LOSS OF OFFSITE POWER ACCIDENT ANALYSIS IN A VVER-1000/V446 NUCLEAR POWER PLANT

by

Mohsen ESFANDIARI [1]**, *Gholamreza JAHANFARNIA*** [1*]**,**
Kamran SEPANLOO [2]**, *and Ehsan ZARIFI*** [2]

[1]Department of Nuclear Engineering, Science and Research Branch, Islamic Azad University, Tehran, Iran
[2] Reactor and Nuclear Safety School, Nuclear Science and Technology Research Institute, Tehran, Iran

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The aim of this study is to present a thermo-hydraulic analysis of the loss of offsite power accident in VVER-1000/V446 nuclear power plant using the RELAP5 code. Loss of offsite power accident would lead to the unavailability of major active safety systems, and that the safety criteria ensuring a secure operation of the nuclear power plant would be violated, resulting in core heat-up with possible core degradation. Therefore, the analysis and investigation of the plant, during this accident, is very important. For this purpose, different behaviors of major components in the primary and the secondary sides of the reactor coolant system are studied. In this paper, the reactor is simulated by using RELAP5 system code. The results show a reasonable agreement with the developed model and also with the Bushehr nuclear power plant final safety analysis reports.

Key words: loss of offsite power, safety analysis, RELAP5, nuclear power plant, VVER-1000

INTRODUCTION

The safe operation and accident recovery of nuclear power plants (NPP) depend heavily on the availability of alternating current (AC) electrical power. Offsite power sources normally supply this essential power from the electrical grid to which the plant is connected [1].

For investigation of the loss of offsite power (LOOP) impacts, status of the plant is essential whether the reactor is critical or shut down [1]. If the plant is on power when a LOOP occurs, then generally a reactor trip occurs, challenging various safety systems designed to bring the plant in to a safe shutdown. Most of the safety systems require AC power, therefore emergency diesel generators (or other emergency AC power sources) must start and run up to supply this power until offsite power is restored to the safety buses. If the emergency AC power sources fail, the plant is still designed to shut down safely via portions of safety systems that can function for a limited period of time without AC power (*e. g*., turbine-driven pumps for coolant injection). Even if the plant is in shut down mode when a LOOP occurs, emergency AC power must be supplied to the residual heat removal systems.

Many studies have investigated the data on LOOP and/or offsite power restoration. Zhiping calculated and compared different LOOP frequency for some plants in different categories [1]. He considered LOOP events occurred during 10 years. Park *et al*. [2] considered LOOP frequency and restoration time at power and shut down operations for the actual LOOP events that had occurred from 2005 to 2012 at NPP in Korea. The LOOP event caused by tornado in Surry NPP was studied by Borysiewicz *et al*. [3]. They used the probabilistic analysis for investigation of the reliability of power plant protection systems when the LOOP event occurred. The results of Volkanovski *et al*. [4] studies about LOOP events in nuclear power plants represent the LOOP events that usually occur during operational mode and continue for two minutes or more. They concluded that prominent cause for the LOOP events are human errors. Also, the trend analysis of LOOP events were performed by Volkanovski *et al*. [5]. This paper presented the LOOP events frequency between 1990 and 2012 using two databases: the IAEA International Reporting System (IRS) and Nuclear Regulatory Commission (NRC) Licensee Event Reports (LER) database. The LOOP Events at China's NPP are investigated by Jiao *et al*. [6]. All LOOP events in NPP of China from 1993 to 2017 were collected and several features for a LOOP event were considered. The main causes identified for the events were equipment failures. Analysis of station blackout event of HTGR-Type Experimental Power Reactor was studied by using PCTRAN-HTR functional simulator code [7]. In this paper, pressure and temperature parameters of fuel

* Corresponding author; e-mail: rezajahan@yahoo.com

and coolant have been presented during the station blackout event. Severe accidents analysis in VVER-1000 reactors were investigated by Tusheva *et al*. [8]. They used the integral code ASTEC for calculation of thermal hydraulic behavior and compared it with real thermal hydraulic behavior.

