ANALYSIS OF THERMAL HYDRAULIC BEHAVIOR OF KONVOI PWR DURING A DESIGN EXTENSION CONDITION

by

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Scientific paper
https://doi.org/10.2298/NTRP201209004H

In nuclear power plants the design extension conditions are more complex and severe than those postulated as design basis accidents, therefore, they must be taken into account in the safety analyses. In this study, many hypothetical investigated transients are applied on KONVOI pressurized water reactor during a 6 inches (182 cm²) cold leg small break loss-of-coolant-accident to revise the effects of all safety systems ways through their availability/non-availability on the thermal hydraulic behaviour of the reactor. The investigated transients are represented through three cases of small break loss-of-coolant-accident as, Case-1, without scram and all of the safety systems are a failure, Case-2, the normal scram actuation with a failure of all safety systems (non-availability), and finally Case-3, with normal actuation scram sequence and normal sequential actuation of all safety systems (availability). These three investigated transient cases are simulated by creating a model using analysis of thermal-hydraulics of leaks and transient code. In all transient cases, all types of reactivity feedbacks, boron, moderator density, moderator temperature, and fuel temperature are considered. The steady-state results are nearly in agreement with the plant parameters available in previous literature. The results show the importance of the reactivity feedback effects in loss-of-coolant-accident on the fallouts power as they are considered the key parameters for controlling the clad and fuel temperatures to maintain them below their melting point. Moreover, the calculated results in all cases show that the thermal-hydraulic parameters are in acceptable ranges and encounter the safety criterion during loss-of-coolant-accident design extension conditions accidents processes. Furthermore, the results show that the core uncover and fuel heat up do not occur in KONVOI pressurized water reactor the design extension conditions simulations, as all safety systems provide adequate core cooling by sufficient water inventory into the core to cover it.

Key words: loss-of-coolant accident, reactor scram, design extension condition, core uncover, thermal-hydraulic phenomena, safety injection four-loop of KONVOI PWR

INTRODUCTION

Design extension conditions (DEC) are postulated accident conditions that are not considered for design basis accidents (DBA), but are considered in the design process for the facility by best estimating methodology. The DEC comprise conditions in events without significant fuel degradation and conditions in events with core melting, according to SSR-2/1. The safety of nuclear power plants (NPP) during DEC is one of the most important topics which must be demonstrated before the issuance of the operating licenses [1]. One of the DEC events is anticipated transient without scram (ATWS). An essential emergency core cooling system (ECCS) is installed to cope with those types of accidents and prevent their propagation to a beyond design basis accident (BDBA). Despite the best estimate codes that are used today in the safety assessment of NPP, the ECCS performance is still assessed against the same criteria, such as the peak clad temperature less than 2200 °F (1200 °C), maximum local clad oxidation less than 17 % and core wide both the oxidation less than 1 % [2-5].

In the German risk study of loss-of-coolant-accident (LOCA), the reactor cooling system (RCS) inventory varies throughout events depending on the number of safety injection pumps (SIP) in operation and the discharge rate via the break, which is itself a function of brake size, RCS pressure and coolant condition at the break location. Many transient conditions in the RCS may temporarily occur in the RCS if the leakage rate at high pressure is greater than the injection rates of the safety-injection pumps as a result of a coolant reduction in RCS inventory. In particular, if the SIP are postulated to operate at reduced availability, the decay heat from the core is then transported to the steam generators (SG) [6].

A small break is sufficiently large that the primary system depressurizes to the high pressure safety injection set point and a safety injection signal is generated, automatically starting the high pressure safety

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injection (HPSI) system. Breaks smaller than 3/8 inch in diameter do not depressurize the reactor coolant system because the reactor charging flow can replace the lost inventory [7]. The leaks with sizes from 40 cm^2 to 380 cm^2 were judged as not controllable with the implemented measures of the core cooling systems and lead to a core melting accident [8]. A simulation of LOCA has been carried out using the TRACE code to investigate the effect of reverse flow restriction device (RFRD) on the flow rate as well as peak clad temperature of BWR fuel bundles during three different LOCA scenarios: small break LOCA (25 % LOCA), large break LOCA (100 % LOCA), and double-ended guillotine break (200 % LOCA). These results demonstrated that the user device could substantially block flow reversal in fuel bundles during LOCA, allowing for coolant to remain in the core during the coolant blowdown phase, and showed that it could retain additional cooling water after activating the emergency systems, which maintains the peak clad temperature at lower levels. Also, the RFRD achieved their flood phase earlier than without the RFRD [9].

In this study, SBLOCA is chosen as a basic transient with a 182 cm^2 break, in one of the cold legs. The transient occurred using investigated transients DEC that are applied for the reactor safety analyses. In the safety analyses, the effects of the reactor total reactivity and the safety injection systems (active and passive) availability, for the reactor safety during DEC are studied. The DEC are simulated with and without scram (ATWS) during SBLOCA. The simulations are performed for the KONVOI PWR reactor by creating a new model using ATHLET code [10]. In ATHLET Model, the neutronic and the thermohydraulic calculations of the reactor are involved in the steady and the transient states for the behavior of the reactor core, over the total reactivity effects.

REFERENCE PLANT

The German 4-Loop PWR (GKN II) type KONVOI which belongs to the Kraftwerk Union (KWU) [11] is the reactor selected to study the thermohydraulic behavior of its core during SBLOCA safety analysis, by creating a simulation model using AHLET code. In the model, the fuel type selected to be core contains 193 fuel assemblies (FA) with thermal power of 3850 MW and electric power of 1300 MW. Each fuel assembly is arranged in 16 × 16 arrays and includes 236 fuel rods. The general information about the reactor, assembly; and fuel is shown in tabs. 1 and 2, respectively.

