STUDY ON SAFETY ANALYSIS OF PWR REACTOR CORE IN TRANSIENT AND SEVERE ACCIDENT CONDITIONS

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1. Le Dai Dien, Nguyen Tu Oanh. Analysis of DVI line break in ATLAS test facility using RELAP5 code. J. Nuclear Science and Technology, No. 4, pp.33-41, VINATOM, 2011.
2. Le Dai Dien, Le Tri Dan. Analysis of Steam Generator Tube Rupture Accident for Korean Reactor APR1400. J. Nuclear Science and Technology, Vol. 3, No. 2, pp.7-14, VINATOM, 2013.
3. Le Dai Dien, Bui Thi Hoa, Vo Thi Huong. Application of MELCOR code to Westinghouse 4-loop PWR Severe Accident and Evaluation of RPV Lower Head Performance. . J. Nuclear Science and Technology, Vol. 4, No.2, VINATOM, 2014
4. Tae Woon Kim, Jinho Song, Vo Thi Huong, Dong Ha Kim, Bo Wook Rhee, Shripad Revankar. Sensitivity study on severe accident core melt progression for advanced PWR using MELCOR code. Nuclear Engineering and Design (2013) http:www.elsevier.com/ locate/nucengdes.

ABSTRACT: The cooperation research project on the "Study on Safety Analysis of PWR Reactor Core in Transient and Severe Accident Conditions" between Institute for Nuclear Science and Technology (INST), VINATOM and Korean Atomic Energy Research Institute (KAERI), Korea has been setup to strengthen the capability of researches in nuclear safety not only in mastering the methods and computer codes, but also in qualifying of young researchers in the field of nuclear safety analysis. Through the studies on the using of thermal hydraulics computer codes like RELAP5, COBRA, FLUENT and CFX the thermal hydraulics research group has made progress in the research including problems for safety analysis of APR1400 nuclear reactor, PIRT methodologies and sub-channel analysis. The study of severe accidents has been started by using MELCOR in collaboration with KAERI experts and the training on the fundamental phenomena occurred in postulated severe accident. For Vietnam side, VVER-1000 nuclear reactor is also intensively studied. The design of core catcher, reactor containment and severe accident management are the main tasks concerning VVER technology. The research results are presented in the 9th National Conference on Mechanics, Ha Noi, December 8-9, 2012, the 10th National Conference on Nuclear Science and Technology, Vung Tau, August 14-15, 2013, as well as published in the journal of Nuclear Science and Technology, Vietnam Nuclear Society and other journals. The skills and experience from using computer codes like RELAP5, MELCOR, ANSYS and COBRA in nuclear safety analysis are improved with the nuclear reactors APR1400, Westinghouse 4 loop PWR and especially the VVER-1000 chosen for the specific studies. During cooperation research project, man power and capability of Nuclear Safety center of INST have been strengthen. Three masters were graduated, 2 researchers are engaging in Ph.D course at Hanoi University of Science and Technology and University of Science and Technology, Korea, respectively.

1. INTRODUCTION

Safety analysis is one of the requirements for the construction and operation of NPP. Understanding of physical phenomena - as well as thermal hydraulics computational simulation is an important tool to confirm safety of NPP in the postulated accidents. The cooperation between VINATOM and KAERI in the safety analysis of PWR has been established since 2009 and the first phase has been successful carried out in 2010. In order to strengthen the capability of researchers at INST, VINATOM the second phase of the project (2012-2013) is supported.

The objectives of the project are as follows:

- Enhancement of capability of implementation of computer codes in the safety analysis work for NPP including system code and sub-channel code.

- Training of young researchers in safety analysis in thermal hydraulics as well as in severe accident study.

For the cooperation, the common objectives are the establishments of application system of thermal hydraulic safety analysis code for PWR, including: Evaluation of system TH code like RELAP5, MARS and MELCOR for severe accidents. The other important objective is exchange of human resources between Korea-Vietnam through on the job training (OJT) for Vietnamese code users and lectures for code technology and safety analysis in the fields of thermal hydraulics and severe accident.

With the above mentioned targets, the research topics focus on main studies which has been shown to be effective through research works:

1. To complete basic problems in safety analysis report (SAR) of APR1400 reactor along with the problems had been done in the first phase (2009-2010) to make a complete safety analysis problems for APR1400 reactor.

