small unstable molten pool volume of steel from 41000s to 49000s as given in Fig.18. From 49000s, the formation of steel molten pool became stable and adjacent small pools merged into together to form a lager molten pool of steel.

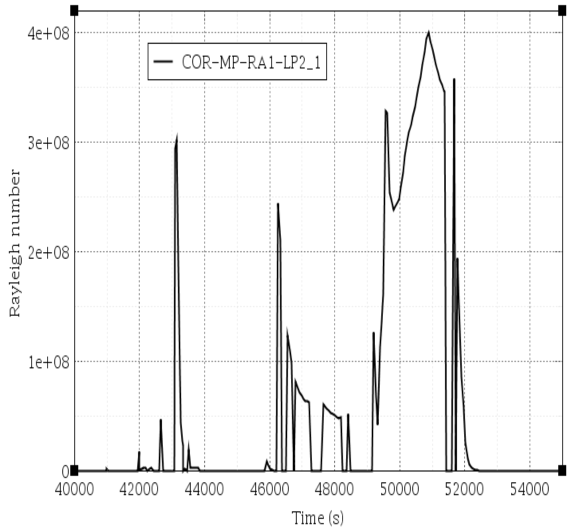

Fig.19: Rayleigh number steel molten pool in lower plenum in case 2A

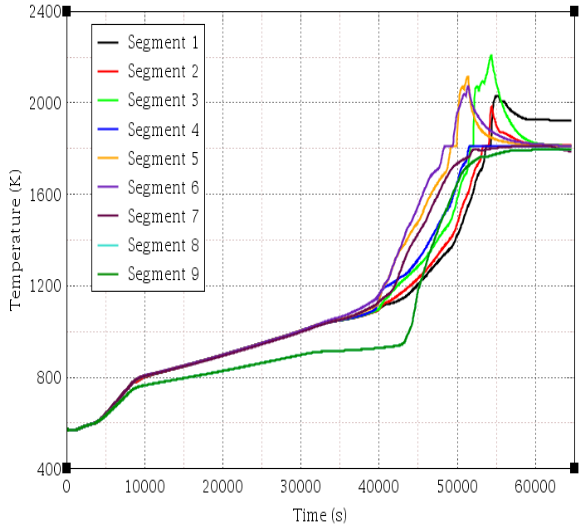

Fig.20: Temperature all segments in case 2A

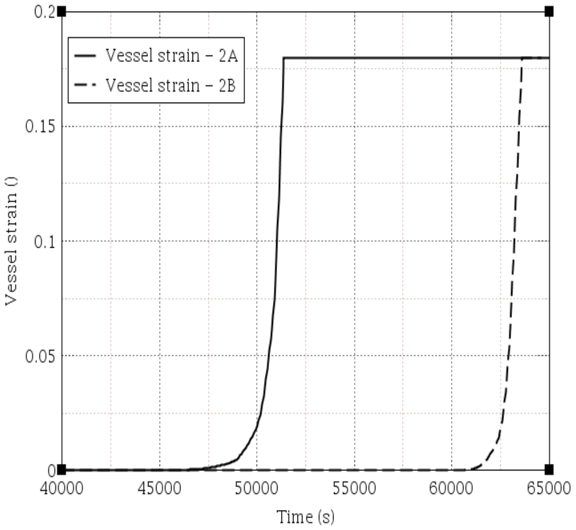

Fig.21: Reactor vessel strain in case 2A (solid line) and case 2B (dash line)

The molten pool seemed to form adjacent to segment 5 and 6 of lower wall, and the heat transfer between molten pool and lower head wall characterized by Rayleigh number (Fig.19) in MELCOR 1.8.6 loaded more heat than steel particles (heat transfer coefficient between particles and lower head wall was set in input file as 1000 W/m^2), therefore, the temperature of low head wall at segments 5th and 6th sharply increased at 50000s as seen in Fig.20. The temperature of segment 5th seemed to increase a little sharper and higher than that of segment 6th , and the results indicated that vessel strain at segment 5th reached 0.18 (Fig.21) at **51375s** firstly causing the failure of lower head vessel in case 2A according to Larson-Miller parameter.