Table 1. General reactor data

Parameter	Value
Power [MW]	3850
Mass flow rate [kgs^{-1}]	19874
Mass flow rate in core (94 %) [kgs^{-1}]	18682
T-prim out [°C]	326
T-prim in[°C]	290
P-prim [MPa]	15.8
Average FA-power [MW]	19.95
P-sec [MPa]	6.4
T-sec [°C]	262

Table 2. Fuel assmbly features

Parameter	FA type	
No of fuel assembly	16 × 16 – (20)	18 × 18 – (24)
No of rods per assembly	256	324
Fuelled	236	300
Unfuelled	20	24
Overall assembly length [m]	4.83	4.83
Overall assembly width [m]	$2.3\cdot10^{-1}$	$2.3\cdot10^{-1}$
Rod outside diameter [m]	$1.08\cdot10^{-2}$	$9.50\cdot10^{-3}$
Rod length [m]	4.40	4.40
Pellet length [m]	$1.10\cdot10^{-2}$	$9.80\cdot10^{-3}$
Pellet outside diameter [m]	$9.11\cdot10^{-3}$	$8.05\cdot10^{-3}$
Pellet density [gcm^{-3}]	10.45	10.45
Clad thickness [m]	$7.25\cdot10^{-4}$	$6.40\cdot10^{-4}$

THE ATHLET MODEL

Establishing nodalization

The simulated reactor is a generic German PWR (KONVOI type), which has four loops and produces 1300 MWel of power. During the creation of the ATHLET Model, the four loops are simulated by only two, three identical loops are represented by intact one loop and the other loop is the broken loop. Each separated loop consists of a steam generator U-tubes that are connected to the hot and cold legs, which a spray line at the top of the pressurizer is simulated. The cold legs are connected to the downcomer of the reactor pressure vessel (RPV), which lies parallel to the core channels and the reflector, all connected at the bottom by the volume of the lower plenum. On top of the core channels, the upper plenum is simulated, as well as the connection to the hot legs as shown in the nodalization scheme fig. 1.

THE REACTOR SAFETY SYSTEMS

The KONVOI pressurized water reactor (PWR) have two safety systems that are represented in the following two sections:

Passive safety system (hydro-accumulators)

The KONVOI PWR have one passive safety system which consists of eight hydro-accumulators,

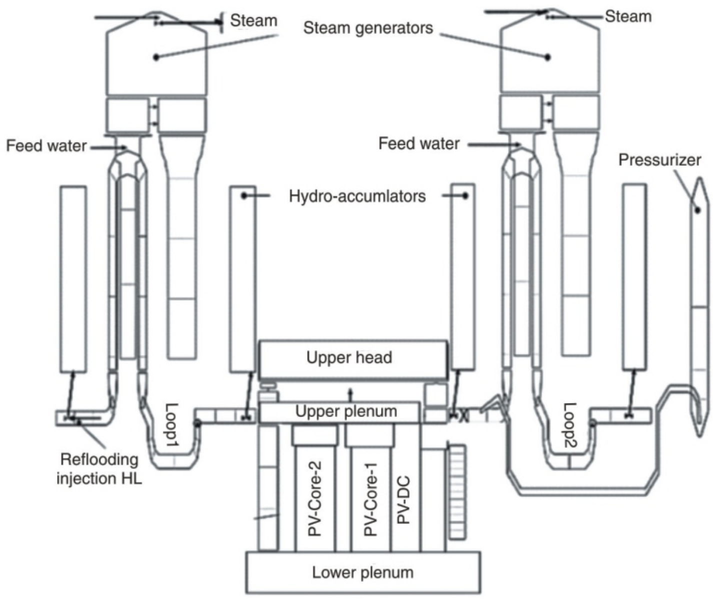

Figure 1. Graphical representation of the simulated reactor structure with an X indicating the location of the leak

one accumulator for each leg, with a water volume of 45 m^3 and a set point valve below 2.6 MPa.

Active safety system, emergency core cooling systems

The reactor has two active emergency core cooling systems (ECCS), high pressure injection system, (HPIS) and low pressure injection system (LPIS) in the primary loops that are connected to the cold legs. The HPIS starts actuation below 11 MPa, with a flow rate of 77 kgs^{-1} per pump where LPIS starts its injection below 1.1 MPa with a maximum flow rate of 325.4 kgs^{-1} per pump. There is an additional active ECCS called extra borating system (EBS) modeled with a constant non-pressure dependent injection mass-flow rate of 2 kgs^{-1} per pump, [11-13]. In the simulation ATHLET Model, all active and passive injection systems are represented as fills.

Steady-state safety analysis

For the reliability of code transient simulation, a Nodalization qualification step is performed. After a steady-state run that is extended for 300 seconds for the stabilization of the model results. The calculated values of the main parameters are compared with the corresponding nominal values of the reference plant [11].

Steady-state conditions for ATHLET model and TRACE code

The reactor Parameters of the steady-state conditions and the comparison are outlined in tab. 3 and the difference as a percentage of the reference value is presented in the last column.

Table 3. The reactor parameters of the steady-state conditions

Parameter	Reference value [11]	Calculated value	Difference [%]
Reactor parameters			
Total power [MW]	3850	3752	–2.54
Core inlet temperature [°C]	290	291.34	0.238
Core outlet temperature [°C]	326	325.94	–0.010
Primary pressure [MPa]	15.8	15.94	0.88
Total coolant flow rates [kgs^{-1}]	8682	18665.24	–0.0897
Steam generator parameters			
Steam flow/SG [kgs^{-1}]	512	512.8499	–0.166
Steam pressure [MPa]	6.4	6.3825	–0.27

Comparison of the steady-state results between ATHLET and TRACE models

Tables 4 and 5 and figs. 2 and 3 show the steady-state results of the two models (ATHLET and TRACE).

The comparison results outlined in tabs. 4 and 5 and figs. 2 and 3 show that there is agreement between the ATHLET Model steady-state results with the corresponding published results of the TRACE model [11].

Transients safety analysis

Accidents description and assumptions

The investigated transients for DEC safety analysis are represented with many hypothetical accidents

Table 4. The ATHLET model steady-state results

Axial nodes	Pressure [Pa]	T_{sat} [°C]	T_{liq} [°C]	T_g [°C]	Velocity [ms^{-1}]	Density [kgm^{-3}]
NODE1	$1.599560 \cdot 10^7$	347.4091	291.346	346.5923	4.644351006	744.2761841
NODE2	$1.599031 \cdot 10^7$	347.3326	291.3445	347.3326	4.619028091	744.2545166
NODE3	$1.598922 \cdot 10^7$	347.3004	294.6073	347.3004	4.57022047	737.9349365
NODE4	$1.597376 \cdot 10^7$	347.222	298.9415	347.222	4.611373425	729.2983398
NODE5	$1.595815 \cdot 10^7$	347.1428	304.0352	347.1428	4.666882515	718.7607422
NODE6	$1.594240 \cdot 10^7$	347.0628	309.4505	347.0628	4.735877514	706.9955444
NODE7	$1.592645 \cdot 10^7$	346.9818	314.7078	346.9818	4.815169334	694.8841553
NODE8	$1.591038 \cdot 10^7$	346.9001	319.3903	346.9001	4.895985603	683.3596802
NODE9	$1.589434 \cdot 10^7$	346.8185	323.2041	346.8185	4.957458019	673.3207397
NODE10	$1.587860 \cdot 10^7$	346.7384	325.9886	347.4091	4.974225044	665.5239868
NODE11	$1.585420 \cdot 10^7$	346.614	325.9789	346.7384	4.957675934	665.4677734
NODE12	$1.585383 \cdot 10^7$	346.6121	325.9801	346.614	4.909605026	665.4630737

