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Thermal hydraulic system of a VVER-1000 nuclear reactor and numerical simulations

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Abstract. This paper presents some results of our study on the numerical simulation of the thermal hydraulic system of the Russian VVER-1000 pressurized water nuclear reactors. The simulations were conducted using the integrated VISA (Visual System Analyzer) and RELAP5 (Reactor Excursion and Leak Analysis Program, version 5) softwares known as VISA_RELAP5. Originally RELAP5 (a thermal hydraulic system code) and then recently VISA (a graphical user interface) were developed for the simulation of the thermal hydraulic system of Western type pressurized water reactors (PWR) undergoing transients. In Vietnam, research on the numerical simulation of the thermal hydraulic system of Da Lat nuclear research reactor and some typical types of PWRs using RELAP5 have long been carried out in our research group along with some of other research institutions. It should be noted that knowledge of the VVER reactor system is still lacking in Vietnam until now. The data that we used in our modelings, simulations and calculations are real data of the Kalinin nuclear power plant (NPP) in Russia. Therefore this research has important practical implications especially for the preparation for the safe operation and proper management of incidents (accidents) in the 1st NPP that will be built in Ninh Thuan, Vietnam. The reactors adopted in the 1st Ninh Thuan NPP will be the Russian VVER PWR. From the point of view of the available information about VVER reactors in Vietnam, our study is immensely useful since Russia has not yet much opened up information about VVER reactors. At this point, our research is a basic important step towards a practical study case.

Keywords: Thermal hydraulic, VISA, RELAP5, Nuclear Reactor, Pressurized Water Reactor (PWR), Nuclear Power Plant (NPP), VVER-1000, Ninh Thuan NPP.

1. Introduction

Although nuclear power industry has experienced some serious accidents in the past (Chernobyl accident in Ukraine and Three Mile Island accident in the US) and recently the Fukushima accident in

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Japan, most people believe that nuclear power is still a very important power source and plays crucial role in our globe. It is clearly stated that "Though nuclear power industry faces enormous safety challenges, it is still an important choice in the 21st century." [1].

Vietnam Government has approved plans to build 2 NPPs in Vietnam that are the 1st and the 2nd Ninh Thuan NPPs [2]. It is therefore mandatory to develop human resources for nuclear industry (especially nuclear power industry), and to promote research on various aspects related to nuclear reactors, nuclear power plants, nuclear safety etc. Among those, the research and safety analysis based on numerical modelings, calculations and simulations of the reactor thermal hydraulic system using computer codes is much important as well.

The best-estimate thermal hydraulic simulation programs (e.g. RELAP5 code [3]) have long been developed. They have been step by step improved to model more accurately the thermal hydraulic system of the nuclear reactor and NPPs. Those programs are also important in the calculation and simulation of important thermal hydraulic phenomenon in NPPs. However the application of those programs tends to be limited among small groups of experts. The rapid development of the capability of the personal computer permits those programs now to be able to run well on personal computers. As a consequence, those simulations program are becoming more and more popular. However, a major restriction that still exists is that the preparation for the input data files is usually very complicated and easy to have errors. Therefore VISA program was developed under the cooperation between KAERI (Korea Atomic Energy Research Institute) and KHNP (Korea Hydro-Nuclear Power), Korea to perform the tasks of a GUI (Graphical User Interface) and to help users to exploit more effectively the thermal hydraulic simulation programs [4]. VISA can be integrated with three thermal hydraulic simulation programs including MARS, RETRAN-3D and RELAP5. Here we use the integrated program VISA and RELAP5 which is called VISA_RELAP5 for short [5]. VISA program has many powerful functions to support the users in the modeling, calculation and simulation of the thermal hydraulic system, and safety assessment of NPPs [6].

