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MELCOR steady state calculation of the generic PWR of 40MWth

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ABSTRACT

After the two most significant nuclear accidents in history – the Chernobyl Reactor Four explosion in Ukraine (1986) and the Fukushima Daiichi accident in Japan (2011) –, the Final Safety Analysis Report (FSAR) included a new chapter (19) dedicated to the Probabilistic Safety Assessment (PSA) and Severe Accident Analysis (SAA), covering accidents with core melting. FSAR is the most important document for licensing of siting, construction, commissioning and operation of a nuclear power plant. In the USA, the elaboration of the FSAR chapter 19 is according to the review and acceptance criteria described in the NUREG-0800 and U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.200. The same approach is being adopted in Brazil by National Nuclear Energy Commission (CNEN). Therefore, the FSAR elaboration requires a detailed knowledge of severe accident phenomena and an analysis of the design vulnerabilities to the severe accidents, as provided in a PSA – e.g., the identification of the initiating events involving significant Core Damage Frequency (CDF) are made in the PSA Level 1. As part of the design and certification activities of a plant of reference, the Laboratory of Risk Analysis, Evaluating and Management (LabRisco), located in the University of São Paulo (USP), Brazil, has been preparing a group of specialists to model the progression of severe accidents in Pressurized Water Reactors (PWR), to support the CNEN regulatory expectation – since Brazilian Nuclear Power Plants (NPP), i.e., Angra 1, 2 and 3, have PWR type, the efforts of the CNEN are concentrated on accidents at this type of reactor. The initial investigation objectives were on completing the detailed input data for a PWR cooling system model using the U.S. NRC MELCOR 2.2 code, and on the study of the reference plant equipment behavior – by comparing this model results and the reference plant normal operation main parameters, as modeled with RELAP5/MOD2 code.

Keywords: **severe accident, MELOR, PWR.**

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1. INTRODUCTION

Introduction must clearly explain the context of the article. State the precise objective and hypothesis to be discussed.

For improving the NPPs safety, the study of severe accidents has become increasingly important [1]. The normal operation and accident in nuclear reactor has been modeled and studied with different codes along of the years [2-5]. The reference [6] demonstrated the importance of the self-initialization of the steady-state on the simulations of transient responses and accidents sequences timing in a PWR modeling on MELCOR. MELCOR is a fully integrated, engineering-level computer code developed by Sandia National Laboratories for the NRC, whose primary purpose is to model the progression of accidents in Light Water Reactor (LWR) NPPs [7-8]. The MELCOR code has great modeling flexibility afforded by a control volume and control functions approach. Its models have been and are still being improved, including enhancements to the code based on results from Fukushima [9-10]. In Brazil, past efforts have been made to test simulation codes and models to advance severe accident phenomenological studies and severe accident management, including the use of the MELCOR code [11]. The main objective of the work presented in this paper was modeling and analyzing the operational parameters such as pressure, temperature, and flow rate, until the "achievement of the steady state of a reference PWR NPP, using the severe accident MELCOR code version 2.2. The results of parameters under normal operating conditions (steady state conditions) are compared with those obtained with RELAP5/MOD2 code [12-13]. Posteriorly, this model will be used to simulate different types of severe accident in this reference plant, supporting deterministic and probabilistic analysis, as required to licensing of new plants, as described in NUREG 0800 [14]. Starting with the description of the reference PWR plant, next sections present the developed model – including the model inputs, the nodalization, the heat structures, and the control functions and tables – and its results for the steady state condition. Before that, however, this paper presents a short description of the MELCOR code.

2. SHORT DESCRIPTION OF THE MELCOR COMPUTER CODE

Several versions of the MELCOR code have been developed by Sandia National Laboratories for plant risk assessment and source term analysis since 1982. In this study, MELCOR version 2.2 is utilized for the fulfillment of the addressed objectives. This code is used to treat the vast spectrum of severe accident phenomena, including thermal-hydraulic response in a reactor coolant system (RCS) and containment, core heat-up, degradation and relocation, and fission product release and transport behavior in LWR, High Temperature Gas Reactor (HTGR) [15] and Spent Fuel Pools (SFP) [16-17].