Table 5. The TRACE code steady-state results [11]

Axial nodes	Pressure [Pa]	T_{sat} [°C]	T_{liq} [°C]	T_g [°C]	Velocity [ms^{-1}]	Density [kgm^{-3}]
NODE1	$1.594357 \cdot 10^7$	347.2	292.9	347.2	4.530	741.0
NODE2	$1.59347 \cdot 10^7$	347.1	295.9	347.1	4.552	743.9
NODE3	$1.59258 \cdot 10^7$	347	299.8	347	4.590	726.7
NODE4	$1.59168 \cdot 10^7$	347	304.8	347	4.642	716.2
NODE5	$1.59078 \cdot 10^7$	346.9	310.5	346.9	4.710	703.9
NODE6	$1.58986 \cdot 10^7$	346.9	315.7	346.9	4.792	691.5
NODE7	$1.58894 \cdot 10^7$	346.9	320.1	346.9	4.878	680.0
NODE8	$1.58801 \cdot 10^7$	346.9	323.3	346.9	4.961	670.9
NODE9	$1.58708 \cdot 10^7$	346.8	325.8	346.8	4.028	663.7
NODE10	$1.58614 \cdot 10^7$	346.8	327.1	346.8	4.083	659.4

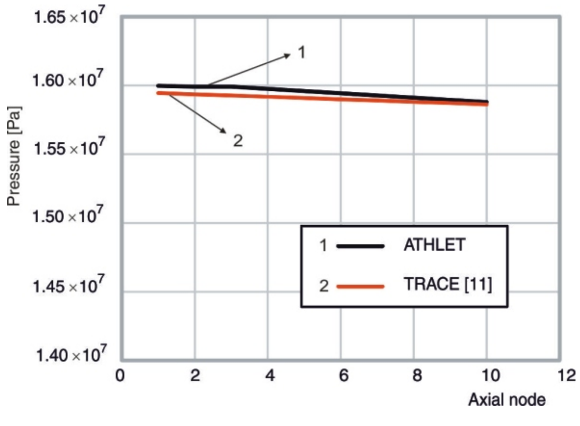

Figure 2. Axial pressure

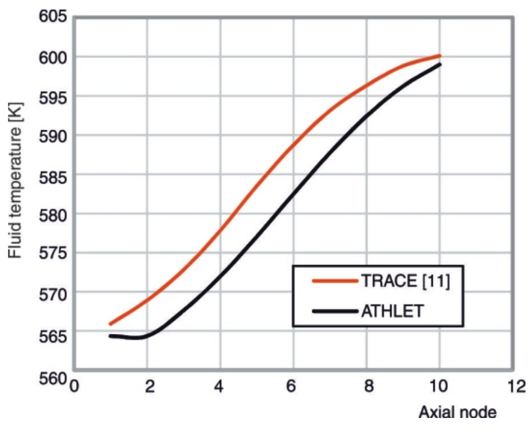

Figure 3. Fluid temperature

that happen in the KONVOI PWR during SBLOCA to study the effects of all safety systems throughout their availability/non-availability on the thermohydraulic behavior of the reactor. SBLOCA is assumed to occur in the cold leg of one of the loops as shown in fig. 1. The break size is a 6 inches in diameter. The transient initial conditions are represented in tab. 6 while the reactor operates at 100 % of the nominal power.

The investigated transients for DEC safety analysis are represented with three cases of hypothetical accidents during SBLOCA as the following three sections:

Case-1: The SBLOCA, without scram and non-availability actuation of all safety systems (active and passive)

In this case, SBLOCA with anticipated transient without scram (ATWS) and without all safety systems (ECCS and accumulators) is simulated, as DEC is con-

Table 6. Imposed sequence of events involved in this transient with their set points

Imposed event	Time/set point
Steady-state normal operation [s]	0-50
Break initiation [s]	at 50
Reactor scram signal	Pressurizer pressure + DP containment
Reactor coolant pump stop/main feed water stop	Reactor scram signal
Main steam valve closure	Reactor scram signal
Low safety injection start [MPa]	Pressurizer pressure1.1
Auxiliary feed water system in the intact or broken loops start/stop	LEV-S2DC
High safety injection start [MPa]	Pressurizer pressure 11 [MPa]
Accumulator injection start [MPa]	Pressurizer pressure 2.6 [MPa]
End of transient [s]	55000

sidered and used the general control simulation module (GCSM), which is a part of the ATHLET model input deck [10].

Case-2: The SBLOCA, with scram and nonavailability actuation of all safety Systems

In this case, SBLOCA with normal scram actuation through control rods insertion is simulated, with nonavailability actuation of all the safety systems as DEC.

Case-3: The SBLOCA with reactor scram and availability actuation of all safety systems

In this case, SBLOCA with normal scram actuation through control rods insertion is simulated, with the availability of all the safety (active and passive) systems as DEC.

RESULTS AND DISCUSSION

Comparison between Case-1 and Case-2

The comparison between the results of Case-1 and Case-2 are represented, to show the effect of reactor scram, with and without the availability actuation of all active and passive safety systems, on the behavior of KONVOI reactor during SBLOCA, DEC.

Comparison of the output scram signals of the reactor and the pump speed

Figure 4 shows the comparison of the output scram signals of the reactor for the two cases. In Case-2 the pumps coast down occurred due to the reactor scram as shown in fig. 5, where the relation of pump speed with time.

Comparison of the primary pressure and the thermal core power of the reactor

The primary pressure and the thermal power results from the two cases comparison are represented in figs. 6 and 7, respectively.

