Nucl. Sci. and Tech., Vol.6, No. 4 (2016), pp. 08-17

Nuclear Science and Technology

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

# Safety Analyses of VVER-1200/V491 reactor for longterm station blackout along with small LOCAs

Bui Thi Hoa[1], Hoang Tan Hung[1], Hoang Minh Giang[1]

[1]*Nuclear Safety Center, Institute for Nuclear Science and Technology*

**Abstract:** Performance of Passive Heat Removal through Steam Generator (PHRS-SG) of VVER-1200/V491 reactor presented in Safety Analysis Report for Ninh Thuan 1 shows that in case of long term station black out (SBO), VVER-1200/V491 reactor can be cooldown and remained in safety mode at least 24 hours based on PHRS-SG performance. Anyway, long term station blackout along with small break in main coolant pipe of VVER-1200/V491 is assumed to be happening as an extension design condition that needs to be investigated. This study focuses on investigation of SBO along with different size of small break of LOCAs with expectation of finding the range of break size that the reactor is still kept in safety mode during 24 hours. During the investigation, some indicators for fuel damage such as the timing of HA1 actuation or mass of coolant inventory discharged are introduced as necessary information contributed to Severe Accident Management Guideline (SAMG).

**Keywords**: *VVER, SBO, RELAP5, SBLOCAs, passive system…*

## I. INTRODUCTION

The advanced VVER's reactors are designed with passive safety systems to deal with design extension conditions such as SBO for core cooling during at least 24 hours. After reactor shutdown, decay heat generation continues and it has the possibility to evolve a severe accident. The Passive Heat Removal System through Steam Generator (PHRS-SG) have to remove this residual heat sufficiently, since active cooling systems need the electric pumps, which are unavailable in case of SBO. The PHRS-SG is used to solve decay heat problem in VVERs. Based on the passive characteristics of the system, the PHRS-SG could remove the residual heat under natural convection conditions. The performance of PHRS-SG for VVER-1200/V491 is verified and validated on a large-scale (1/110) stand [1] at the Scientific-Industrial Association for Research and Design of Power-Generation Equipment (NPO TsKTI, St. Petersburg). The experiments show that the system performs reliably and effectively as designed. These experimental data were used to verify the thermal hydraulic codes KORSAR and SOKRAT, which are used to model the thermal hydraulic processes in the system at the Leningrad nuclear power plant. As illustration of PHRS-SG to deal with design extension conditions, a scenario of SBO for 90000 seconds is presented in SAR for VVER-1200/V491 reactor [2]. However, during long term SBO, leakage from primary system can be occurred due to loss of component cooling system for main coolant pumps and that effect can be considered as small break at pipeline of primary system. Thus, it is presented a practical proposal to investigate the performance of PHRS-SG in case of SBO along with small break in pipe line of primary system. In this study, at the first, to consider the appropriate simulation modelling using RELAP5/MOD3.3, the calculated results of SBO scenario in about

https://doi.org/10.53747/jnst.v6i4.171
Received 19 December 2016, accepted 10 January 2017

24 hours are compared with those in SAR. Then, the performance and capability of PHRS-SG in a spectrum of SBO along with different small break of main coolant pipe (SBLOCAs) is investigated. It is expected that the outcome from the study provides additional analysis to SAR and contribute information to Severe Accident Management Guideline (SAMG).

## II. SIMULATION MODEL FOR RELAP5/MOD3.3

### A. Initial and boundary condition

The general thermal and hydraulic input data accepted for calculation presented in [3] are used to develop simulation model in steady state. For verification of transient simulation, the assumption are followed SBO scenario mentioned in [2] for loss of all alternative current electric power supply sources for 24 hours. As a result of SBO, all Main Coolant Pumps (MCPs) are shut off, turbine generator stop valves closed and Steam Generator (SG) feed water supply is turned off. Once safety systems sections are de-energized and diesel-generations failed to start, due to the occurrence of the input event, PHRS-SG is activated to remove the reactor residual heat and cool down the SGs. The main reactor data presented in [2]are mentioned in Table I. Figure 1 shows the nodalization of primary system with two channel of the core. The average channel consists of 162 fuel assemblies and the hot channel includes one fuel assembly.

