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Research paper

Thermal hydraulic analysis of the effect of emergency safety procedures for mitigation of accident conditions of pressurized water reactor nuclear power plants

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Abstract

The prime objective of the nuclear safety is the complete prevention of the release of radioactive materials anywhere. One of the great concerns to the public is the nuclear safety of nuclear power reactors, especially after the two catastrophic accidents of Chernobyl and Fukushima. If a nuclear power plant (NPP) accident occurs, it is difficult to protect the environment and public from the radiation exposure. The safety systems performance of the nuclear plants is a very vital and decisive factor in enhancing the use of nuclear energy. The present research work offers thermal hydraulic simulation analysis of loss of coolant accident and station blackout conditions of a pressurized water reactor (PWR). In this study, the thermal hydraulic is performed by using RELAP and PCTran codes to establish a model of this PWR which could simulate its primary system, with good accuracy in the case of loss of coolant accident and station blackout conditions. These simulations depict the thermal hydraulic analysis and the verification of the response and efficiency of the accident management procedures in make certain that adequate, efficient, and effective emergency core cooling and auxiliary cooling safety systems are sufficient to submit a powerful cooling capacity of the reactor core in the case of accident conditions and that it makes an active contribution to the mitigation of the significant undesirable consequences, and the importance of the emergency and auxiliary safety systems during the accident conditions.

Keywords: *RELAP and Patron Codes; Nuclear Safety; Station Blackout; LOCA; Thermal Hydraulic.*

1. Introduction

The safety of nuclear power plants rely greatly on the availability of amenable and continuous source of cooling water during all modes and situations of the plant operation. Nuclear energy remains an attractive available energy source which is capable of producing the large amount of energy required to satisfy the ever increased energy demand. However, safety of nuclear power plants will remain the most severe and serious concern to all.

In current designs of nuclear reactors the adopted safety standards and features are immensely strict and of predominant importance in order to reduce significantly the expected frequency of serious core accidents. Therefore, researches which supply more knowledge and information in reactor safety become of important plausibility. The present research lavishes knowledge in this field.

The utmost imposing events of nuclear power plants (NPPs) are the Station Blackout (SBO) condition which expresses the loss of all sources of AC electrical power to all plant equipment, which is decisive and essential to assure that the cooling system is capable and efficient enough to remove continuously the decay heat generated by the nuclear fuel in the reactor core after reactor shutdown, and the condition of loss of coolant that exhibits the loss of the primary loop cooling water in the reactor core[1], [2].

Station Blackout (SBO) leads to a prompt reactor trip with the control rods to be inserted by gravity consequent to loss of power. This action promptly shuts down the nuclear reactor [3].

One of the most limiting design-basis accidents is the loss of coolant accident (LOCA) in which the loss of coolant results in a failure to remove heat from the nuclear fuel in the core. Even small losses of fluid (or loss of coolant flow) could have undesirable serious consequences. The most serious severe accident is the total loss of the coolant [4]. This can lead to either partial or complete core meltdown.

This paper presents a comprehensive analysis of the behavior of a PWR NPP in two cases: the emergency core cooling system (ECCS) behavior of the reactor core for the primary circuit loss of coolant accident (LOCA), and Station blackout (SBO) initiating event occurring with reactor at full power and with the nuclear fuel at equilibrium condition. The analysis is done by employing PCTran code calculations. The rupture area is 200 cm^2. The results are significantly useful in the most important field of nuclear safety. Safety is the main serious concern for the world. Reactors are chosen mainly and primarily according to their sound reliable robust safety and control measures and systems.

2. Specifications of the reference PWR NPP

Pressurized Water Reactors are the most widely used worldwide for the production of electrical power (about 62% as per numbers and about 68% as per electric output power). The present selected reference PWR plant is a Westinghouse design one [5]. The overall power cycle consists of three generally independent closed cycles: primary, conventional, and tertiary [5].

The primary main coolant loop includes the reactor vessel where the energy is generated, reactor coolant pump, steam generator, pressurizer, and connected pipes between them [5]. The secondary conventional loop is the steam heating cycle where the steam generator produces the steam which is expanded in the turbine to convert thermal energy into mechanical power and then to electrical power by a generator [5].