2. To start studing the severe accident in NPP from fundamental phenomena to some key issues such as skills using MELCOR code, expanding the scope of the research to VVER-1000 reactor, thereby helping staff to participate in the activities that support to Ninh Thuan 1 projects in future.

3. The study results demonstrated by the thematic research activities, simulations using computer codes (RELAP5, MELCOR, COBRA, MARS, ANSYS FLUENT, ANSYS CFX) for some specific problems. There have been some reports in national scientific conferences and the results demonstrated in Master thesis as well as contribution to doctoral thesis under progress.

The contents of the study are shown systematically in Figure 1. including roadmap for research towards building expertise in thermal hydraulics safety analysis and severe accident.

Thermal hydraulics phenomena are important in most of accidents in DBA as well as BDBA. The thermal hydraulic safety concerns with:

- Safety analysis of DBA for evaluating the adequacy of the design to cope with transient and accident conditions

- Safety analysis of BDBA for evaluating if consequences can be considered as acceptable

- Safety analysis of Severe accident and Accident management (AM) development to prevent or mitigate accident consequences

To address these safety concerns, thermal hydraulic codes are studied for simulation of Korean APR1400 reactor. The main problems in SAR have been studied during the years 2009-2010 [1] and continued to study in order to make a full set of safety analysis including LOCA,

LOCA and SBO, REA, FWB, LOFA, SGTR, MSLB. Based on these studies, the group of researchers in thermal hydraulics safety analysis has been set up.

The studies in severe accident have been intensively performed. The training on the basic phenomena in severe accident with the lectures presented by experts from KAERI was held in INST.

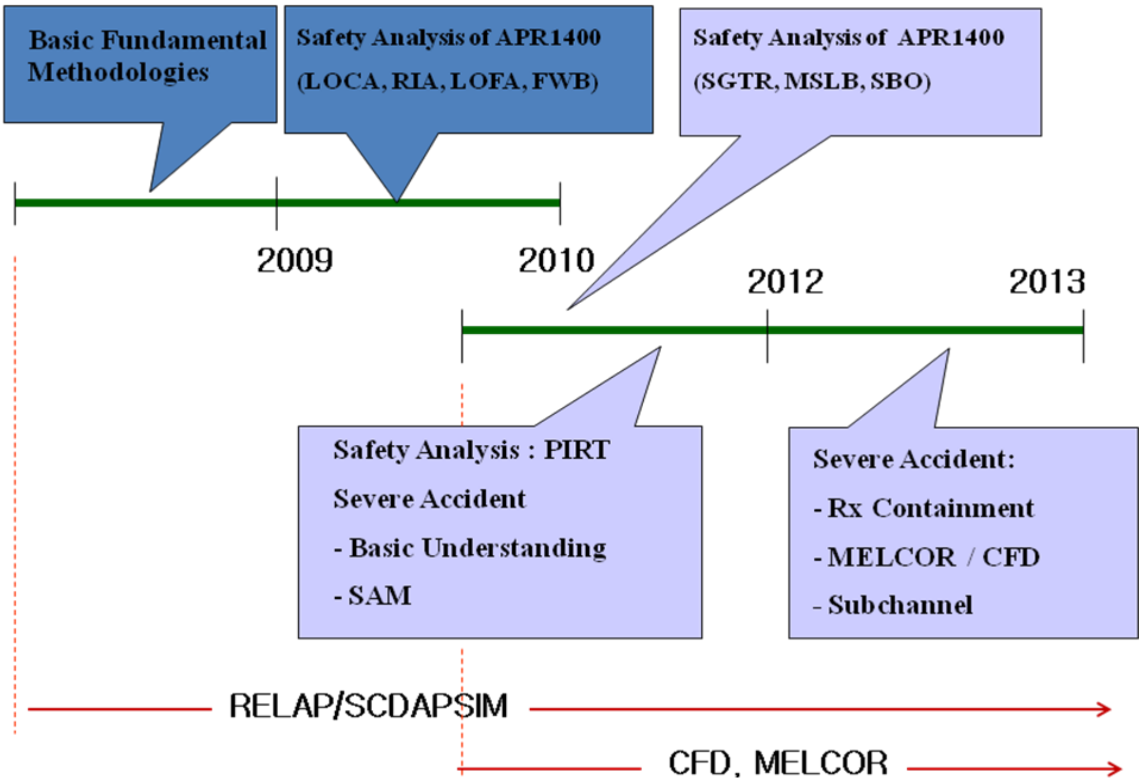


Figure 1: Implementation strategy of computer codes and research works for the cooperation.