The MELCOR code structure is constructed on separated packages. Control Volume Hydrodynamics (CVH) package calculates the thermal/hydraulics of control volume including one phase or two-phase flow. The Heat Structure (HS) package models the mass and heat transfer across its boundary surfaces into control volumes of the intact and solid structures as walls, bottoms, and ceilings. The Flow Path (FL) package models in connection with the CVH package, the flows. The FL package is used to simulate equipment as valves, check valves and pumps. The pipes are simulated with flow paths or by control volumes (CV). Each CV is connected to another CV by a FL. The Control Function (CF) package evaluates user-specified system of functions that controls opening/closing valves – and controlling plot writing, defining plot variables, etc. [7-8].

The other packages can be divided on three general structures: 1) Basic physical phenomena: hydrodynamics, heat and mass transfer to structures, gas combustion, aerosol and vapor physics; b) Reactor-specific phenomena: decay heat generation, core degradation, ex-vessel phenomena, sprays, and; c) Engineered Safety Features (ESF) and support functions: thermodynamics, equations of state, other material properties, data-handling utilities, equation solvers [7-8].

The building of a reactor input model requires input for Core Thermal Response (COR) package. Ejection of the molten material from the reactor vessel to the cavity is modeled by Energetic Fuel Dispersal Interaction (FDI) package through transfer of materials between packages. The Interaction between molten material and cavity concrete are simulated by the Cavity (CAV) package. Gas combustion is modeled by BUR package and Radio Nuclide Release, Transport, and Chemistry are modeled by (RN) package. The Containment Spray (SPR) package models heat and mass transfer to spray system [7-8].

3. DESCRIPTION OF THE REFERENCE PWR PLANT

The nuclear power plant considered is a 48MWth two-loop PWR in the design phase. In this plant, the pressure vessel, steam generators (SG), the reactor coolant pumps (RCPs), and the pressurizer (PZR) are enclosed in a steel containment, which is surrounded by a water pool used as shielding and ultimate heat sink. A confinement building houses the steel containment and a secondary system with two turbo-generators. The core consists of 21 fuel elements with lattices of 17x17 and an active fuel height of 0.987 m. The nominal plant operating parameters are given in Table 1.

Table 1: Nominal operating parameters.

Parameter	Nominal value
Primary side pressure (MPa)	1,31
Hot Leg temperature (K)	558,08
Cold Leg temperature (K)	537,95
Feed water flow rate per steam generator (SG) (kg/s)	9,45
Secondary side pressure (MPa)	0,377
SG liquid level (m)	2,53
SG steam flow rate (kg/s)	40,0

As illustrated in Fig.1, this plant includes two single loops, and comprises the reactor pressure vessel (RPV), two hot legs feeding U-tube type SG respectively, two cold legs –each with three primary reactor coolant pumps –, and a PZR. The scheme presented in Fig.1
is better explained in the next section.