3. Nuclear power applications engineering computer codes

The RELAP thermal-hydraulic code is designed to calculate the thermal-hydraulic response of the overall reactor primary coolant system (RCS), core damage progression, release of fission products, and transport criteria during severe conditions and accidents. The code is capable to deal with simulations of steady-state, transient, and accident conditions which include Loss of Coolant Accident (LOCA) and many types of transients condition in Light Coolant Water Reactors (LWRs) [6]. The RELAP5 thermal-hydraulic code is based on two-fluid models allowing for unbalanced temperatures and velocities of the fluids [7].
PCTran offers comprehensive analyses of nuclear reactor accidents. The cladding, coolant, and containment shielding radioactive leakage is detected by controls and animated monitors. This expresses in depth understanding of the science and technology of nuclear reactors and energy with the aim to protect the plant and public [8].
Software PCTran package can simulate many transient and accident conditions for nuclear power plants. A mimic of the Nuclear Steam Supply System (NSSS) and containment monitors the status of important elements and parameters and allow simulations of operator actions by interactive control [8].PCTran is a fruitful reliable simulator for all types of nuclear reactors including PWR, BWRAP1000,ABWR, and ESBWR [8].

4. Model description

The reference nuclear power plant considered in this work is a Pressurized Water Reactor (PWR) type. It is a Westinghouse design reactor [5]. Figure 1 depicts description of the nuclear power plant cycles and details of nuclear fuel components. The plant electrical power output is about 1000 MWe, in which light water duplicates as coolant water and moderator of the reactor. The PWR is designed as a four-loop plant but the simulation combines each two loops in one. The main technical data of the reference plant are given in Table1.

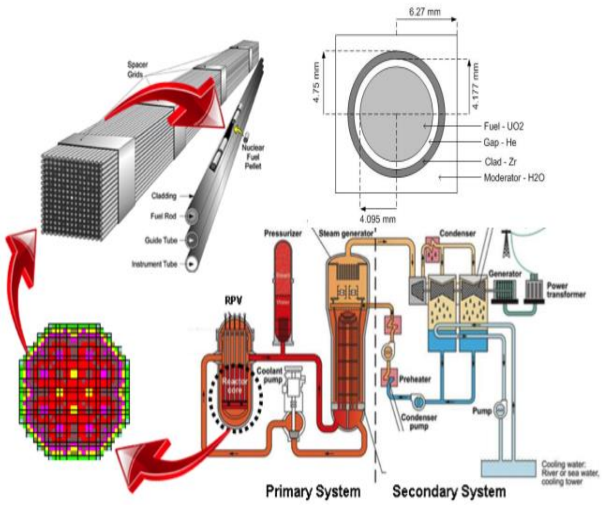

Fig. 1: The Nuclear Power Plant Cycles and Nuclear Fuel Details [5].

Table 1:.Main Technical Data of The Reference Plant [5]

Parameter	Value	Unit
Thermal power reactor output	3322.3	MWt
Electrical power output	1000	MWe
Reactor pressure	155	Bar
Reactor temperature	565.7	K
Total coolant mass flow rate	18630	kg/s
Core mass flow rate	17700	kg/s
Fuel active length	3.66	M
Fuel rod diameter	10.6	Mm
Total number of fuel rods in the core	50952	Fuel rods
Number of fuel assemblies	193	Fuel ass.
Number of fuel rods per assembly	264	Fuel rods

5. Nodalization of the thermal hydraulic model

The RELAP nodalization diagram of the cycle in the present simulation is shown in Fig.2 for the reactor pressure vessel and the primary loop and it is based on the design and operating data of the specific component.
Figure 2 illustrates the nodalization diagram of the PWR NPP RELAP code model. The calculated results from the RELAP code, in the steady state, were confirmed with the main reference design technical parameters in Table 1. The nodalization diagram is modeled for the components of the PWR NPP during steady state condition. The components include the reactor pressure vessel, 2 hot legs, 2 cold legs, 2 Reactor Coolant Pumps (RCPs), 2 Steam Generators (SGs), one Pressurizer (PZR), surge line pipe, and main feedwater and auxiliary feedwater systems.
Our model contains 87 hydrodynamic nodes and 5 heat structures (HSs). All the loops are self-reliant modeled. All loops have one steam generator (SG) that includes both the secondary and primary sides with heat exchange structures. The steam generator has both inlet and outlet plenums which are modeled as separate branches. Nodalization of the secondary side is limited to the SG downcomer and riser, the SG dome, and the main live steam line. Each loop has a reactor coolant pump (RCP) which is like for each loop with all actual characteristics. The coolant water flow rate through the core is about 17,700 kg/s. The model of primary system has one pressurizer in the first loop as indicated in the Fig. 2, it also has two valves, one relief valve, safety valve, and connections between components of hot, cold and intermediate legs and between the outlet of SG and RCP.