This study presents results of an accident analysis for a hypothetical LOOP scenario in a reference Bushehr-1 VVER-1000 reactor [9]. The initiating event with complete loss of AC power, belongs to the typical, beyond design basis accidents for which the time of plant survivability, without severe fuel damage, depends solely on built-in safety features.

The occurrence of the specific thermal-hydraulic phenomena, appearing during such an event, is being investigated with the thermal-hydraulic system code RELAP5 [10]. The LOOP scenario is characterized by complete unavailability of all active safety systems, except the battery supplied steam dump to atmosphere (BRU-A) valves. This event represents a classical high pressure accident scenario, and if no safety systems are activated, or any accident management measures are applied, it could lead to the failure of the reactor pressure vessel (RPV) under high pressure, which could cause challenges to the containment. Early controlled depressurization of the primary circuit applied as a mitigative measure could prevent failure of the RPV under high pressure. Moreover, the high pressure melt ejection often accompanied by the phenomena known as direct containment heating, represents a serious threat to the containment integrity. Taking into consideration the specifications of the VVER-1000 reactors, a comparative analysis has been done.

METHODS AND MATERIALS

Bushehr-1 VVER-1000 is a pressurized water reactor (PWR) with a thermal power of 3000 MW and gross electric output of 1000 MW. The unit has four circulation loops, each including a main circulation pump and a horizontal steam generator (SG). The pressurizer is connected to one of the main circulation loops.

In this modelling all major components of the primary and the secondary side of the reactor coolant system, the necessary reactor protection and safety injection systems, are simulated with RELAP5 code. The adopted nodalizations of the VVER-1000 reactor are shown in fig. 1. In this modeling, the actual parameters and dimensions are used to describe the flow areas, volumes, hydraulic diameters, elevations, heat transfer area, and heat structure masses. On the primary side, the reactor core, the RPV, main circulation pumps, main circulation pipes, the pressurizer, and the relief and safety valves, have been modelled. On the secondary side, special attention has been paid to the modelling of the SG and their related safety systems.

Before performing a transient simulation, a steady-state calculation for adjusting the boundary conditions, necessary for the analyses of the discussed accident sequences, has been performed. In that way, the initial plant conditions such as reactor power, SG power, temperatures, water mass, and mass flow rates are calculated. The steady-state calculation has been performed for 100 seconds. In tab. 1. The conditions at the end of the steady-state simulations have been summarized.

In case of total loss of all AC power supply sources, including failure of diesel-generators, the accident is more serious as all active parts of safety system, such as the emergency feed water and emergency core cooling system (ECCS) water, fail. However, if AC electrical power either from the grid or from the diesel-generators is not restored quickly, the consequences to the plant and the public can potentially be extreme. As there are no means to remove decay heat from the primary circuit, the accident develops and leads to at high primary pressure and periodical opening and closing of the pressurizer safety valves. The loss of primary coolant through the pressurizer safety valve leads to the core dry out and further heat-up and transition of the accident into severe stage which includes:

- Switch off of all main coolant pumps (MCP).
- Reactor scram (control rods drop) due to loss of three from four MCP.
- Switch off of the main and auxiliary feed water systems of the secondary side.
- Switch off makeup / letdown system of the primary system.
- Disconnection of pressurizer (PRZ) system power supply (PRZ heaters).
- Closing of turbine stop valve (TSV).
- The BRU-K connection is blocked- due to loss of condenser vacuum; fast-acting steam dump valve with discharge to turbine condenser (BRU-K).

Initial conditions and availability of systems are:

- The NPP at normal operating conditions (100 % reactor power).
- The SG pressure regulation is available; BRU-A (fast-acting steam dump valve with discharge to atmosphere).
- For VVER-1000: BRU-A stops at 7200 seconds (batteries depletion).
- Pressurizer relief and safety valves are available.
- Active ECCS (High Pressure Injection System. Low Pressure Injection System are not available.
- Passive ECCS (accumulators) are available.