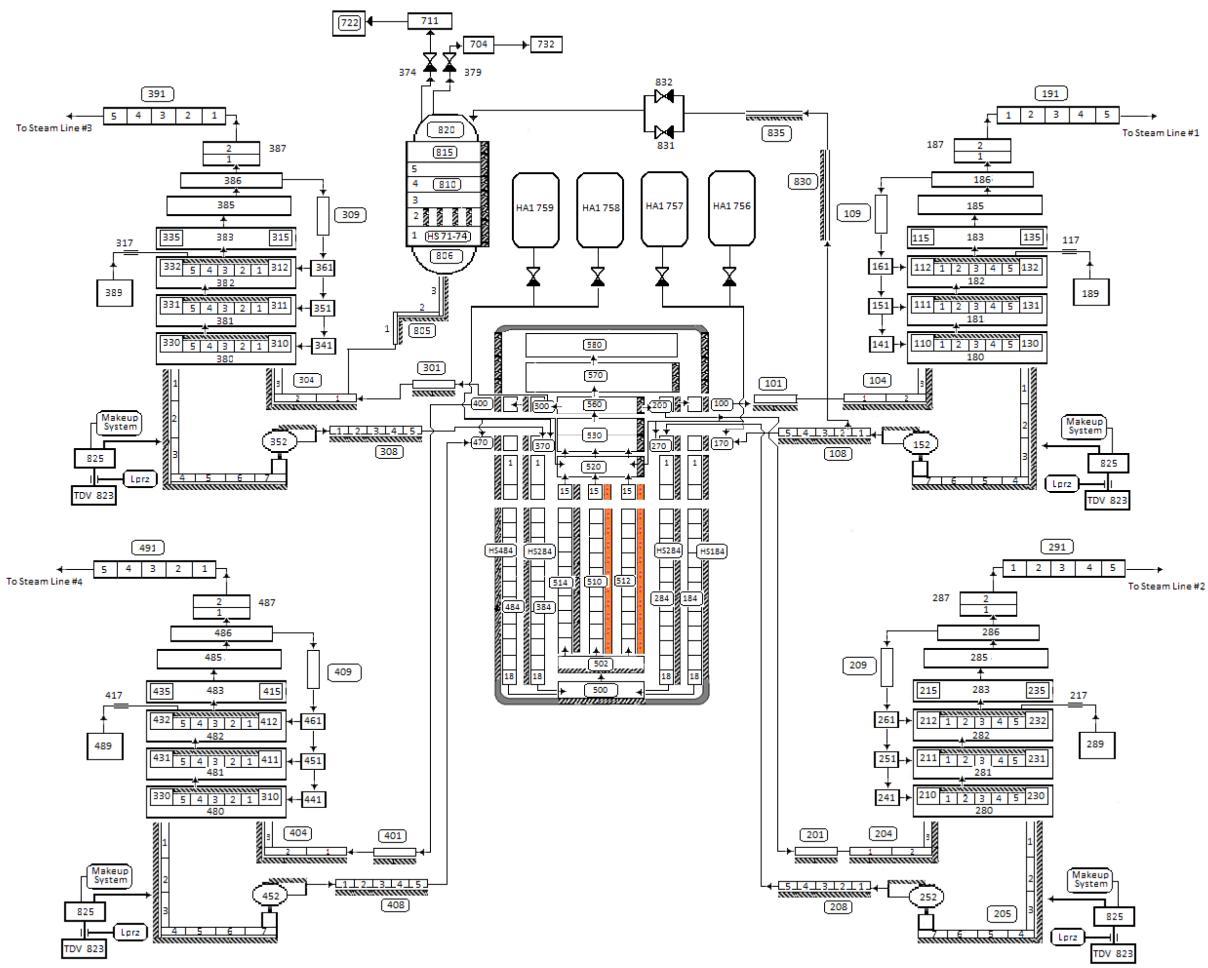


**Fig.1.** Nodalization of primary system of VVER-1200/V491

**Table I.** The main data used to develop simulation using RELAP5

| Parameter Value | Value |
|---|---|
| Reactor thermal power, MWt | 3200 |
| Reactor coolant flow, $m^3/h$ | 88000 |
| Coolant pressure at core outlet, MPa | 16.2 |
| Coolant temperature at reactor inlet, °C | 298.2 |
| Steam pressure in SG steam collector, MPa | 7.0 |
| Feed water temperature, °C | 225.0 |

The thermal and hydraulic input data for PHRS-SG are presented in [4] with main characteristics as following. The system consists of four independent channels, connected to the steam and water volumes of corresponding SG's. Each channel of the passive heat removal system includes the main component as following:

- One emergency heat removal tank (EHRT);
- Sixteen sections of Emergency cooldown heat exchangers (ECDHE) heat exchangers;
- Valves for distribution of flows;
- Steam and condensate pipelines;

The full water capacity of the EHRT for each channel is 600 $m^3$ (at T = 30 °C) at that the effective usable EHRT volume is 535 $m^3$.Heat exchanging sections (ECDHE) are located underwater (H = 5.54 m) in the lower EHRT section. The heat exchanging bundle of every ECDHE section consists of 140 bent tubes 16x2 mm. Heat exchanging bundles are connected by upper inlet and lower outlet manifolds. Distance between manifolds is 1.95 m. Surface of external heat-transferring surface of piping in every ECDHE section is 14.1 m2. Therefore, the total heat transferring surface of every of four PHRS-SG channels is around 239 $m^2$.

