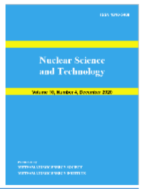
Nucl. Sci. and Tech., Vol.10, No. 2 (2020), pp. 01-14

Nuclear Science and Technology

Journal homepage: https://jnst.vn/index.php/nst




# Analysis of in-vessel accident progression in VVER1000 NPP during SBO accident with external reactor vessel cooling method

Doan Manh Long[1], Nguyen Van Thai[2], Tran Chi Thanh[3]

*[1)] Nuclear Training Center, 140 Nguyen Tuan, Thanh Xuan, Hanoi*

*[2)] Department of Nuclear Engineering and Environmental Physics, School of Engineering Physics, Hanoi University of Science and Technology, 01 Dai Co Viet, Hai Ba Trung, Hanoi*

*[3)] Viet Nam Atomic Energy Institute, 59 Ly Thuong Kiet, Hoan Kiem, Hanoi*

*Email[(1)]: doanmanhlong@vinatom.gov.vn*

**Abstract:** In this study, the MELCOR v1.8.6 code was utilized to perform an analysis of the in-vessel accident progression in VVER1000 reactor during the Station Black-Out (SBO) accident with and without external reactor vessel cooling (ERVC) strategy. The analysis presented the predictions of the main phenomena during the accident such as failure of fuel cladding, collapse of lower core support plate, relocation of core debris to lower plenum and mass of debris components in lower plenum, and provided comparisons between two cases in term of main parameters such as integrity time of reactor and structure components of molten pool. These parameters are very important inputs for further research on the application of external vessel cooling strategy for VVER1000 reactor.

**Keywords:** *External Reactor Vessel Cooling, SBO, VVER1000, In-Vessel Melt Retention, Severe Accident.*

## I. INTRODUCTION

The external reactor vessel cooling strategy through submerging reactor vessel into water has been well known as a novel severe accident management strategy with the aim at preventing lower head vessel from failure in case of core melt accident, called the In-Vessel melt Retention (IVR) strategy. This strategy has been successfully adopted as severe accident management strategy for low power reactors such as VVER440 [1, 2] and AP600 [3]. Subsequently, the strategy has become a crucial research issue in severe accident management strategy for high power reactors as AP1000 [4], APR1400 [5] and HPR1000 [6].

Right from the first, the severe accident codes have greatly contributed to analysis of IVR strategy. Since the analytical methods [7, 8] and numerical methods [9, 10, 11] require initial bounding conditions which mostly provided by severe accident analysis codes such as RELAP5/SCDAP and MELCOR code, in order to evaluate the thermal response of lower head wall under heat load from molten pool. The RELAP5/SCDAP was used to provide bounding conditions including mass of molten debris and its components, temperature, and decay heat deposited in molten debris in an extensive series of severe accident calculations for AP1000 [4] and APR1400 [5]. And the MELCOR code was applied to analyze the in-vessel accident and the capability of IVR strategy for a large scale pressurized water reactor [12].

https://doi.org/10.53747/jnst.v10i2.32
Received 20 February 2020, accepted 08 May 2020

Recently, the preliminarily studies on the feasibility of the IVR strategy for VVER1000 reactor have been analyzed through benchmark calculations under a very conservative scenario of Large Break Loss of Coolant Accident (LBLOCA) by using different severe accident codes as MELCOR, ASTEC, PROCOR and MAAP [13]. The results of the benchmark calculations have confirmed that the IVR strategy could prevent VVER1000 lower head vessel from failure and provided bounding conditions for stand-alone calculations by using computational codes. Although, the benchmark calculations also showed that the large discrepancies were found in the predicted mass components of debris bed and heat flux distribution on the external surface of lower head wall in different codes, the results provided a bunch of necessary data in order to specify a range of bounding conditions for IVR analysis in case of LBLOCA scenario for VVER1000 reactor.

The IVR strategy adopted for VVER1000 reactor was also analyzed through the Station Black-Out (SBO) accident by using ASTEC code and ANSYS Fluent [14] which showed the external vessel cooling was capability for the stabilization of the vessel and the retention of the molten corium inside VVER1000 vessel. However, the study did not provide in details the mass of debris components which relocated to lower plenum and presented decay heat power in debris bed clearly, which would not benefit for further studying on heat load imposing to lower head vessel by using numerical and analytical methods. In addition, the results obtained from one code are not enough for studying IVR application for VVER1000 reactor during SBO accident, there need additional number of studies by other codes in order to contribute additional data for specifying the range of bounding conditions.

In the present study, an analysis of in-vessel accident progression during SBO accident combined with externally vessel cooling for VVER1000 reactor was performed by using MELCOR v1.8.6 code. The study provided predictions on the key parameters for further studies on IVR application for VVER1000 reactor, such as: the timing of main events as collapse of fuel cladding and lower core support plate; relocation of core debris and mass of debris components into lower plenum; and the decay heat deposited in debris bed as well.

## II. MELCOR INPUT FOR VVER1000 REACTOR

The MELCOR v1.8.6 code [15] is a recent version of MELCOR code with taking full of advanced features which have been capability of simulating a broad spectrum of severe accident phenomena in light water nuclear power plants such as thermal hydraulic response of the reactor coolant system; core heat-up, degradation and relocation; hydrogen production, transport, and combustion; fission product release and transport behavior. The MELCOR v1.8.6 code has an improvement in modeling formation of molten pool in lower plenum and thermal-interaction of lower head wall and molten pool. These features enable the version to apply for studying the IVR strategy.

The VVER1000 reactor is a Russian pressurized water reactor which produces 3000MW thermal power and 1000MW electrical power. It comprises of four cooling circulation loops. 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 the fourth loop and the spray lines of pressurizer are connected to cold leg of the first loop, and safety valves and relief valves are mounted on the top of PRZ. 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.