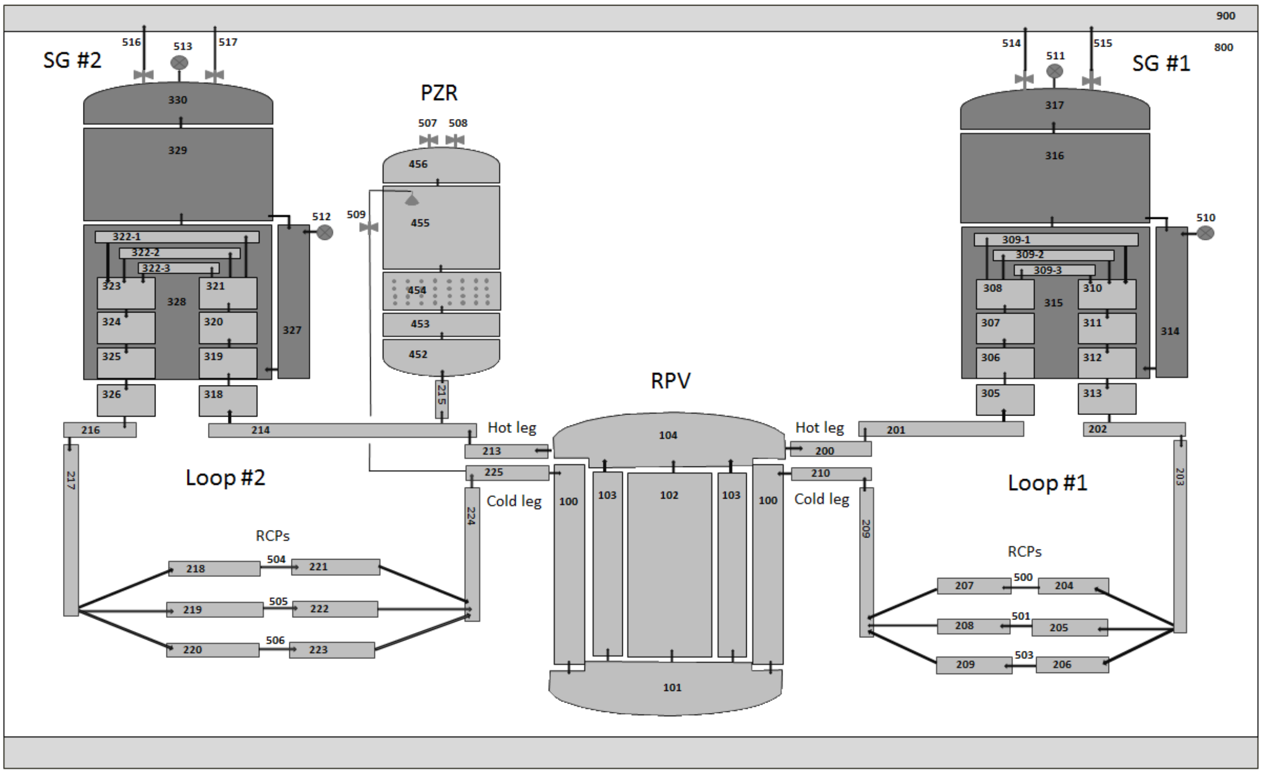

Figure 1: Scheme for the reference NPP model in CVH and FL packages

4. MODEL FOR THE REFERENCE PLANT REACTOR COOLING SYSTEM

In the model illustrated in Fig.1, hot legs are lumped together with the RPV upper head and the reactor core volumes, and the cold legs with the reactor annular descending channel. The hot leg pipe is simulated by 2 control volumes CV 200 and CV 201 to the Loop #1, and CV 213 and CV 214 to the Loop #2, that are used for the SG piping. The coolant is injected through the cold leg pipe into the RPV, and that consists of 11 control volumes for each loop, as indicated in Table 2.

Table 2: Control volumes – Legs of Loop #1 and #2

CV	Name
CV 200, CV 201	hot leg#1
CV 213, CV 214	hot leg#2
CV 202 - CV 210	cold leg #1
CV 216 – CV 225	cold leg #2

These CVs are connected through the flow path represented by one arrow for mass and energy transference. The arrows in Fig. 1 represent FLs indicating the direction of positive flow (not the flow itself). Loop #2 is connected to the PZR. As indicated in Table 3, FLs (FL 500, FL 501, FL 502, FL 503, FL 504, and FL 505) are associated with the six reactor coolant pumps. These pumps are simulated with FANA pump model – available in the MELCOR code [7-8]. The water injection and the vapor consumption in the SGs are considered constant in this model.

Table 3: Flow path – Pumps of Loop #1 and #2.

FL	Name
FL 500, FL 501, FL 502	reactor coolant pumps #1
FL 503, FL 504, FL 505	reactor coolant pumps #2

The PZR contains the atmospheric release valve (FL 506) in the upper end and spray valve (FL 507) to handle pressure excursions in excess. Also, the PZR is connected with a surge line (CV 215) to primary Loop #2 on its hot leg. These PZR valves are presented in Table 4 – the CV 215 will be discussed later in this paper, with the nodalization of PZR. When the PZR reaches the set point value pressure, the atmospheric release valve discharges steam, so the pressure is kept approximately constant. Also, spray valve of relatively cool water is turned on inside the PZR, lowering the coolant temperature in the PZR and thereby lowering the pressure.