Discussion on results of case 2B

For case 2B, thank to reactor vessel cooling, severe accident consequence in lower plenum occurred later as compared to case 2A, for detail: the failure of support system happened at 44447s compared to 39775s and the appearance of core debris was at 45000s compared to 40000s. The core debris directly contacted and transferred heat to lower head wall, therefore, the temperature of all segments started to increase at 45000s as seen in Fig.22 and Fig.23 showing the average temperature of the 1st layer and the 6th layer of 9 segments. The formation of the molten pool in case 2B locally started forming at 52500s and became a larger pool nearly at 60000s (Fig.24) and the Rayleigh number of the molten pool was presented in Figure 25. As seen in Fig.20 and Fig.21, before the appearance of core debris, the temperature of all segment were well below 800^{o}K, however, after 45000s, the massive relocation of core debris to lower plenum (Fig.17) caused the increase of temperature of all segments. Figure 21 also pointed that only the outermost temperature of segment 8th and 9th were kept at saturation temperature of water, while others were far above. Through Fig.22 and Fig.23, we can see that the molten pool seemed to directly contact to layer 3th, 4th and 5th when temperature of three layers increased fastest after 60000s, and the fastest of three was of segment 5th.

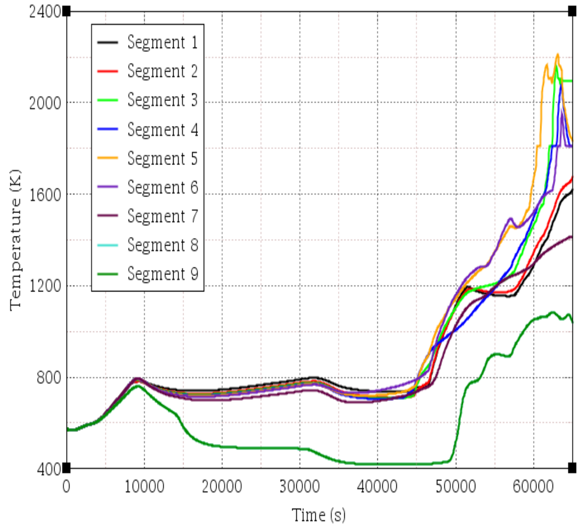

Fig.22: Temperature layer 6th of all segments

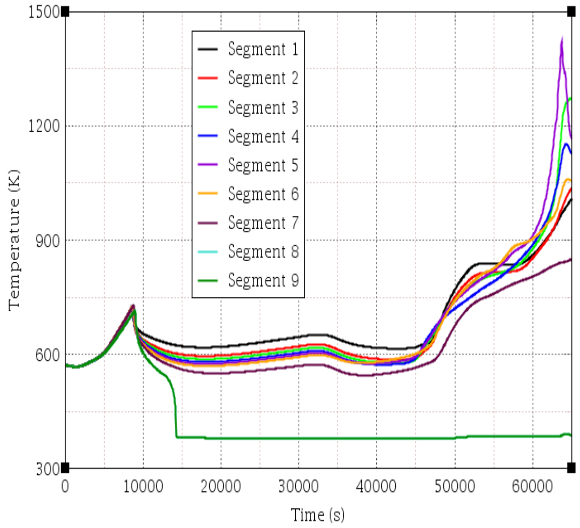

Fig.23: Temperature layer 1st of all segments

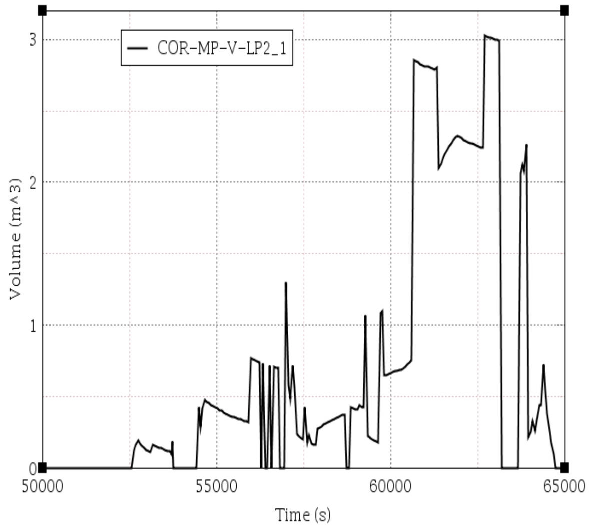

Fig.24: Volume of steel molten pool in lower plenum

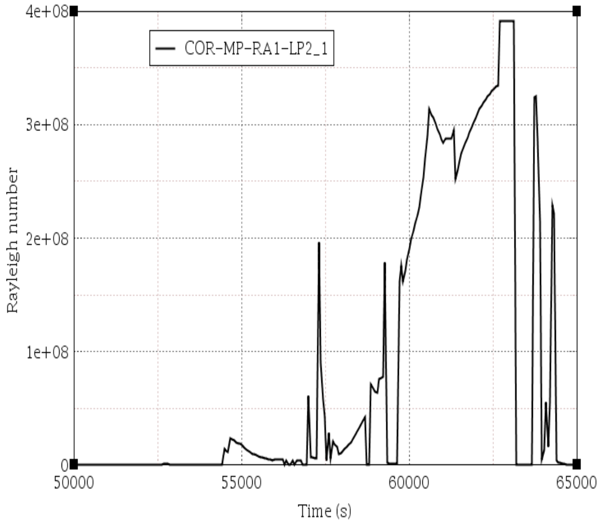

Fig.25: Rayleigh number steel molten pool in lower plenum

Fig.23 showed that after 10000s the temperature of external surface of lower head wall were well above the saturation of water in cooling channel, therefore, the heat transfer between lower head wall and water might be at the transition boiling between the fully developed and film boiling regimes. Before the appearance of core debris in lower plenum at 45000s, the film boiling locally occurred at external surface of lower head wall was not dangerous when circulation of water as characterized