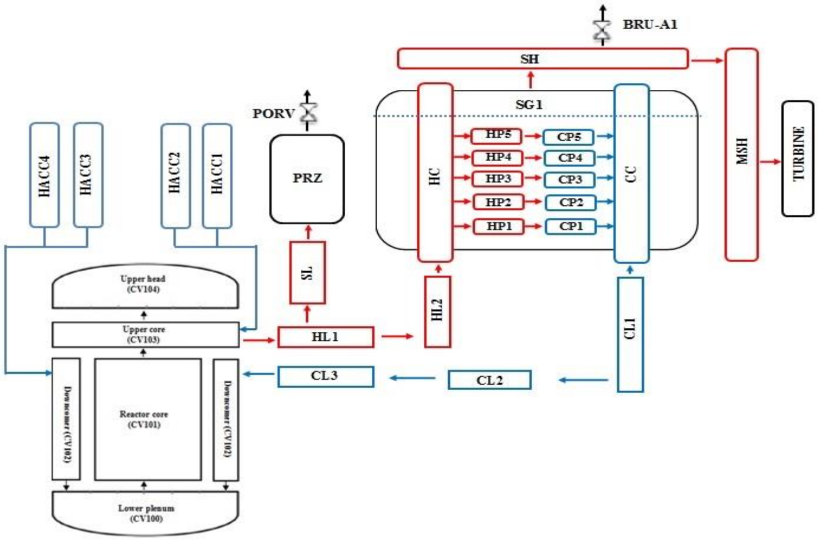



**Fig. 1.** A MELCOR nodalization scheme for thermal hydraulics volumes

The VVER1000 reactor was input into MELCOR code through a nodalization step. Fig.1 displays a nodes scheme for thermal hydraulics volumes of reactor vessel and a circulation loop including the pressurizer systems. The internal volume of reactor vessel consists of lower plenum, reactor core, downcomer, upper plenum and upper head which were assigned numerical identifiers as CV100, CV101, CV102, CV103 and CV104 respectively. In an individual circulation loop, the hot leg was divided into two control volumes (CV110 and CV111) and cold leg was divided into three control volumes (CV130, CV131 and CV132); heat exchanger tubes inside steam generators were divided into five groups which were split into hot part (HP) with red boundary and cold part (CP) with blue boundary; hot collector and cold collector were modeled as two control volumes; pressurizer systems including a surge line, pressurizer, safety and relief valves; and relief tank were also modeled. Secondary part of steam generator was modeled as a control volume. The passive core cooling system, comprising of four Hydro-ACCumulators (HACCs), was also included.

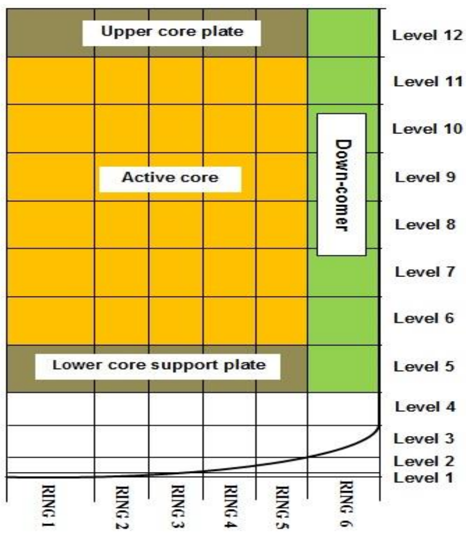



**Fig. 2.** A nodalization scheme for structures in reactor core, lower plenum and downcomer

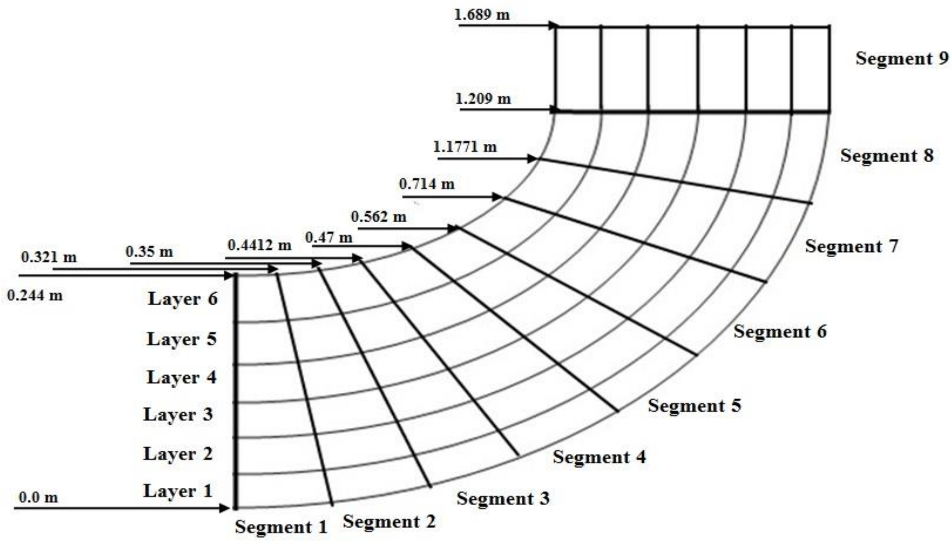



**Fig. 3.** A node model for lower head wall

Fig. 2 demonstrates the nodalization scheme of structures in reactor core, lower plenum and downcomer which were divided into six radically concentric rings and twelve vertically levels. The lower head wall was divided into nine segments and each segment was sliced into six thickness-equaling layers, is presented in Fig. 3.

## II. FAILURE MODEL OF LOWER HEAD

In this calculation, the Larson-Miller creep rupture model in MELCOR code [15] was used to evaluate failure of lower head vessel. The model evaluates cumulative damage in lower head wall following time in response to mechanical loading at elevated temperatures based on the Larson-Miller parameter and a life-fraction rule.