The most important parameters' behavior, sequence of events and systems operation, are presented in tab. 2. The calculation is performed up to 10000 s when cladding temperature reach to the safety criteria.

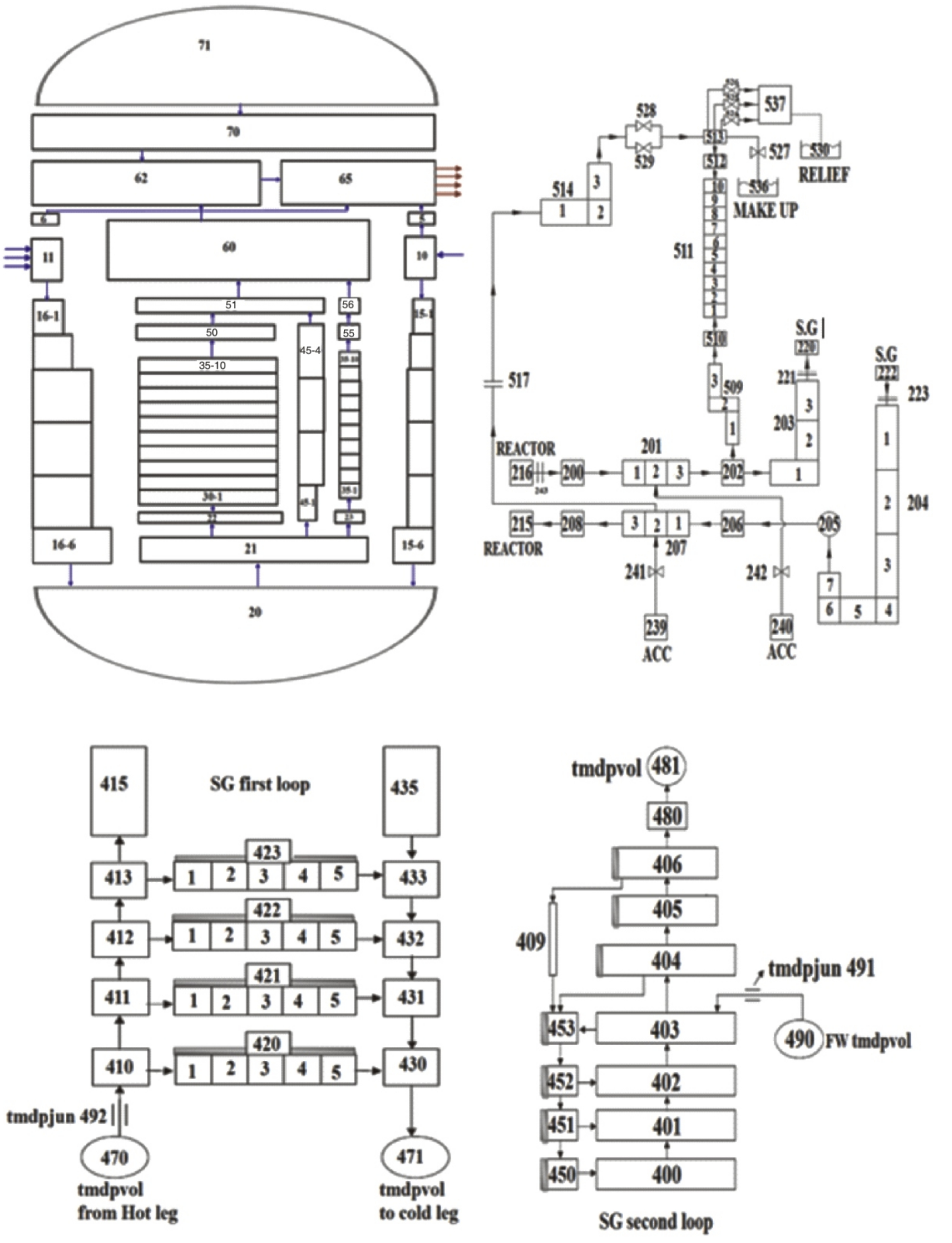

Figure 1. The VVER-1000 reactor/ Nodalization scheme

RESULTS AND DISCUSSION

The relative reactor power is shown in fig. 2. The initial event leads to switch off of all MCP, the reactor protection system actuates after 1.5 seconds, due to *Three of Four MCP switched off* and after this signal all control rods drop in 4 seconds to the bottom of the core so, the reactor power suddenly decreases but due to the decay heat, the power generation is continued.

The primary and secondary pressures, for the discussed scenario, are given in figs. 3 and 4, respectively. Shortly after the event, the secondary pressure increases to the set point pressure thresholds of the steam dump to atmosphere (SDA), and after opening

Table 1. Steady-state parameters of the VVER-1000 plant

Parameters	Value
Core power, [MWth]	3000
Primary pressure, [MPa]	15.7
Average coolant temperature at reactor outlet, [°C]	321.0
Maximum coolant temperature at reactor inlet, [°C]	291.0
Mass-flow rate through one loop, [kgs^{-1}]	4400.0
Pressure in SG, [MPa]	6.27
Steam mass-flow rate through SG, [kgs^{-1}]	408.0