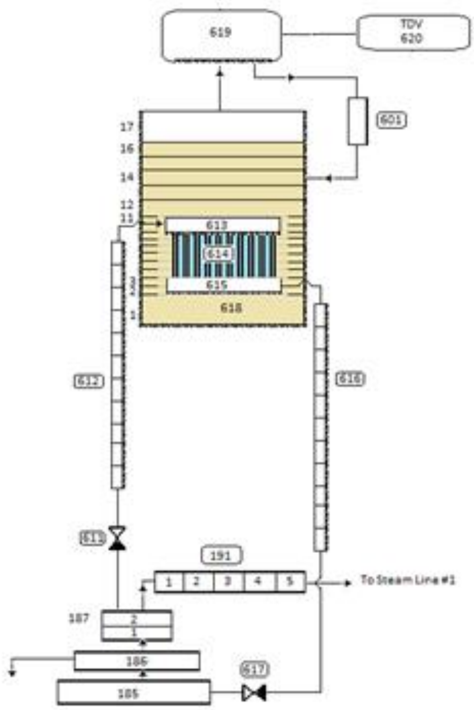


**Fig.2.** Nodalization for one channel of PHRS-SG system

Every PHRS-SG channel is connected to the steam and water volume of a corresponding steam generator. The steam supply pipeline to heat exchanger with diameter 273x20 mm is connected to a special SG steam header nozzle, and the condensate outlet pipeline from the heat exchanger with diameter 108x9 mm is connected to a special nozzle of SG water volume. The “small” distribution valve with the nominal diameter DN50 is connected to the main condensate down-comer pipeline. Maximal capacity of one PHRS-SG channel when opening the “small” valve and water temperature around 30 $^{o}$C of EHRT in operation modes is envisaged by the design is 28 MW. Figure 2 shows nodalization of one channel of PHRS-SG system with steam pipeline connected at SG steam header nozzle and condensate pipeline connected to a special nozzle of SG in upper plenum. As mentioned in [4], in case of SBO, the condensate pipeline connected to SG using “small” distribution valve with the nominal diameter DN50. The EHRT is simulated by a pipe with 17 control volumes in which 16 lower volumes filled with water and the top volume filled with air as assumption of initial conditions. During transient evaporation occurs and top volume of EHRT filled with steam. The evaporation steam then rise up to ambient environment denoted by control volume 619 in which the thermal hydraulics property is affected by boundary condition from environment (control volume 620) and EHRT (control volume 603).

**B. Results comparison between simulation modeling and SAR**

A scenario of SBO presented in [2] is calculated using the study’s simulation modelling to verify the overall of system performance, especially in PHRS-SG’s removing decay heat through steam generator. The sequence events occur as following.

(1) Transient is initiated at 0.0 s because of station blackout;

(2) Main coolant pumps and turbine trip at 0.0 second;

(3) Feed pumps trip at 1.0 second;

(4) Reactor trip at 1.9 seconds

(5) Starting of PHRS-SG triggering at 36.9 seconds as a result of a sequence events: SBO (0,0s), reactor shutdown and start of diesels (1.9s), failure of start diesels within 30 seconds (31.9s) and 5 seconds of delay time to open PHRS-SG valve (36.9s);

(6) Accumulator’s injection starts at RCS pressure falls below 5.9 MPa;

(7) Active safety injection systems are not available due to SBO.

The results for steady state of this study and those calculated in SAR [2] are presented in Table II with thermal power and feed water temperature are 3200 MW and 498 $^{o}$K, respectively.

**Table II.** Comparisons of main parameters for steady state calculation

| Characteristics | Unit | SAR Values [3,5,7] | Calculation Values | Deviation (Percentage) |
|---|---|---|---|---|
| Coolant pressure at the reactor core outlet | MPa | 16.2[(a)] | 16.22 | 0.12% |
| Coolant temperature at the reactor inlet | $^{o}$K | 571[(a)] | 571.06 | 0.01% |
| Coolant temperature at the reactor outlet | $^{o}$K | 601[(a)] | 602.3 | 0.22% |
| Differential pressure across the core | MPa | 0.147[(b)] | 0.152 | 3.40% |
| Average speed of coolant in core | m/s | 5.7[(b)] | 5.71 | 0.18% |
| Coolant flow through the reactor | $m^3$/h | 88000[(a)] | 86168 | 2.08% |
| SGs water level | m | 2.7 ± 0.05[(c)] | 2.52 | 6.67% |
| PRZ level | m | 8.17[(c)] | 8.25 | 0.98% |
| Pressure at SG outlet | MPa | 7.0[(a)] | 7.1 | 1.43% |

*(a) (See Ref. [3]), (b)(See Ref. [5]), (c) (See Ref. [7])*

As mentioned in [6], the sequence of SBO scenario events is listed in Table III along with timing of SAR and the present study. It is observed that at beginning of SBO, all main coolant pumps are trip, reactor is shutdown. Thus, pressure and temperature at core inlet and out let decrease as illustrated in Figure 3, Figure 4 and Figure 5. Due to SBO, all active safety systems are unavailable. Decay heat in this case is removed by PHRS-SG system with maximum power of each channel around 30 MW (Figure 7) at the beginning of accident. The performance of PHRS-SG system is maintained cladding temperature under 350°C up to 90000 seconds (Figure 6).