In the world now, the RELAP5 code (developed in the US) is superior to other thermal hydraulic simulation codes in the nuclear industry and nuclear research. Therefore in this research, we chose the VISA_RELAP5. However the most important and crucial task is the mastering of RELAP5 program. That has long been conducted in the Department for Industrial and Environment Fluid Dynamics, Institute of Mechanics, Vietnam Academy of Science and Technology (VAST) through the numerical modelings, calculations and simulations of Da Lat nuclear research reactor in Vietnam and some typical PWRs using RELAP5 [7-14].

The application VISA_RELAP5 to NPPs with Russian VVER reactors is one of the following steps to contribute to raising our capability in thermal hydraulic research and safety analysis, and to mastering the technology of VVER reactors that will be transferred to Vietnam in near future. Preliminary results of this research have been reported at the IX National Conference on Nuclear Science and Technology held in August 2011 in Ninh Thuan province, Vietnam [15]. Through our communication with other nuclear research groups coming to the conference from almost all of nuclear research institutions in Vietnam and from abroad, our study would be the first of this kind in Vietnam. Given the fact that knowledge of Russian VVER reactor system is still seriously lacking in Vietnam (perhaps to some extent, even in the world [16]), our research hopefully will provide preliminary relatively detailed information about VVER reactors. This type of reactors has some characteristics different from the Western type PWR reactors. VVER reactors have horizontal steam generators (rather than vertical ones in PWRs), circulation cooling loops having isolation valves that can be closed to isolate one (or several) loop(s) if necessary (e.g. when the reactor operates at low power level or in case of emergency etc.) [17, 18].

2. Brief of VISA and RELAP5 codes

2.1. VISA

The VISA graphical user interface program was originally developed under the joint effort of some research institutions and energy industries in South Korea [4-6]. This program is designed to be integrated with some thermal hydraulic simulation programs such as RELAP5, MARS and RETRAN etc. through the use of new written or modified input/output functions (not calculation-related functions) of the thermal hydraulic simulation programs. So the connection between VISA interface and the simulation programs is by the help of only input/output functions (the calculation-related functions of the simulation programs are kept intact). The connection is via the dynamic link libraries (DLLs) of the simulation programs [4-5].

Main functions of VISA include:

- Manage input/output files, the graphic files; select unit (SI or British etc.) for the output results (Project functions);
- View, edit and change the value of the parameters of the control system, the geometrical parameters of thermal hydraulic system (Pre-processor functions);
- Graphically view the calculated results directly during the calculation process and review the previously calculated results by using graphics and mimics (Graphic and Mimic functions);
- View the output results through graphical windows, graphs of variables (normally the variables of time); monitor the status of trips (Graphic interface and Trip functions).
- Simulate actual operations of the plant operators; the control of thermal hydraulic system of the reactor and the NPP is carried out via four types of controls including controls of trips (on/off), controls of valve area, controls of flow-rate in pipes, and controls of reactor power (Interactive control functions).

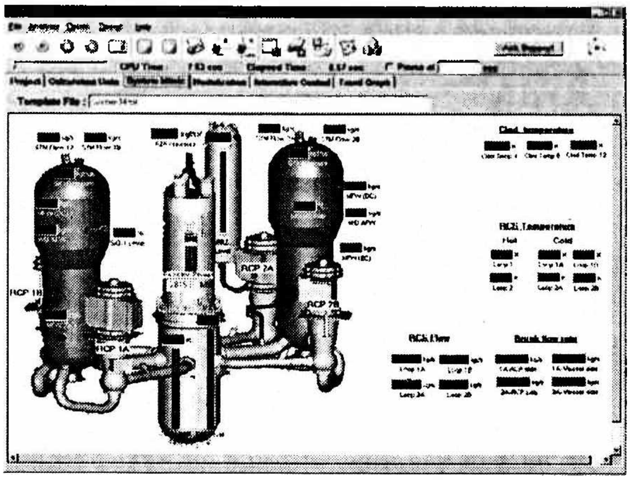

Fig. 1. Graphical simulation of a typical Western PWR (Mimic functions) (figure from [5]).