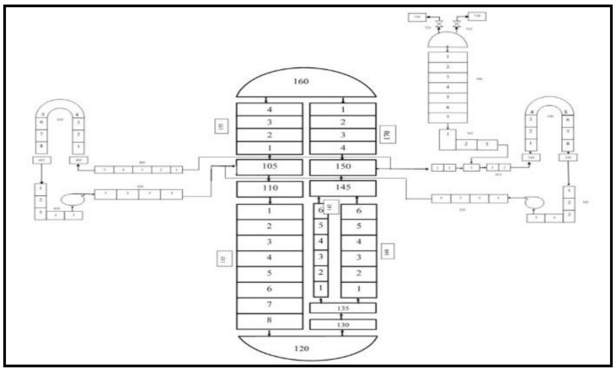

Fig. 2: Simplified Nodalization Diagram of the Primary System.

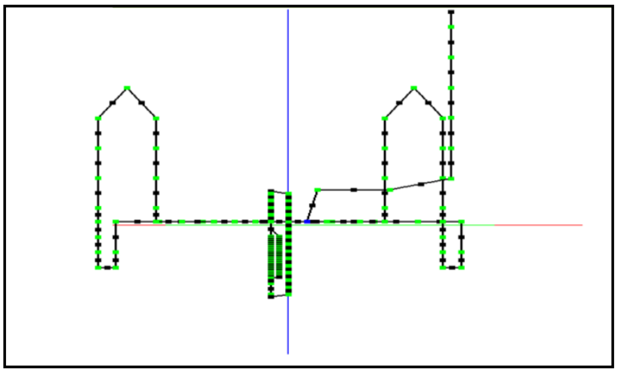

Fig. 3: The RELAP Code Nodalization Model of the System.

6. Layouts of PCTran code PWR components

A picture of the layouts of the PWR components on PCTran for the present simulation are shown in Fig.4.The estimated results from the PCTran code, in transient condition, were endorsed with the reference design parameters in Table 1. The PCTran layout diagram is modeled for the required components during LOCA condition, which contains the reactor vessel, hot legs, cold legs, Reactor Coolant Pumps (RCPs), Steam Generators (SGs), Pressurizer (PZR), surge line pipe, Auxiliary Feedwater Lines, Turbine island emergency feed water systems (EFWS), emergency core cooling systems (ECCS), High pressure injection system (HPI), Low pressure injection system (LPI), and Accumulator (ACC)).

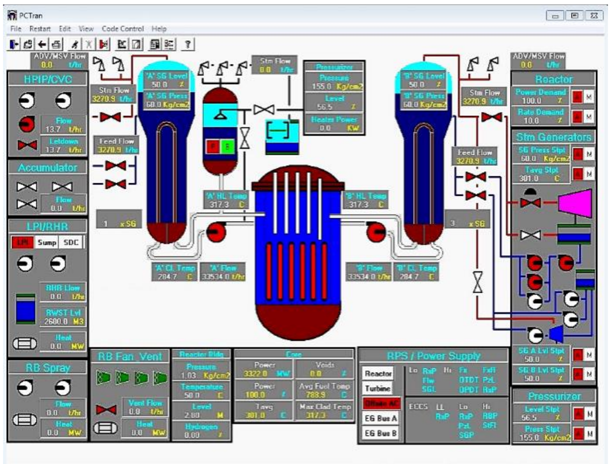

Fig. 4: The Layout of PWR PcTran Components [8].

7. Results and discussion

7.1. LOCA simulations

The accident simulation was conducted by means of modeling the PWR and the emergency safety systems deploying the PCTran code in order to describe the emergency core cooling systems (ECCS) response and core behavior under severe accident conditions. The time sequence of events for accident progression is tracked in Table 2.