Table 4: Flow path – PZR Valves

FL	Name
FL 506	Atmospheric release valve
FL 507	Spray valve

The reactor containment is simulated with single control volume and the model is simplified. It is represented with one control volume (CV 800). The last control volume is a water pool providing ultimate heat sink (UHS) conditions for the outside surface of containment (CV 900). The heat structures in the containment (CV 800) are associated within the water pool (CV 900) (See Table 5).

Table 5: Control volumes – Containment

CV	Name	HS	Name of connected
CV 800	Inner containment	x	CV 100, CV 104, CV 900
CV 100CV 900	Water pool outside of containment	x	CV 800

Fig. 2 shows the PWR reactor core and pressure vessel model, as modeled in CVH package. The inner RPV is divided into five control volumes that represent the annular descending channel (CV 100) which is connected with the cold leg, the lower plenum (LP) (CV 101), the reactor core (CV 102), the core bypass (CV 103) and the upper head (CV 104) which connects with the hot leg. The simulated HS with associated CVs for the RPV are presented in Table 6.

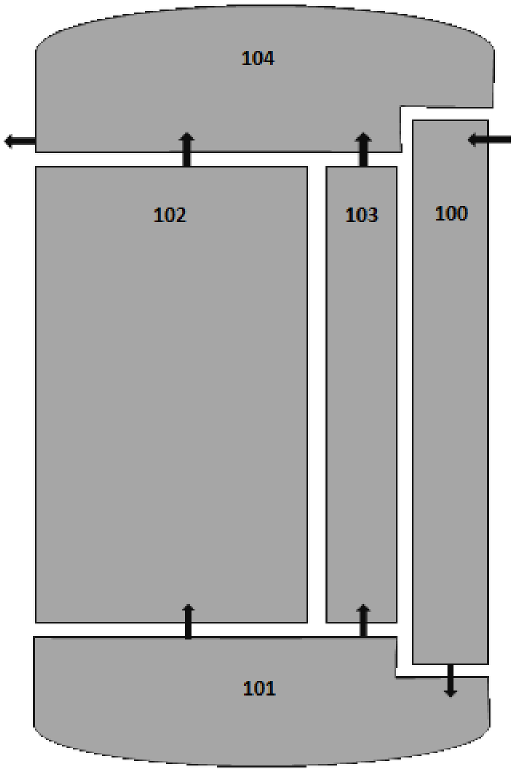

Figure 2: RPV model in CVH and FL packages

Table 6: Control volumes – RPV

CV	Name	HS	Name of connected
CV 100	annular descending channel	x	CV 101, CV 210, CV 225
CV 101*	LP	x	CV 100, CV 102, CV 103
CV 102*	reactor core	x	CV 101, CV 104
CV 103	core bypass	x	CV 101, CV 104
CV 102	upper head	x	CV 102. CV 103, CV 200, CV 213

()*Simulated in COR package

In the COR package, the reactor core and LP of the RPV are divided axially and radially into cells. A ring in fueled region represents a group of assemblies. The core nodalization developed consists in 3 radial rings (as illustrated in Fig.3) and 12 axial levels (as illustrated in Fig.4). The number of fuel elements and mass in each of radial rings for the axial levels 6-10 (the fuel active region) are shown in Table 7.

The core is built on unequal elevations of axial nodes. The width of each level is made on the base of estimation of the code capabilities to calculate material mass distribution.

Table 7: Distribution of fuel elements and mass in each of radial rings (axial levels 6-10).