SG total water mass for one SG, [kg]	47000.0

Table 2. Sequence of events [9]

Time [s]	Event	Interlocks, set point for actuation or other reason
0.0	Trip of all RCP sets Trip of the main and auxiliary feed water systems of the secondary side Trip of makeup-blowdown system of the primary system BRU-K disconnection Disconnection of PRZ system power supply	Loss of all AC off-site and on-site power supply sources
0.6	Closing the turbine generator stop valves	Turbine emergency protection action
1.4	Scram signal generation	The NPP blackout
1.7	The onset of control rod motion	Emergency protection action
5.0	The BRU-A opening	Reaching SG pressure of 7,15 MPa
2800.0	The SG drainage	–
7000.0	Onset of the core heat-up	–
10000.0	End of calculation	–

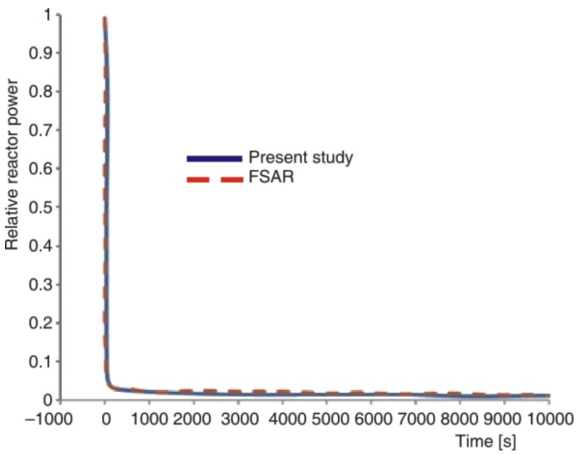

Figure 2. Relative reactor power

of the BRU-A valve this causes continuous decreasing of water inventory on the secondary side of SG.

During the first seconds of the accident, the primary pressure is dropping due to decreasing of the core decay heat. The BRU-A valve is operating and it maintains the secondary pressure at 6.67 MPa. Likewise, for the reference German PWR the pressure regulation for the secondary side is activated and shortly after that, the partly cool-down procedure for the secondary side is actuated leading to a cool-down rate of 100 Kh^{-1} at 7.5 MPa.

