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Research paper

The performance and response of additional passive auxiliary cooling system for safety of pressurized water nuclear reactors at unanticipated transients conditions

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Abstract

The main function of nuclear safety of nuclear power reactors is to furnish assured removal of decay heat to an ultimate heat sink after safe shutdown of the reactor. Nuclear safety is the most challenging issue in nuclear energy; it is in fact of the utmost concern to all. Station Blackout (SBO) is one of the representative important accidents related to the electric power source in nuclear power plants. This research paper is concerned with thermal-hydraulic simulation of an additional secondary passive auxiliary feedwater system of a pressurized water reactor nuclear power plant (PWR NPP). The purpose is to analyze the performance, efficiency, and response of the behavior of the present additional secondary passive auxiliary feedwater system in case of a station blackout (SBO) unanticipated transients conditions. SBO could be one of the worst nuclear accidents. In this study, the thermal hydraulic analysis is performed by using RELAP thermal hydraulic code to develop a model of this PWR in the case of station blackout unanticipated transient conditions. The main conclusions of this study are that the additional secondary auxiliary passive feedwater safety system is adequate, efficient and effective as a heat sink to provide a reliable strong cooling capability of the secondary steam generator in the case of beyond design basic conditions. The newly added auxiliary cooling system affirms its capability to make an active contribution to the mitigation of the significant undesirable consequences during the station blackout accident. The presence of this cooling system is rather important for more safety assurance of nuclear reactors.

Keywords: *Thermal-Hydraulic; RELAP Code; Nuclear safety; Station Blackout; PWR.*

1. Introduction

New design options are currently being pursued to improve tremendously the safety systems of future nuclear power plants to reduce the probability of accident occurrence to almost nil. It must be noted that nuclear accidents are the most intimidating in the field. The increase share of nuclear energy in global energy consumption is intimately connected, among other factors, with nuclear safety and control. The chief concern in new designs of nuclear reactors is nuclear safety. Emphasis is on simpler and easier operating systems with passive or intrinsic characteristics which would ensure continuous cooling of fuel and its containment systems. These features would enable the nuclear power plant to survive potential severe accidents without fuel damage and human intervention. Safe shutdown of the reactor is an important safety function which must be absolutely ensured to avoid and mitigate accidents of severe nature. Most of the shutdown systems rely on active components like instrumentation signals and power signals although made fail safe and partially passive [1].

The primary job of the nuclear safety is the prevention of the release of radioactive materials, and ensuring that the operation of nuclear power plants does not contribute significantly to individuals and social health risk. The main specific issue of the nuclear safety is the need for removing the decay heat, which is necessary even after reactor shutdown [2].

The performance of nuclear safety systems is a very important factor in evaluating sustainability of nuclear energy, as it can improve the environmental indicator used to evaluate the overall nuclear energy continuality. The basic principle of nuclear safety is to ensure that nuclear reactors do not damage public and/or individual health plus protect the nuclear power plant and its employees [2].

One of the representative accidents related to the electric power in nuclear power plants is a Station Blackout. SBO which is initiated by a loss of all offsite power with a concurrent failure of both emergency diesel generators. With no alternate power source, most of the active safety systems that perform safety functions are not available. To prevent scenarios of SBO from becoming worse, a safety class Alternative AC power is installed as a redundancy to provide electricity source to the vital equipment that perform safety function [3].

After the Japanese Fukushima Daiichi nuclear accident, mitigation measures against extended SBO sequences were investigated around the world [4]. After this accident, the station blackout of nuclear power plants has widely attracted researcher's concerns of the residual heat-removal problem again [5].

The present research is concerned with the subject of nuclear reactor safety. The main objective of the present work is to study the effect of a newly additional secondary passive auxiliary feedwater system (ASPAFS) on the nuclear fuel cooling in case of a station blackout unanticipated transients conditions of a PWR. SBO if occurred can lead to a catastrophic accident. This adds value to the present work. This analysis is carried out by using the best estimate thermal hydraulic computer code.

