PSA-ORIENTED ANALYSIS OF TRANSIENTS IN ATUCHA UNIT II PHWR WITH A RELAP MODEL

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Abstract

This paper presents a model of the Atucha II (PHWR) nuclear power plant developed for RELAP5/MOD3.3. The nodalization was implemented to comply with the Probabilistic Safety Analysis required in the licensing, commissioning, and operating process. A data management platform is implemented that processes the geometric and physical data for nodalization and generates the code input. The steady state is analyzed by comparing calculated variables with their respective design values and previous calculations performed with other models/tools. Finally, a case of loss of heat sink caused by an electrical supply failure is analyzed.

1. Introduction

The Atucha II nuclear power plant is a two-loop Pressurized Heavy Water Reactor (PHWR) under construction in Lima, Argentina, next to the Atucha I nuclear power plant [1].

A Probabilistic Safety Analysis (PSA) is being carried out for the licensing process. In this context, several sequences derived from postulated initiating events are studied from probabilistic and deterministic points of view. The analysis is based on a RELAP5/MOD3.3 model, with a thorough documentation of all system data used as a requirement to guarantee the construction procedures, licensing, commissioning, and operation. The fulfilment of this requirement makes it easier to achieve the level of quality assurance needed in the nuclear industry.

The objective of this paper is to show the application of an integral platform of data and nodalization management as an important tool to generate the RELAP model for the analysis of transients for PSA in the Atucha II licensing process.

The resulting steady state using this nodalization is analysed by comparing calculated variables with their respective design values and previous calculations performed with other models/tools. Additionally, an application of the model to the simulation of Emergency Power Case (EPC) scenarios is presented. In these cases, the loss of both external and internal power lines is involved.

2. Reactor description

The reactor is cooled and moderated by heavy water. Its design is based on Atucha I and the modern German Pressurized Water Reactor (PWR). The primary circuit, which is shown in Fig.1, is similar to a two-loop PWR. However, the reactor core consists of vertical channels surrounded by a large volume of heavy water acting as a moderator. Figure 2 shows a schematic view of the pressure vessel: The coolant coming from the Steam Generators (SGs) enters the Reactor Pressure Vessel (RPV), falls down through the annular downcomer (the space between the RPV wall and the outer wall of the moderator tank), reaches the lower plenum, and enters the coolant channels where the fuel assemblies are located. The coolant is heated by the fuel assemblies, reaches the upper plenum, and then leaves the RPV flowing to the SGs through the hot leg. Figure 2 represents one of the 451 coolant channels, which are surrounded by the moderator tank.

The moderator is thermally separated from the primary system, although it is maintained at the same pressure by pressure equalization openings. The moderator must be kept at a lower temperature for neutronics reasons (170ºC, that is, 130ºC lower than the primary system). During full power operation, 95% [~2000 MW(th)] of the total fission power is released in the fuel, and the remaining 5% [~100 MW(th)] is released in the moderator because of neutron moderation and gamma absorption. Additionally, another 5% of the thermal power generated in the core is transferred by conduction from the reactor coolant to the moderator through the channel and moderator tank walls. To remove this power the moderator is cooled by four independent loops, as shown in Fig.3. The removed heat is used to preheat the SGs feedwater.