Radial rings	Number of Fuel elements	% Mass	Mass of UO2 (Kg)	Mass of clad (Kg)	Mass of supporting structure (Kg)	Mass of NS (Kg)	Mass of Ag-In-Cd (Kg)
1	1	5%	146,43	41,64	34,02	9,97	5,89
2	6	29%	878,57	249,83	204,11	59,82	35,32
3	14	67%	2050,0	582,93	476,27	139,59	82,41

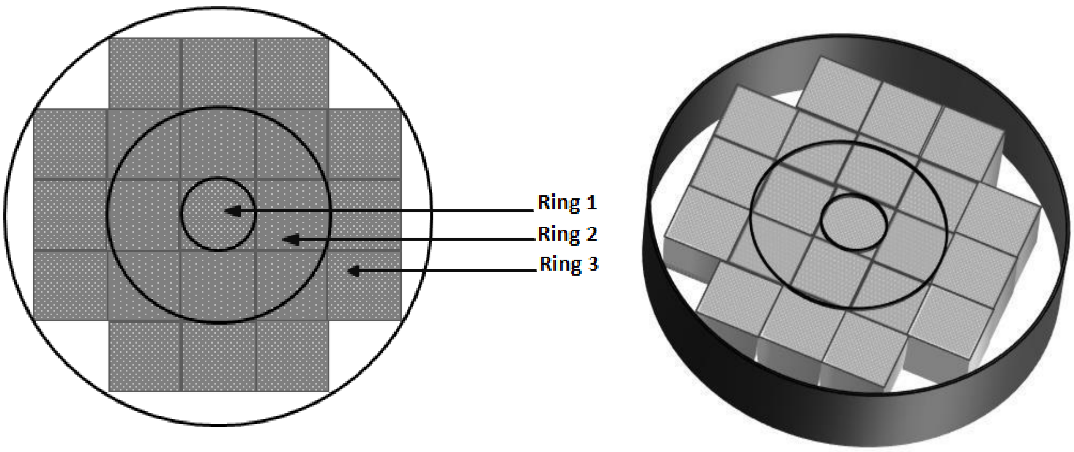

Figure 3: Radial rings and axial levels of reference reactor core.

At the COR package, the rings represent the fuel elements with cladding, supporting structures and non-supporting structures. Figure 4 shows the nodalization scheme of the reactor core including lower head wall.

The lower head wall directly below the core was divided into 6 segments as illustrated in Fig. 4. The support plate is included in the axial level 3. The level 4-5 and 11-12 represent the inactive bottom and top of the reactor core, respectively.

For the RPV, some heat structures were modeled, as follows: a) between the core (CV 102)and the bypass (CV 103), corresponding with the 4 to 12 axial levels – i.e., nine heat structures were considered; b) between cylindrical portion of the bypass (CV 103) and the descending channel (CV 100); c) between the descending channel (CV 100) and the hemispherical portion of the LP (CV 101); d) among the core (CV 102), the bypass (CV 103) within hemispherical portion of upper head (CV 104) – i.e., two heat structures were considered; e) between the hemispherical portion of the upper head (CV 104) and the containment (CV 800), and; f) between the cylindrical portion of descending channel (CV 100) and the containment (CV 800). Note that between the LP (CV 101) and the containment (CV 800) there are no heat structure, rather it is modeled in MELCOR as part of the COR package.