Table 2: The Time Sequence of Events of the Accident Progression

Event	Time
Break Beginning	0
Reactor scram from RCS pressure (pressure ~ 13.22 MPa)	5
High pressure safety injection starts (HPSI) (pressure ~	10
Accumulators injection start (ACC) (pressure ~ 43 bar)	130
Low pressure safety injection start (LPSI) (pressure ~ 11.36	410
Calculation terminated	2000

When the reactor coolant system (RCS) is at a pressure of about 13.22 MPa, the reactor protection system signal is generated to trip the reactor, and this takes place at 5.5 sec. The control rods are inserted to begin the fast trip of the reactor (i.e. terminates the fission process and the reactor thermal power generation), and the ECCS starts to supply water to the RCS. The mass flow rate of the initial break is higher than the injection rate of the emergency safety injection systems, therefore, the coolant reserve of the RCS is reduced continuously, which leads to a collapsed liquid levels in the reactor and the pressurizer.

Accordingly, the emergency core cooling systems (ECCSs) begin to supply water when the RCS pressure drops to 12.97 MPa, then the initial pressure of the high pressure safety injection (HPSI) system starts to inject cooling water into the RCS at 10 sec. When the RCS pressure falls down to the accumulators initial pressure of 43 bar, the accumulators start to inject borated water into RCS at 130 sec. Following that, and when the RCS pressure drops to 11.36 bar, which is the initial pressure of the low pressure safety injection (LPSI) system, LPSI starts to inject cooling water into the RCS at 410 sec. The safe reactor and pressurizer levels are recovered after about 600 seconds after the beginning of transient.

After about 2000 sec transient condition initiated, and the volume of water injected by the ECCS is appropriate to compensate for the loss of coolant water through the break leakage. The ECCS injection is now enough to keep the fuel and cladding temperatures within the safe limits. At the beginning of the transient, the cladding temperature starts to increase, reaching a peak of 789.3 °C at 10 sec, associated with fast prompting of protection and control system. This temperature increase does not reach rates beyond the allowed temperature limits and thus, the reactor core integrity is safeguarded and secured.

Figure 5 exhibits the reactor total megawatt thermal power with time. Figure 6 gives the thermal core power level and power neutron flux level during LOCA conditions with time. The reactor trip is done by inserting the control rods in the reactor and stops the fission chain reaction and hence the reactor thermal power generation. The decay heat generation in the reactor core produced from the fission products continues, but then decreases by cooling the reactor core.

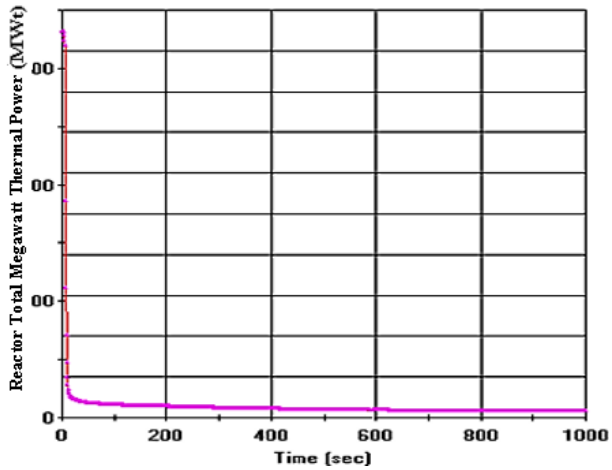

Fig. 5: The Reactor Total Megawatt Thermal Power During LOCA Accident Conditions.

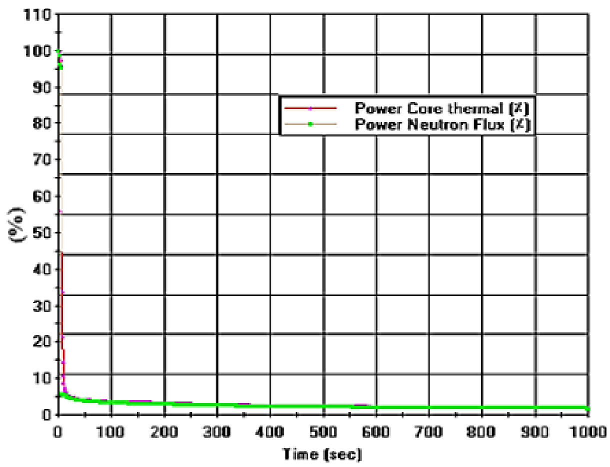

Fig. 6: The Thermal Core Power Level, And Power Neutron Flux Level during LOCA Accident Conditions.