Due to the heat transfer from primary to secondary side, the secondary side water inventory is continuously evaporating, decreasing the liquid level on the secondary side of SG, see fig. 5. With decreasing levels in the SG, the SG power is also decreasing and after reaching its minimum the primary pressure starts to increase. Later on, the primary pressure reaches the threshold for opening of the PRZ relief valve and around 3000 seconds there is a blow-down through the valve. At 6500 seconds the minimum SG level is reached, due to the larger water inventory in the secondary side. After SG's depletion, the heat transfer from primary to secondary side breaks down and the pressurizer relief valve opens and closes much faster,

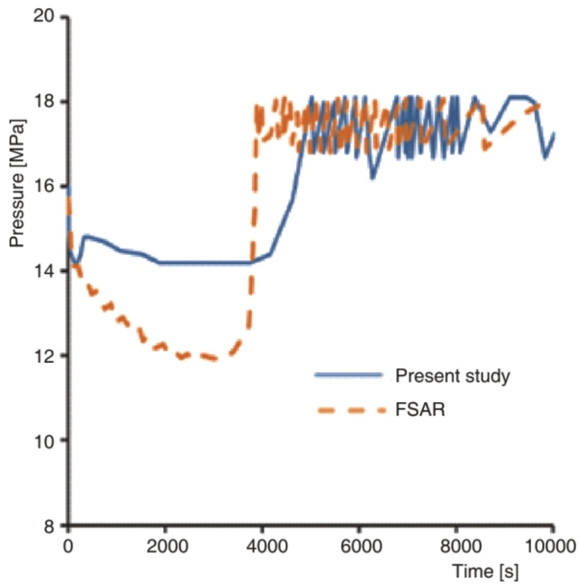

Figure 3. Primary side pressure

see fig. 6. With the actuation of the pressurizer relief valve the pressurizer level is increasing and when the level reaches the position of the relief valve, a transition from two phase to single-phase water flow through the valve can be observed.

Due to the increasing loss of primary coolant through PRZ relief valves, the primary mass inventory decreases and a RPV cover bubble. When the water level