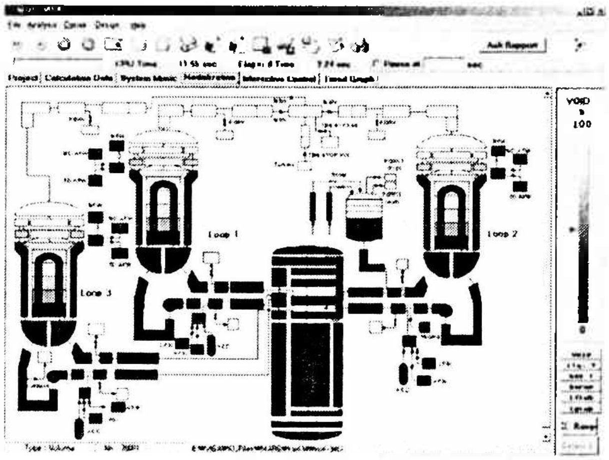

Fig. 2. Graphical presentation of the PWR nodalization model and the calculated void in the system during calculation process (Pre-processor and Graphic functions) (figure from [5]).

2.2. RELAP5

The RELAP5 is a best-estimate transient simulation code for the simulation of light water reactor coolant system during postulated accidents. Coupled behavior of the reactor coolant system and the reactor kinetics is implemented. RELAP5 model includes separate models for all of the components of the reactor thermal hydraulic system (i.e. fuel rods, reactor core, control rods, reactor vessel, pump, heat conduction structures, pipe, valves, control systems etc.). Originally the code was based on a homogeneous equilibrium model (HEM) of the two-phase flow process. Then the code was totally rewritten with the use of a two-fluid, nonequilibrium, nonhomogeneous, hydrodynamic model for transient simulation of the two-phase system behavior. The version used in this research is RELAP5/MOD2 which employs a full nonequilibrium, six-equation, two-fluid model.

Study and applications of RELAP5/MOD3.2 code was conducted in many of our previous researches and will not be shown here. Details of the system of equations, the solution methods, the program flow chart, test calculations, and applications etc. can be found in [3, 7-15, 17].

3. Modeling, calculation and simulation of the thermal hydraulic system of a VVER-1000 reactors using VISA_RELAP5

VVER reactors are of the PWR design developed in Russia which exhibit many similarities to Western PWRs. Russia has made a lot of effort to improve many aspects of this reactor type and to promote export to the international market. One of the potential market is Vietnam whose government has signed contracts with Russian companies to build a nuclear power plant in Ninh Thuan province (the 1st NPP of Vietnam). Therefore the reactor technology, numerical modeling, calculation, simulation and safety analysis of the thermal hydraulic system of VVER nuclear power plant are urgently needed in Vietnam.

3.1. Structure of the VVER reactor system

a. Overview

VVER reactor system studied in this research is a VVER-1000/V-338 reactor (version V-338, the original design of the VVER-1000 nuclear reactor series with 1000 MW electric power). Fig. 3 below shows the typical thermal hydraulic system of the VVER-1000 reactor. Compared with Western PWR (Fig. 1), ones can see clearly some differences. VVER reactors are equipped with isolation valves (Main Gate Valve - MGV) and horizontal steam generator (Fig. 2). Table 1 below shows some of the main parameters of the VVER-1000/V-338 reactor [18]. In general, the VVER nuclear reactor type has some advantages over Western PWRs. Detailed discussions can be referred further to in [19].