2. THE MAIN RESULTS AND DISSCUSION

2.1. Improvement of capability of using RELAP5

Three safety analysis problems including LOCA and SBO, SGTR and MSLB have been performed for APR1400 reactor. Especially in the SGTR analysis, the simulation results has been compared with the simulated ones by MARS-3D reported by KAERI [2].

Figure 2 shows the primary and secondary system pressures during the simulation for the SGTR event. When a steam generator tube is ruptured, the reactor coolant system pressure immediately drops as a tube break and the PRZ backup heater is actuated as designed. After the ECCS is actuated, water level in PZR increases again. The water levels in PZR and in both steam generators (intact and broken) are shown in figure 3. It is also noted, that the starting point used in [2] included steady state (run for 300s) as indicated in the figure. The results simulated by RELAP5 performed by us are in good agreements with the KAERI report [2].

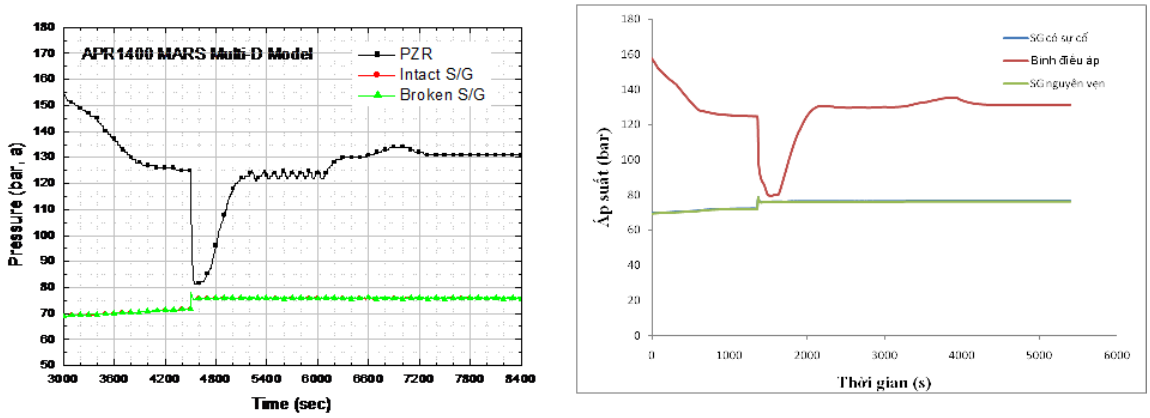


Figure 2: Pressure changes in the primary and secondary loop in SGTR accident in comparison between calculations by RELAP5 (Right) and MARS-3D(Left)[2].

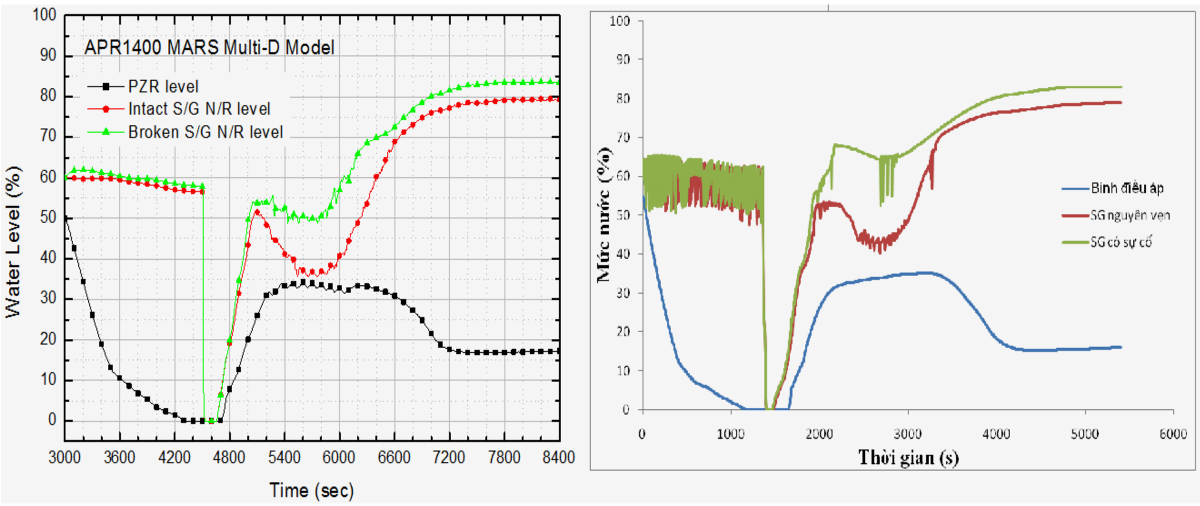


Figure 3: Collapsed water levels in PZR and narrow range of SG (broken and intact) in SGTR accident in comparison between calculations by RELAP5 (Right) and MARS-3D(Left)[2].