2. Characterization of the reference pressurized water reactor

Pressurized Water Reactors are the most dominantly used in the world to produce electrical power. The chosen reference PWR plant is taken from one of the Westinghouse designs. The total power cycle comprises three generally independent closed cycles: primary, conventional, and tertiary.
The primary cycle includes a reactor vessel where the energy produced from the controlled fission reaction is transformed to the coolant water. The hot coolant is pumped to the steam generator (SG) where the thermal heat is transferred to a secondary loop. A Pressurizer is connected to the primary hot leg to keep the system pressure above the saturation pressure to avoid water boiling (in contrary to Boiling Water Reactors). The secondary cycle is the heat cycle utilization where the steam generated in the SG flows to a steam turbine to produce mechanical energy and then electrical power.
Thermal-hydraulic codes for the purposes of the safety analysis for nuclear power plants are used as crucially important codes. Analyses done on the fundamentals of the computer simulation results are those of the steady and transient operational states. The goal of such analyses is to indicate if the analyzed nuclear object with the available safety systems is able to withstand an accident sequences and what are the potential consequences of an accident, along with a related timescale [6].
The simulation results play a key role in designing, licensing and operating the nuclear power plant. The codes are required because nuclear power plant systems work at a highly sophisticated level that surpasses the capabilities of the human mind and simple, basic theoretical models. Safety analysis relies on conservative principles and requirements for system design and operation. Meeting those requirements ensure a high reliability level, stating that the risk associated with plant operation for workers and society is reasonably low and feeble. With the increasing quality of data and the models implemented, it is possible to create a more realistic thermal-hydraulic analysis, which uses data from probabilistic codes to choose the most probable accident scenario [6].
The thermal-hydraulic codes used for safety analysis need to be adequately verified and validated in order to ensure their credibility and reliability. Verification describes the accuracy of the translation of physical equations to the computer code language. Validation determines the correctness of the mathematical models, which have to be a realistic representation of the system. Validation is usually performed by comparing the results obtained from the model and experiments. The validation process shows uncertainties and inaccuracies in models, which need to be taken into consideration later in the safety analysis process. In thermal-hydraulic codes, the aim is to determine modeled system parameters, in terms of fluid mechanics and heat transfer of the materials in the analyzed domain [7].
The RELAP is one of the significant thermal hydraulic codes which is designed to express thermal hydraulic modeling of the reactor primary coolant system (RCS), progression of core damage, release of fission products, and transport during severe conditions and accidents [8]. This code is able to perform simulations of steady-state, transient, and accident transient conditions which include Loss Of main Coolant Accident (LOCA) and many transients' types in Light Water Reactors (LWRs) [8].

3. System description and modeling

The reference nuclear plant in this work is a pressurize water reactor type. The plant is equipped with a pressurized water cooled reactor with an electrical power output of about 1000 MWe, in which light water serves as both coolant and moderator. The PWR NPP configuration is designed as a four loop plant but simulation combines each two loops in one loop (take into account the new dimensions of hot and cold leg pipes, steam generators tubes, RCP flowrate, and pressure head, results from this combination). A simplified diagram of the nuclear power plant in the present work is illustrated in Fig. 1. The main technical data of the plant are depicted in Fig. 1 [9]. The proposed additional passive auxiliary cooling system is shown in the figure.
Modeling diagram of the RELAP code of the PWR cycle adopted for the simulation is depicted in Figure 1 for the primary loop and it's components based on the design and operating data of the component.
The nodalization diagram is modeled for the components of the PWR NPP during transient condition, which contains 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 and auxiliary feedwater systems.
The current model consists of 87 hydrodynamic components and 5 heat structures (HSs) (between the core fuel and coolant, and the secondary and primary steam generators). All loops have one SG that contains both the secondary and primary sides with heat exchange structures. The SG has both outlet and inlet plenums which are modeled as separated branches. Nodalization of The secondary side is limited to the SG down comer and riser, the dome, and the main steam line. Each loop has an identical RCP and contains actual characteristics. The cooling water flow rate through the core is about 17,700 kg/s. The model of the primary system has one pressurizer in the first loop which has two valves, relief valve and safety valve.