Table 1. Main parameters of the VVER-1000/V-338 reactor

Thermal / Electric power	3000 MWth / 1000 MWe
Coolant Pressure (in the Primary system)	15.7 MPa
Number of cooling loops	4
Coolant flow rate through the reactor core	84800 m^3/h
Coolant temperature inlet (to reactor core)	289.7 °C
Coolant temperature outlet (from reactor core)	320.0 °C

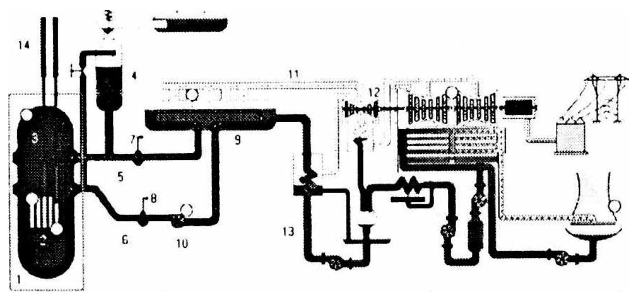

Fig. 3. Overview of the thermal hydraulic system of a typical VVER reactor (horizontal steam generator).

The main components of the thermal hydraulic system of the VVER reactor including (Fig. 3):

1 - Reactor vessel
2 - Reactor core
3 - Control rods
4 - Pressurizer
5, 6 - Hot and cold legs (primary system)
7, 8 - Main Gate Valves
9 - Horizontal Steam Generator
10 - Main circulation pump (primary system)
11 - Steam line
12 - Turbines
13 - Cold leg (secondary cooling system)
14 - Control rod driving mechanisms

b. Reactor vessel

The design of the reactor vessel and the vessel's main dimensions are shown in Fig. 4. Main parameters are shown in Table 2 below.

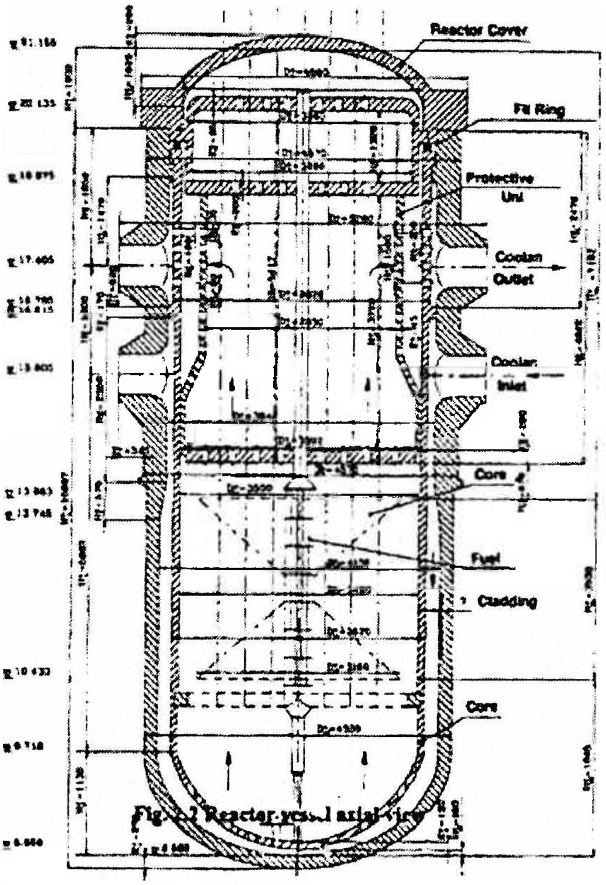

Fig. 4. VVER-1000/V-338 reactor vessel.

Table 2. Main parameters of the VVER-1000/V-338 reactor vessel

Length (m)	10.88
Diameter, external on the cylindrical part of the reactor (mm)	4535
Thickness of the cladding layer (mm)	7
8 nozzles, diameter (mm)	850
Reactor vessel weight (ton)	304
Reactor vessel total volume (m^3)	110
Core volume (m^3)	13.7
Core heat transfer surface (m^2)	5130
Mass of fuel in the core (kg)	80100
Wall thickness (mm)	199.5
Number of fuel assemblies (hexagonal)	163
Number of fuel rods in 1 fuel assembly	312

c. Cooling system

The reactor cooling system includes a primary side and a secondary one. They are shown in Fig. 3 above. Main parameters are shown in Table 3 below.