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

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

Where T is the temperature of segment, and $P_{LM}$ is the Larson-Miller parameter that is given as:

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

Where $\sigma_e$ is the effective stress (Pa) and calculated as:

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

Where $\Delta P$ is the pressure difference across the lower head; $\rho_d$ and $\Delta z_d$ are the density and the depth of the debris in lower plenum; g is gravitational acceleration; $R_i$ 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:

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

Due to the division of $\frac{\Delta t}{t_R}$, the plastic strain expressed in Eq.4 is a dimensionless value. And following the rule of the model, the failure of lower head vessel was declared when the value of $\varepsilon_{pl}$ reaches to the value of 0.18 [15].

## III. INITIAL CONDITIONS AND SCENARIOS

Before initiation of the accident, the VVER1000 reactor was assumed to be operating at steady state condition with full thermal power. The steady state was simulated in MELCOR code and the key parameters are compared to design values in Table I. The steady state parameters obtained from MELCOR simulation were mostly in oscillating limitation of design values, except the value of maximum coolant temperature at reactor inlet was little higher. Although, there were not oscillating limitations for mass flow rate through reactor core and steam mass flow rate at steam generator outlet, the values obtained from MELCOR were close to these design values. Generally, the steady state conditions obtained from MELCOR are in good agreement with the design values.

The total Station Black-Out accident was initiated at zero second due to loss of offsite and onsite power including diesel generator and batteries except batteries for BRU-A valves of SGs. The total SBO caused failure of all active safety systems including Emergency Feed Water systems. The additional assumptions of the scenario were made as follows:

- Pressure of PRZ was controlled by relief valves and safety valves with characteristics in Table II;
- There was not taken into account main coolant pumps seal leakages;
- Pressure of SGs was controlled by BRU-A valves to maintain pressure in SGs below 6.7MPa;
- Four Hydro-ACCumulators (ACCs) were available;
- The depressurization for reactor vessel by opening safety valves on top of PRZ and cavity flooding strategy were triggered when temperature of steam in reactor core exceeded 650ºC based on Accident Management Measures (AMM) [14].

**Table I.** Comparison of parameters at steady state condition

| Parameters | MELCOR value | Design value |
|---|---|---|
| Core power (MW) | 3100 | 3000+210 |
| Primary pressure (MPa) | 15.6 | 15.7±0.3 |
| Maximum coolant temperature at reactor inlet (K) | 567 | 559.15±2.0 |
| Average coolant temperature at reactor outlet (K) | 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 II.** The characteristics of the pressurizer relief and safety valves

| Name | Characteristics | Design value | MELCOR value |
|---|---|---|---|
| Relief valves | Opening pressure (MPa) | 16.00 | 16.00 |
| | Closing pressure (MPa) | 15.70 | 15.70 |
| Safety valves Stage 1 | Opening pressure (MPa) | 18.11 | 18.11 |
| | Closing pressure (MPa) | 16.67 | 16.67 |
| Safety valves Stage 2, 3 | Opening pressure (MPa) | 18.60 | 18.60 |
| | Closing pressure (MPa) | 17.07 | 17.07 |

## III. RESULTS AND DISCUSSION

### A. The base case without ERVC

The SBO accident was initiated at 0 second when the VVER1000 reactor was operating at full power capacity. Because of total loss of onsite and offsite power systems, the active safety systems were entirely neutralized. The passive safety system, which consists of four hydro-accumulators (HACCs), was available and only initiated when pressure in primary loop decreased to below 5.8 MPa. The main events of the accident are presented in Table III.

The pressure of primary loop sharply dropped because of a sudden reduction of thermal power to decay heat power (Fig. 4). At early phase, the decay heat was removed via steam generators through dump valves. Afterward, the steam generators nearly dried out at 1000 seconds (Fig. 5), the pressure of primary loop drastically increased (Fig. 4). The primary pressure was controlled by relief and safety valves with the opening and closing set point listed in Table II. Therefore, the primary pressure was maintained at safety threshold below 20 MPa (Fig. 4).

Since the decay heat kept on generating, together with leakage through the opening of relief valves, the water volume inside reactor vessel constantly decreased. Fig. 6 displaying volume of water in reactor core and lower plenum points out the uncovery of reactor core started at 2300 seconds and the reactor core was totally uncovered at 4100 seconds, the lower plenum even ran out of water before reactor core at 3800 seconds.

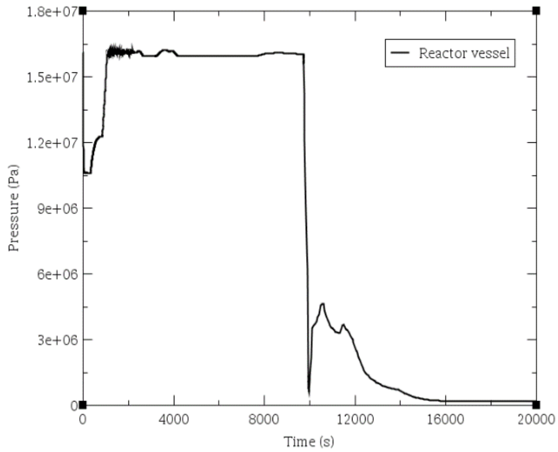



**Fig. 4.** Pressure of primary loop

**Table III.** Main events of in-vessel accident progression

| **Main events** | **Time** (second) |
|---|---|
| Initiation of accident | 0.0 |
| Reactor tripped | 1.6 |
| Begin of core uncovery | 2300 |
| Total core uncovery | 4100 |
| Initiation of primary depressurization | **9727** |
| Begin of water injection from HACCs | 9910 |
| Stop of HACCs injection | 9990 |
| Start of oxidation | 10000 |
| Start of fuel cladding failure | 10300 |
| Failure of lower core plate | 10846 |
| Failure of lower head vessel (CREEP-RUPTURE) | **21938s** |

The temperature of vapor in reactor core over 650$^{o}$C was predicted at 9727 seconds which triggered the opening of relief and safety valves on the top of PRZ to depressurize primary loop. The depressurization created the sharp drop of pressure in the primary loop at 9727 seconds (Fig. 4), and the pressure dropped below 5.8 MPa at 9910 seconds which initiated the water injection from four hydro-accumulators. The water supply from HACCs was stopped when their water volumes reduced below 6 $m^3$ in order to prevent noble gases from being transported into reactor vessel. The result showed the water volume in HACCs dropped below 6 $m^3$ at 9990 seconds (Fig. 7), hence the water supply from HACCs only lasted 80 seconds. The additional water from four HACCs only helped the reactor core recovering in a very short time (Fig. 6). Afterward, the volume of water in reactor vessel quickly decreased and the reactor core was uncovery again at 12300 seconds (Fig. 6). And due to the evaporation of additional water, the pressure of primary loop slightly increased (Fig. 4).