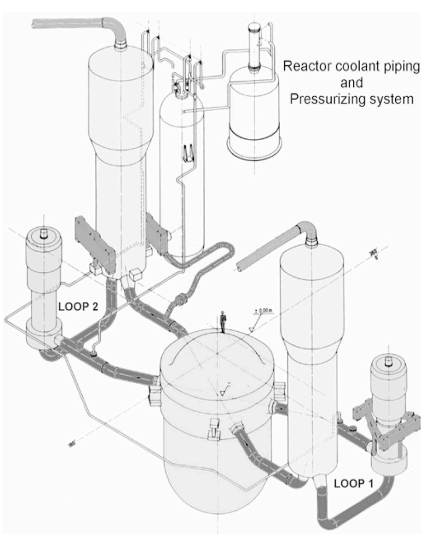

Fig. 1-Reactor primary system

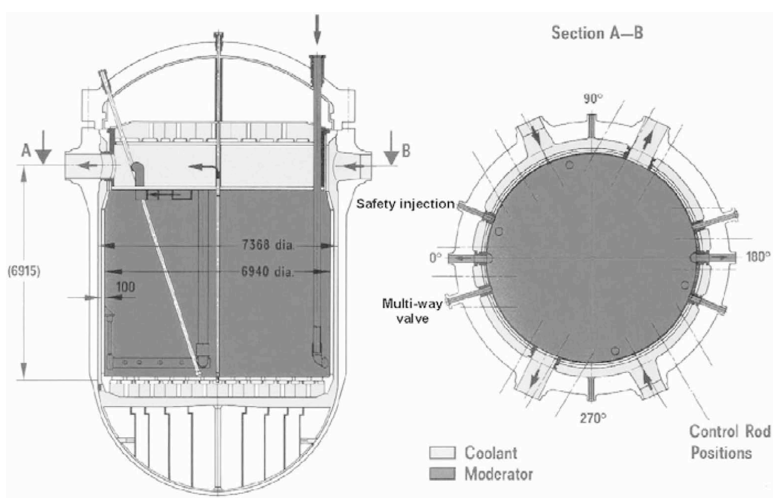

Fig. 2-The RPV and coolant channels

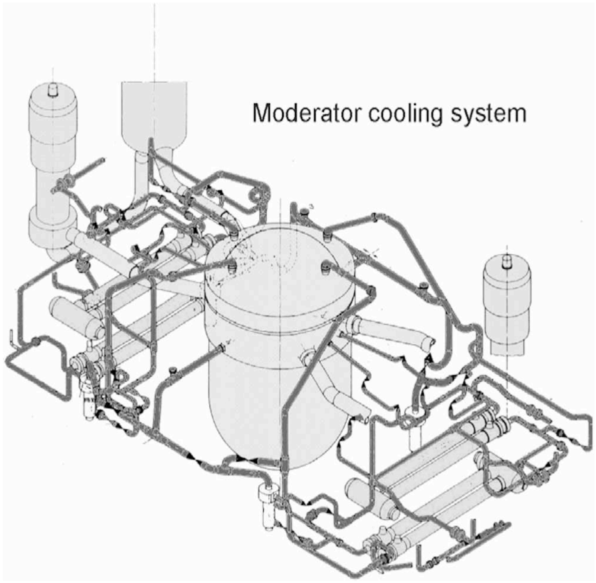

Fig. 3-Moderator cooling system piping

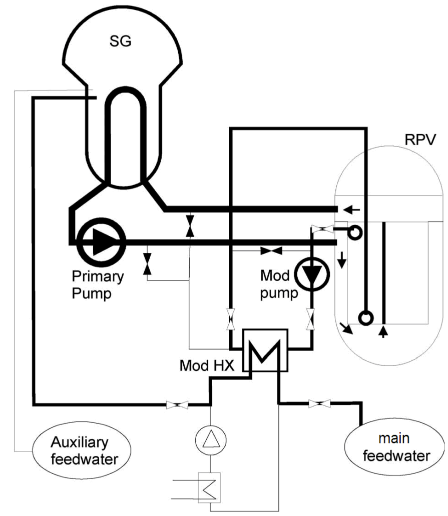

Fig. 5-Configuration in normal operation

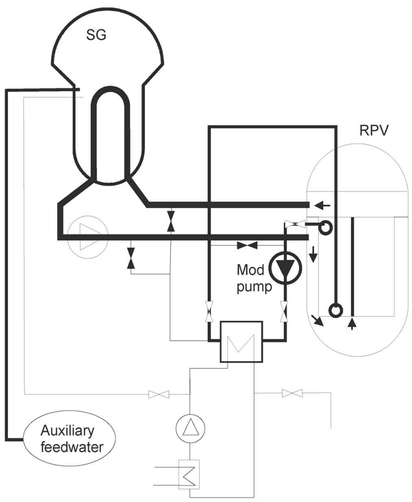

Fig. 4-Configuration for RHR by SGs/AFWS

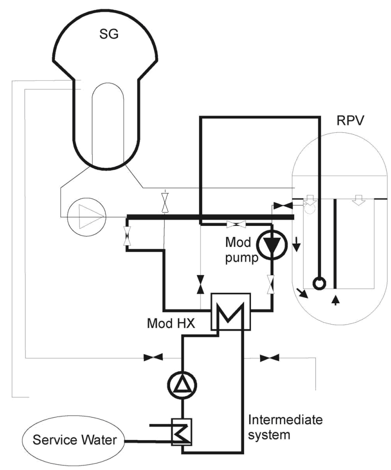

Fig. 6-Configuration for RHR by Moderator RHR chains

All safety systems are provided with four redundancies, each one with 50% of total required capacity (i.e., two out of four are needed to perform a safety function). In particular, the moderator heat exchangers are used as a Residual Heat Removal (RHR) system to extract core power in case of safe hot shutdown or accidental sequences.

Another way to extract the core power during shutdown conditions is through the SGs using the Auxiliary Feed Water System (AFWS), which feeds both SGs from the feedwater tank using four independent pumps (two out-of-four design).

Because of the low excess reactivity available in a natural uranium reactor, a continuous refuelling is required in normal operation. This action is performed by the refuelling machine, which operates at the top of the RPV inserting new fuel elements and taking out the exhausted ones from the coolant channels.

To control the reactor power Atucha II has 18 control rods, which enter diagonally to the reactor core, in order to leave free space above the reactor for the refuelling machine. Additionally, a fast boron injection system, as a second independent shutdown system, is provided, which injects boric acid into the moderator tank.

2.1 Operational modes

In order to remove the generated power under different conditions, the systems are arranged using different configurations, which are explained below.

2.1.1 Normal operation

As mentioned before, the moderator must be maintained at a relatively low temperature for reactivity reasons. The heat removal is achieved by the moderator cooling system, which in normal operation acts as an independent system, preheating the SGs feedwater (Fig.4; only one primary loop and one moderator loop are shown). Consequently, the moderator and primary systems are refrigerated separately. The coolant heated in the core is cooled in the SGs as in a classical PWR. The moderator, in turn, is collected from the top of the moderator tank, cooled down in the moderator heat exchangers, and injected back to the bottom of the tank.

2.1.2 Residual heat removal by SGs (AFWS)

Simultaneously with reactor trip, turbine trip is actuated, and the set point of the secondary pressure in the SGs is reduced from 56 to 22.2 bar (271 to 220ºC) at a rate of 72 bar/h (100K/h), which is the hot shutdown state. This depressurization is performed by SGs relief valves or by the turbine bypass valves. At the same time, moderator cooling is stopped to balance as fast as possible the coolant and moderator temperatures to reach the hot shutdown condition. After balancing the temperatures and reaching the hot shutdown state, the removal of the decay heat continues via the SGs, fed by the AFWS or by the main feedwater system, as shown in Fig.5.