**Table III.** Timing in SBO scenario

| Event | Time (s) SAR values [6] | Time (s) Calc/Input Values |
|---|---|---|
| SBO | 0.0 | 0.0 |
| MCP tripped | 0.0 | 0.0 |
| Turbine stop valves closed | 0.6 | 0.6 |
| Feed water supply to SGs is completely stopped | 1.0 | 1.0 |
| Reactor tripped | 1.9 | 1.9 |
| Starting of PHRS-SG | 36.9 | 36.9 |
| Starting of boric solution supply from HA1 | 5758.4 | 6680.0 |
| Termination of reading time | 90000 | 90000 |

Figure 3, 4 and 5 show behavior of pressure and temperature in the primary system from SAR and this study. It is observed that at 10000 seconds from beginning, the calculated pressure blowdowns faster than those in SAR while calculated temperature of reactor inlet and outlet decrease a little slowly, but in the duration from 10000 to 90000 seconds, good agreement between calculated pressure and temperature and those in SAR is obtained. Resulting from similar behavior of pressure and temperature in primary system in range (10000 – 90000 seconds), the maximum cladding temperature is similar in both calculated results and those in SAR. Thus, the comparisons of results illustrated from Figure 3 to Figure 6 show that good agreement between the study simulations of system with those mentioned in SAR in the range of 10000 to 90000 seconds. Therefore, the study's simulation modelling is suitable to investigate behavior of VVER-1200/V491in case of SBO along with small break LOCAs in long term duration from more than 10000 to 90000 seconds.