Figure 6 shows that, after the initiation of the break which is extended for a period of around 1000 seconds a sharp decrease in the primary pressure in Case-1 faster than which occurred for the pressure in Case-2, but after this time the pressure in Case-1 is higher than in Case-2. These results have occurred although there was no scram of the reactor in Case-1, the main pumps still running and the availability of the scram in Case-2 as shown in fig. 6. Moreover, fig. 7 illustrates that in Case-1 the reactor power is decayed after a period of 247 seconds due to the reactivity feedbacks (boron, doppler, and coolant) but, in Case-2 the power is decayed immediately (76 seconds) due to the reactor scram. At the same time, the two figures show that in Case-1 the run of the model calculations does not extend to the end time (5500 seconds) but stops at 3800 seconds, owing to the inability of the model for no scram.

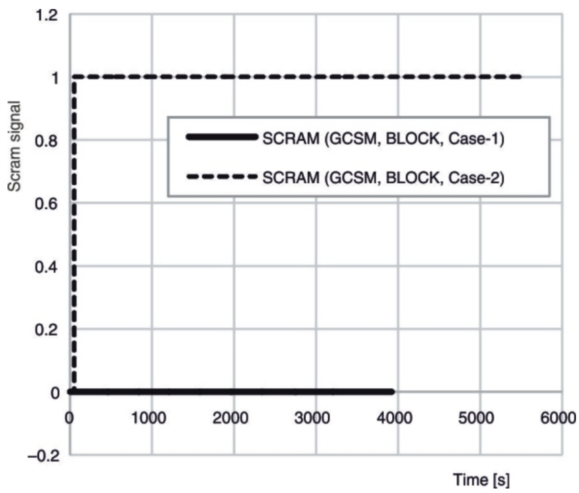

Figure 4. Scram signals

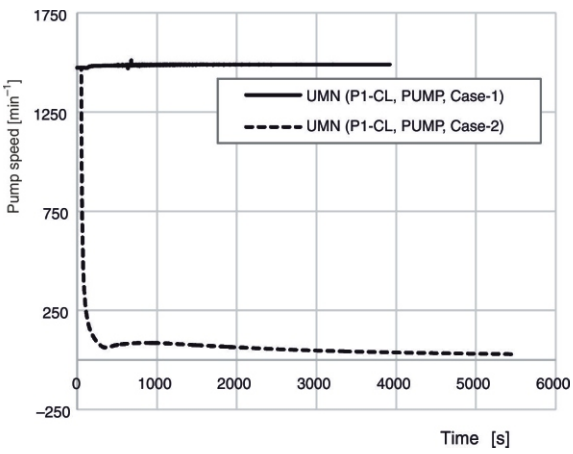

Figure 5. Pump speed

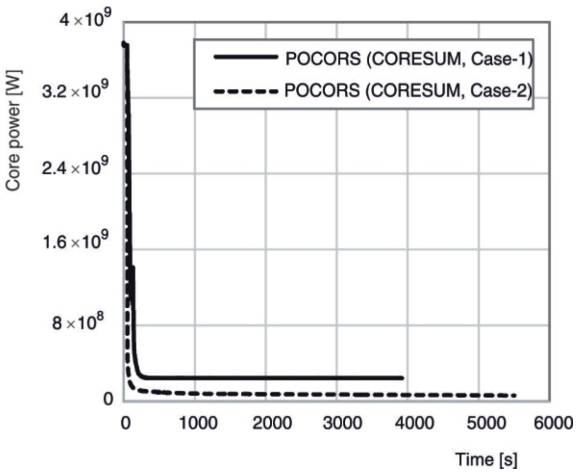

Figure 6. Primary pressure

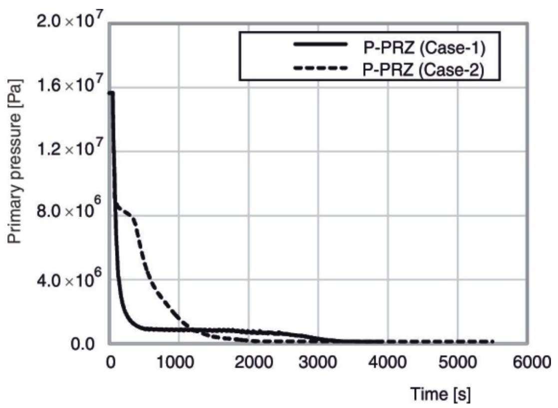

Figure 7. Core power

Comparison of the maximum clad temperatures of the reactor

Figure 8 represents the comparison for maximum clad temperature between the two cases with different run times 2000 seconds and 5500 seconds.

Figure 8(a) shows that after the initiation of break that is extended to 1800 seconds a slight increase in the maximum clad temperature in Case-1 lower than which occurred in Case-2 then it fast increases from 800 seconds to 2000 seconds with the values 300 °C to 760 °C, respectively. But in Case-2, it starts to increase sharply from 750 seconds to 2000 seconds with the values 350 °C to 2500 °C, respectively. The two figures show that the maximum clad temperatures in Case-1 still increase more than in Case-2. The results, as shown in figs. 6 and 7, occurred owing to the no scram of the reactor but the main pumps still running for about (247 seconds) in Case-1 and the availability of the scram in Case-2. Moreover, fig. 8(b) which the run time is extended to 5500 seconds illustrates that in Case-1 the maximum clad temperature kept rising then the model stopped to fail as discussed in the next section but in Case-2 the model succeeded in calculation all the run time with the maximum clad temperature.