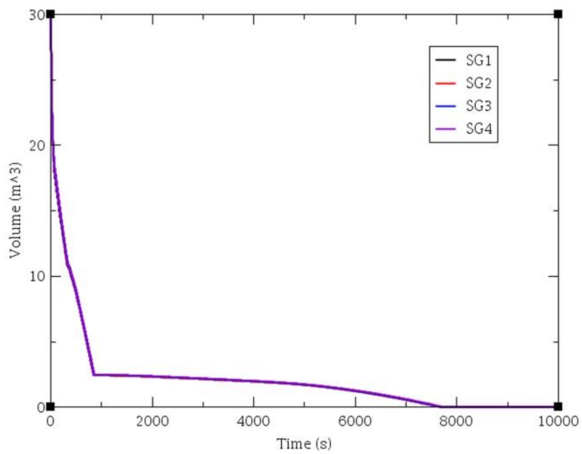



**Fig. 5.** Water volume of SGs

Before the initiation of depressurization, the high pressure condition in primary loop did not provide good condition for structural oxidations. Only after the depressurization and the water injection from HACCs brought a beneficial condition for structural oxidations, and the oxidations were predicted to happen at 10000 seconds. Fig. 8 presents mass of hydrogen which was generated from oxidations of steel and zircaloy, and total mass of generated hydrogen was 372 kg.

The occurrence of oxidations boosted the reactor core heating. The temperature of fuel claddings in ring 1 are displayed in Fig. 9 which shows the temperature of fuel claddings from level 7 to 11 rapidly increased right after the happening of oxidations, even at this time the reactor core was submerged by additional water. The failure of fuel claddings occurred at 10300 seconds.

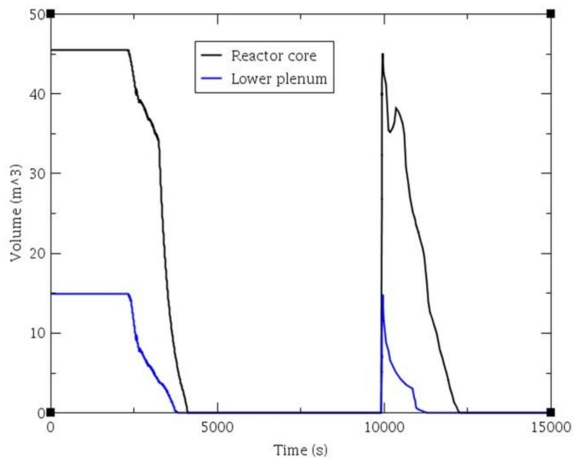



**Fig. 6.** Volume of water in reactor core and lower plenum

The failure of fuel claddings marked the collapse of reactor core and created debris containing heat source. The core debris relocated to the bottom of the reactor core area and imposed thermal load on lower core support plate. The simulation showed the failure of lower core support plate occurred at 10846 seconds, only 846 seconds (14 minutes) and 546 seconds (9 minutes) after the depressurization and failure of fuel cladding respectively. The failure of lower core plate triggered massive relocation of core debris to lower plenum. Fig. 10 presents evolution of mass of core debris in lower plenum.

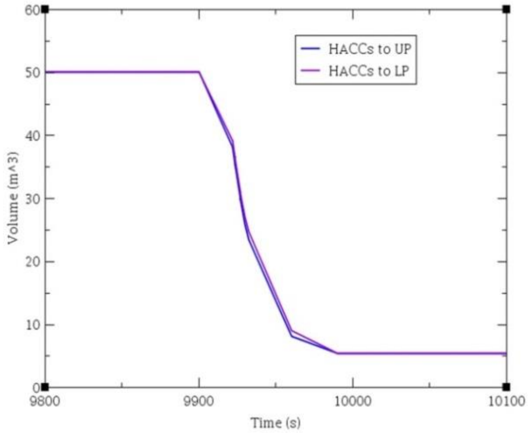



**Fig. 7.** Volume water of HACCs

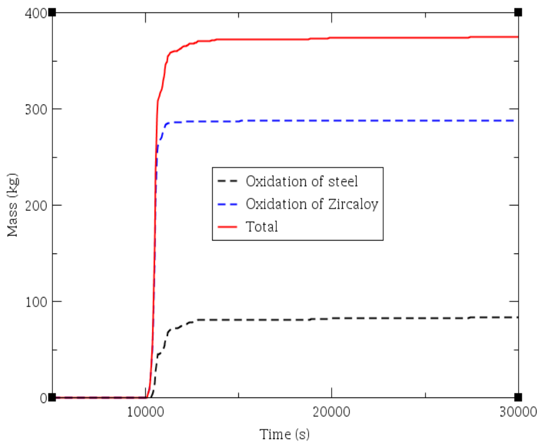



**Fig. 8.** Mass of generated hydrogen

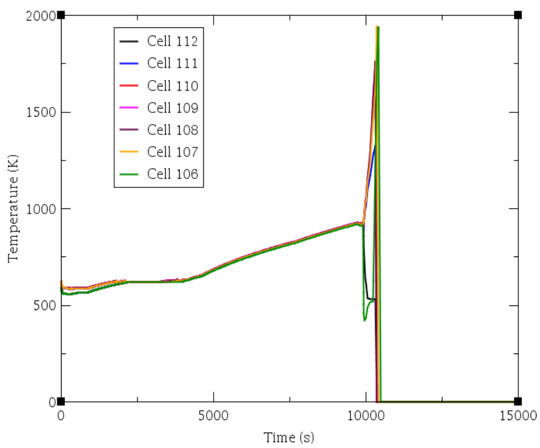



**Fig. 9.** Temperature of fuel claddings in ring 1

At 10864 seconds, the time of core debris relocation to lower plenum, water volume in lower plenum was about 4 m$^3$ (Fig. 6). Under the massive amount of hot debris, the remaining water in lower plenum was quickly boiled off, and the area ran out of water again at 11000 seconds.