2.1.3 Residual heat removal by Moderator RHR Chains

In the event of total loss of feedwater to SGs, the decay heat is removed by the Moderator RHR chains. In this case, the moderator loops are reconfigured to cool down not only the moderator but also the coolant. This is shown in Fig.6 (for simplicity only one out of four RHR chains is shown in the diagram). In this mode, the water is removed from the lower ring header of the moderator tank; then, it is cooled in the moderator heat exchangers transferring the heat to an intermediate system, which in turn transfers it to the ultimate sink (river water). The cooled moderator water is re-injected into the primary system to cool the channels containing the fuels. The circuit is closed through the pressure equalization holes located between the upper part of the moderator tank and the upper plenum.

3. Data management

As mentioned, an integral platform of data and nodalization management was implemented for the RELAP5/MOD 3.3 reactor transient simulation code [2]. The system includes geometry and process input data, calculation of related parameters, data processing for nodalization development, and automatic generation of the input file for the RELAP code. A system of data tracking was implemented to support quality assurance of the model and to minimize error occurrence. The platform is based on Microsoft Excel® worksheets, which allows data tracking and Visual Basic® programming to perform the automatic generation of the RELAP input file

Geometric and process data for each plant system are stored in separate Excel files. For each system file, several work sheets storing these data are saved. Some of them are listed below:

General information: Variables definition and dates of modifications are included in the system.
Diagram and nodalization scheme: The hydraulic circuit, its components, and the corresponding RELAP nodalization numbers are detailed.
Original data: Original geometrical and process information about the system modeled is saved.
Nodalization: The user performs specific calculations for the proposed nodalization scheme.
Tables: Materials thermal properties, values for control variables, etc., are stored.
Control variables: All the internal control variables are specified.
Output data: The file name and directory are specified.
Hydrodynamic components (branch, mtpljun, pipe, snglvol, tmdpvol, sngljun, tmdpjun, valve, pump): Data for all components of the nodalization scheme proposed are computed.
Trips: All activation and control trips for different components are specified.
Structures: Geometry and characteristics of thermal structures are described.
Permanent cards: General information such as time step control cards and expanded plot variables is stored.