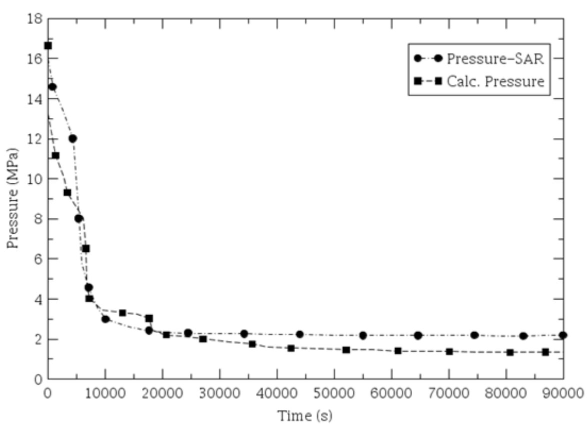


**Fig. 3.** Pressure at core inlet and outlet

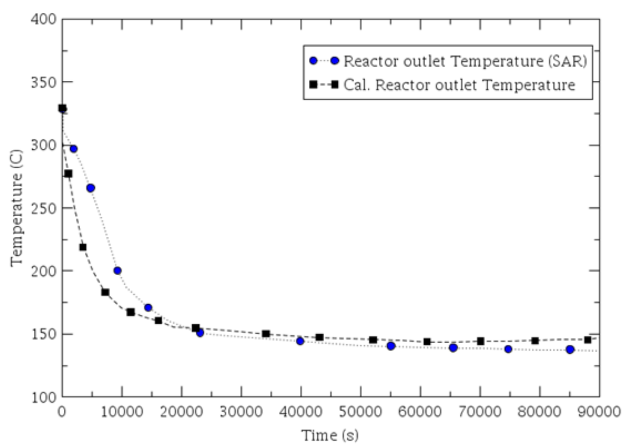


**Fig. 4.** Temperature at core outlet

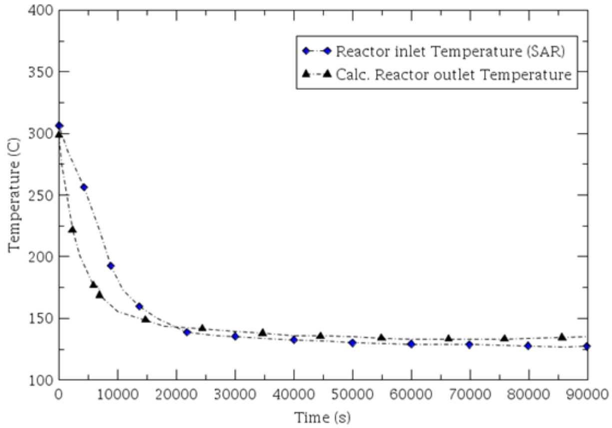


**Fig. 5.** Temperature at core inlet

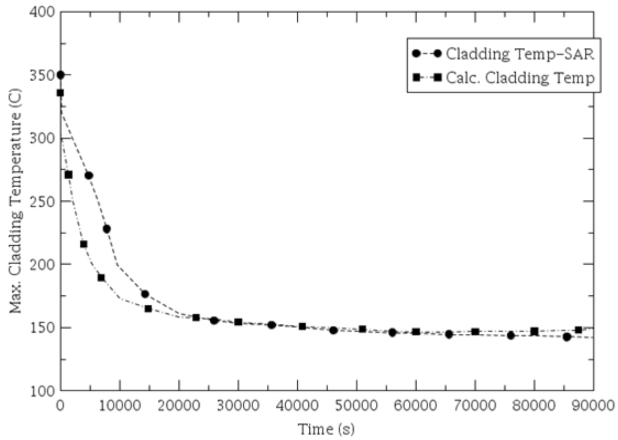


**Fig. 6.** Maximum cladding temperature

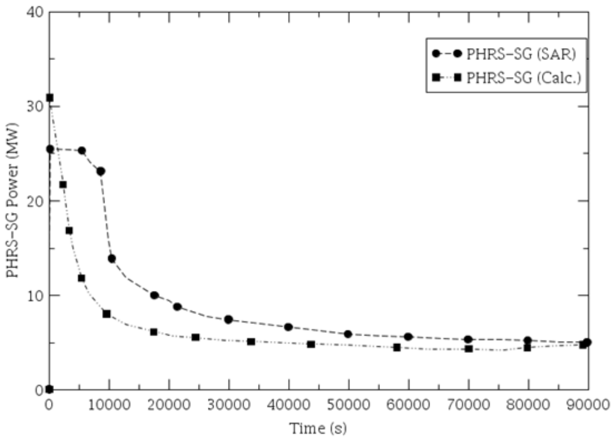


**Fig. 7.** Power of each channel of PHRS-PG system

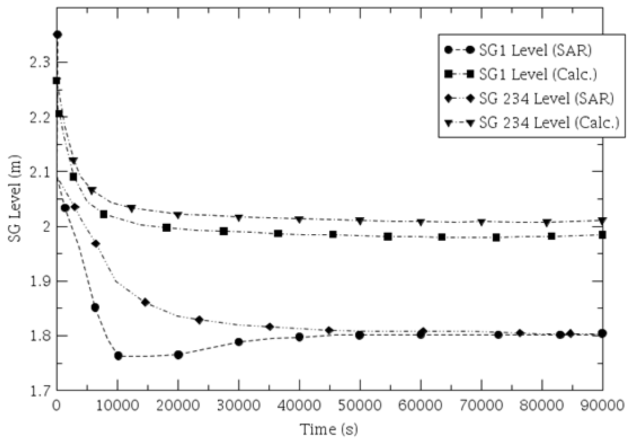


**Fig. 8.** Coolant weighting level in SGs 1-4

## III. RESULTS AND DISCUSSION

### A. Analyses for spectrum of break size along with SBO

It has possibility to assume that if SBO scenario occurs, then leakage from primary system can be happened. For example, the leakage can occur at main coolant pump seals due to loss of component cooling system which operate based on active systems. The results from analysis of SBO scenario without leakage show that performance of PHRS-SG system can maintain safety of core after shutdown until at least 24 hours. However, for stress test of VVER-1200/V491, it is needed to investigate the fuel cladding temperature behavior in case of different break at primary system along with SBO. Thus, a spectrum of beak size with SBO is studied with corresponding break equivalent diameter of 25, 30, 50 and 100 mm. The timing of event sequence of the analysis is given in Table IV.

**Table IV.** Timing of event sequence with different break size

| Event | Break size (mm) | | | | |
|---|---|---|---|---|---|
| | 25 | 30 | 35 | 50 | 100 |
| SBO along with LOCA (s) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |

| | | | | | |
|---|---|---|---|---|---|
| MCP tripped (s) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Turbine stop valves closed (s) | 0.6 | 0.6 | 0.6 | 0.6 | 0.6 |
| Feed water supply to SGsstopped (s) | 1.0 | 1.0 | 1.0 | 1.0 | 1.0 |
| Reactor tripped (s) | 1.9 | 1.9 | 1.9 | 1.9 | 1.9 |
| Starting of PHRS-SG (s) | 36.9 | 36.9 | 36.9 | 36.9 | 36.9 |
| Starting of boric solution supply from HA1 (s) | 1320.0 | 1070 | 910.0 | 670.0 | 399.0 |
| Core melting begins ($T_{cl}$>1500K) | - | - | 80750 | 28150 | 12320 |
| Termination of reading time (s) | 90000 | 90000 | 90000 | 50000 | 15000 |

It is observed that, the different break size causes different blowdown of pressure during the first 5000 seconds as Figure 8. Then the collapsed core water level decrease with different rate in Figure 9 corresponding with pressure blowdown. The total inventories for different break size discharged are illustrated in Figure 10. The fuel damage indicated by maximum cladding temperatures increase over 1500 $^{o}$K are shown in Figure 11. During the accidents, the water levels in ultimate heat sink EHRT decrease as Figure 12.