Table 3. Main parameters of the cooling system

Primary loops (from 1 to 4)	
Horizontal Steam Generator (Primary side)	
Main Circulation Pump (Capacity: m^3/h)	20000
Two main gate valves	
Hot leg length (m)	16.94
Cold leg length (m)	28.72
Pipe inner diameter (mm)	850
Pipe outer diameter (mm)	990
Secondary loops (from 1 to 4)	
Feed water system, Steam generator (secondary side)	
Steam lines, Turbine generator, Condenser system	

d. Steam generator

VVER reactors all use the horizontal steam generators. Fig.5 shows a typical horizontal steam generator..

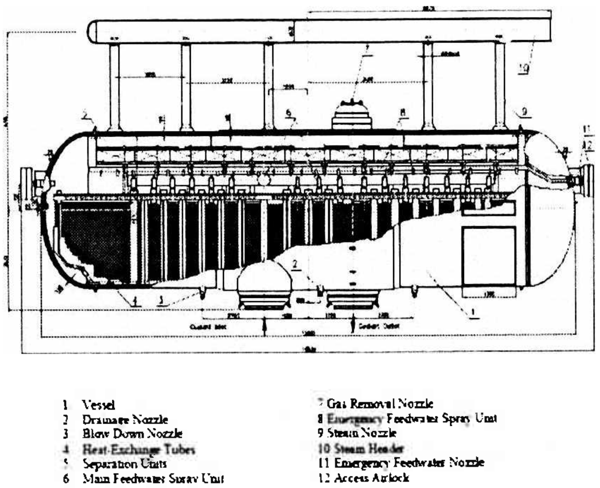

Fig. 5. Horizontal steam generator (figure from [20]).

Compared with vertical steam generators in Western PWRs, horizontal steam generators used in VVER reactors have some typical advantages as shown below [20-21]:

- moderate steam load and simple gravity-based steam separation mechanism;
- moderate velocity of the cooling water in the second loop of the steam generator thus preventing any danger of vibrations of the heat-exchange tube system;
- validated longtime serviceability of the steel tubes (the maximum time of operation is 38 years for PGV-440 and 23 years for PGV-1000 type steam generators);
- vertical arrangement of the first-loop collectors, preventing accumulation of sludge deposits on their surfaces, thereby decreasing the danger of corrosion damage to the heat-exchange tubes in the region where the tubes are built into the tube sheet;
- larger volume of water in the second loop, enabling the ability to cool the reactor via the steam generator in the case where normal and emergency water feeding has stopped;

- in this steam generator design, it possible to maintain an allowable concentration of dissolved impurities in the critical zones and increasing the reliability from the viewpoint of corrosion effects;
- horizontal arrangement of the heat-exchange surface, enabling reliable natural circulation of the first-loop coolant even with a massive water level below the top rows of the heat-exchange tubes;
- convenient access to the tube sheet for servicing and checking from the first- and second-loop sides; there are no heat-exchange tubes at the bottom of the housing, so that sludge is more easily removed through the purge system;
- presence of equipment for disconnecting the collectors from the main circulation pipelines, making it possible to decrease the time required to perform maintenance work and to increase the capacity utilization factor by performing work simultaneously on several steam generators and refueling the reactor.

3.2. Modeling

Similar to other numerical modeling systems, the numerical simulation of the thermal hydraulic system of nuclear reactors using VISA_RELAP5 requires the system be divided into control volumes, or thus nodalization model for every component in the system (pump, valve, pipe, reactor core etc.). Based on the practical data of the Kalinin nuclear power plant [18], the nodalization model for the numerical simulation has been developed (Fig. 6).