by mass flow rate of water at channel inlet presented in Fig.26 and flow rate of steam at outlet Fig.27, can accommodate to keep the temperature of external surface around 600^{o}K as seen in Fig.23. The circulation in channel suddenly deceased after 30000s because there was a slight decline of temperature of lower head wall due to the sudden decrease of pressure in reactor vessel after 30000s (Fig.12). For whole scene, we can see that the circulation in cooling channel gradually decreased, it can be explained that the formation of vapor film at low segments might obstruct water to enter into channel. This also explains the decrease water level in cooling channel (Fig.14) following the time, whereas the mass flow rate of steam at outlet still increased. Due to the decline of cooling circulation and the increase heat load from hot debris and molten pool afterward, the heat transfer might turn into stable film boiling from 55000s while mass flow rate of steam at outlet became stable as seen in Fig.28.

After 60000s, the lower head wall at segment 5th became very thin, and its thickness was about 4 cm because there was only layer 1st which temperature was below melting point (1600^{o}K).Whereas, heat load from molten pool sharply increased as characterized by the increase of Rayleigh number in Fig.18 from 62500s, caused the increase of heat flux on the outer face of segment 5th. When the heat flux on outer face of segment 5th reached to value 0.33 MW/m^2 might cause the boiling crisis at cooling channel which can be explained by the sudden decrease of steam flow rate at outlet of channel at 63585s as seen in Fig.27 and the decrease of heat flux after reaching the peak. The occurrence of boiling crisis at outer surface of segment 5th caused the failure of lower head vessel when its strain reached to value 0.18 according to Larson-Miller parameter (Fig 21).

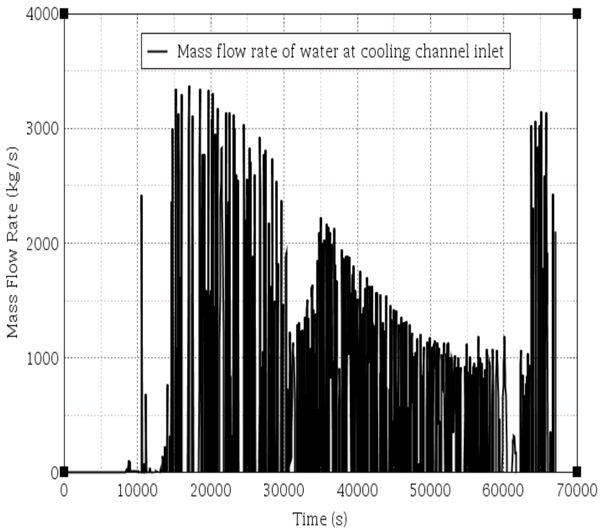

Fig.26: Mass flow rate of water at channel inlet

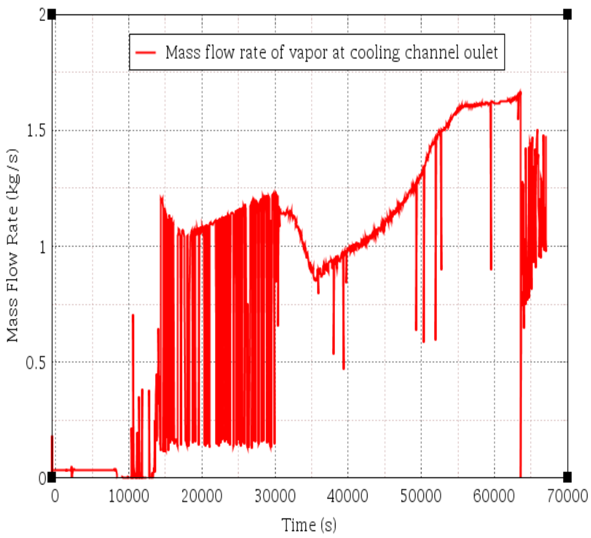

Fig.27: Mass flow rate of steam at channel outlet

Table 6: Mass of debris located to lower plenum in case 2A and 2B

Debris	Mass in case 2A (kg)	Mass in case 2B (kg)
UO2	80600	80600
Zr	22100	23100
ZrO2	1900	515
Steel	41400	41700
Oxide of steel	40	17