With adequate boundary conditions, each system can be tested individually considering its links to other systems of the plant. A RELAP input file can be generated for any system, and the results can be tested with its design parameters.

After this individual system test, the total integration of all the systems can be performed to obtain a single RELAP input file for the whole plant. The boundary conditions for each system are replaced by the links between adjacent volumes of the connected systems. The user can specify, in a separate work sheet, the working directory, the name of the RELAP input file, and the name of the files containing the systems involved in the coupling. Using the integrated RELAP input file based on the plant data stored in the data sheet of each system, the steady-state parameters in normal operation can be obtained and tested with the corresponding design values.

4. System nodalization

Each system nodalization includes the tracking of geometric and process data from the design manuals of the plant and are stored using the Integral platform of data and nodalization management.

Every system is modelled in detail. For example, the 451 coolant channels were grouped into 36 channels (6 radial and 6 azimuthal zones). They were modelled with RELAP pipe components, and with heat structures to simulate the channel walls and the fuel elements. The division in six radial zones corresponds to the five hydraulic zones in the core with similar channel mass flows, corresponding to the radial distribution of the generated power. The central zone is partitioned into two regions. The azimuthal division criterion is adopted in order to take into account the fact that the core is divided into two halves, each one associated with one cold leg inlet. Each half is divided into three zones, the central one of 80 deg (centred with the cold leg inlet), and two of 50 deg at both sides. This division was chosen because the hot leg outlets from the upper plenum are rotated 50 deg with respect to the cold leg inlet (this causes a redistribution of the mass flow coming from one loop of the primary circuit to the other). In each zone, an average channel is simulated with a pipe component. It was divided into 24 axial zones (1 for the inlet, 1 for the non active portion, 20 for the active zone, and 2 for the outlet extended to the top of the moderator tank). This axial division was chosen to distribute in the pipe unions the located friction losses introduced by the separators of the fuel elements. This division is extended later to downcomer and moderator tank nodalization. The heat structures simulating the fuel elements are divided into 20 axial zones, in correspondence with the active portion of the channel nodalization. The coolant channels are connected to the lower and upper plenum systems. A point-kinetics model and radiation heat transfer from channel walls are included.

In a similar way, all systems of the plant are nodalized in detail; some of them are upper and lower plena and gaps, downcomer and moderator tank structure, primary system piping and pressurizer, moderator circuits, steam generators, normal and auxiliary feedwater system, steam system, and Moderator RHR system (four redundancies) [3].

5. Integral model test

5.1 Steady-state hydraulic parameter comparison

The friction loss coefficients were adjusted to reproduce the design steady-state values (friction losses and mass flows) calculated previously with the DRUD2O code [4].This code was developed by Kraft Werk Union (KWU) and calculates in detail friction losses in the primary circuit, SGs, pipes, RPV, and coolant channels. The RELAP results were compared with the corresponding ones of DRUD2O and showed good agreement. In Table I, a comparison for the values obtained for total primary circuit mass flow and percentage differences across different regions can be seen. A good agreement between the two codes was observed. The largest differences are observed in the primary pipes (hot and cold legs and loop seal). This is caused by the equivalent lengths used by DRUD2O in the accessories (curves) of these components. As there is no information about these equivalent lengths, the pressure losses calculated by RELAP were considered correct.