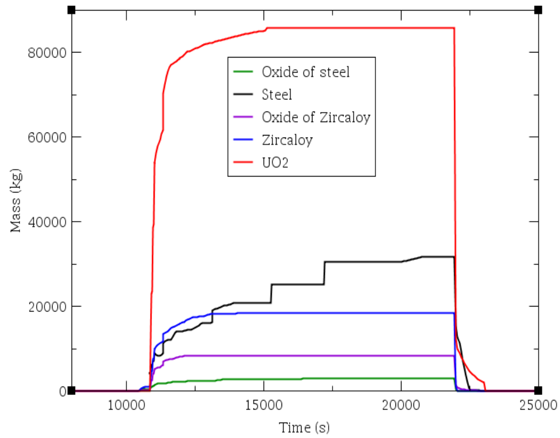



**Fig. 10.** Mass of debris components in lower plenum

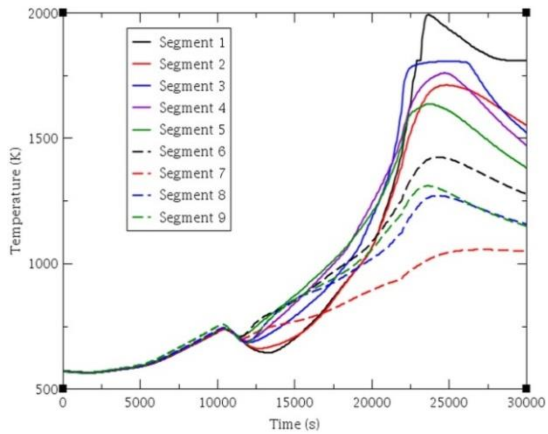



**Fig. 11.** The average temperature of 9 segments

Because there were none of any cooling measures from internal and external reactor vessel, the decay heat generated in debris directly imposed to lower head vessel. Fig. 11 demonstrating the average temperatures of 9 segments shows the average temperatures of the segments slightly decreased after the depressurization, they, however, started to increase right after appearance of core debris in lower plenum. Generally, the first five segments were imposed thermal load more than others which led to higher evolution of their temperatures compared to others. Since 20000 seconds, temperatures of the first five segments were over 1000°K which initiated the thermal strain of the lower head vessel at the segments. The evolutions of thermal strain fraction at 9 segments are displayed in Fig. 12. Among the first five segments, the 3rd segment seemed to be imposed the highest heat hence its temperature was the strongest increase. Due to experience of the highest thermal load, the evolution of elastic strain of the 3rd segment was the sharpest increase and reached to the value of 0.18 at 21938 seconds (6 hours) since initiation of the accident, marked the failure of lower head vessel.

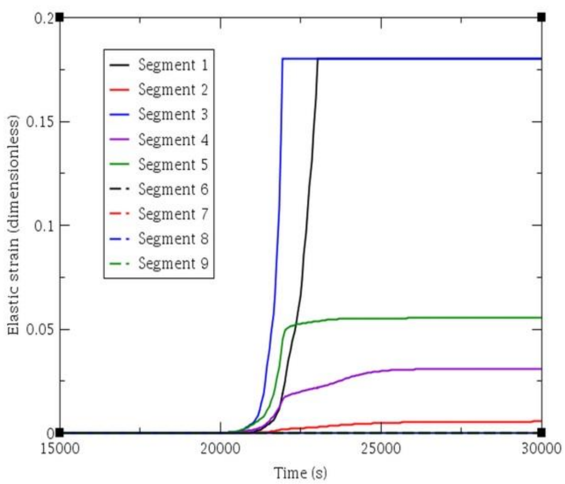



**Fig. 12.** Elastic strain of 9 segments

**Table IV.** Mass of debris in lower plenum

| **Debris** | Mass (kg) |
|---|---|
| UO2 | 80600 |
| Zr | 18150 |
| ZrO2 | 8800 |
| Steel | 31000 |
| Oxide of steel | 2750 |

Fig. 10 indicates the relocation of core fuels started from 10864 seconds and stopped at 15000 seconds, and together with other components such as $ZrO_2$, Zr and oxides of steel was stable after 15000 seconds. Meanwhile, the evolution of steel debris kept on going due to the collapse of supporting structure in lower plenum and stabilized after 20000 seconds. The results showed all core fuels relocated to lower plenum and according to time relocation of core fuels the decay heat power deposited in debris bed with elimination of decay power of volatile gases varied from 32 MW to 29 MW. At the time of lower head failure, also end of simulation time, total mass of debris components, predicted by MELCOR v1.8.6, are listed in Table IV.

A code-to-code comparison for the main events of in-vessel accident progress between MELCOR code in this work and ASTEC in [14] was presented in Table V. The comparison shows the considerable differences of results obtained in both codes. The prediction of ASTEC on relocation of core melt to lower plenum was at 9705 seconds which is 1141 seconds earlier than MELCOR prediction which was at 10846 seconds. As a result, the failure of lower head vessel predicted by ASTEC was at 20886 seconds which is 1052 seconds earlier that MELCOR prediction which was at 21938 seconds. However, MELCOR results predicted a larger mass of corium in lower plenum than ASTEC results which were 131 tons and 73.3 tons respectively.