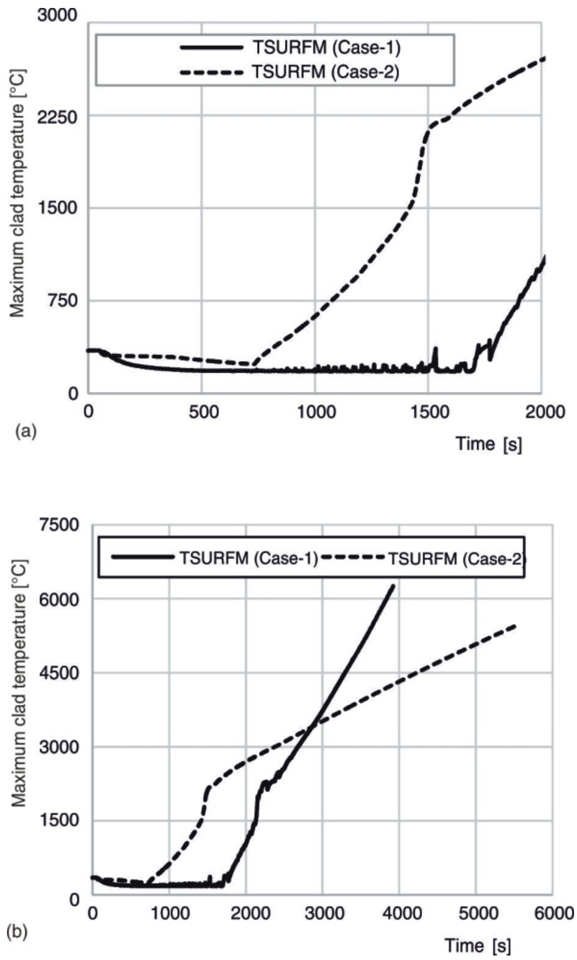

Figure 8. Maximum clad temperature with a different run time

Comparison of the primary and secondary pressures

Figures 9(a) and 9(b) represent the comparison of the output primary and secondary pressure between Case-1 and Case-2.

Due to the availability of the main pumps the primary pressure remains higher than the secondary pressure as shown in fig. 9(a). Consequently, the effectiveness of the SG as a heat sink and the establishment of natural circulation in the primary loop have continued and the clad temperature reached 750 °C at 1800 seconds. In Case-2 the primary pressure decreases gradually and becomes lower than the second pressure so the SG become inactive, a heat sinks as shown in fig. 9(b) since the Maximum clad temperature reached is 750 °C at 1000 seconds.

Comparison of the reactor voids

Figures 10 and 11 show a void formation Case-1 and Case-2 during depressurization on the primary side. A void formation in the RPV dome may occur if the fluid in the RPV, as shown in fig. 10, is still hot and

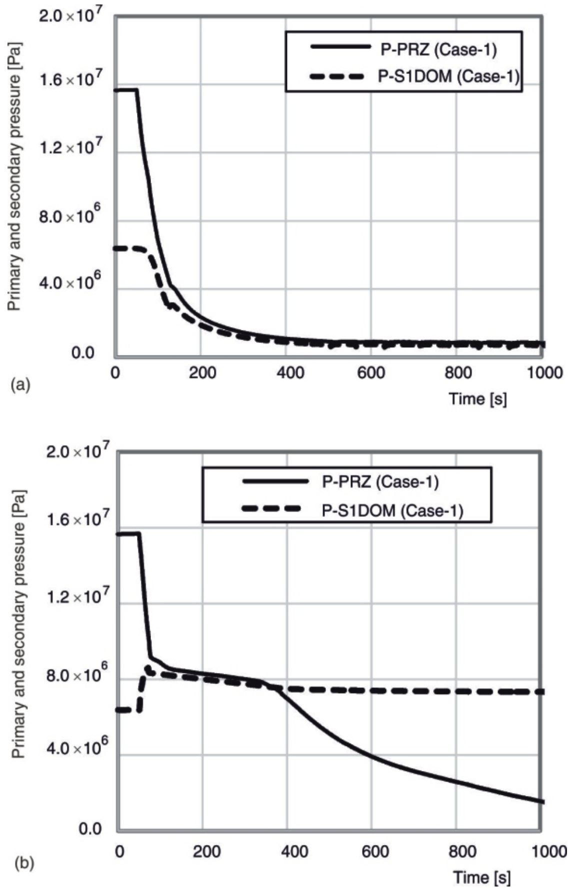

Figure 9. Primary and secondary pressure in Case-1 and Case-2

enters flash evaporation as the RCS pressure drops. It no longer seems possible that the void expansion in the RPV dome could progress up to the point where the steam volumes enter the reactor coolant line (RCL) hot leg as shown in fig. 12.

From the previous figs. 11 and 12, the void is formed faster in Case-1, but it reached the max value in Case-2 at 688 seconds before Case-1 which the max value of core void takes place at time 1690 seconds.

Comparison of the break discharge and the normalized core level for the reactor

Figures 12 and 13 show in Case-1 and Case-2, due to the break a large discharged flow out, so the primary pressure depressurization and the higher of the primary coolant temperature, the core collapsed water level decreases rapidly especially in Case-2 faster than Case-1 as shown in fig. 13.

Comparison of the core outlet temperature

Figures 14(a) and 14(b) show the comparison between the two cases for the core outlet temperature. In Case-1 where the scenario of the accident is without scram the main pumps are still running so the core outlet temperature is safe for 2000 seconds from the transient time. In Case-2 the core outlet temperature is still