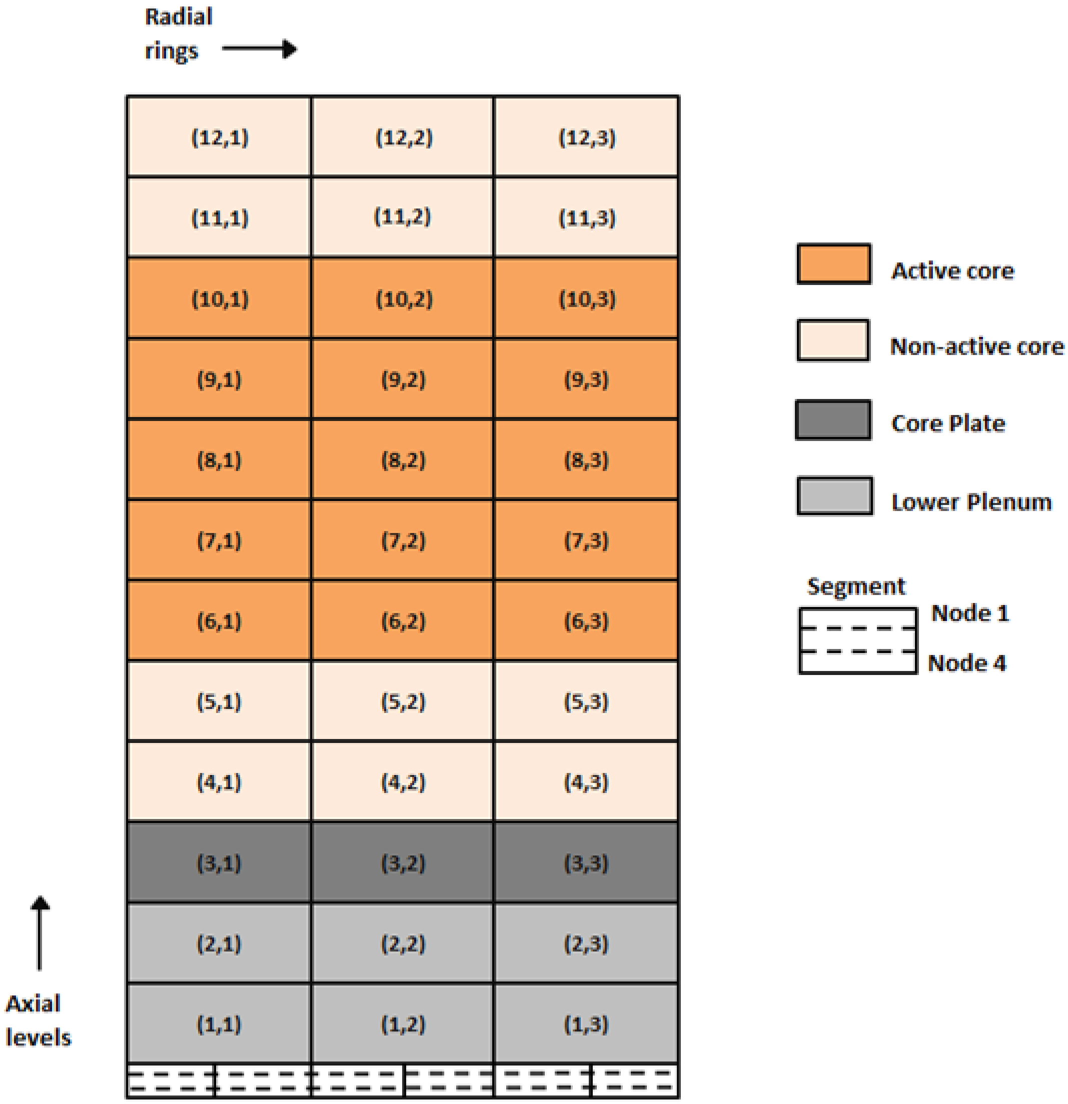

Figure 4: LP and reactor core nodalization.

Fig. 5 shows the SG (Loop #2) and PZR model. The two SGs have been divided in 15 CVs, two of them are representing the collectors cameras of reactor inlet (CV 305 and CV 318) and outlet (CV 313 and CV 326), one is representing the riser & moisture separator (CV 316 and CV 329), one representing the steam dryers & steam dome (CV 317 and CV 330), one representing the barrel (CV 315 and CV 328), one representing the down comer (CV 314 and CV 327) where feed water enters, and the other are representing the U-tube where the primary coolant flows. The U tube of the each SG is represented by 9 CVs representing ascending and descending side (CV 306 - CV 312 and CV

319 - CV 325) respectively, and the curve part (CV309 and CV 322). The simulated HS with associated CVs for the SG (Loop #2) are presented in Table 8. The SGs are modeled with 18 heat structures between the U-tubes CVs and the barrels CVs.