Region	RELAP [kg/s]	DRUD2O [kg/s]	Difference [%]
Total primary	10565	10566	0.0
Core bypass			
DC-Upper plenum gap	1	1.01	0.9
DC-Hot leg	0.6	0.58	-3.3
DC-Moderator-Safety Injection system	0.83	0.81	-2.6
Lower plenum-Moderator tank	0.53	0.52	-1.0
Core			
Zone 1	2.71	2.74	1.0
Zone 2	4.1	4.14	0.9
Zone 3	6.17	6.25	1.2
Zone 4	18.65	18.85	1.1
Zone 5	68.36	68.03	-0.5

Table I-Steady-state hydraulic parameters adjustment

Parameter	RELAP	KUMOTEI	Unit
Reactor thermal power	2159.9	2161.0	MW
Aver. moderator temp.	170.6	170.5	ºC
Primary side			
Steam Generators			
Inlet temp.	313.2	312.5	ºC
Average temp.	295.8	295.1	ºC
Outlet temp.	278.3	277.6	ºC
Mass flow	5327.1	5303.5	kg/s
Power	983.9	975.9	MW
kA	49434	51300	kW/K
Mod. heat exchangers			
Inlet temp.	193.8	201.5	ºC
Average temp.	165.9	171.4	ºC
Outlet temp.	138.0	141.4	ºC
Mass flow	220.9	220.9	kg/s
Power	51.5	55.7	MW
kA	2826	2404.1	kW/K
Secondary side			
Steam Generators			
Steam pressure	55.9	55.9	bar
Inlet temp.	173.3	175.5	ºC
Outlet temp.	271.0	271.0	ºC
Mass flow	478.5	478.5	kg/s
Mod. heat exchangers			
Inlet temp.	122.3	121.0	ºC
Outlet temp.	172.7	175.4	ºC
Mass flow	239.0	239.3	kg/s

Table II-Steady-state thermal parameter adjustment

5.2 Steady-state thermal balance

In this case the steady-state design thermal parameters were compared with previous values calculated by the KUMOTEI code [5]. In Table II a comparison for values of temperatures and power in primary and secondary circuits is shown. A good agreement between the compared values can be seen. There is a small discrepancy between the inlet and outlet temperatures of the moderator heat exchangers (primary side), but the global thermal balance is accurate.

Once the steady state was adjusted with the design values, all transient analysis for the PSA and licensing process can be carried out.

6. Sequence analysis

As an example, an EPC transient, which is the initiating event with the highest probability of occurrence, is analyzed. In this case, all systems are disconnected (except the ones connected to batteries). The normal procedure is to start up the diesel generators, which supply power only to the essential systems needed to guarantee the basic safety functions. These systems are reconnected in a sequential order. Nevertheless, the higher "energy consumers" components used during normal operation become unavailable: primary circuit pumps, SGs main feedwater pumps and turbine bypass and condenser. On the other hand, the main systems available (power supplied by diesel generators) are: AFWS, Moderator RHR chains, pressure control system and main steam relief and safety valves.

The implemented strategy consists in producing the reactor shutdown and then removing the energy from the primary system by means of the SGs, using the AFWS and venting the generated steam to the atmosphere through the relief valves. The secondary system pressure is decreased to the hot shutdown set point (27bar) with a ramp of 100K/h. If heat removal by the SGs fails, the Moderator RHR system is demanded. Depending on the sequences, this system can be triggered by very low level in both SGs, high temperature in moderator or coolant systems, or Loss-Of-Coolant-Accident (LOCA) signal. In both cases, pressure control in the primary system (spray, or in case of its failure, pressurizer safety valves) is required.

After the EPC event, the transient evolution depends on the success or failure of the demanded systems. Classical event trees are constructed in order to study all possible sequences and final stages. In order to give support to the event tree construction, deterministic simulations are carried out. Within this context, systems are evaluated considering that they provide the minimum capability for assuring the success of their corresponding safety function. For this reason, success sequences may differ from design basis, and fewer redundancies could be needed.

The success sequences can be grouped into two kinds: the ones with the core cooled by SGs (AFWS) and the ones with the core cooled by Moderator RHR chains. On the other hand, in sequences with failure of heat removal, the expansion of the coolant and moderator drives the system to a LOCA situation. Although their progression depends also on the intervention of injection systems, the most relevant is the blackout sequence, in which safety systems are not available at all.

