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Evaluation of VVER-1200/V-491 Reactor Pressure Vessel integrity during large break LOCA along with SBO using MELCOR 1.8.6

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Abstract: After Fukushima accident and stress test recommended by IAEA for existing reactors, higher safety requirements are enforced upon nuclear power plants during design extension and severe accident conditions. Based on those arguments, Vietnam Government requests a lot of effective safety solutions, in designs proposed for the nuclear power plants in Ninh Thuan province of Vietnam, which can prevent the accident progression toward severe accidents and mitigate severe accident consequences. One of safety requirements is related to delay time of core melt during design extension condition. Especially, if the worst case of accidents occurs, the reactor vessel integrity must be maintained at least 24 hours from the beginning of the accident. With the aim at investigation of Reactor Pressure Vessel (RPV) integrity, in this study, MELCOR 1.8.6 code is used to evaluate the integrity of RPV lower head for VVER-1200/V-491 reactor during a Large Break Loss of Coolant Accident (LBLOCA) in combination with Station Blackout (SBO) event. The study figures out several parameters related to melt down progress such as: rupture position and rupture timing, the amount of hydrogen generated. Availability of the second stage hydro-accumulators (HA2) in the VVER-1200/V-491 is assumed as an additional improvement to delay the timing of core melt as well as to maintain the vessel integrity for long-term.

Keywords: *VVER-1200/V-491, severe accident, LBLOCA, SBO, RPV, SAM*

I. INTRODUCTION

Integrity evaluation for the RPV lower head during severe accident progress plays an important role in Severe Accident Management (SAM). Therefore, a variety of studies have been carried out to predict a number of characteristic parameters related to core degradation and corium relocation during the accident progress. For example, the lower head rupture timing, the amount of hydrogen generated and radioactive productions released are discussed in many studies with different accident scenarios and reactors. It is observed that, Longze Li *et al* [1] studied vessel rupture in various scenarios of SBO along with failure of steam generator (SG) safety relief valve for Chinese CPR1000 reactor. In addition, the effect of break size spectrum of LOCAs that impacts on vessel rupture timing, amount of hydrogen generated, corium and fission products released for VVER-1000/320 was presented by B. Chatterjee *et al* [2], Palin Groudev *et al* [3] and in other studies.

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Due to the fact that VVER-1200/V-491 reactor is a candidate for the first nuclear power plant (NPP) project in Vietnam, the investigation on failure mechanisms of lower head and evaluation of RPV integrity in severe accident scenarios for the reactor play an important role in assessing reliability and safety of the reactor. Thus, in this study, a severe accident progress and the vessel integrity for VVER-1200/V-491 reactor are investigated by simulation using MELCOR 1.8.6 based on the scenario of simultaneous events: LBLOCA and SBO. The LBLOCA is initiated by cold leg double-ended rupture with break size of 850 mm on the fourth loop in which pressurizer is connected. This break ensures the conservative assumption.

Since SBO occurrence, the active safety injection systems are not available, therefore, only passive safety systems including the first hydro-accumulators and passive heat removal system (PHRS) through SG are available. The study focuses on prediction of the position and timing of vessel rupture for the accident as well as the amount of generated hydrogen inside the reactor vessel. Another case study is investigation based on the assumption in which the second stage hydro-accumulator system (HA2) is available as additional design feature in VVER-1200/V-491 with the discharge flow rate the same as in VVER-1200/V-392M. The objective of this assumption is to evaluate the efficiency of HA2 in maintaining the RPV integrity.

II. MELCOR MODELING FOR VVER-1200/V-491 REACTOR

NPP of VVER-1200/V-491 reactor is designed by the Organization of General Designer “SPb Atomenergoproekt” (St. Petersburg). A lot of safety and economy achievements, which base on evolutionary development and feedback from long-time operation of VVER-1000, are integrated into VVER-1200/V-491 design. The VVER-1200/V-491 reactor reaches nominal thermal power of 3200 MW with four coolant loops.

MELCOR 1.8.6 is a fully integrated system computer code developed by Sandia National Laboratories for the United States Nuclear Regulatory Commission which is capable to simulate severe accident progressions in light water reactor NPPs. A broad spectrum of severe accident phenomena in the nuclear reactors is treated by MELCOR 1.8.6 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.