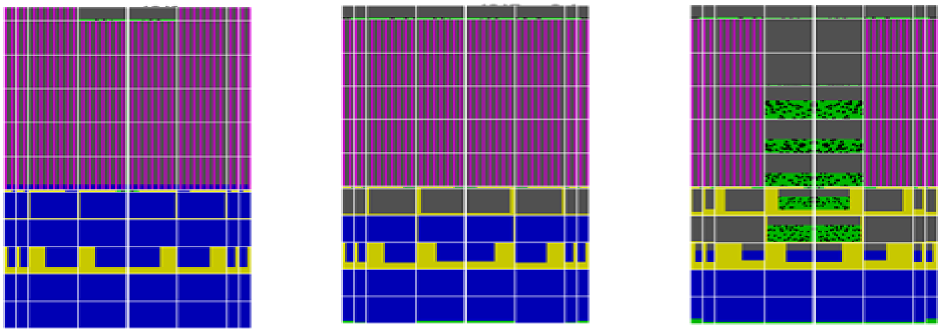
2.2. Analysis of severe accident in NPP

This study refers to the phenomena and processes occurring in severe accident, the safety and prevention systems to minimize the accident consequences, the researchers are initially equipped with the basic knowledge not only in phenomena review but also in study of MELCOR code by the help of KAERI experts. The modeling of Westinghouse 4-loop PWR was simulated and SBO with RCP seal leakage is simulated. According to [3], the results reported in WASH-1400 indicated that breaks of an equivalent diameter in the range of 0.5 to 2 inches in the RCS pressure boundary are an important events which may lead to core-melt. The overall probability of core-melt due to SBLOCA could be dominated by events such as RCP seal failures was also interested.

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

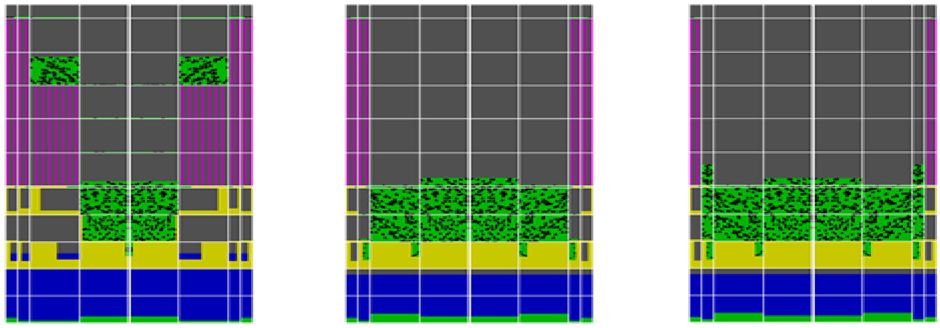
The cladding temperature heat-up and exceeds the melting temperature, the cladding failure starts to occur from top of ring 1 at 8.16 hours after that it spreads to other areas as shown in figure 4.

The simulations performed in this study are comparable with another simulations by MELCOR [4].



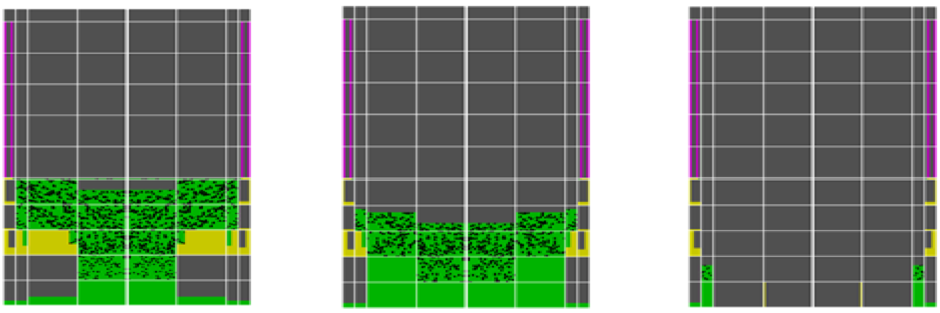
T = 29400 s (8.17h) | T = 31620 (8.78 h) | T = 36200 s (10.06h)

Collapse in ring1 began | Debris in Lower Head | Collapse of ring 1



T = 36450 s (10.13h) | T = 38200 s (10.6h) | T = 38230 s (10.62h)

Failure in ring 2 | Collapse of ring 2 | Collapse of ring 3



T = 53050 s (14.7h) | T = 63720 s (17.7h) | T = 90000 s (25h)

Debris reaches lower head | Melt injected to cavity | Core damage after 25h

Figure 4: Accident sequences in reactor core and lower head.