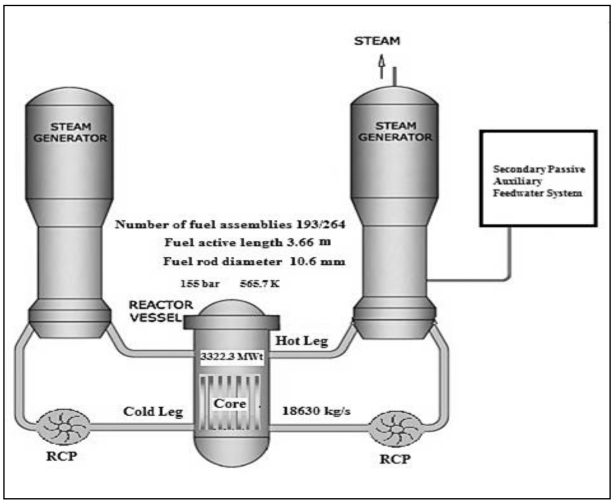

Fig. 1: Simplified Schematic Diagram of the PWR Plant and Its Main Technical Data.

4. Description and operation of the additional passive auxiliary feed water system

The additional secondary auxiliary passive feedwater system and its nodalization diagram are indicated in Fig.2. This system is a passive one which means that it does not need any power to operate. This system is active, powerful, and sufficient during unanticipated and anticipated accidents conditions specially station blackout which means loss of all sources of electrical power and steam flow sources. This system operates during unanticipated station blackout condition when there is no any cooling source available, thus this system operates under the effect of pressure difference across the valves which open when the pressure increases above the pressure set point of the valve. Here the cooling water goes through the tank to the steam generator as heat sink to remove the heat from the primary cooling water which removes the decay heat generated in the reactor core and ensure the stability and integrity of the core. The system components include tank , piping system connections, and set of valves as described in Fig. 2.

The additional secondary auxiliary passive feedwater system model nodalization, which is shown in Fig. 2, consists of the tanks (600) which are connected to pipe components with ten axial sub volumes (905). Feedwater is distributed to two trip valves ((910) and (915)). The trip valves are connected to pipe components with ten axial sub volumes (920) which distribute feedwater to a mechanical valve (925) which controls the water flowrate then connected to the SG to provide the steam to the generator secondary side cooling to remove the heat generated in the nuclear fuel in the core which is transferred to the steam generator through the reactor coolant system.

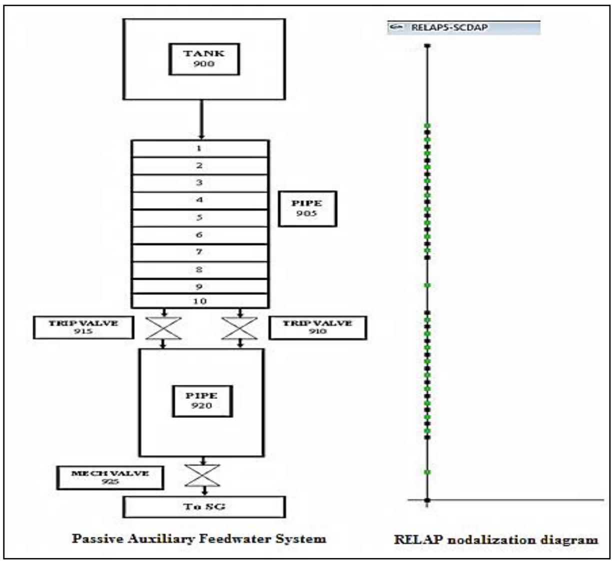

Fig. 2: Simplified Diagram and RELAP Nodalization Diagram of the Additional Passive Auxiliary Feedwater System.

5. Results and discussions of the thermal-hydraulic simulation of unanticipated transients conditions

The results obtained by the RELAP thermal hydraulic computer code are presented in the subsequent figures for studying the effect of the additional auxiliary passive secondary feedwater system on the nuclear fuel cooling in case of a station blackout unanticipated transients conditions of the considered PWR.