In this study, the reactor coolant systems are modeled as illustration in Fig. 1 with 152 control volumes, 175 flow paths and pressurizer connected to the hot leg in loop 4. Hydrodynamic materials in one control volume include different phases of coolant: pool (water), vapor, fog and non-condensable gases. Each flow path connects two control volumes. The conservation laws are applied to calculate hydrodynamic changes inside the control volume and between the control volumes.

The core and lower plenum regions of the reactor vessel are divided into 6 concentric radial rings and 12 axial levels as shown in Fig. 2. The first six levels are for the lower plenum region and core support plate, the next five levels are for active core fuel region and the last level is for upper core plate.

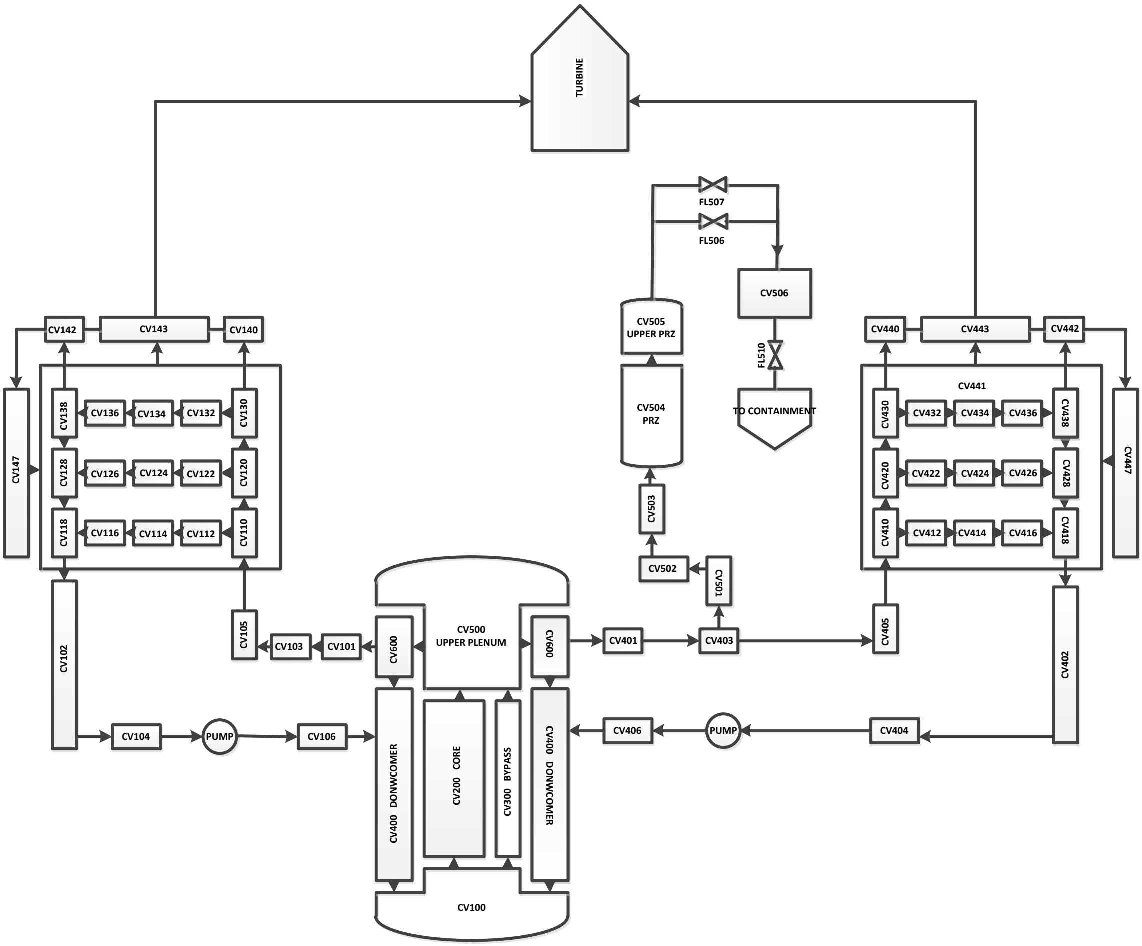
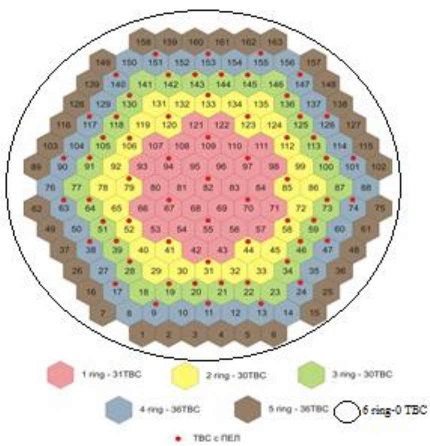