Figure 7 demonstrates the coolant flow rate of break during LOCA conditions with time. The flow through the break reaches a maximum value of approximately 3300 kg/s after 5 sec when in the beginning the reactor coolant pressure is high, later on, it has a decreasing trend as the steam depressurizes as shown in the figure.

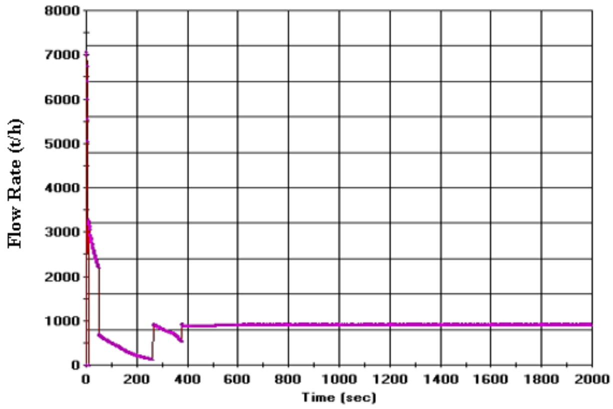

Fig. 7: Flow of Break during LOCA Accident Conditions.

Figure 8 presents the total flow of ECCS during LOCA conditions with time. The ECCS contains three main systems: HPI, accumulator, and LPI. The ECCS injection was ample and sufficient to keep the fuel and cladding temperatures within the required safe limits. The combined coolant flow of these ECCS systems is observed in the figure.

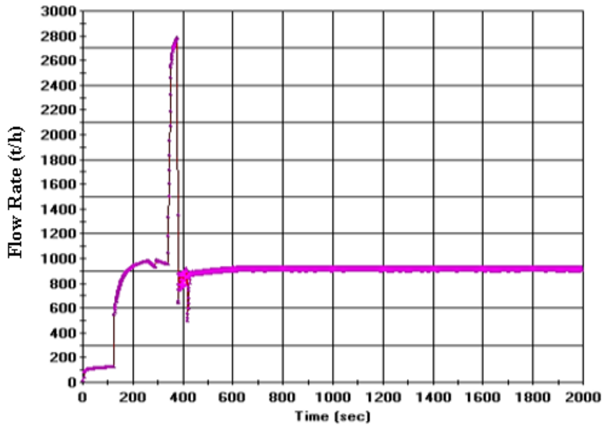

Fig. 8: Total Flow of ECCS during LOCA Conditions.

Figure 9 shows the reactor pressurizer level during hot leg LOCA accident conditions with time. The pressurizer level drops sharply from its steady state value of 56.5% to zero at 30 sec, and the pressure reaches 120 bar, then the volume of the system's cooling water decreases as a result of the break leakage in the leg. The emergency core cooling systems operate and supply cooling water to the RCS, and this action refills the pressurizer and increases its level after 600 sec.

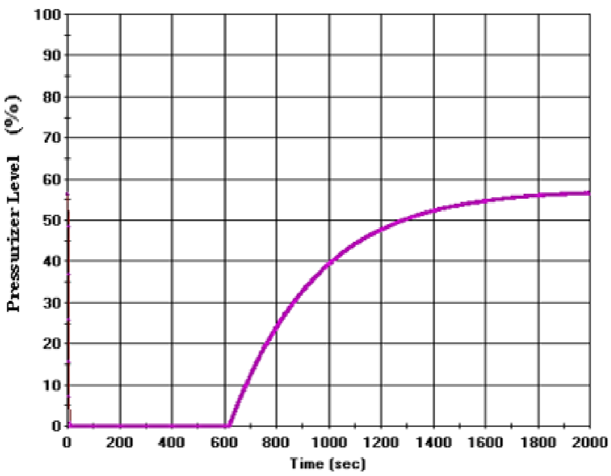

Fig. 9: Pressurizer Level during LOCA Conditions.

Figure 10 represents the average fuel and clad temperatures during LOCA conditions with time. As seen, these temperatures are decreased after the LOCA accident occurrence because of the sufficient safety measures adopted and the operation of ECCS.