The differences in the time of main events were caused by considerable differences in calculation models in two codes. Besides the reactor trip assumption of ASTEC also contributed to differences in predictions when it was assumed to happen at the same time with initiation of SBO accident which is unrealistic.

In addition, the difference in model of core debris relocation to lower plenum caused a significant difference in prediction of mass of $UO_2$ relocating to lower plenum. In MELCOR code, the relocation of core debris was triggered when the lower core support plate was failed by both mechanical and thermal load, and when the lower core support plate in a ring failed and lost capability of supporting, its entire structure and all material resting on it would totally collapse and relocate to lower plenum. Meanwhile, ASTEC code [16] did not predict the mechanical failure of the lower core support plate but only its thermal failure, and the core debris only relocated to lower plenum by melting the plate and crossing through the plate hole to lower plenum. In this case, results of MELCOR simulation showed the lower core support plate in five rings completely collapsed, therefore all core fuel relocated to lower plenum. In case of ASTEC, the mass of core debris relocating to lower plenum depended on the size of the hole in the plate hence somehow the size was not large enough for all core fuel relocating to lower plenum in this scenario.

**Table V.** MELCOR and ASTEC main events

| **Main events** | **Time** (second) | |
|---|---|---|
| | **MELCOR** | **ASTEC** |
| Initiation of accident | 0.0 | 0.0 |
| Reactor tripped | 1.6 | 0.0 |
| Opening of steam-dump to atmosphere valves (BRU- A) | 8.0 | 45.0 |
| Begin of core uncovery | 2300 | - |
| Heating up of the core | 4000 | 8767 |
| Total core uncovery | 4100 | - |
| Steam dry-out | 8000 | 3800.0 |
| Closing of BRU-A valves | 8000 | 7200 |
| Initiation of primary depressurization | **9727** | **10341** |
| Begin of water injection from HACCs | 9910 | 10514 |
| Stop of HACCs injection | 9990 | 17545 |
| First material slump in lower plenum | 10846 | 9705 |
| Failure of lower head vessel (CREEP-RUPTURE) | **21938** | **20886** |
| Total mass of corium in lower head (tons) | **131** | **73.33** |

### B. Deployment of IVR strategy with ERVC

In this study, the external vessel cooling strategy was adopted in order to cool down the hot debris from external surface of the VVER1000 lower head vessel by injecting water into the cavity. The water injection was initiated when the vapor temperature in reactor core exceeded 650°C. The water level in reactor cavity was maintained at the height of cold leg. Until now, there has not been official design of ERVC for VVER1000 reactor. Therefore, the study proposed a simplified scheme of ERVC strategy for VVER1000 reactor which included a cooling channel formed by reactor vessel and a frame structure, and an unlimited water resource. Fig. 13 demonstrated a nodalizational scheme of the ERVC in MELCOR code.

The SBO accident was re-simulated with deployment of ERVC strategy. The results showed the performance of ERVC strategy did not affect the in-core accident progress. The evolution of the accident remained the same as the scenario without implementing ERVC strategy. A comparison of main events between two SBO scenarios with and without deploying IVR strategy is presented in Table VI.

The initiation of ERVC strategy was simultaneously happened at 9727 seconds with the primary depressurization when vapor temperature in reactor core exceeded 650°C. Fig. 14 demonstrated water level in reactor cavity and the water level reached the height of cold leg at 12500 seconds, then it was maintained at the level. Figs. 15-17 displayed the temperature of 9 segments of VVER1000 lower head wall in two both scenarios. The solid lines and the dash lines present the results of the scenario without ERVC strategy and with ERVC strategy respectively. The figures indicate the decrease of temperature of lower head walls in both cases happened right after the initiation of depressurization at 9727 seconds. However, in case of implementing ERVC (dash line) the temperature of lower head at 9 segments decreased deeper.

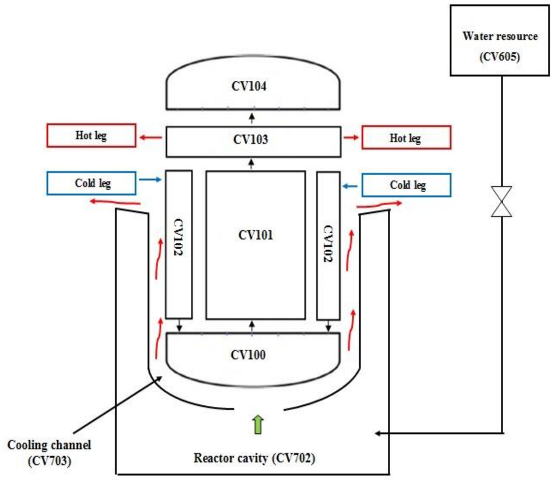



**Fig. 13.** A nodalization for external reactor vessel cooling strategy

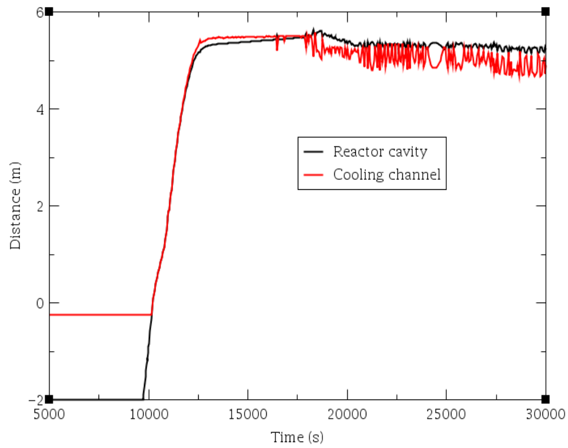



**Fig. 14.** Water level in cooling channel and cavity

**Table VI.** Main events of in-vessel accident progression

| **Main events** | **Time** (seconds) | |
|---|---|---|
| | **With IVR** | **Without IVR** |
| Initiation of accident | 0.0 | 0.0 |
| Reactor tripped | 1.6 | 1.6 |
| Begin of core uncovery | 2300 | 2300 |
| Total core uncovery | 4100 | 4100 |
| Initiation of primary depressurization | **9727** | **9727** |
| Start of cavity flooding | **9727** | **9727** |
| Begin of water injection from HACCs | 9910 | 9910 |
| Stop of HACCs injection | 9990 | 9990 |
| Start of oxidation | 10000 | 10000 |
| Start of fuel cladding failure | 10300 | 10300 |
| Failure of lower core plate | 10846 | 10846 |
| Failure of lower head vessel (CREEP-RUPTURE) | **27341** | **21938** |