Fig. 1 Nodalization scheme of reactor coolant system for VVER-1200/V-491



	Ring 1	Ring 2	Ring 3	Ring 4	Ring 5	Ring 6	
12	112	212	312	412	512	612	Upper core
11	111	211	311	411	511	611	fuel
10	110	210	310	410	510	610	fuel
9	109	209	309	409	509	609	fuel
8	108	208	308	408	508	608	fuel
7	107	207	307	407	507	607	fuel
6	106	206	306	406	506	606	Lower core plate
5	105	205	305	405	505	605	Lower plenum
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 MELCOR modeling for the core and the lower plenum

The passive safety systems consist of four HA1 tanks, and four PHRS-SGs. The HA1 is modeled as a simple control volume which is connected to the reactor vessel through direct vessel injection line or through downcomer. Water volume of each HA1 tank is 50 m^3 and discharge pressure is 5.9 MPa. Fig. 3 presents the simulation principle for HA1.

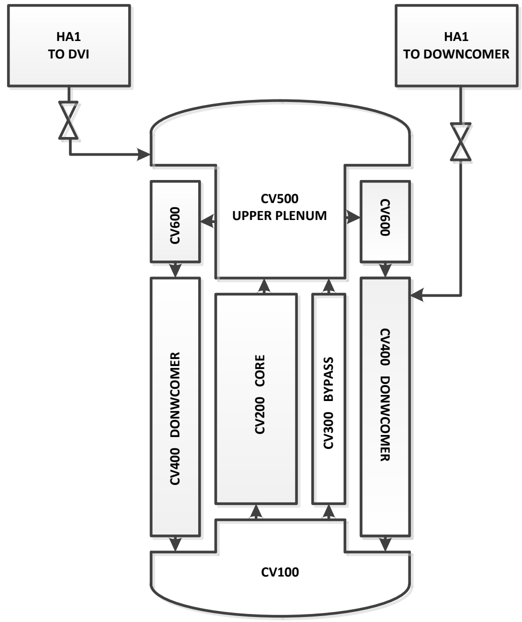


Fig. 3 Modeling of the HA1 systems

Fig. 4 shows the scheme of modeling one PHRS-SG. As in design, the surge line CV151 is steam duct that transports steam from secondary side of steam generator to collectors: CV152 and CV153. The heat exchanger includes primary side that consists of control volumes CV154 and CV155. The tube bundles are modeled by CV154 and CV155 from which heat is transferred to secondary side of heat exchanger that denotes by CV160. The SG PHRS uses air as ultimate heat sink and is described by CV160 properties.

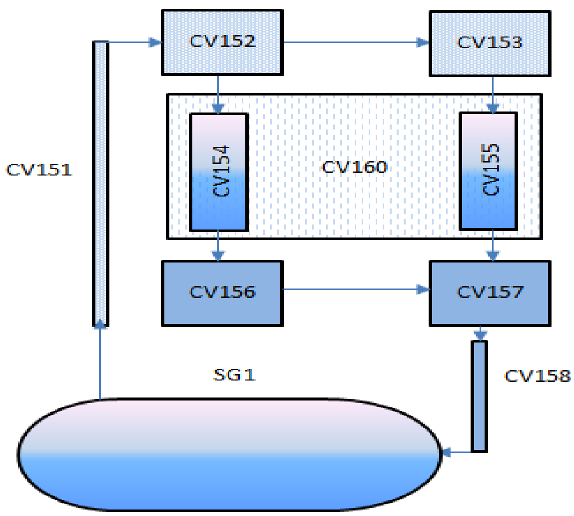


Fig. 4. Modeling of the PHRS-SG for one SG

III. INVESTIGATION ON SIMULTANEOUS OCCURRENCE OF LBLOCA AND SBO

Simultaneous occurrence of LBLOCA and SBO is assumed at 0.0 second with double ended break on cold leg of the fourth loop in which the pressurizer is connected to hot leg. The equivalent diameter of the break is 850 mm. Values of important parameters at initial condition are given Table I.