2.3. Implementation of CFD in reactor T/H

The application of CFD is studied by practice to use ANSYS software (Copyright Research Academy version) in collaboration with ANSYS staff from Hanoi University for Science and Technologies (HUST). Based on basic exercises, ANSYS FLUENT and CFX have been used to simulate PSBT experiments.

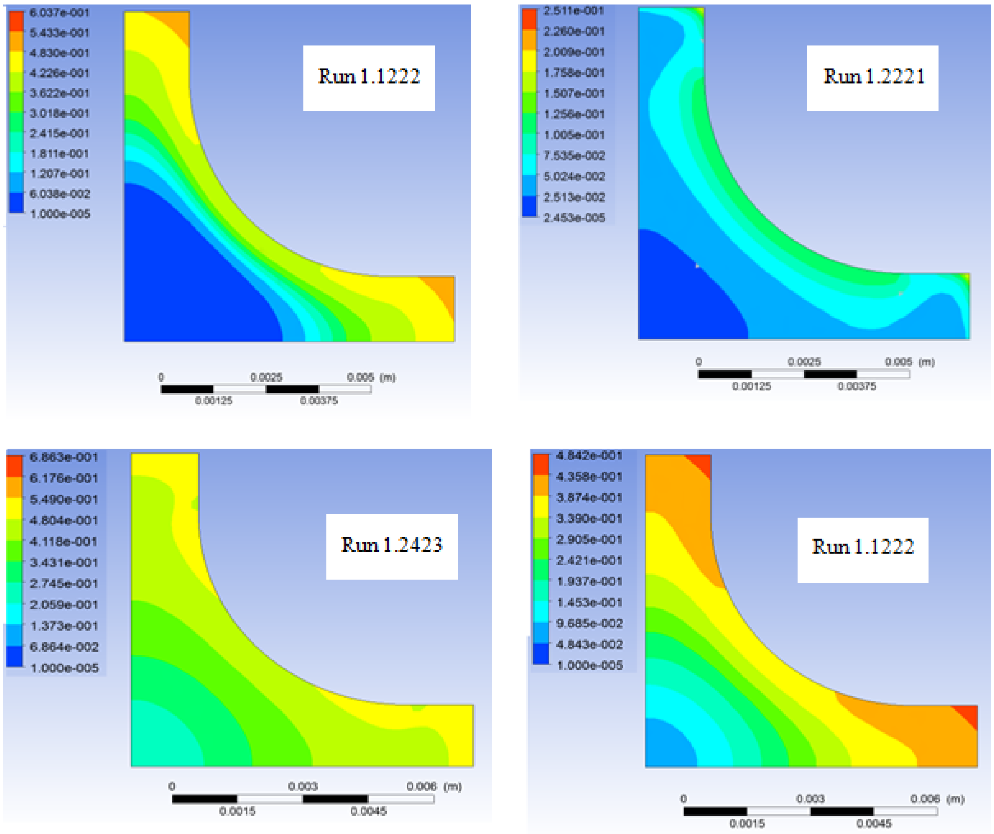


Figure 5: Cross view of the void fraction at measurement position.

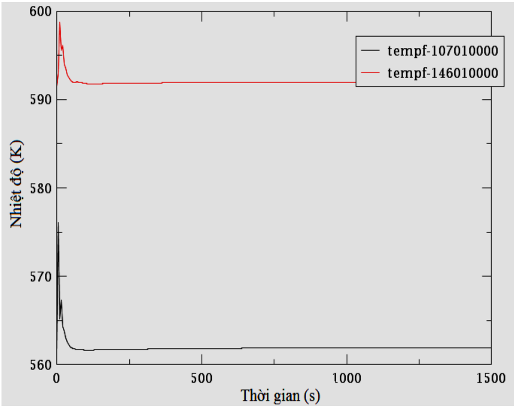
Simulation of two phase flow in a channel is still hard problem with CFD. For the S1 exercise of the benchmark problem mentioned in this paper there are a lot of study which are introduced. This study introduces the utilization of two phase flow simulation with additional sub model of MUSIG which is available in Ansys CFX 14.5. Simulations are presented in figure 5.