In the following sections, some of the sequences presented above are analysed. The events taking place during the first seconds in the EPC sequences are present in all of them. The first and second SCRAM signals are triggered because of low flow in the moderator circuit and low rpm in the primary pumps, respectively. Some common boundary and initial conditions imposed to this transient are listed below:

1. Initial power is 100%.
2. Initial average temperature in moderator is 170ºC.
3. Decay power is based on ANSI/ANS-5.1-1979 data, isotopes ^{235}U, ^{238}U, and ^{239}Pu [6].
4. Core burn up is in equilibrium.
5. Both main feedwater lines and the separate line to SG2 are disabled (one out of four AFWS pumps). The separate line to SG1 is assumed available.
6. Control rods are not available for reactor SCRAM. Shutdown is performed by the fast boron injection system (two out of four redundancies considered). The first SCRAM signal is not taken into account. Reactivity inserted by the second shutdown system is modelled as a function of the volume of the boron injected into the moderator.

7. The spray in the pressurizer is configured at its minimum capability (KBA40 system unavailable. One out of two KBA80 pumps is assumed to be operating, when demanded, for spray in the primary system).
8. Mass inventory control system is assumed to fail.
9. There are no manual actions.

6.1 Refrigeration by AFWS

This case considers the most probable sequence: the success of all demanded systems. In this case, the system configuration is shown in Fig.5. The RHR is achieved by SGs fed by the AFWS. In the simulated sequence, only one out of two SGs is assumed to be fed by one out of four auxiliary feedwater pumps. The steam is vented to the atmosphere by one out of four relief valves (condensers are not available because of the initiating event). The pressure on the secondary side is controlled this way (Fig.7), being depressurized with a ramp of 72 bar/h (100 K/h) to hot shutdown condition, corresponding to 27 bar in the secondary side.

Because of the loss of heat removal from the moderator heat exchangers, the moderator temperature increases (Fig.8). At the same time, because of the depressurization on the secondary side, the coolant temperature decreases. Nevertheless, in the medium term the expansion of the moderator counteracts the contraction of the coolant, causing the pressurizer level to rise (Fig.9), until temperatures on the moderator and primary sides are balanced. This rising level compresses the steam space, causing the pressure to increase. Pressure control is achieved by means of the auxiliary spray (or safety valves in case of unavailability of the auxiliary sprays) (Fig.7). Figure 10 shows the power removal of the SGs. The power in the SGs initially decreases sharply because of the reactor and turbine trip. Then, it increases because of the depressurization of the secondary side, following the 100 K/h ramp. When this ramp ends, the removed power stabilizes. In the long term, when the decay power is lower than the removed power, the level in the pressurizer tends to decrease, and pressure control is not required any more. Figure 11 shows both SGs levels. As mentioned before, only one SG is fed by the hypothesis. The level in both SGs decreases until the low-level signal is achieved and the AFWS is triggered (conservatively, it was assumed the AFWS to be started late by a reactor protection signal on SG level<5m, disregarding an earlier in-service signal on low feedwater flow rate). Then, the level on the fed SG is recovered. The other SG can still remove power until it dries out at ~5000 s.

6.2 Refrigeration by Moderator RHR chains

This group of sequences derives from a failure of the AFWS. In this case, the system configuration is shown in Fig. 6. In the simulated sequence, only one out of four Moderator RHR loops is assumed to be available. The power removal by the SGs and Moderator RHR chains is shown in Fig.12. The Moderator RHR is activated because of low level in both SGs, at ~3000s. The SGs continue removing power until they dry out. In the long term, the Moderator RHR system removes the decay power. The reactor is driven to a safe shutdown state, and the temperature in the primary and moderator systems is driven to a hot shutdown value. As a consequence, the level in the pressurizer can be controlled, as shown in Fig. 13.

6.3 Blackout

In case of blackout, all active components are unavailable, except the safety valves of the pressurizer and the relief valves of the SGs. A conservative hypothesis of closed failure of the SGs relief and safety valves and unavailability of the accumulators was assumed. Therefore, the only energy reservoir is the primary and moderator heavy water inventories. The primary pressure is controlled by cycling the pressurizer safety valve (Fig.14). In this sequence, cycling of the pressurizer safety valve is assumed until the pressurizer gets full of liquid (Fig.15). Afterwards, it is assumed that the valve fails opened. In this case, the moderator volume acts as a heat sink, increasing its temperature until it equalizes with the primary side (Fig.16). When the core dries out, the level in the moderator tank is still high (Fig.17). In this way, part of the power generated in the fuel elements can be transferred by thermal radiation and convection through the steam to the channel walls and from this point to the moderator water. The fuel temperature becomes critical after ~5500s (Fig.18).