Table I. Initial condition

Parameters	Design value	Calculated value
Core power, MW_t	3200	3200
Primary pressure, MPa	16.2	16.3
Average coolant temperature at reactor inlet, ^oC	298.2	300.7
Average coolant temperature at reactor outlet, ^oC	328.9	329.3
Pressure of steam at SG outlet, MPa	7.0	6.97
Steam temperature, ^oC	287	285.7
Steam mass flow rate through one SG, kg/s	445.0	460.3

If the accident occurs, the reactor coolant pumps and turbine are tripped at 0.0 s and the reactor is tripped at 1.6 s due to delay time. However, the reactor core still generates a huge amount of decay heat. In the secondary system, main feedwater pumps are tripped at 0.0 s due to SBO. The first-stage accumulators (HA1) system can work to make up the inventory of reactor coolant system, to compensate for the amount lost by the break when the reactor coolant system pressure reaches the opening set-point of 5.9 MPa. The active safety injection systems are unavailable due to SBO. The PHRS-SG is in operation but ineffective due to LBLOCA. It is assumed that BRU-A valves (fast-acting atmospheric relief valves) fail and SG safety valves operate. They are active and passive systems, respectively. Table II shows the chronology of this accident analyzed by MELCOR 1.8.6.

Table II. Sequence of main events

Main events	**Time (s)**	**Comment**
LBLOCA along with SBO	0.0	Initiating event
Main coolant pumps trip	0.0	Due to SBO
Feedwater pumps stop	0.0	Due to SBO
Reactor trip	1.6	Delay time
First-stage accumulators start working	7.0	Reactor Coolant System (RCS) pressure decreases below 5.9 MPa
PHRS-SG starts working	30	Delay time
First-stage accumulators exhausted	186	Water level of one accumulator falls below 1.2 m
Zr oxidation starts	75	
Fuel cladding melting begins	955	Cladding temperature increases to 2500 K
Corium relocation into lower head begins	2486	Core plate support failure
Vessel lower head failure	9726 (2.7 h)	Creep-rupture at segment 5 ring 4
Other parameters		
Hydrogen generation (kg)	311.19	
Total corium mass injection to cavity (ton)	186.311	
Corium mass inventory in the lower plenum after the vessel failure (ton)	69.972	

Due to a large amount of coolant in the RCS discharged into containment through the breaks, reactor core dries out rapidly and RCS pressure drops sharply (Figs. 5 and 6). At 7.0 s, the primary pressure reaches 5.9 MPa, the HA1 system starts injecting coolant into the reactor to recover the core. The injection regime of the HA1 system is shown in Fig. 7. After the HA1 system is exhausted, the reactor core is uncovered again, water level in the core decreases as in Fig. 8 and fuel temperature increases rapidly. At temperature around 1173 K, fuel cladding begins to react chemically with steam, which produces hydrogen and reaction heat. It leads to more rapid temperature increase in the core. At the top of active core, where it is uncovered firstly, fuel elements begin melting at 955 s and the melting expands to other parts (Fig. 9).

At the moment of 2486 seconds, failure of core support plate occurs, corium begins relocating into the lower plenum and forming the melt pool. Temperature of lower head increases due to heat transferred from corium. Then the plastic strain of the lower head occurs and it increases to the failure limit of reactor vessel [4] as shown in Fig. 10. The reactor vessel failure occurs at 9726.0 seconds (2.7 hours) as a result of creep-rupture in lower head. Then, the corium begins injecting into the cavity of the containment through the vessel breach. Fig. 11 illustrates the mass change of materials in the reactor vessel after its failure. Fig. 12 shows total cumulative hydrogen generated in reactor core during the accident progress.