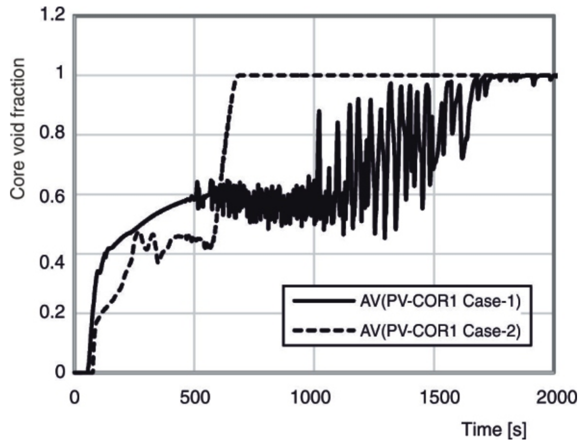

Figure 10. Core void fraction

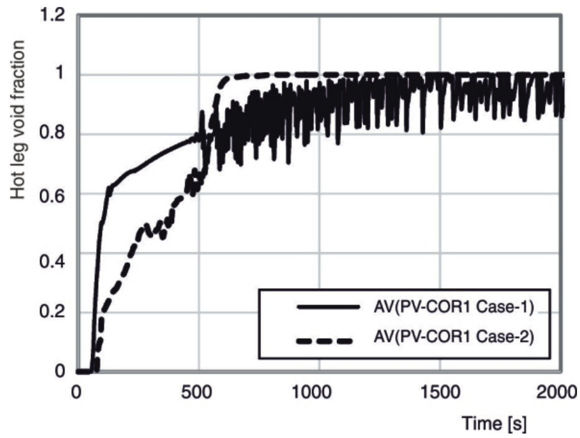

Figure 11. Hot leg void fraction

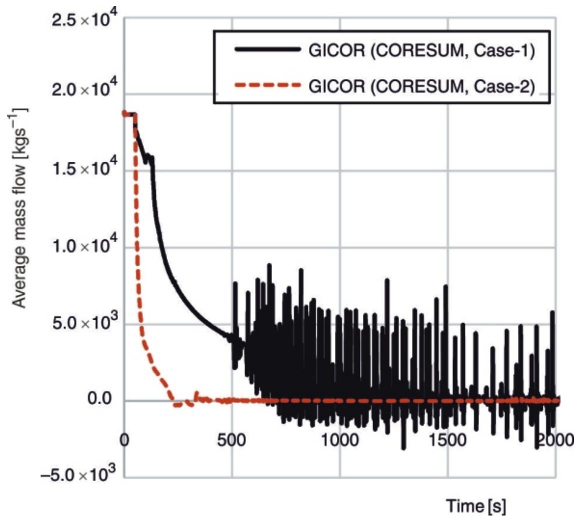

Figure 12. Average mass flow

safe only for 800 seconds, where the pumps are not running. Figure 15(a) shows that after the initiation of break and extended for a brief period of around 1800 seconds a small increase of the core outlet temperature in Case-1 lower than which occurred in Case-2 that is a fast increase from 800 seconds until 1800 seconds, but after this time it started to decrease in case 2. These results occurred as shown in figs. 6 and 7 and figs. 8(a) and 8(b) due to the no scram of the reactor and the main

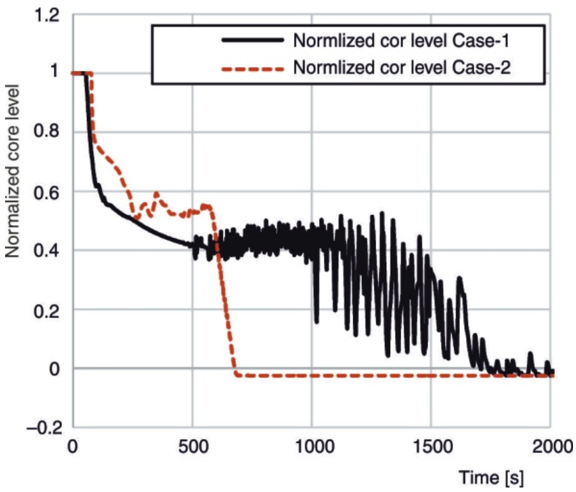

Figure 13. Normalized core level

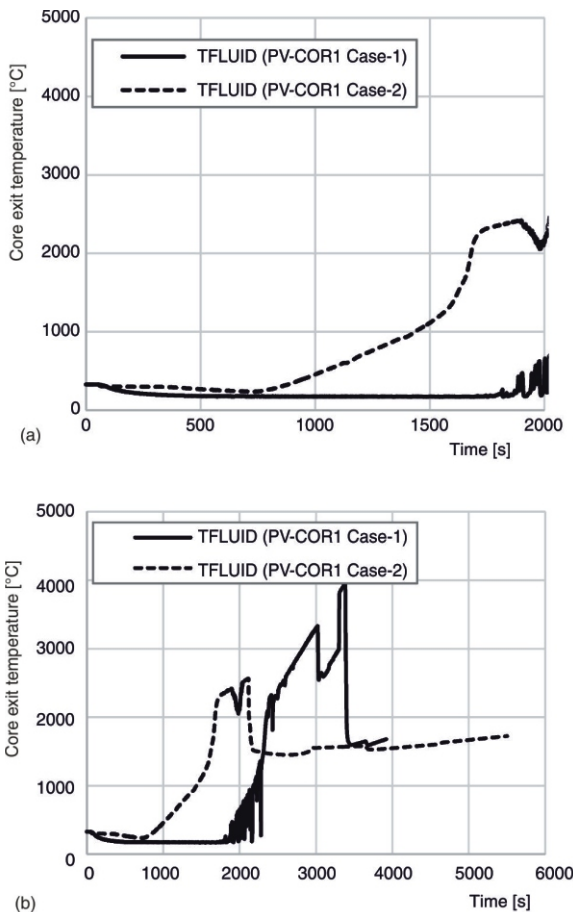

Figure 14. Core outlet temperature

pumps still running in Case-1. Moreover, fig. 14(b) which the run time is extended to 5500 seconds illustrates that in Case-1 the core outlet temperature kept increasing until it reached 3800 °C, then the Model stopped to fail as discussed before, but in Case-2 the Model succeeds in calculation all the run time with the maximum core outlet temperature 2500 °C.

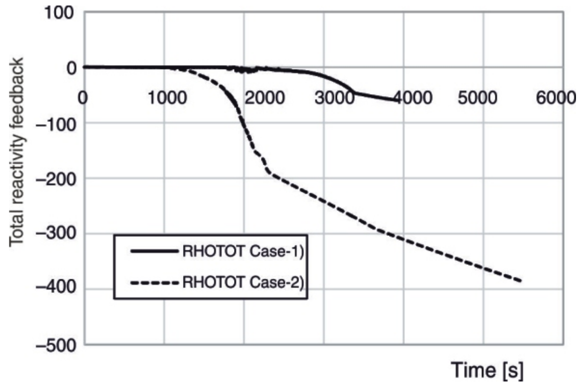

Figure 15. Total reactivity feedback

Comparison of the total reactivity at the reactor

Figure 15 shows that the total reactivity feedback in Case-1, where the transient of the accident without scram, is considered with the sum of only change of the feedback from boron, coolant density and fuel temperature, but in the Case-2 the total reactivity feedback increases due to adding of the control rod (external reactivity feedback).

The results of the comparison between Case-2 and Case-3

To show the effect of safety systems on the behavior of the KONVOI reactor during SBLOCA, the comparison between the results of Case-2 (without actuation of safety systems) and Case-3 (with actuation of safety systems) is completed.

Comparison for the density of the coolant system and core void fraction

The comparison of the results is concerning the parameters of the pressurized water reactor as the density of the coolant system, as shown in fig. 16 and core void fraction in fig. 17. Figure 16 shows that the density is increasing in the coolant system in Case-3 due to the actuation of the safety systems that reverses in Case-2. These results confirmed the core void fraction decreasing as shown in fig. 17.

Comparison for the normalized core level and the maximum clad temperature

The comparison between the Case-2 and Case-3 for the normalized core level and the maximum clad temperature, respectively, are represented in figs. 18 and 19.