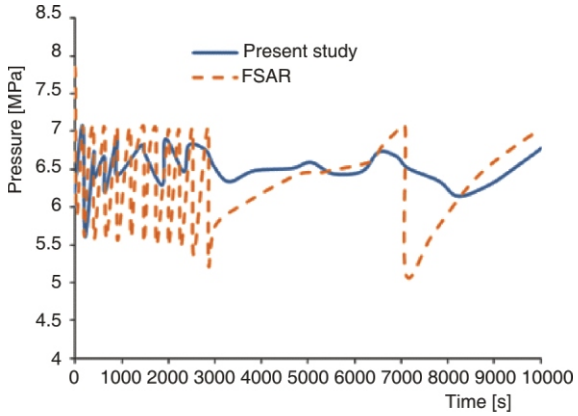

Figure 4. Secondary side pressure

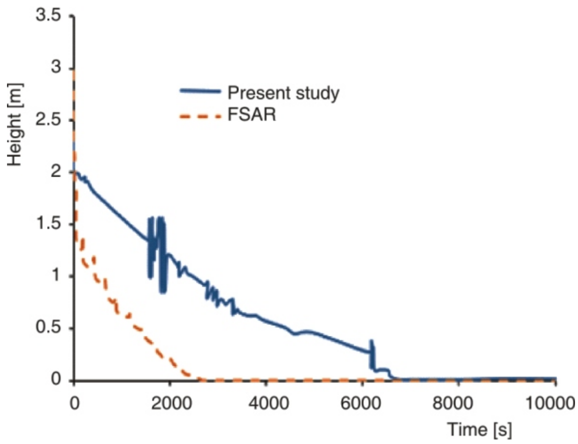

Figure 5. Water level of SG

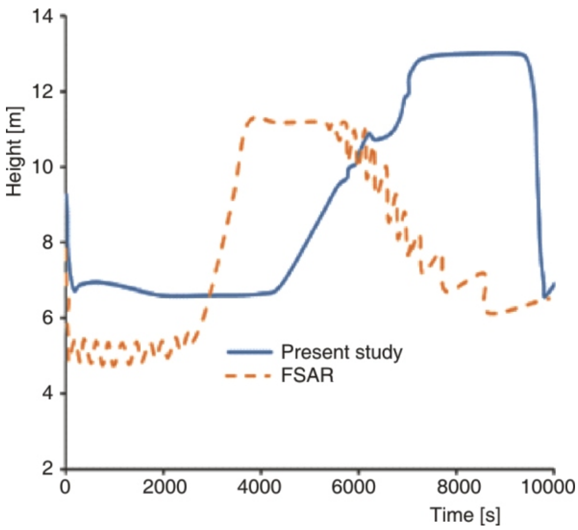

Figure 6. Water level of pressurizer

in reactor vessel drops below hot nozzles elevation, the natural circulation in the primary system is interrupted.

The results indicate that although there is a difference between the present study and the references, the behavior of the systems, as observed in diagrams, is nearly the same.

The main reason for these deviations is the use of different codes and models in these simulations. Also,

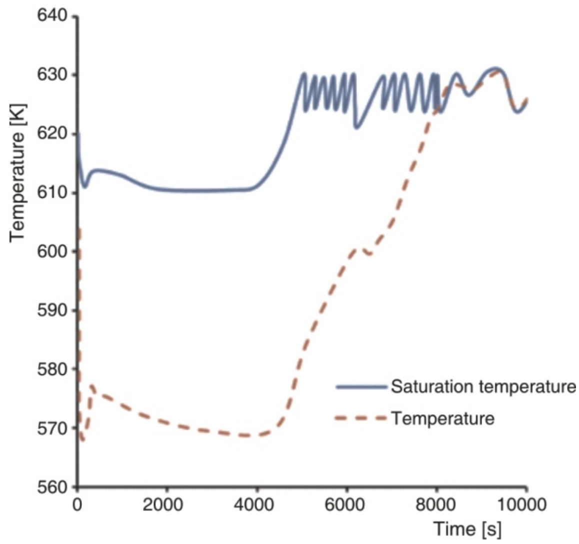

Figure 7. Temperature of hot leg of RCS and margin to the boiling temperature

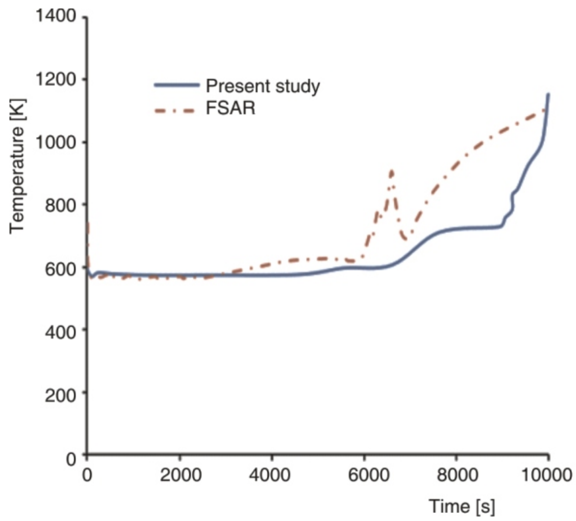

Figure 8. Fuel cladding temperature

in the present model, especially in the secondary side, only the SG and their safety systems are modeled, while in the reference, other components of the secondary side, including turbines, are also modelled.

As there is no water supply to the primary circuit, the core starts to heat-up, fig. 7. The reduced primary system mass leads to core uncover and its consequent dry out. But, the fuel cladding temperature does not exceed the safety margin of 1200 °C, fig. 8.

CONCLUSIONS

The study presented in this paper aims at the investigation of the thermal-hydraulic behavior of a generic VVER-1000 reactor in case of the total loss of AC power accident. The initial event leads to reactor scram, turbine trip, total loss of feed water, and trip of all main coolant pumps. As a consequence, the scenario results in a core heat-up under high pressure conditions. With the help of an accident management measure (primary side depressurization), the primary pressure can be reduced, so that the passive emergency cooling systems (accumulators) can inject water into the primary system and thus the start of an extended core dry-out can be significantly delayed.