The PZR is a vertical, cylindrical vessel with hemispherical top (CV 456) and bottom heads (CV 452) that connects the PZR with the primary circuit. As presented in Fig. 5 submerged electrical heaters (CV 454) are installed between the lower compartment (CV 453) and upper compartment (CV 455) to increase the pressure in the reactor coolant system. The simulated HS with associated CVs for the PZR are specified in Table 9. The heat structures in the PZR are in the portion of the submerged electric heaters (CV 454) modeled as solid cylindrical geometry.

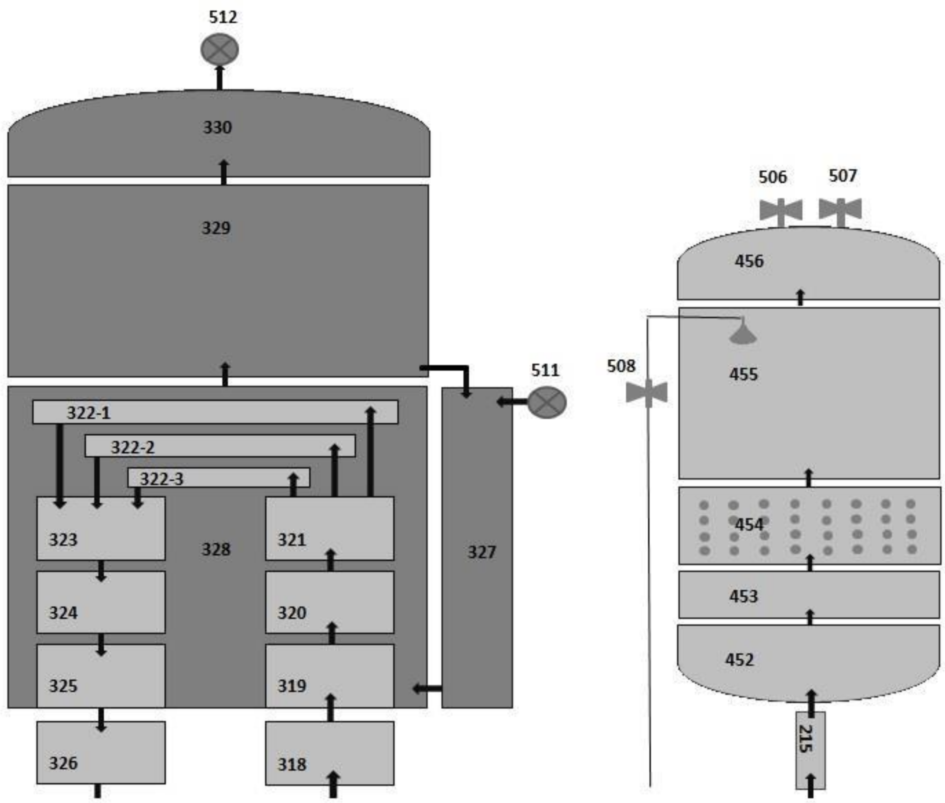

Figure 5: GV and PZR models in CVH and FL packages

Table 8: Control volumes – SG of Loop #2

CV	Name	HS	Name of connected
CV 318	collectors cameras of reactor inlet		CV 214, CV 319
CV 326	collectors cameras of reactor inlet		CV 325, CV 216
CV 319-CV 325	U tube	x	CV 328
CV 322	curve part of U tube	x	CV 328
CV 329	riser & moisture separator		CV 327, CV 328, CV 330
CV 327	down comer		CV 328, CV 329, CV 511
CV 511	feed water		CV 327
CV 512	steam release		CV 330

Table 9: Control volumes – PZR

CV	Name	HS	Name of connected
CV 456	hemispherical top		CV 455
CV 452	bottom heads		CV 453, CV 215
CV 453	lower compartment		CV 452, CV 454
CV 454	submerged electrical heaters	x	CV 453, CV 455
CV 455	upper compartment		CV 454, CV 456
CV 215	surge line		CV 452, CV 214

4.1. Control Function (CF) and Tabular Functions (TF)

In the model presented in this paper, the logic of opening and closing of the relief and safety valves of the PZR, the spray valve of PZR, and the logic of switching on and off banks of electric heaters were controlled by CF and TF.