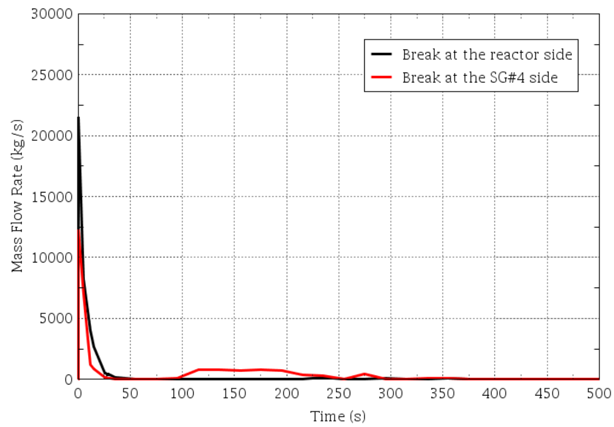


Fig. 5 Coolant flow rate into the containment

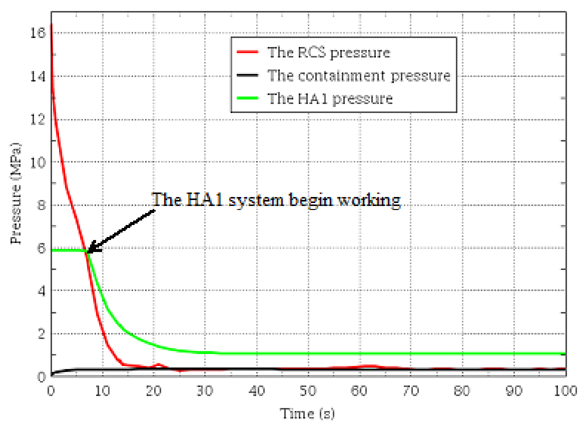


Fig. 6 Pressure change in the RCS, containment and HA1 tank

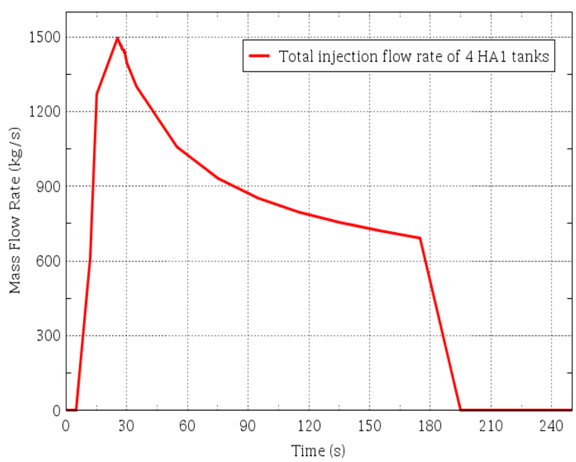


Fig. 7 The injection regime of 4 HA1 tanks

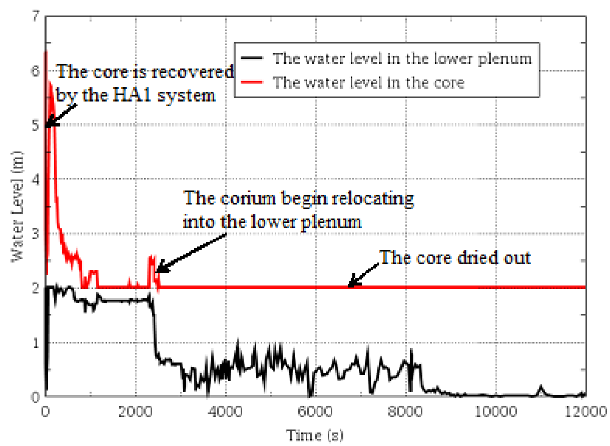


Fig. 8 Water level in reactor core and lower plenum

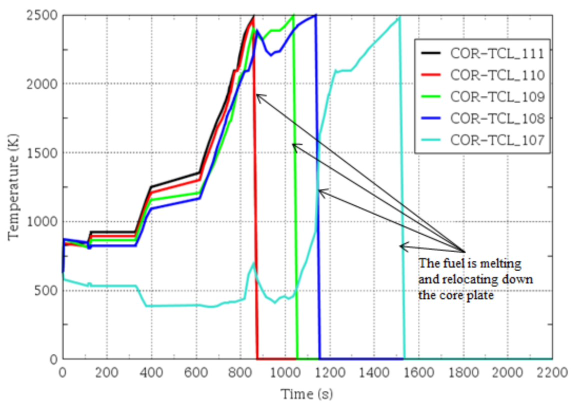


Fig. 9. Cladding temperature change in core cells

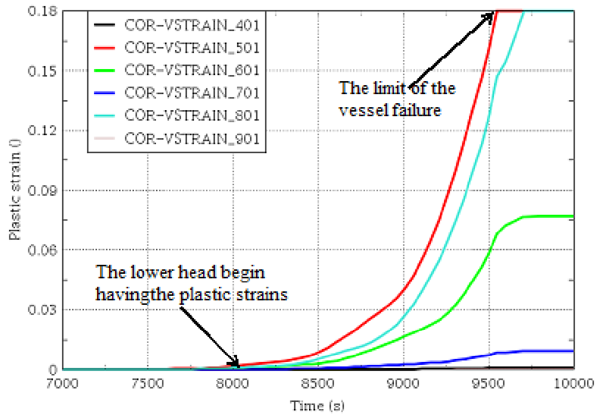


Fig. 10. Plastic strains of the lower head nodes

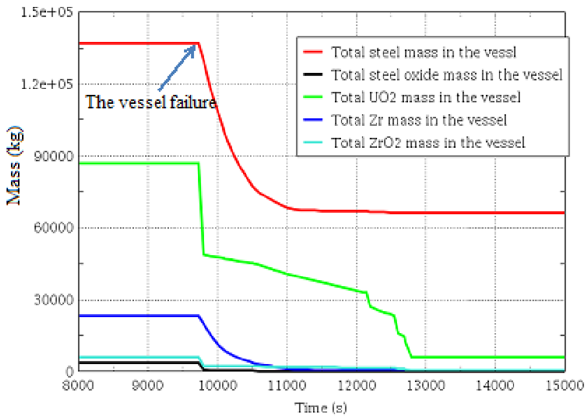


Fig.11. Mass change of some materials in the vessel

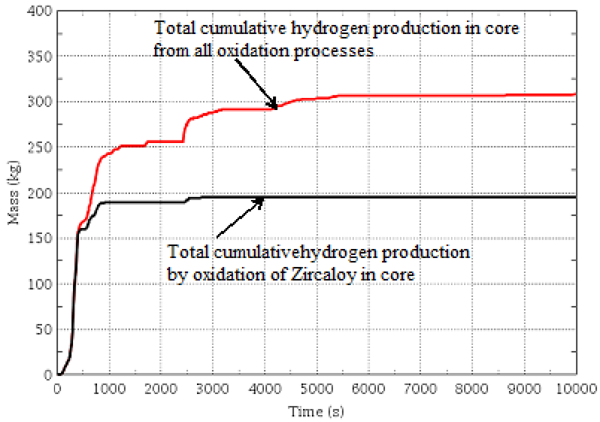


Fig.12. Total mass of hydrogen generated in the vessel

The results from analysis of LBLOCA and SBO in simultaneous occurrence mentioned in this study show that, for the original design of the VVER-1200/V-491, reactor vessel integrity is only maintained for 2.7 hours from the beginning of the accident. Therefore, it can be concluded that safety design of VVER-1200/V491 does not satisfy the requirement for nuclear power plants in Vietnam.

IV. EFFECTIVE ANALYSIS OF HA2 SYSTEM AS NEW ADDITIONAL IMPROVEMENT IN DESIGN OF VVER-1200/V-491 REACTOR

With the aim to delay time of core melting as well as vessel failure for the VVER-1200/V-491 reactor, a second-stage accumulator system (HA2) is assumed as an additional feature in reactor design which is similar with the HA2 system in VVER-1200/V-392M reactor design [5]. The HA2 system includes 8 tanks which inject coolant into the reactor through cold legs and DVI (directly vessel injection) when RCS pressure decreases and reaches 1.5 MPa. The discharge regime of HA2 system is given in Table III.

Table III. The discharge regime for one HA2 tank

Time (s)	100-5430	5431-10860	10861-29000	29001-86400
Mass flow rate (kg/s)	5.0	2.5	1.65	0.8

The same scenario of LBLOCA (with double break on cold leg) and SBO in simultaneous occurrence is studied with purpose to evaluate capability of the HA2 system in cooling the reactor core and maintaining vessel integrity for long-term.