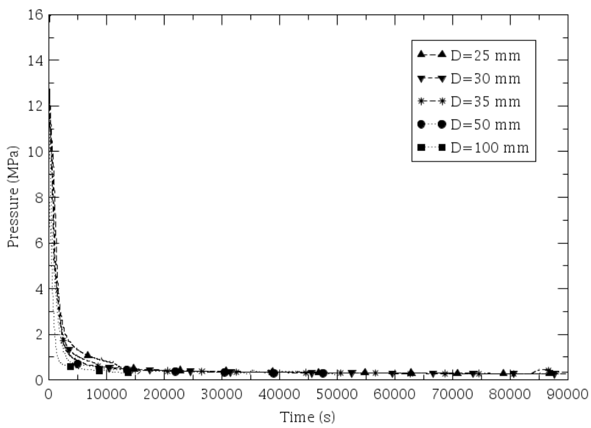


**Fig. 8.** Pressure with different break size

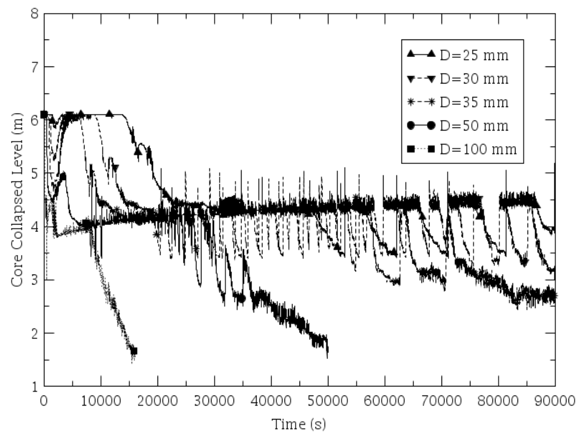


**Fig. 9.** Core Level with different break size

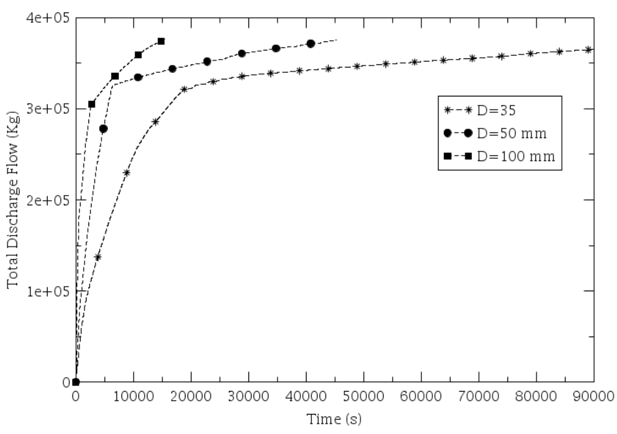


**Fig. 10.** Total inventories discharged

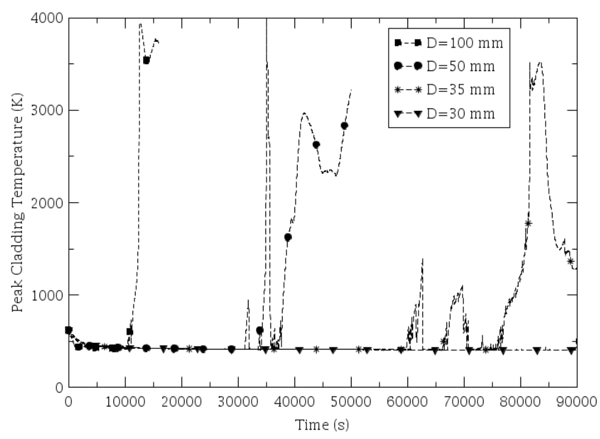


**Fig. 11.** Max cladding temperature with different break size

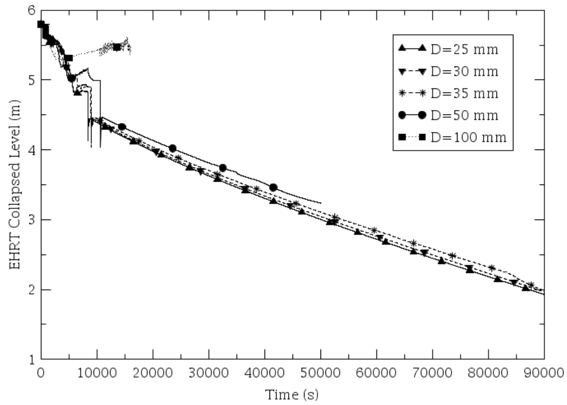


**Fig. 12.** EHRT level with different break size

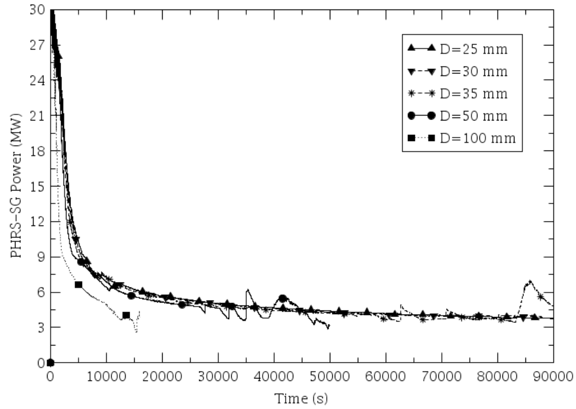


**Fig. 13.** PHRS-SG power with different break size

Figure 13 show the PHRS-G power for each break size and it is observed that the lowest power is happened with largest break size (100mm). Thus, performance of PHRS-SG is not significant effective in this case and fuel damage around 3 hours after accident occurring as illustrated in Figure 11. Similarity, with the break size of 50mm and 35mm the PHRS-SG can delay fuel damage more than 11 hours and 22 hours, accordingly, as illustrated in Figure 11 and Figure 13. For small break size less than 30mm, PHRS-SG performance is satisfied to keep fuel cladding in safety mode at least 24 hours.