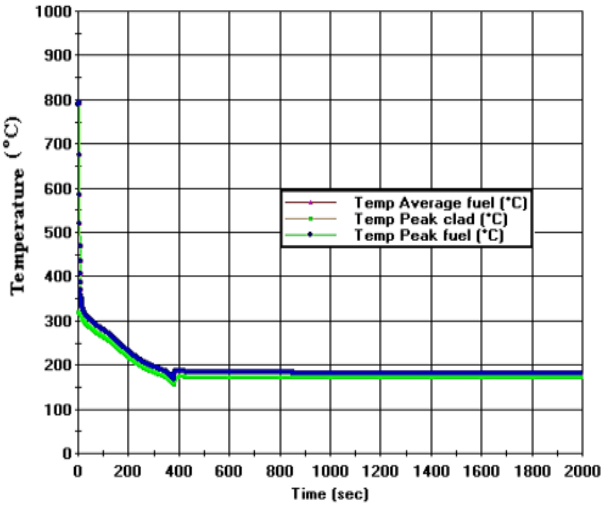

Fig. 10: Average Fuel and Clad Temperatures during LOCA Conditions.

7.2. Station blackout simulations

For thermal hydraulic simulations of the PWR NPP, the RELAP code is used for thermal power of the plant of about 3322.3MWt. The following figures show the results given by the thermal hydraulic computer RELAP code for the battery capacity time which provides electrical power source to the control systems.

Figure 11is a plot of the power level of the reactor, (reactor fission power, total reactor power, and reactor power from decay of fission products and actinides), during station blackout condition and reactor trip time. Control rods are dropped in the reactor core to shut down the reactor power and stopping the fission chain reaction. The decay heat from the fission fragments continues to generate but decreases with cooling the reactor core.

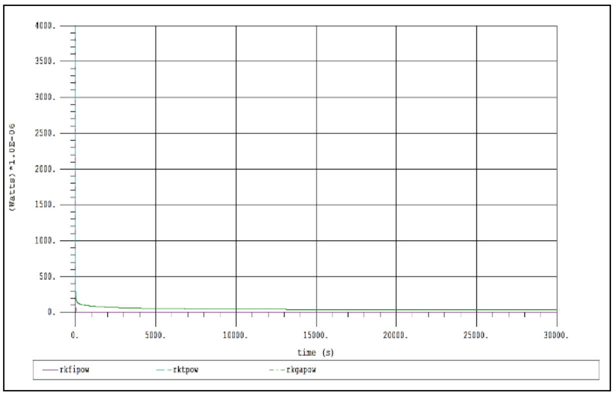

Fig 11: Reactor Fission Power, Total Reactor Power, and Total Reactor Power from Decay of Fission Products and Actinides During SBO.

Fig. 12 describes the leakage rate of the secondary circuit with time after SBO condition. Opening safety relief valves allows leakage of an amount of steam from the system.

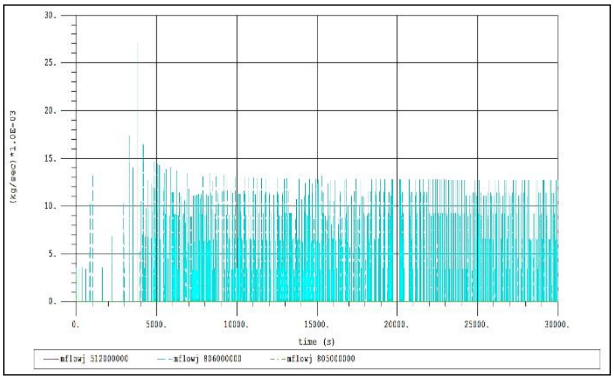

Fig. 12: The SG Safety Relief Valves A Leakage Rate During SBO.

Figure 13 shows the pressure fluctuation of the secondary circuit with time. When the pressure increases until it reaches that of the valve set point, the SG safety relief valves open. The pressure increases due to feedwater temperature rise.

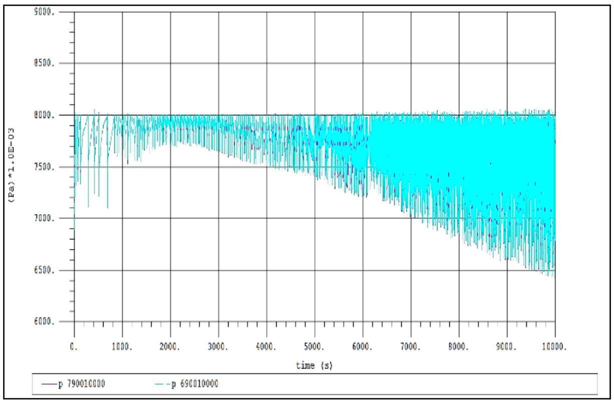

Fig. 13: Pressure Fluctuation of Secondary Circuit after SBO Condition and the SG Safety Relief Valves Opening.