As a result from the analysis, when pressure drops under given set point, HA2 system injects coolant into the core with design mass flow rate. Therefore, core is being cooled for long term with cladding temperature below melting point until HA2 system is exhausted. Thus, the timing of core melting and the RPV rupture are delayed for more than 24 hours. The analysis results of two cases (without or with HA2 system operation) are presented in Table 4. After the HA2 injection finishes, the core dries out quickly again and cladding temperature increases up to melting point that causes the core begins melting at 27.8 hours. The corium relocation into the lower head begins to occur at 29.1 hours.

Table IV. Comparison of the main parameters between results in this study and the ISAR report [6]

Main events	Time (s)		ISAR report
	HA1	HA1&HA2	HA1&HA2
LBLOCA along with SBO	0.0	0.0	0.0
Main coolant pumps trip	0.0	0.0	0.0
Feedwater pumps stop	0.0	0.0	0.0
Reactor trip	1.6	1.6	1.9
First-stage accumulators (HA1) start working	7.0	7.0	5.0
Second-stage accumulators (HA2) start working	No	22.0	23.0
PHRS-SG starts working	30.0	30.0	30.0
First-stage accumulators (HA1) exhausted	186	186	176
Second-stage accumulators (HA2) exhausted	No	86400 (24 hours)	24.5 hours
Fuel cladding melting begins	955	100100 (27.7 hours)	27.5 hours
Corium relocation into lower head begins	2486	1.05304E+05 (29.1 h)	111240 (30.9 hours)
Vessel lower head failure	9726 (2.7 h)	> 30 hours	115200 (32.0 hours)
Calculated termination	5 hours	30 hours	

The calculation results for the design of VVER-1200/V-491 with an assumption of HA2 system operation are also compared with those in ISAR [6] for VVER-1000/V-392 in which HA2 system is employed. As shown Table IV, the calculated results are conformable to those mentioned for VVER-1000/V-392 [6] about the accident progress and the timing of sequent events. However, coolant inventory versus time inside reactor vessel, with the amount about 25 tons from this study, is smaller than that in ISAR [6] as illustrated in Fig. 15. Therefore, fuel temperature in top core cell (green line) is higher than the average fuel temperature mentioned in Ref. [6] (black line), meanwhile, fuel temperature in the bottom core cell (red line), which is covered by water, is lower (Fig. 16) and hydrogen from oxidation reaction is generated earlier, as shown in Fig. 15.

Calculated result for coolant inventory inside the vessel is similar with the results given in Ref. [7], which investigated VVER-1000/V-392 reactor by RELAP5 code. From Ref. [7], it is also observed that with design mass flow rate for HA2, the core is still uncovered and cladding temperature at the average channel increases higher than 1500 K in long term. Thus, both simulation results, calculated by MELCOR and by RELAP5, are similar but different with those presented in ISAR which may be calculated by computer codes from Russia. Therefore, in future, it is needed to compare physical models between MELCOR and Russian code through a benchmark similar to this study.