Figure 18 shows that in Case-3 the core is fully covered owing to the continuous cold and hot legs injection of emergency core coolant (ECC), also accumulator is activated in the set point of each system dur-

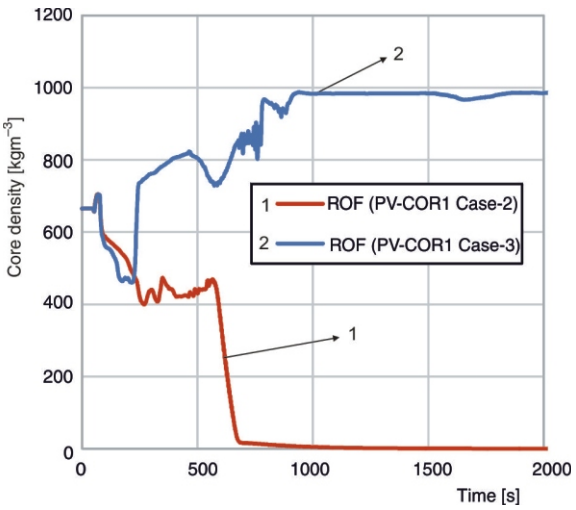

Figure 16. Density of the core coolant

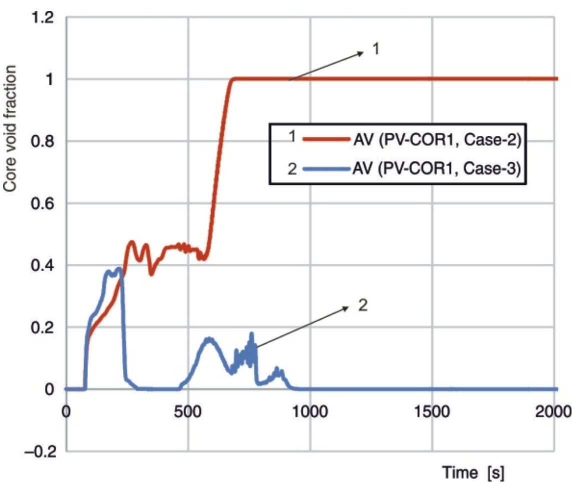

Figure 17. Core void fraction

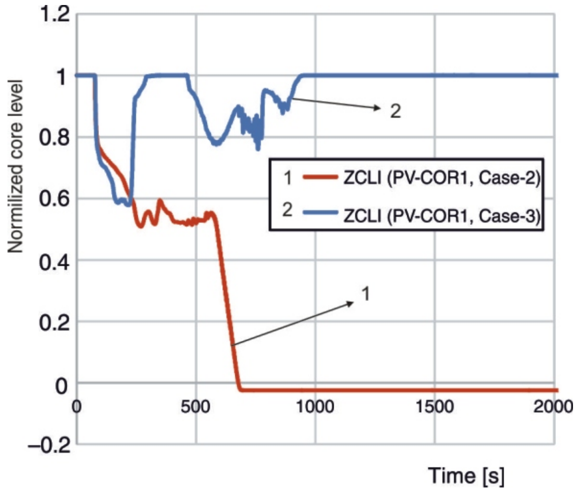

Figure 18. Normalized core level

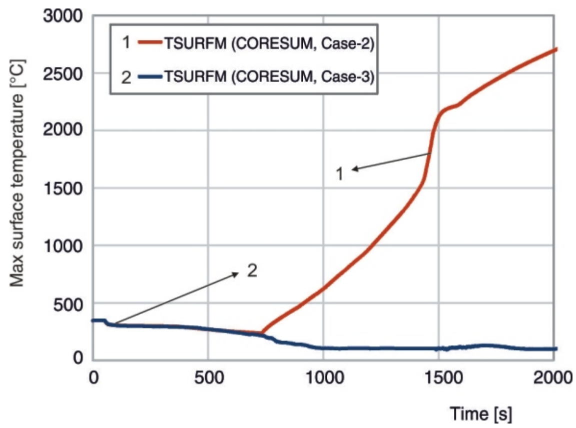

Figure 19. Maximum clad temperature

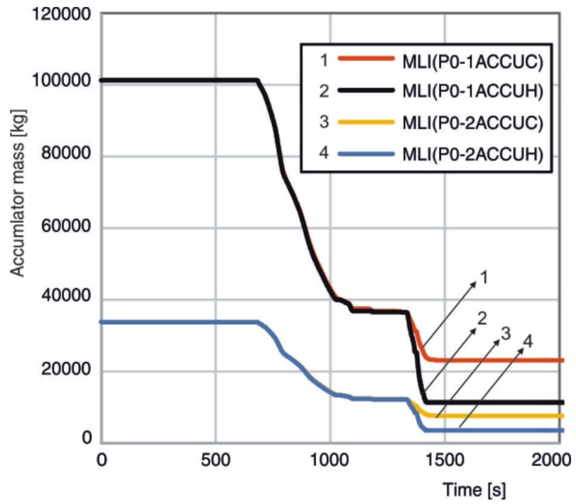

Figure 20. Accumulator's inventory

ing the SBLOCA, DEC. Consequently, the cladding surface temperatures are reduced as shown in fig. 19.

Accumulator mass and the break and safety system flow rate in Case-3

During Case-3, where the availability of safety systems, the decrease in the primary pressure occurs rapidly enough due to SBLOCA for early intervention of HPIS, accumulators and LPIS to overcome the break discharge and the core uncover and heating up occur for a very short time. Figure 20 shows the decreasing patterns of the accumulator's inventory during accident time. When the pressure drops below 11 MPa after 9 seconds, the HPI starts automatically and injects water into the three intact hot legs and the cold leg of the broken loop. The hydro-accumulator check valves open below a system pressure of 2.6 MPa, which is reached at 650 seconds after the leak opening, as shown in fig. 20 because the injection rate of the HPI is not sufficient to overfeed the leakage. After 810 seconds and below 1.1 MPa, the LPI is started, which then cools the core by injecting water into both the hot and the cold legs as shown in fig. 21, the cladding surface temperature is reduced in as shown in fig. 19.