7. Conclusions

A model of the Atucha II PHWR plant for the RELAP5/MOD3.3 code was developed based on an integral platform of data management. This tool makes easier the storage of geometry and process data, their prosecution for calculations needed for the plant nodalization, and finally the automatic generation of the code input. The model was oriented to support the PSA, required in the licensing, commissioning, and operation process. The integral platform developed allows a complete tracking of data, from design reports to the input deck, which facilitates quality assurance of the model by independent reviewers. The main components and systems of the plant were included in the nodalization, enabling the model to correctly represent the expected physical behavior.

The results obtained for steady state were compared with the correspondent design values and previous calculations performed with other codes and showed good agreement not only for the hydraulic parameters but also for the thermal values in the primary and secondary systems. Each auxiliary system was tested individually to adjust their design operating values (mass flows and thermal balance).

An example of using the model for transient analysis for PSA was studied in an EPC case, which is the initiating event with the highest probability of occurrence. The operational modes are described for RHR by the SGs and Moderator RHR system. The main aspects of the plant dynamic are presented and analyzed. The common initial events for all sequences in this initiating event are stated. Representative variables of selected sequences are shown, and the possible final stages are identified.

The model will be further used for PSA level I transient and accident analysis.

8. References

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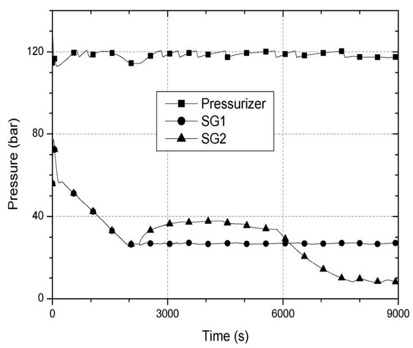