Figure 3 gives the RCP flow system resulting from the sequences of the unanticipated station blackout condition with time. Station blackout condition, which means loss of all sources of (AC) power, leads to RCP trip and stops the flow rate of the cooling water of the reactor primary loop.

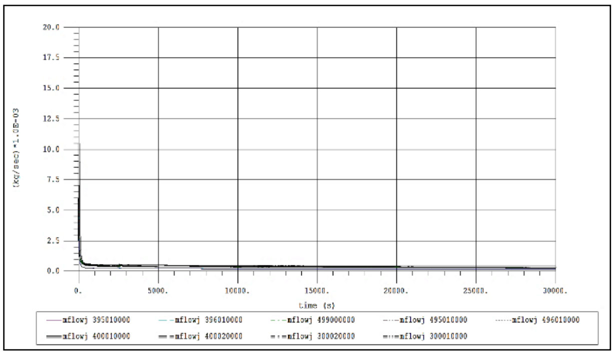

Fig. 3: RCP Primary Coolant Flow System.

Figure 4 indicates the power reactor level with time, (total reactor power, reactor fission power, and total reactor power from decay of fission fragments and actinides), during unanticipated station black-out condition. Here the reactor trip results on that control rods drop in the core and shutdown the reactor and stops the fission process. The heat decay from the reactor core fission products continues to generate, but decreases with cooling the reactor core.

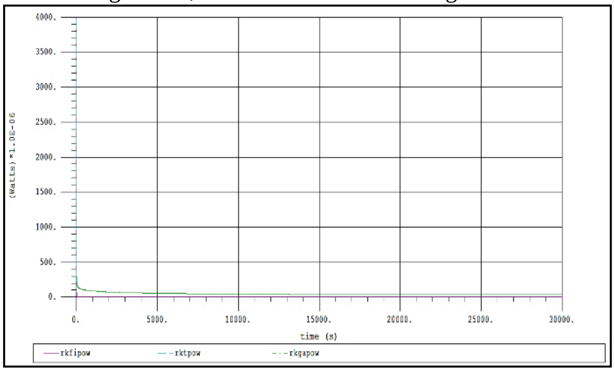

Fig. 4: Total Reactor Power, Reactor Fission Power, and Total Reactor Power from Decay of Fission Fragments and Actinides during SBO.

Figure 5 illustrates the primary cycle pressure and the consequence of opening of the pressurizer relief valves and SG dump valve with time. The pressure increases after the SBO condition because of the rise of the coolant temperature and then decreases when the relief and safety valves open as they reach their set points, and hence decrease the pressure. This action also operates the ASPAFS to cool the SG secondary side.

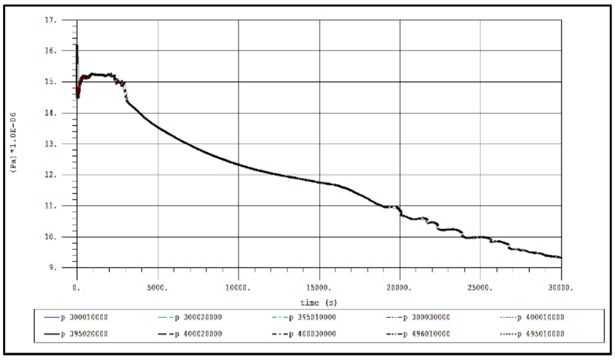

Fig. 5: Reactor Primary Pressure during Station Blackout Accident.

Figure 6 shows the leakage rate of the secondary loop after the SBO condition and the amount of leakage during the opening of the relief safety valves with time, where the heat was transferred from the primary loop coolant to the secondary feedwater. As the primary coolant temperature increases, due to the heat decay generated in the reactor core, the feedwater temperature increases in the SG secondary side which leads to increasing the pressure in the SG. The SG relief safety valves open if the pressure increased until attaining the valves set point which leads to leak of an amount of SG steam.