As shown in Fig. 6, nodalization model of the VVER-1000/V-388 reactor shown in VISA_RELAP5 interface consists of four circulation loops, these are a reactor vessel, four cooling loops, a pressurizer, four steam generators, steam lines, a steam collection and a steam storage system, safety valves, an exhaust valve of steam generator system (Fig. 6).

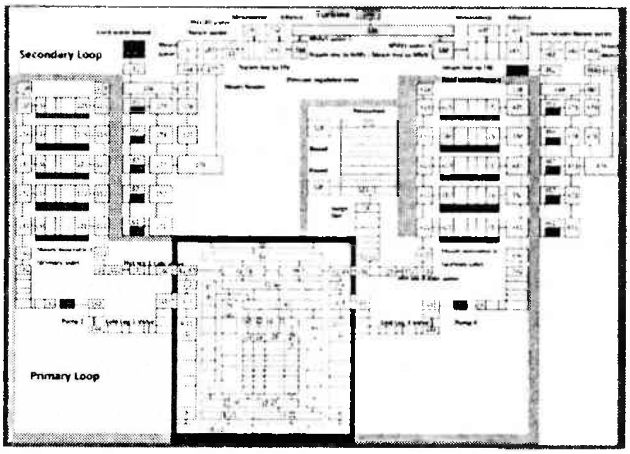

Fig. 6. Nodalization model of the thermal hydraulic system of the VVER-1000/V-338 reactor.

The Mimic functions of VISA_RELAP5 integrated system allow viewing the calculated thermal hydraulic parameters directly during the simulation process or after the simulation finishes (Replay mode). In this research, our Mimic model of the VVER-1000 reactor was developed as shown in Fig. 7.

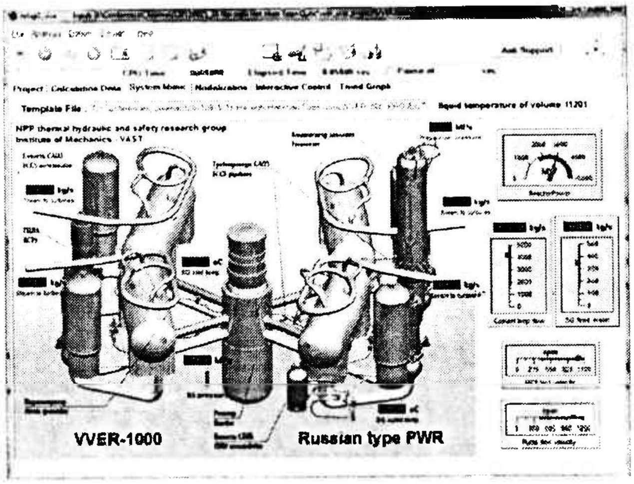

Fig. 7. Graphical simulation of the thermal hydraulic system of the VVER-1000 reactor using Mimic functions.

3.3. Results of the numerical simulation of the thermal hydraulic system of the VVER-1000 reactor using VISA_RELAP5

Calculation and simulation was initially carried out for the steady state of the reactor operation at constant thermal power 3000 MWth (the nominal power of the reactor). The calculation shows that the system reaches steady state after about 100 seconds. Fig. 8 shows reactor thermal power at steady state 3000 MWth.

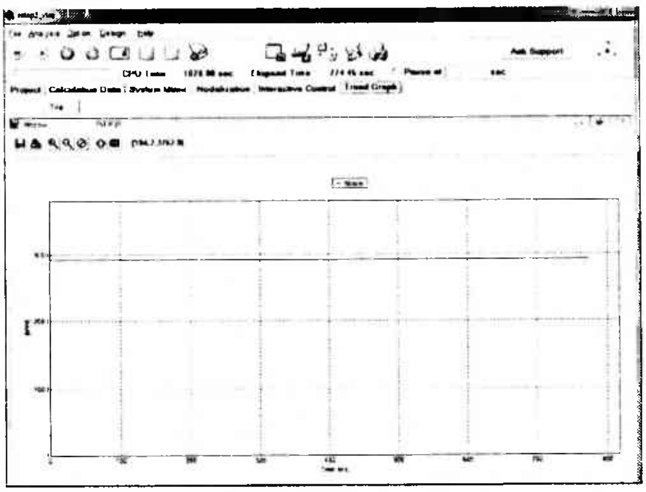

Fig. 8. Reactor thermal power at steady state.