Two important parameters related to timing of fuel damaged are observed. Those arethe timing for actuation of emergence core cooling accumulators (HA1) system and the timing for total inventory discharged reach 33000 kg. Figure 14 show the relation between timing of HA1 actuation and fuel damaged. Similarly, Figure 15 give inventories discharged reached 33000 kg versus fuel damaged timing.

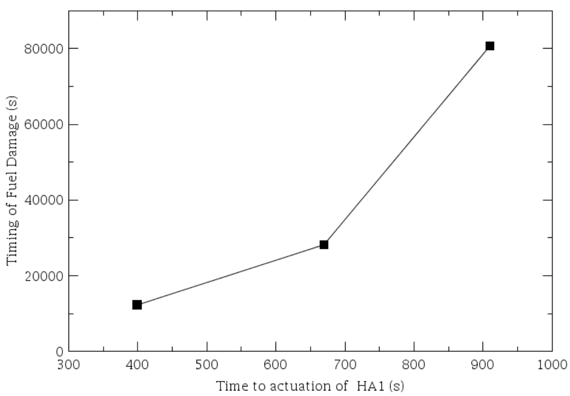


**Fig. 14.** HA1 actuation vs Fuel Damaged timing

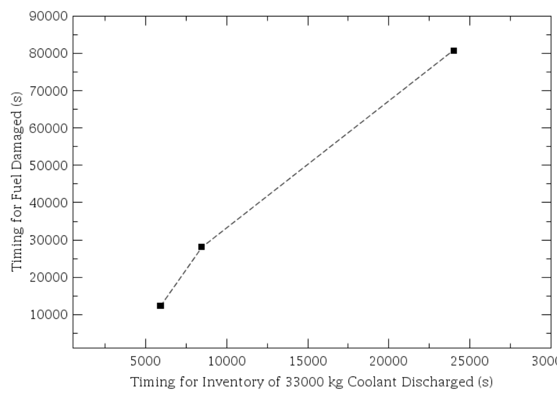


**Fig. 15.** Inventories discharged vs Fuel Damaged timing

**B. Sensitivity study with EHRT temperature**

It is known that the different siting of NPP causes different ambient temperature. The study in section 3.1 assumes that ambient temperature around 30 $^{o}$Cand this value is specified to EHRT water temperature. However, it is needed investigate PHRS-SG with more conservative condition with higher environment temperature. In this study, an additional analysis with ambient temperature and EHRT water temperature of 45 $^{o}$C is also investigated. Table 5 show the timing of fuel damage with different size and ambient temperature. It is previously observed that the higher ambient environment, the sooner fuel damage occurs. The timings that cause fuel damage sooner for different break size of 100, 50and 35mm are around 2, 20 and 200 minutes, accordingly.