The PZR pressure is mainly regulated by adjusting the heater power and the spray valve opening. If the pressure goes below a certain set point the electrical heaters switches on. There are three kinds of heaters, proportional heaters with medium power, and back up heaters that can add power to a large total rate. Heater power as a function of pressure is regulated by a CF.

Material properties information was based on what is available in MELCOR. The exception was the material for the SGs. Using TF, the material Inconel 800 properties were added in for the SGs. Other TF also was developed to insert a fixed flow rate for the SGs inlet and outlet and for steam injection and release for the SGs.

5. RESULTS FOR THE STEADY STATE CONDITION

The steady state calculation is an important objective for the model development to later allow studying the accidents progression. A steady state calculation using the model for the reference plant reactor cooling system – presented in the previous section (Section 4) – was conducted to verify the success achieved in the use of MELCOR code. This verification was performed by comparing the model results with the nominal values reported in the RELAP5/MOD2 model [12]. The results are presented in Table 10 to Table 13.

Table 10: Comparing the steady state simulation and the reference data – RPV.

Reactor Pressure Vessel parameter	Relative Error (%)
Reactor thermal power (100% of full power), MWt	0,00
Primary pressure (MPa)	1,27
Average temperature in vessel (K)	-2,52
Average heating of coolant at the inlet and outlet of the vessel (K)	-11,74

Table 11: Comparing the steady state simulation and the reference data – SG.

Steam generator parameter	Relative Error (%)
SG outlet pressure (100% of full power) (MPa)	0,00
Feed water temperature (K)	0,00
Generated steam vapor temperature (K)	0,15
Average temperature at GV (K)	0,55
SG liquid level (m)	4,34
Steam flow rate (kg/s)	0,00
Feed water flow rate (kg/s)	0,00

Table 12: Comparing the steady state simulation and the reference data – PZR

Pressurizer parameter	Relative Error (%)
Operation pressure (MPa)	0,01
Operation temperature (K)	0,45
PZ liquid level (m)	0,55

Table 13: Comparing the steady state simulation and the reference data – Primary Circuit.

Primary circuit parameter	Relative Error (%)
Temperature in hot leg (K)	-1,56
Temperature in cold leg (K)	1,99
Primary flow rate (kg/s)	-0,53

Table 10 presents the parameters for the RPV. The most significant deviation was found for the parameter related to average heating of coolant at the inlet and outlet of the vessel. As presented in Table 11 – dedicated to the parameters for the SG –, the resulting SG level is between the minimum and the maximum levels allowed in the RELAP5/MOD2 model [12]. Table 12 and Table 13 present the parameters for the PZR and for the Circuit, respectively.

Considering the consulted literature [17-20], the comparison based in relative error shows a good agreement between the reference values presented in the RELAP5/MOD2 model and MELCOR calculation – i.e., the evaluated error is in the same magnitude order as the consulted papers [21] –, confirming that the MELCOR input model represents with good accuracy the reference plant.

6. CONCLUSIONS

An input model of the reference PWR NPP was developed with MELCOR code version 2.2. It is not possible to validate the model due to the lack of design data and experiments. The calculations for the steady state main parameters from this model were compared to the results of a model developed with the RELAP5/MOD2 code in order to gain experience in MELCOR code.

The most significant relative error was observed for the average heating of coolant at the inlet and outlet of the vessel (11.74%). Nevertheless, the relevance of this difference is attenuated by the lower relative error observed for the temperatures in the hot legs and the cold legs (-1.56% and 1.99%, respectively). The general results show a good agreement between the reference data and MELCOR calculation (i.e., considering the results of the consulted literature) – except for the difference between the temperature of coolant at the inlet and outlet of the vessel, the relative error were lower than 5% for the rest of the parameters.

This simulation is the first study that is performed with the severe accident MELCOR code in LabRisco/USP. The developed model is preliminary, not all systems of the reference NPP were included. For future studies, a better representation of the reference plant will be made to simulate different types of transients and severe accidents and verify the performance of safety and mitigation components.

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