Figure 14 depicts the axial distribution temperature progression of fuel and cladding temperatures at different axial levels for the heat structure with time. As shown, these temperatures get increased after the occurrence of the SBO accident, and then become stable and decrease because of the adequate safety measures acquired together with the operation of TD AFWS in this case.

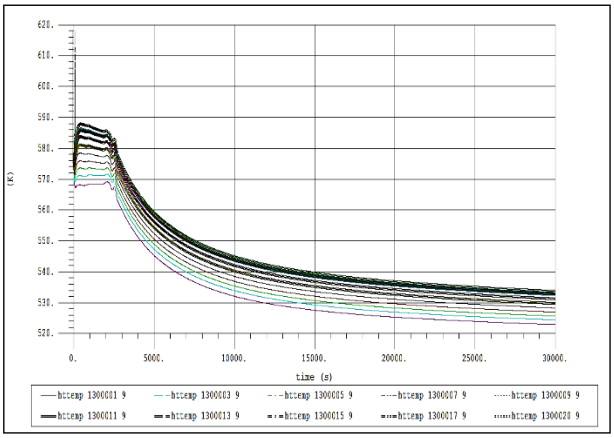

Fig. 14: Time Progression of Fuel and Cladding Temperature at Different Axial Levels.

8. Conclusions

RELAP thermal hydraulic model for the primary system of the studied pressurized water reactor is deployed. The model is tremendously important for the studies of the safety analysis for nuclear power plants. The present results obtained from the reactor simulation's behavior during LOCA and SBO conditions can demonstrate and reveal the reactor behavior in real life. The results are prominently important in the field of safety of nuclear reactors. The target of these simulation analyses is to prove whether the analyzed nuclear reactor with the safety systems is able to comply and cope with the accident sequences and what are the prospective consequences of an accident. The thermal hydraulic simulation results play a supreme role in designing, licensing, and operating the nuclear power plant.

Management mitigation procedures have been evaluated against LOCA consequences. Therefore, the LOCA of the PWR is examined to evaluate and check the efficiency and the effectiveness of the mitigation procedures. The study of the severe accident loss of coolant accident (LOCA) concentrates on the performance and response of emergency safety systems of PWRs. The simulation analysis of the 200-cm^2 break explains the reactor behavior during hot leg loss of coolant accident conditions.

The analysis of the hot leg break reveals that the ECCS can provide enough cooling to avoid damage or meltdown of the reactor's core. In the long term, the ECCS keeps RCS filled to remove the decay heat partly by the break flow. Furthermore, the core must be kept responsive to cooling during and after the event. The results confirm the good actuation of the ECCS, insuring the integrity of the nuclear reactor core.

The results of the analysis of the station blackout accident indicate that the TD-AFWS can offer adequate cooling to avoid damage of the reactor core. Therefore, TD-AFWS of PWRs are decisive and main action to mitigate the station blackout undesirable serious consequences.TD-AFWS provide additional time for the operator to salvage and restore the electrical power source to prevent the damage of the reactor core. It is recommended that the extension of the power capacity of batteries can be an active measure to continue the operation of AFWS in order to mitigate the extended time of station blackout.

The RELAP and PCTran codes proved adequate and sufficient to simulate the behavior of PWR NNPs during accident conditions LOCA and SBO.

The present results are beneficial in the foremost concern subject in nuclear energy which is nuclear safety.

Additional procedures should be adopted to mitigate the adverse effects of accident consequences. For instance the new reactor designs use safety injection systems to cool down the reactor core, e.g. the use of coolant water spent fuel pool, or redundancy and diversity kinds of electrical power sources.

9. Nomenclature

HS	Heat structures
INEEL	Idaho National Engineering and Environmental Laboratory
LOCA	Loss of Coolant Accidents
LWRs	Light Water Reactors
NPP	Nuclear power plant
NRC	Nuclear Regulatory Commission
PWR	Pressurized water reactor
RC	Reactor coolant
RCPs	Reactor coolant pumps
RCS	Reactor coolant system
RPV	Reactor pressure vessel
SG	Steam generator
UO_2	Uranium Oxide
ZrO_2	Zirconium oxide

10. Conflict of interests

The authors declare that there is not conflict of interests regarding the publication of this paper

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