**Table V.** Timing of fuel damaged with different ambient temperature

| Event | Break size (mm) | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | 30 | 30(+) | 35 | 35(+) | 50 | 50(+) | 100 | 100(+) |
| SBO along with LOCA (s) | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 | 0.0 |
| Starting of solution supply HA1 (s) | 1070 | 1070 | 910 | 910 | 670 | 670 | 399 | 399 |
| Fuel damaged begins ($T_{cl}$>1500K) (s) | - | - | 80750 | 68630 | 34900 | 33720 | 12320 | 12180.0 |
| End of calculation (s) | 90000 | 90000 | 90000 | 90000 | 50000 | 50000 | 15000 | 15000 |

*(+) notes ambient temperature equal 45 $^o$C*

## IV. CONCLUSIONS

The performance of PHRS-SG system of VVER-1200/V491 is investigated in case of different break size along with SBO. It is concluded that with the break size of equivalent diameter below 30 mm the autonomy of reactor within 24 hours is confirmed. However, with break size larger than 35 mm the fuel damaged will occur during 24 hours with specific timing depended on corresponding break size. The useful information for SAMG is timing of HA1 actuation and total inventories discharged from break.The conservative ambient temperature also causes accident early around 20 to 200 minutes with small break from 50 to 35 mm.Anyway, this study still needs more improvement due to that fact that the results comparison between study simulation modelling and SAR's results show some difference within 10 000 seconds from beginning of accident and it may cause from simulation of PHRS-SG system. In future, the simulation of experiment test for PHRS-SG such as SPOT-PG is needed to improve simulation modelling.

## ACKNOWLEDGMENT

The authors of this paper wish to express their appreciation for the financial support from Ministry of Science and Technology (MOST) through the Ministry R&D Project: "Study on safety of VVER-1200/V491 in LOCA scenarios along with failure of ECCS systems using RELAP5", code DTCB.03/16/VKHKTHN.

## REFERENCES

[1].V.O. Kukhtecvick, V.V.Bezlepkin, S.V.Svetlov, V.G.Sidorov, S.B.Alekseev, V.A.Ilyin and N.V.Lyapin, "Experimental studies of thermohydraulic processes for the Passive heat removal system at the Leningradskaya nuclear power plant", Atomic Energy, Vol.108, No.5, 2010.

[2]. Ninh Thuan Nuclear Power Project Management Board, NINH THUAN 1 NUCLEAR POWER PLANT PROJECT, Feasibility Study, PART 1, Feasibility Study Description, VOLUME 3, Specialized reports, Report 1, Feasibility Study Safety Analysis, Section 7.5 Deterministic analysis Subsection 7.5.7 Consideration of design capability for beyond design basis accidents. NT1.0-3.101-FS-01.03.01.07.05.07.00.

[3]. Ninh Thuan Nuclear Power Project Management Board, NINH THUAN 1 NUCLEAR POWER PLANT PROJECT, Feasibility Study, PART 1, Feasibility Study Description, Vol. 3, Specialized reports, Report 1, Feasibility Study Safety Analysis, Chapter 7 Safety analysis Section 7.5 Deterministic analysis, Subsections 7.5.1, 7.5.2, 7.5.3, 7.5.4, 7.5.5. pp. 13-23.

[4]. Ninh Thuan Nuclear Power Project Management Board, NINH THUAN 1 NUCLEAR POWER PLANT PROJECT, Feasibility Study, PART 1, Feasibility Study

Description, Vol. 3, Specialized reports, Report 1, Feasibility Study Safety Analysis, Section 6.4 Engineered safety features. NT1.0-3.101-FS-01.03.01.06.04-rev02. pp. 94-101.

[5]. Ninh Thuan Nuclear Power Project Management Board, NINH THUAN 1 NUCLEAR POWER PLANT PROJECT, Feasibility Study, PART 1, Feasibility Study Description, Vol. 3, Specialized reports, Report 1, Feasibility Study Safety Analysis, Chapter 6 Description and conformance to the design of plant systems. Section 6.2 Reactor. NT1.0-3.101-FS-01.03.01.06.02-rev02. pp. 123, 22.

[6]. Ninh Thuan Nuclear Power Project Management Board, NINH THUAN 1 NUCLEAR POWER PLANT PROJECT, Feasibility Study, PART 1, Feasibility Study Description, Vol. 3, Specialized reports, Report 1, Feasibility Study Safety Analysis, Chapter 7 Safety analysis Section 7.5 Deterministic analysis, Subsection 7.5.7 Consideration of design capability for beyond design basis accidents. NT1.0-3.101-FS-01.03.01.07.05.07.00. pp. 17.

[7]. Ninh Thuan Nuclear Power Project Management Board, NINH THUAN 1 NUCLEAR POWER PLANT PROJECT, Feasibility Study, PART 1, Feasibility Study Description, Vol. 3, Specialized reports, Report 1, Feasibility Study Safety Analysis, Chapter 7 Safety analysis Section 7.5 Deterministic analysis, Subsection 7.5.6 Anticipated operational occurrences and design basis accidents. pp. 183 – 184.