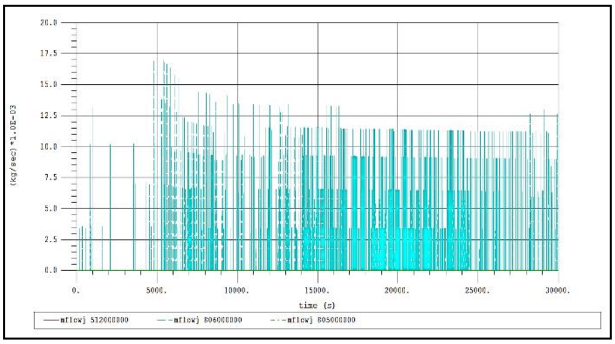

Fig. 6: The SG Relief Safety Valves A Leakage Rate During SBO Event.

Figure 7 exhibits the pressure fluctuation with time. When the primary coolant temperature increases due to the heat decay generated in the reactor core which results in feedwater temperature increase in the SG secondary side which leads to increase in the pressure in the SG. The SG relief safety valves open if the pressure increased until attaining the valves set point.

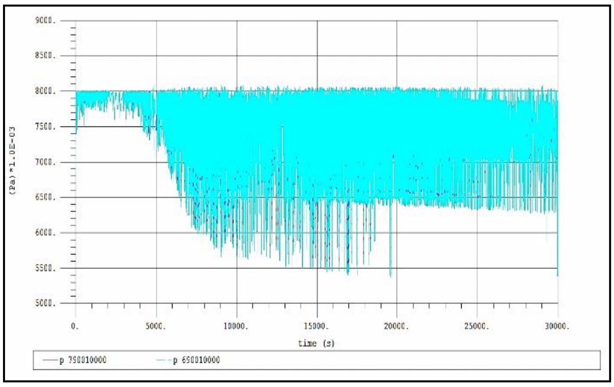

Fig. 7: Pressure Fluctuation of Secondary Circuit after SBO Condition and the SG Relief Safety Valves Opening.

Figure 8 gives the outlet and inlet temperatures of the pressure vessel with time. The difference results in the heat decay of the reactor core. The temperatures decrease due to the additional passive feedwater safety system which remove the heat generated in the primary coolant.

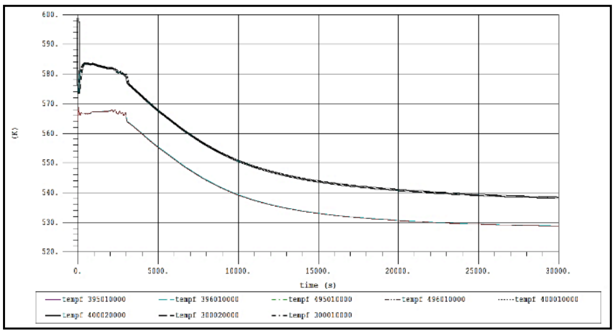

Fig. 8: Outlet and Inlet Temperatures of the Reactor Pressure Vessel.

Figure 9 represents the axial distribution temperature progression of the cladding and fuel temperatures at different axial levels (from bottom to top of fuel rod) for the heat structure with time. As seen, these temperatures increased after the SBO accident occurrence, and then become stable and then decrease because of the sufficient safety measures taken and the operation of ASPAFS in this case.

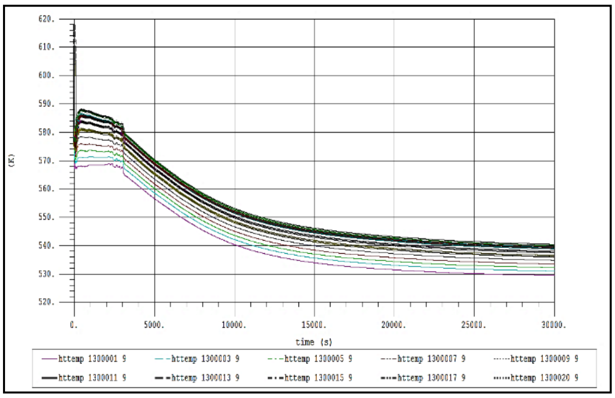

Fig. 9: Time Progression of the Cladding Temperature at Different Axial Levels.