7. CONCLUSION

In the long-term high pressure SBO scenarios, the integrity of VVER1000/V320 lower head vessel could not be secured, the high pressure and high temperature caused creep-rupture failure of VVER1000/V320 lower head vessel following the Larson-Miller parameter. The external cooling only delay the failure of lower head vessel, not prevent it from failure. The results, therefore, in the long-term high pressure SBO scenarios confirmed again the action of depressurizing for reactor vessel as a prerequisite requirement for implementing of IVR strategy for a light water reactor in general and for VVER1000/V320 particularly.

For the SBO scenarios with taking action of reactor vessel depressurization, the failure of lower head vessel was also creep-rupture mode following the Larson-Miller parameter, but caused by the thermal load from adjacent molten pool, not by pressure as in the long-term high pressure scenarios, and the implementation of IVR strategy only brought into play to delay the failure of VVER1000/V320 lower head. The results of the scenarios showed that the failure of VVER1000/V320 lower head vessel happened before a steady sate molten pool established in lower plenum which indicated that the failure of VVER1000/V320 could happened in transient period and the results also confirmed again the importance of severe accident progression analysis for high power reactors in term of IVR strategy.

MELCOR code is a kind of system codes which is good for providing overall picture of severe accident and early warning in severe accident progression. The Lumped Parameter method have been used in MELCOR to simulate heat transfer in molten pool and to calculate the transient conditions from relocation of core melt to formation of molten pool in lower plenum. However, the limitations of the Lumped Parameter method are using the average correlations and only applying for simple geometry so that it cannot be applied for analysis of specific effects of flow characteristic in term of heat transfer in molten pool. CFD simulation has been developed and applied to study turbulent natural convection in a volumetrically heated melt pool, provided important insights into the thermo-fluid mechanisms. Although using CFD for molten pool heat transfer simulation is very computationally expensive, it is very useful in providing insights into flow physics and flow behavior, moreover, CFD simulation can be used for analysis in complicate geometry. Especially, CFD combined with PECM will save time of calculation, but providing precise results.

Therefore, for study IVR, only using MELCOR code is not enough. The results obtained from MELCOR will be improved and updated following up analyses of more precise simulation by integral calculation by using CFD tool (FLUENT) combined with PECM for more precise estimation of thermal load to lower head vessel in the future.

ACKNOWLEDGEMENT

The author thanks to Ministry of Science and Technology, Viet Nam Atomic Energy Institute and Nuclear Training Center for supporting the project “Study and Calculation In-Vessel Melt Retention strategy for VVER1000”, ID: DTCB.04/18/TTNTC.

REFERENCES

1. T.G.Theofanous et al., “In-Vessel Coolability and Retention of Core Melt,” DOE/ID-10406 Volume 1, October 1996.
2. O. Kymalainen, H. Tuomisto, T.G.Theofanous, “In-vessel retention of corium at the Loviisa plant,” Nuclear Engineering and Design 167(1997) 109-130.
3. T.G.Theofanous and S. Angelini, “Natural convection for in-vessel retention at prototypic Rayleigh numbers”, Nuclear Engineering and Design 200 (2000)1-9.
4. T.N.Dinh, J.P.Tu, T.Salmassi and T.G.Theofanous, “Limits of coolability in the AP1000-ralated ULPU-2400 configuration V facility,” the 10th International Topical Meeting on Nuclear Thermal Hydraulics (NURETH-10), Seul, Korea, 2003.
5. J.L.Rempe et al., “*In-vessel retention of molten corium: Lession learned and Outstanding issues*,” Nuclear Technology, Vol.161, 2008.
6. SANGIORGI Marco et al., “*In-Vessel Melt Retention (IVMR) Analysis of a VVER-1000 NPP*,” JRC Technical Report, European Commission, 2016
7. Sandia National Laboratories, MELCOR computer code manuals, Ver.1.8.6, Rev. 3 vol. NUREG/CG 6119, 2005.
8. Polina Tusheva et al., *“Analysis of severe accidents in VVER1000 reactor by using the integral code ASTEC,”* Proceedings of the 17th International Conference on Nuclear Engineering ICONE17 July 12-16, 2009, Brussels, Belgium.