The results show that there is a significant improvement of the convergence for the runs being studied. However, for various case of the two phase flow it is needed to study more correlations for selection of appropriate key parameters for model simulation.

2.4. Thermal-hydraulic analysis for PWR

Several of important topics related to advanced light water reactor like critical heat flux, two phase flows, departure from nucleate boiling (DNB) etc. have been studied. The updated knowledge in thermal hydraulics safety analysis is addressed so that the following studies are intensively performed:

The APR1400 and VVER-1000/V392 reactors have been simulated using RELAP5 and the steady state results are given. The nodalization scheme and steady state simulations of APR1400 has been used since 2009 by the authors [1]. Followings are simulated results for VVER-1000/V392. One nodalization based on OECD benchmark noted as “Simulation #2” and the other developed by us noted as “Simulation #1”. Both nodalizations and steady state simulations are satisfactory as indicated in figure 6 and table 1.



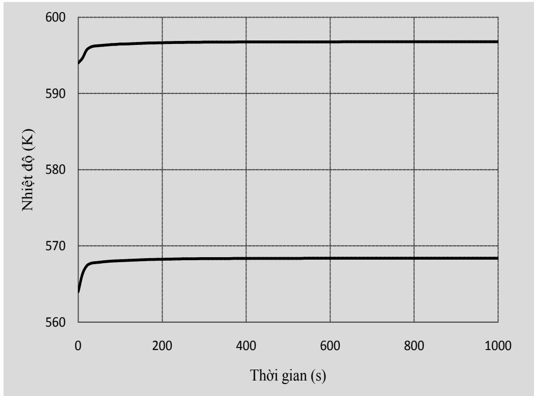


Figure 6: Water temperature at the inlet and outlet calculated by the RELAP5 using different nodalizations: "Simulation #1"(left) and "Simulation #2"(right).

The water temperature at the inlet and outlet calculated by the different nodalizations are presented in Table 1.

Table 1: Thermal hydraulics parameters of RCS of VVER1000/V392 in the normal power operation conditions.

Main parameters	**Design [7]**	**Calculated #1**	**Calculated #2**
Thermal power, MW	3000	3000	3000
Mass flow rate through the core, m^3/h	86000 ± 2600	86532	86029
Mass flow rate / nhánh, m^3/h	21500± 1000	21633	---
Primary pressure (in PZR), MPa	15.7±0.3	15.73	15.8
Water temperature at inlet, ^0C	291 (+2)(-5)	295.3	288.8
Water temperature at outlet, ^0C	321±5	324.3	318.8
Pressure drop in core, MPa	0.148	0.177	---
Pressure drop in RPV, MPa	0.387	0.38	---
Bypass flow rate, %	3	3.1	---
Feedwater temperature, ^oC	220 ± 5	220	220.15
Water level in PZR, m	8.17	8.12	---
Water level in SG (secondary side), m	2.7 ± 0.05	2.7	2.63
SG exit steam pressure, MPa	6.27 ± 0.1	6.08	6.27
Steam temperature, ^oC	278.5	276.7	---

The steady state calculations are performed by simulations in system codes like RELAP5 for VVER-1000/V392 in the normal power operation. The thermal hydraulics parameters calculated in our simulations are generally in good agreements with the design. It is also noted that this simulations are used not only in verifying the provided design data, but also in safety analysis in transient and accident conditions.

3. CONCLUSION

Safety Analysis in which T/H studies are very important needs to be addressed not only in NPP projects going on in Vietnam now, but also in strengthening of our understanding of safety characteristics of NPP systems. Through joint research project, the APR1400 reactor has been studied and safety analysis problems were performed. Based on the experience in first phase [1], the VVER-1000 has been intensively studied, not only in core thermal hydraulics, but also in severe accident including the containment, core catcher features etc. The severe accident phenomena are introduced and simulation of PWR using MELCOR code is supported by KAERI. These are highly appreciated.

The international cooperation is recognized as important factor for HRD in nuclear safety research. The human resource development in the field of safety analysis now is under the request. It is not only requirements of the number of researchers, but also higher qualification of researchers as well as research works.

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