Then major thermal hydraulic parameters which were calculated using VISA_RELAP5 were compared with the measured data during the plant operation at the same operational condition (Table 4).

Table 4. Main thermal hydraulic parameters (measured during the plant operation and calculated using VISA_RELAP5)

Parameters	*Measured*	*VISA_RELAP5*
1. Reactor thermal power, MWth	2917.0	2917.5
2. Coolant flowrate, kg/s	18471.0	18471.2
Loop 1	4253.6	4253.8
Loop 2	4800.0	4800.2
Loop 3	4892.4	4892.1
Loop 4	4524.9	4525.5
3. Primary side pressure, MPa	15.68	15.65
4. Reactor vessel inlet/outlet pressure drop, KPa	326.0	328.6
5. Coolant temperature at reactor vessel inlet/outlet, K		
Loop 1	559/589	559.3/589.2
Loop 2	559/586	559.1/586.5
Loop 3	560/587	560.2/588.2
Loop 4	559/588	559.5/588.6
6. Feedwater temperature, K		
Steam generator 1	437.3	436.5
Steam generator 2	437.3	436.6
Steam generator 3	436.3	437.0
Steam generator 4	436.7	437.9
7. Feedwater flowrate, kg/s		
Steam generator 1	345.6	345.2
Steam generator 2	345.0	345.1
Steam generator 3	360.0	360.3
Steam generator 4	354.4	354.6
8. Steam generator pressure, MPa		
Steam generator 1	5.89	5.88
Steam generator 2	5.86	5.86
Steam generator 3	5.89	5.89
Steam generator 4	5.86	5.86

It is obvious that the calculated results using VISA_RELAP5 with the nodalization model developed are in good agreement with the measured data during the normal operation of the plant at nominal thermal power and in steady state.

The modeling developed should be fully adequate for further studies on the transient behaviors of VVER-1000 reactors. There exist a broad range of thermal hydraulic safety issues related to the VVER reactors that need to be further investigated and addressed such as the RIA transient (Reactivity Insertion Accident), LOCA (Loss of Coolant Accident), LOFA (Loss of Flow Accident), FWLB (Feed Water Line Break accident), Loss of Offsite Power accident etc.

4. Conclusions

The thermal hydraulic system of a VVER-1000 nuclear reactor has been investigated. Basically the VVER reactors are similar to Western type PWRs. Though they exhibit some differences and have some advantages over their counterpart. The VISA_RELAP5 integrated software has been explored and applied in this research to model, calculate and simulate the thermal hydraulic system of a practical VVER PWR NPP, i.e. the Kalinin NPP in Russia. The nodalization and mimic models in VISA_RELAP5 have been developed. Calculations and simulations for the steady state operation of the reactor at its maximum nominal power have been carried out. The calculated results reproduce well the measured ones in steady state. Therefore the numerical model is believed to be proper and appropriate to be used in further simulations of thermal hydraulic transients in the system.

This research on application of VISA_RELAP5 to the thermal hydraulic simulation of VVER reactors is a pioneer initiative in this subject in Vietnam. It has important implication in the promotion of thermal hydraulic safety research and analysis particularly in Institute of Mechanics (VAST), and more generally, to some extent, in Vietnam. In addition, this research hopefully will contribute to the exploration of the VVER reactor technology and to the acquisition of the knowledge about VVER reactors and NPPs. Especially given the fact that the VVER reactors will be installed in the first NPP of Vietnam in Ninh Thuan province.

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