Figure 10 introduces the core temperature behavior along the fuel assembly after the SBO condition with time. Normal operation of the ASPAFS provides a heat sink source to remove the heat from the primary to the SG secondary side and keep the core covered with the reactor coolant. During the ASPAFS operation period, the operators have ample time for restoration of the (AC) electrical power source.

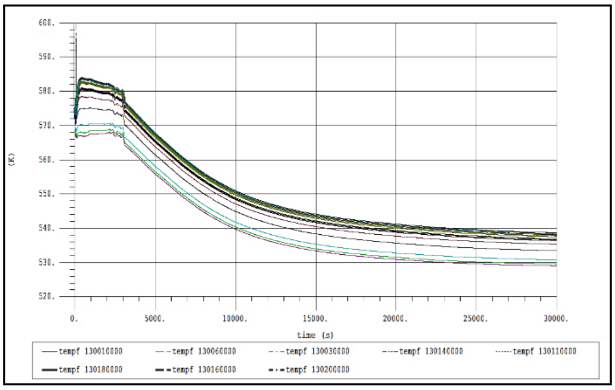

Fig. 10: The Core Temperature along Fuel Assembly after SBO Condition and the ASPAFS Operation.

Table 1 compares the main parameters in cases of non-existence of the auxiliary cooling system [10] with the present case of the additional passive auxiliary feedwater system of the PWR nuclear power plant during the unanticipated loss of AC electrical power source. After the loss of all AC power electrical sources, all components of the primary loop stop functioning, so the control rods drop in the reactor core and terminate the chain reaction and hence the power generation. Despite stopping the fission process, still there is the decay heat generation from the fission fragments due to which the temperature and the pressure of the primary coolant increase without any safety measures. When the additional passive auxiliary feedwater system operates the temperature and pressure decreases due to the removal of the decay heat generated in the reactor primary coolant. This is clear from the results in Table 1.

Table 1: Comparison of Main Parameters for with and Without the Auxiliary Cooling System

Parameters	Without auxiliary system	With auxiliary system
Thermal Power Level (%)	100 – 5	100-5
RCS pressure (bar)	155 – 170	155- 10
RCS Temperature (K)	585- 620	585-540

6. Conclusions

Thermal-hydraulic simulation model of unanticipated transients conditions for the pressurized water reactor primary system was developed by using engineering and technical data from the reference Westinghouse pressurized water reactor. Simulations of the reactor performance during unanticipated conditions can describe its behavior in real operation. For the simulated case, the analyzed and examined parameters demonstrated that the model could successfully depict the reactor performance in transient conditions. The results are incredibly important in the field of nuclear reactors safety.

The simulation results indicate that the additional secondary passive auxiliary feedwater system is capable of removing the decay heat in the reactor core by passive means with good sufficient safety margins. The cladding surface temperature was found to remain well within the safety limits during the whole transient conditions. Furthermore, the core must be kept responsive to cooling during and after the event. The results show the good actuation of the second auxiliary passive feedwater system, guaranteeing the integrity of the nuclear power reactor core. It is recommended that the present auxiliary cooling system to be incorporated in nuclear reactors as an extra safety measure for mitigation or even prevention of severe nuclear accidents.

It is foreseen that new designs of reactors should investigate anticipated and unanticipated accident conditions to ensure sufficient cooling of the reactor core and reach steady state conditions.

7. Nomenclature

AC	Alternative current
DC	Direct current
DDP	Diesel driven pumps
EDG	Emergency diesel generators
HS	Heat structures
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
SBO	Station blackout
TD-AFWS	Turbine driven auxiliary system
TDP	Turbine driven pumps
UO2	Uranium oxide
ZrO2	Zirconium oxide

8. Conflict of interests

The authors declare that there is no conflict of interest regarding to the publication of this paper

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