Comparative analyses for the LOOP accident with the thermal-hydraulic code RELAP5 have been done for a reference VVER-1000, taking into account the plant specifics. The comparison of the results shows, that the general behavior, with respect to the main events and thermal-hydraulic phenomena, is very similar. The differences in the reactor power and in the construction of the SG (orientation, mass inventory, steam velocity, and steam mass-flow rate) are directly responsible for the different timing. A preliminary result of the comparative study is that the operators in the VVER-1000 NPP have more time to be prepared for accident management measures to prevent or mitigate possible core damage. Further analyses are needed to determine the differences in timing more precisely. Therefore, the influence of the SG model (especially the nodalization in vertical direction) and the influence of the accumulator injection (condensation rates, effect on primary pressure) on the course of events have to be investigated in more detail.

An additional simulation for the VVER-1000 reactor, with the earliest possible time for starting primary side depressurization, has been performed. For the simulation, the temperature criterion to start the procedure has been modified. With a core outlet temperature of about 350 °C, the fluid temperature reaches the saturation temperature and steam appears in the RPV. With the modified criterion primary side depressurization starts much earlier. Caused by the higher decay heat in the early phase of the transient, the primary pressure reduction after initiation of primary side depressurization is not so effective compared to the first simulation, of primary side depressurization at a core outlet temperature of about 650 °C. Compared to the first simulation, the accumulator injection starts earlier and at a higher level of the decay heat. As a consequence, the accumulators cannot inject the full amount of water and with the changed criterion, the cladding temperatures start much earlier to rise and then limit the safety margin of 1200 °C. As a conclusion from the results of the two simulations, it should be mentioned, that a start of the primary side depressurization procedure later in time is more effective.

Finally, it has to be pointed out, that without any active emergency cooling system a core dry-out, with increasing cladding temperatures, cannot be avoided *at all*. With the help of the primary side depressurization procedure, the start of an extended core dry-out can be significantly delayed and the core cooling is ensured for at least two hours, but to prevent severe core damage, the recovery of the electricity supply and the start of active emergency cooling systems, after depletion of the accumulators, is an essential safety requirement for both nuclear power plants.

AUTHORS' CONTRIBUTIONS

The computational models of the RELAP5 were developed by M. Esfandiari under the direction E. Zarifi, and figures were prepared by M. Esfandiari and E. Zarifi. All authors participated in the preparation of the final version of the manuscript.

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Мохсен ЕСФАНДИАРИ, Голамреза ЈАХАНФАРНИА,
Камран СЕПАНЛО, Ехсан ЗАРИФИ

АНАЛИЗА АКЦИДЕНТА ГУБИТКА СПОЉНЕ ЕНЕРГИЈЕ У НУКЛЕАРНОЈ ЕЛЕКТРАНИ VVER-1000/V446

Сврха овог рада је приказ термохидрауличке анализе акцидента губитка спољне енергије у VVER-1000/V446 нуклеарној електрани применом програмског пакета RELAP5. Губитак спољне енергије може довести до нерасположивости главних активних сигурносних система и до нарушавања сигурносних критеријума који обезбеђују сигуран рад нуклеарне електране, што би за последицу имало загревање и могућу деградацију језгра. Стога је веома битна анализа и испитивање рада електране у случају овог акцидента. У овом циљу, применом програмског пакета RELAP5, реактор је симулиран и испитина су различита понашања главних компоненти у примарном и секундарном колу система за хлађење реактора. Резултати показују разумну сагласнот са развијеним моделом и са коначним сигурносним извештајем нуклеарне електране Бушер.

Кључне речи: губитак спољне енергије, анализа сигурности, RELAP5, нуклеарна електрана VVER-1000