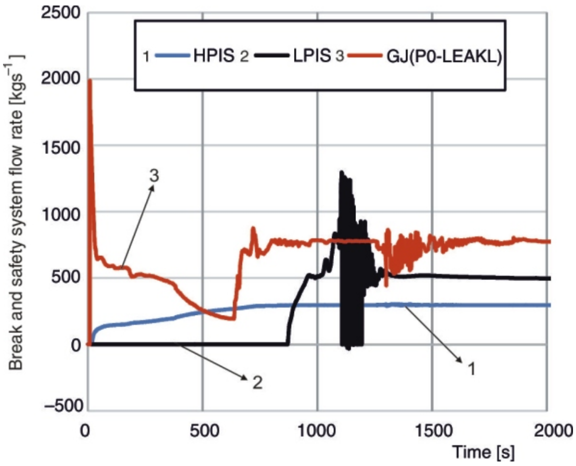

Figure 21. Break and safety system flow rate

CONCLUSIONS

In this study, ATHLET model for KONVOI PWR is created to simulate many investigated transient conditions that are more complex and more severe than those postulated as DBA, so they are DEC. The effect of various investigated transient accident conditions has been analyzed in SBLOCA sequence in one of the cold legs in different three cases. The three cases are considered including SBLOCA as DEC without scram and all of the safety systems are a failure, the normal scram actuation with failure of actuation of the accumulators, and ECC, and with normal actuation scram sequence with normal sequential actuation of all safety systems. The steady-state results are nearly in agreement with the plant parameters available in the literature. Moreover, the results indicate that the actuation of passive, (ACC) and active (HPSI and LPSI) safety systems as in Case-3 could mitigate the accidental consequence of LOCA effectively. Furthermore, the key thermal-hydraulic parameters are in the acceptable range and meet the safety criterion. Since, during the hypothetical investigated transient accidents process, the core uncovers and fuel heat up do not occur thus the safety of KONVOI PWR, during a 6 inches (0.1524 m), cold leg small break LOCA is verified.

NOMENCLATURE

ACC	–	accumulator
ATHLET	–	analysis of the thermohydraulics of leaks and transients
ATHLET-CD	–	analysis of the thermohydraulics of leaks and transients-core degradation
ATWS	–	anticipated transient without scram
CD	–	core degradation
DEC	–	design extension conditions
DBA	–	design basis accidents
ECCS	–	essential emergency core cooling system
EFW	–	emergency feed water
EBS	–	extra borating system
GKN II	–	the German four-loop PWR
GRS	–	gesellschaft für anlagen- und reaktorsicherheit, germany (company)
HPSI	–	high-pressure safety injection
LAcc	–	level in accumulator
LOCA	–	loss of coolant accident
LOOP	–	loss of offsite power
LP	–	low pressure
Kraftwerk Union (KWU)		
LSG	–	level steam generator
PCoolant	–	primary/coolant pressure
NPP	–	nuclear power plant
PSG	–	secondary pressure/ pressure steam generator
PWR	–	pressurized water reactor
PZR	–	pressurizer
RCL	–	reactor coolant line
RCS	–	reactor cooling system inventory
RHR	–	residual heat removal
RFRD	–	reverse flow restriction device
RPS	–	reactor protection system
RPV	–	reactor pressure vessel
SAM	–	severe accident management
SBLOCA	–	small break loss-of-coolant-accident
SBO	–	station blackout
SCRAM	–	emergency reactor shutdown (safety cut rope ax man)
SIP	–	safety injection pump
T_g	–	gas temperature
T_{lig}	–	liquid temperature
T_{sat}	–	saturation temperature

AUTHORS' CONTRIBUTIONS

The idea of the research of the safety analyses of NPP under Design Extension Conditions was a partnership between both S. Helmy and M. Kandil. The computational model of the ATHLET code was verified by M. Kandil. The computational model of the ATHLET code developed and validated through many comparisons was made by S. Helmy. The computational model of the ATHLET code validated with many comparisons was conducted by A. Refaey. The conceptualization, methodology, and formal analysis were made by all authors. All authors participated in the discussion and analysis of the results and made the original draft and the finalization of the manuscript. M. Kandil, S. Helmy, and A. Refaey equally contributed as the main contributors in this paper.

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Received on December 9, 2020
Accepted on March 3, 2021

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АНАЛИЗА ТЕРМОХИДРАУЛИЧКОГ ПОНАШАЊА KONVOI PWR РЕАКТОРА ТОКОМ ПРОДУЖЕНОГ РАДА

У нуклеарним електранама околности продуженог рада сложенији су и строжи од оних који се претпостављају као основни пројектни акциденти, стога се морају узети у обзир у анализама безбедности. У овом раду многи хипотетички истражени прелазни процеси примењују се на KONVOI реактору са водом под притиском током малог акцидента губитка хладиоца ради ревизије ефеката свих сугурносних система преко њихове расположивости или нерасположивости на термохидраулично понашање реактора. Истражени прелазни процеси представљени су као три случаја губитка хладиоца са малим прекидом рада: случај 1 – без хаваријског заустављања и свим сигурносним системима у квару; случај 2 – са нормалним хаваријским заустављањем и отказом свих сигурносних система услед недоступности; и на крају, случај 3 – са нормалним редоследом активирања при хаваријском заустављању и нормалним секвенцијалним активирањем свих сигурносних доступних система. Ова три случаја истражних транзијената симулирани су стварањем модела уз коришћење кода за анализе термохидраулике цурења и транзијената. У свим транзијентним случајевима узимају се у обзир све врсте повратних реакција, присуство борам густина модератора, температура модератора и температура горива. Резултати стабилног стања су готово у складу са параметрима постројења доступним у претходној литератури. Резултати показују важност ефеката повратне спреге реактивности у случају губитка хладиоца на губитак снаге јер се сматрају кључним параметрима за контролу температуре кошуљице и горива како би се одржале испод тачке топљења. Штавише, израчунати резултате у свим случајевима показују да су термохидраулички параметри у прихватљивим опсезима и да испуњавају критеријум сигурности током акцидента губитка хладиоца у случају продуженог рада реактора. Даље, резултати показују да се у KONVOI реактору са водом под притиском не појављује водом непокривено језгро и прегревање горива према тезама у пројектним симулацијама услова продуженог рада, будући да сви сугурносни системи обезбеђују одговарајуће хлађење језгра довољним залихама воде у језгру да га покрије.

Кључне речи: акцидент губитка хладиоца, хаваријско заустављање реактора, продужени пројектовани рад, откривено језгро, термохидраулички феномен, убризгавања четири-петље KONVOI PWR