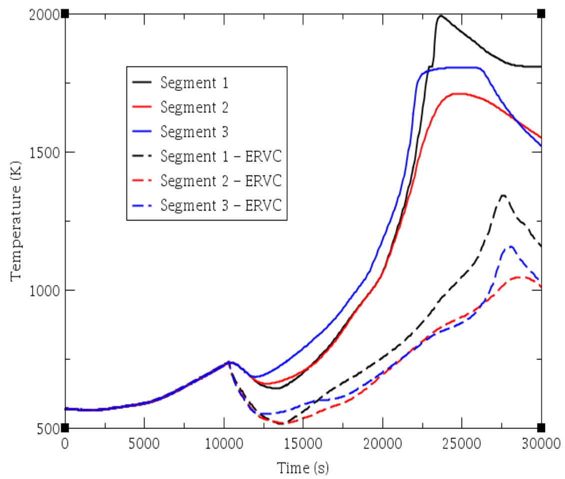



**Fig. 15.** Temperature of segments 1-3

Fig. 15 and Fig. 16 indicate the temperature at segments 1-6 started to increase at 15000 seconds, and it strongly increased at 17000 seconds. Meanwhile, Fig. 17 shows the temperature of lower head wall at segments 7-9 was efficiently cooled and kept at low temperature below 600ºC. Together with the increase of temperature of segments 1-6, the heat transfer between lower head vessel and water drastically increased as well, which caused the vibration of water level in cooling channel (Fig. 14). Fig. 16 shows the temperature of segments 4-6 was kept under 1000ºK (727ºC).

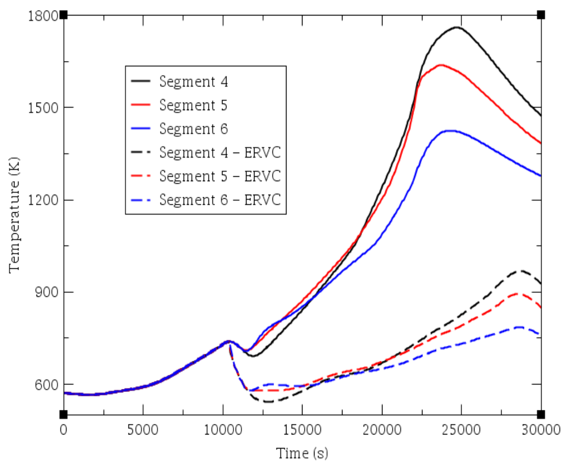



**Fig. 16.** Evolution of temperature of segments 4-6

However, Fig. 15 indicates the temperature of segments 1-3 increased beyond 1000ºK, among them the temperature of segments 1 and 3 were beyond 1000ºK at 24000 and 26000 seconds respectively which caused the occurrence of thermal strain of the segments. Fig. 18 displayed the evolution of thermal strain fraction of all segments. The results show the thermal strain of lower head at segment 1 reached the value of 0.18 at 27341 seconds which marked the failure of VVER1000 lower head reactor vessel.

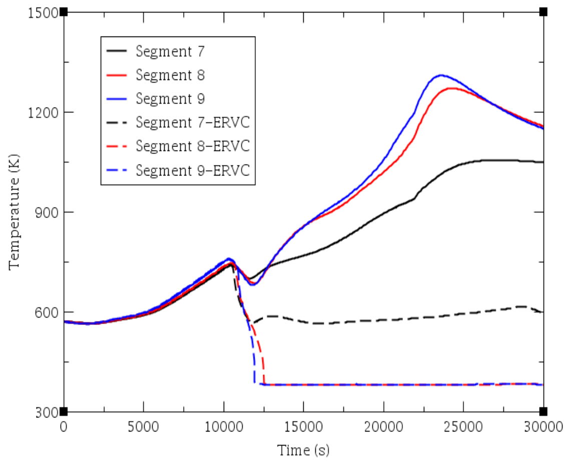



**Fig. 17.** Evolution of temperature of segments 7-9

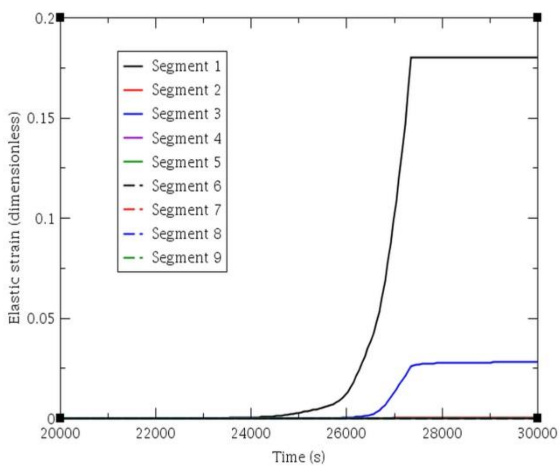



**Fig. 18.** Evolution of thermal strain of segments

## IV. CONCLUDING REMARKS

In this paper, the in-vessel accident progress during the Station Black-Out (SBO) accident combined with externally vessel cooling was analyzed for VVER1000 reactor by using MELCOR v.1.8.6 code. Some conclusions were given as following:

- Under SBO accident, only water from four hydro-accumulators could not prevent the collapse of reactor core and relocation of core debris to lower plenum. The collapse of reactor core marked by failure of fuel cladding was occurred at 10300 seconds (2.86 hours), the appearance of first core debris in lower plenum was at 10846 seconds (3.01 hours), and all core fuels relocated to lower plenum at 15000 seconds (4.2 hours). The decay heat power deposited in debris bed locating in lower plenum was estimated from 32 to 29 MW with taking the elimination of decay power of volatile gases into consideration. All the outcomes would be significant information for further study on thermal behavior of debris bed/molten pool in lower plenum and thermal response of VVER1000 lower head vessel as well;

- In the base case without implementing IVR strategy, the VVER1000 lower head vessel was failed at 21938 seconds (6.09 hours) due to thermal creep-rupture. Meanwhile, in the case of IVR strategy deployment, the results indicated the strategy could not prevent VVER1000 lower head vessel from failure caused by thermal creep-rupture and only prolonged the existence of VVER1000 lower head vessel for 5437 seconds (1.5 hours) compared to the base case;

- The flat shape of VVER1000 lower head vessel also raises a concern about the occurrence of failure at low position when the critical heat fluxes at the low positions were low as seen in results of ULPU [7]. The results of the study also showed the failure of VVER1000 lower head vessel happened at low positions, therefore, it is necessary to optimize geometry of cooling channel for VVER1000 reactor;

- In addition, the comparison on the main phenomena during accident progress between MELCOR and ASTEC suggested it is necessary to take further investigation on in-vessel accident progression in case of SBO accident by other severe accident codes in order to set up a range of key parameters for the bounding configuration of molten pool in VVER1000 lower plenum and specify decay heat power deposited molten pool as well.

## ABBREVIATIONS

| | |
|---|---|
| AMM | Accident Management Measures |
| BRU-A | A kind of atmospheric dump valves of steam generators |
| CC | Cold Collector |
| CL | Cold Leg |
| CP | Cold Part |
| CV | Control Volume |
| ERVC | External Reactor Vessel Cooling |
| HACCs | Hydro-ACCumulators |
| HC | Hot Collector |
| HL | Hot Leg |
| HP | Hot Part |
| IVR | In-Vessel melt Retention |
| LBLOCA | Large Break Lost of Coolant Accident |
| LP | Lower Plenum |
| MSH | Main Steam Header |
| NPP | Nuclear Power Plant |

| | |
|---|---|
| PORV | Pilot-Operated Relief Valve |
| PRZ | PRessuriZer |
| SBO | Station Black-Out |
| SG | Steam Generator |
| SL | Steam Line |
| UP | Upper Plenum |

## ACKNOWLEDGEMENT

The authors are grateful for financial funding of Ministry of Science and Technology and Vietnam Atomic Energy Institute. This work was performed under ministerial project with contract number 05/HĐ/ĐTCB signed on 05/01/2018.

## REFERENCES

[1]. Tuomisto, H., and Theofanous, T.G. "A consistent approach to severe accident management". Nuclear Engineering and Design 148, 171-183, 1994.

[2]. Kymäläinen, O., et al. "In-vessel retention of corium at the Loviisa plant". Nuclear Engineering and Design, 169, 109-130, 1997.

[3]. Theofanous, T.G., et al. "In-vessel cool-ability and retention of a core melt", DOE/ID-10460, Volume 1, Revised October, 1996.

[4]. Esmaili, H., and Khatib-Rahbar, M. "Analysis of in-vessel retention and ex-vessel fuel coolant interaction for AP1000". Energy Research, Inc., ERI/NRC 04-21, NUREG/CR-6849, 2004.

[5]. Rempe, J.L., et al. "In Vessel Retention strategy for higher power reactors". Final Report, INEEL/EXT-04-02561, 2005.

[6]. Ji Xing et al. "HPR1000: Advanced pressurized water reactor with Active and Passive safety". Engineering 2 79-87, 2016.

[7]. Theofanous, T.G., et al. "In-vessel cool-ability and retention of a core melt", DOE/ID-10460, Volume 1, Revised October, 1996.

[8]. Esmaili, H., et al. "An Assessment of Ex-Vessel Steam Explosions in the AP600 Advanced Pressurized Water Reactor", Energy Research, Inc., ERI/NRC 95-211, 1996.

[9]. Bui and Dinh. "Modeling of heat transfer in heated-generating liquid pools by an effective diffusivity-convectivity approach", In: Proceedings of 2nd European Thermal-Sciences Conference, Rome, Italy, pp. 1365–1372, 1992.

[10]. Tran, C.T., and Dinh, T.N. "The effective convectivity model for simulations of melt pool heat transfer in a light water reactor pressure vessel lower head". Part I&II. Progress in Nuclear Energy 51, 849–871, 2009.

[11]. Dombrovskii, L.A., et al. "Numerical Simulation of the Stratified-Corium Temperature Field and Melting of the Reactor Vessel for a Severe Accident in a Nuclear Power Station". Thermal Engineering, Vol. 45, No. 9, pp. 775-765, 1998.

[12]. Yue Jin et al. "In-and ex-vessel coupled analysis of IVR-ERVC phenomenon for large scale PWR". Annals of Nuclear Energy 80 322-337, 2015.

[13]. SANGIORGI Marco et al. "In-Vessel Melt Retention Analysis of a VVER1000 NPP". EUR 27951; doi 10.2790/62596, 2016.

[14]. Polina Tusheva et al. "Analysis of severe accidents in VVER1000 reactors using integral code ASTEC". Proceeding of the 17th International Conference on Nuclear Engineering, Brussels, Belgium, 2009.

[15]. Sandia National Laboratories, "MELCOR computer code manuals", Ver.1.8.6, Rev.3 vol. NUREG/CG 6119, 2005.

[16]. Belon, S., et al. "Insight of core degradation simulation in integral codes throughout ASTEC/MELCOR crosswalk comparisons and ASTEC sensitivities studies". The 8th European Review Meeting on Severe Accident Research, Warsaw, Poland, 2017.