Fig. 7-Primary and secondary system pressure for AFWS heat removal

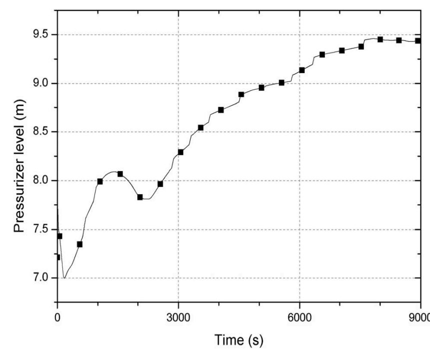

Fig. 9-Pressurizer collapsed level for AFWS heat removal

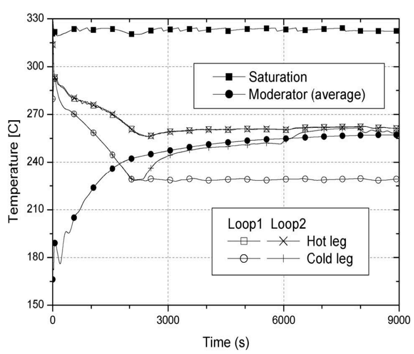

Fig. 8-Primary and moderator temperatures for AFWS heat removal

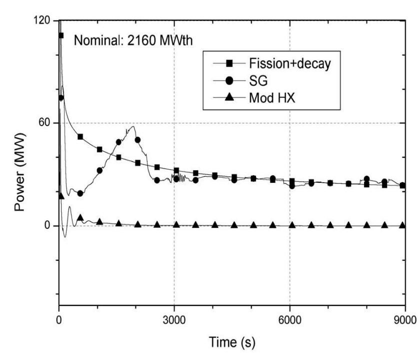

Fig. 10- Plant power for AFWS heat removal

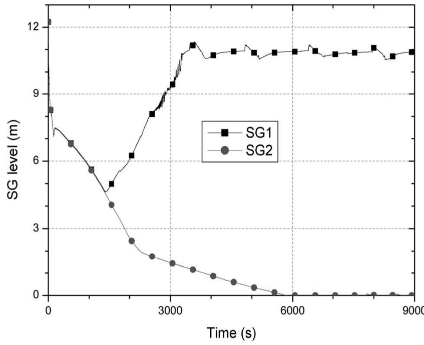

Fig. 11- SGs collapsed level for AFWS heat removal

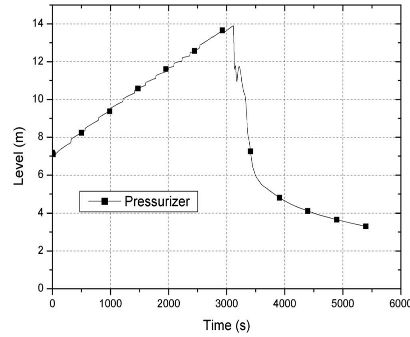

Fig. 15-Pressurizer collapsed level for a blackout case

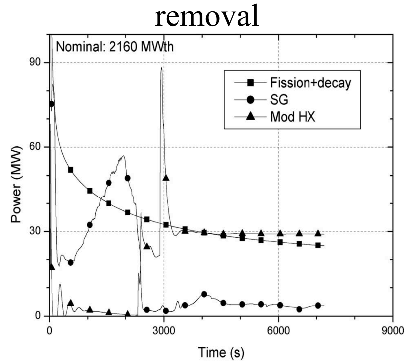

Fig. 12-Plant power for cooling by Moderator RHR chains

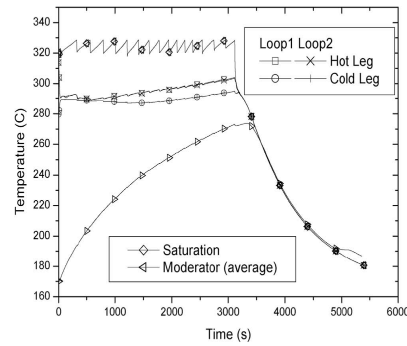

Fig. 16-Primary and moderator temperatures for a blackout case

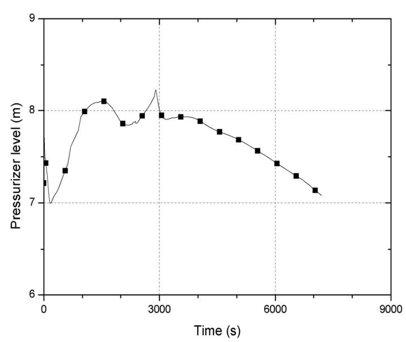

Fig. 13-Pressurizer collapsed level for cooling by Moderator RHR chains

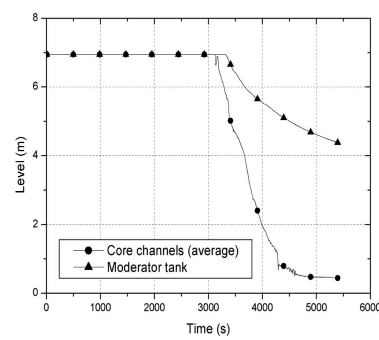

Fig. 17-Core channels and moderator tank water level for a blackout case

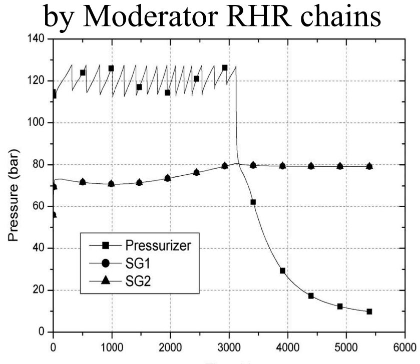

Fig. 14-Primary and secondary system pressure for a blackout case

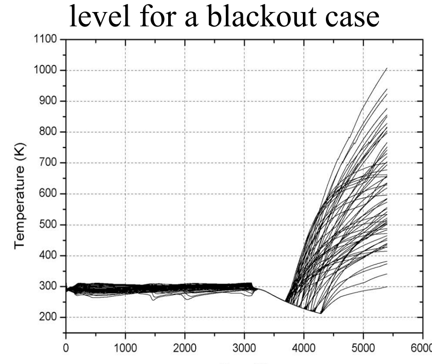

Fig. 18-Fuel clad temperatures (at different locations) for a blackout case