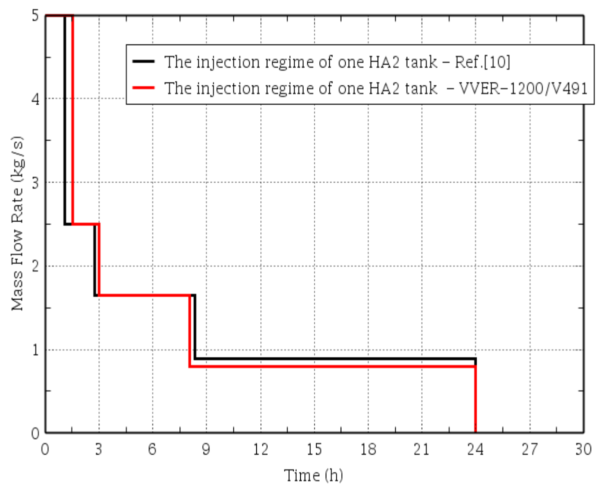


Fig.13. The injection regime of one HA1 tanks

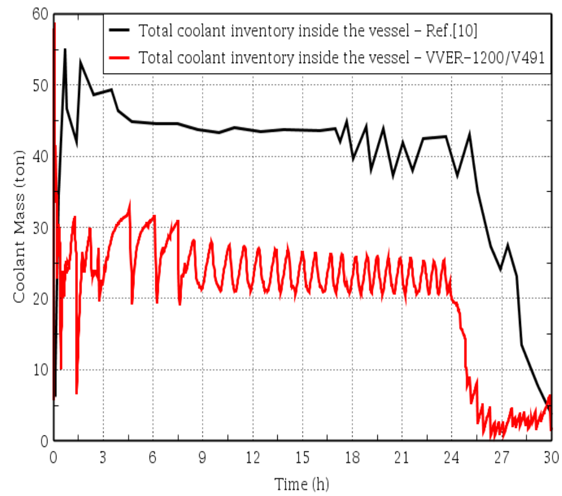


Fig.14. The coolant inventory in the vessel

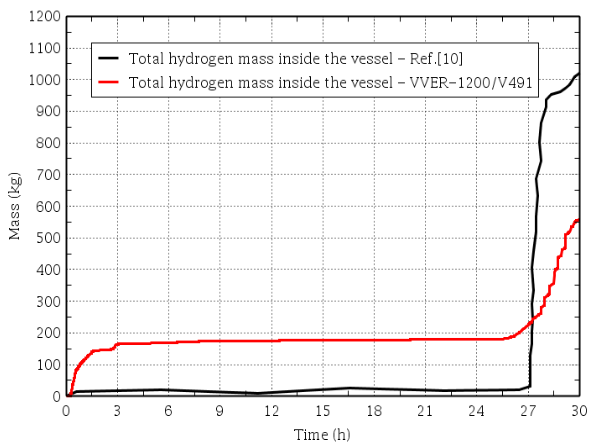


Fig.15. Cumulative hydrogen mass inside the vessel

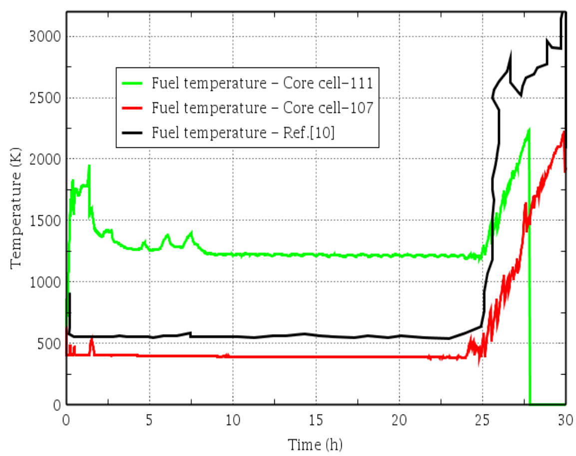


Fig. 16. Fuel temperature change in the core cells

To sum up, the calculation results demonstrate the effect of the additional HA2 system for long-term core cooling and it is shown that the timing of core melting is delayed over 24 hours from the beginning of the accident. Therefore, with effectivesnes of HA2 system as in discussion, it fulfills the safety requirement of maintaining reactor vessel integrity for at least 24 hours during severe accidents for the nuclear power plant (NPP) in Ninh Thuan 1 project in Vietnam.

V. CONCLUDING REMARKS

VVER-1200/V-491 reactor, a candidate for the site Ninh Thuan 1, is investigated in the simultaneous occurrence scenario of LBLOCA and SBO using simulation models of MELCOR 1.8.6. Two cases for that scenario are analyzed: (a) the original reactor design with response from HA1 system and (b) the design modification with an assumption of HA2 system as additional feature.

For the original reactor design, RPV integrity is only maintained in about 2.7 hours. Hence, it does not satisfy safety requirement about maintaining reactor vessel integrity for at least 24 hours during severe accidents for nuclear power plants in Vietnam.

Whereas, because of performance of 2nd stage hydro-accumulators, the timing of core melting is delayed more than 24 hours, the timing of vessel failure extends longer than 30 hours and, obviously, it enhances the possibility of maintaining RPV integrity due to timely availability of additional SAM measures and meets the safety criteria required in Vietnam. The results raise an issue to discuss about the necessity of HA2 system in the VVER-1200/V-491 reactor design to support the SAM and to mitigate severe accident consequences.

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