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Loss of coolant accident in pressurized water reactor. Prediction of a 6-inch cold leg break with Relap5 and Cathare 2

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Abstract

This paper describes the approach to model Loss of Coolant Accident (LOCA) in the 900MWe Nuclear Power Plant Westinghouse PWR reactor.Two thermal-hydraulic system codes were applied: RELAP5 and CATHARE 2 to simulate steady state conditions and a transient evolution. The purpose of this paper is to analyse a 6-inch cold leg break in both codes and compare calculations performed in frame of the benchmark procedure between RELAP5 and CATHARE 2 codes. Obtained steady state and transient results for both codes are comparable and predict similar accident sequences. The results are not final, and the benchmark procedure is ongoing.

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Keywords: RELAP5, CATHARE2, PWR, IB-LOCA, Nuclear Safety

1. Introduction

There is a strong need to understand the characteristics of the Loss Of Coolant Accidents (LOCA) in order to predict the behavior of the protection and safeguard systems designed to withstand any break size and break location in the primary circuit of the NPP. The NPP's supplier is obligated to prove that the plant parameters during LOCA should not violate the acceptance criteria [1]. For this objective numerous thermal-hydraulic codes have been developed in different countries. Among the most popular computer codes used for the safety analysis of the NPP are the RELAP5 (USA) and the CATHARE 2 (France). Both codes are used to simulate the same Design Basis Accident (DBA)

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describing a Loss Of Coolant Accident in a so called benchmark calculation. The work shown in this paper has been prepared in scope of the SARWUT project at Warsaw University of Technology.

2. Benchmark procedure

Benchmark calculation between two thermal-hydraulic codes requires a deep understanding of the accident physics, code's specifications and experiences in the creation of plant model. The analytical procedure should involve the following steps:

- Step1: preparation of equivalent inputs

Remove the inconsistencies in the nodalization scheme, initial conditions and model's options as far as possible.

- Step2: performing the steady state and transient calculations

Discrepancies in results are discussed in details to avoid the impact of unequal models in input (break simulation and option, local pressure losses, main coolant pump characteristic, decay power history).

- Step3: definition of a new basic input deck

Checking for inconsistencies of the new inputs which have not been considered in step 1. Performing new calculations.

- Step 4: Comparison of results and qualifying the capability of the code based on the simulation of an accident. Comparison and assessment of results accordingly to the code's governing equations, specific models and features.

The work presented in this paper describes such a benchmark procedure between CATHARE 2 v.2.5_2 mod8.1 and RELAP5 mod 3.3 Patch04.

3. System description

All over the world a considerable number of PWR nuclear reactor concepts have been proposed over the second half of the twentieth century and the first decade of twenty-first century. Over half of the 400 nuclear power plants in operation worldwide are PWR finished by Westinghouse or its licensees or former licensees. Those power plants share almost identical primary circuit design, same RPV, same fuel assemblies with differences in fuel composition due to effort in increasing a time of fuel campaign. One of the most representative design concepts is Westinghouse 900 MWe 3-loop which is at the centre of this paper.

Reactor core approximate thermal output is 2785 MWth with electrical 900MWe, steam pressure is 66 bar, reactor vessel internal diameter is 398.8 cm. Hot leg internal diameter is 73.7 cm, cold leg internal diameter is 69.9 cm. Core consists of 157 fuel assemblies, fuel assembly array is set 17 by 17 fuel rods, fuel length 365.8 cm [2].

Safety Systems: The primary and secondary side are equipped with emergency cooling systems which can be used during abnormal work conditions. Those systems are:

On the primary side:

- Lower head safety injection (LHSI) – 2 injection systems (on unbroken loop and pressurizer loop)
- Accumulators injection – four acculumators each having 31.74 m3 of liquid opening at pressure of 45 bar.

On the secondary side:

- Emergency steam generator feedwater system.

4. Code description and applied nodalization:

Most currently used two-phase flow models in available "best estimate" thermal hydraulic system codes are based on the "two-fluid model". Phases are treated as interpenetrating continua and separate ("macroscopic") balance equations for each phase are obtained by space and/or time averaging of the local instantaneous simple flow equations, with their source terms representing the interfacial transfers for mass, momentum and energy. Because of averaging, information on local flow phenomena, in particular at the interface separating the two phases or at the near-wall region, is lost and has to be recompensated by further modeling . For the spatial "and or" time averaging of the instantaneous phase equations for quasi-1D pipe flows, the "macroscopic" balance equations of the two-fluid model can be formulated for vapour (i=g) and liquid (i=l) as follows:

$$\frac{\partial}{\partial t}(\alpha_i \rho_i) + \frac{1}{A}\frac{\partial}{\partial x}(\alpha_i \rho_i \vec{v_i} A) = \Gamma_i \qquad (1)$$

With

$$\sum_{i=g,l} \Gamma_i = 0 \tag{2}$$

Momentum:

$$\frac{\partial}{\partial t}(\alpha_i \rho_i \vec{v_i}) + \frac{1}{A}\frac{\partial}{\partial x}\left(\alpha_i \rho_i \vec{v_i}^2 A\right) + \alpha_i \frac{\partial p}{\partial x} \Delta p^{int} \frac{\partial \alpha_i}{\partial x} = F_i^{ext} + F_i^{wal} + F_i^{int} + \Gamma_i(v^{int} - \vec{v_i}) + F_i^{vm} \tag{3}$$

Energy:

$$\frac{\partial}{\partial t}\left(\alpha_i \rho_i [\![(u)]\!]_i + \vec{v_i}^2\right)\right) + \frac{1}{A}\frac{\partial}{\partial x} + p\frac{\partial \alpha_i}{\partial t} = Q_i^{ext} + Q_i^{wal} + Q_i^{int} + \Gamma_i\left(h_i + 0.5\vec{v_i}^2\right)^{int} + F_i^{ext}\vec{v_i} + F_i^{wal}\vec{v_i} + F_i^{int}\vec{v_i}^{int} \tag{4}$$

Each code uses their own method of solving those equations and incorporating the aforementioned additional models. Only the critical flow model will be described since it has most bearing on the calculation results in case of LOCA simulations.

<u>*RELAP5 thermal-hydraulic code:*</u> The RELAP5/MOD3.3 code has been developed for best-estimate transient simulations of light water reactors coolant systems during postulated accidents. The code may be used to model the coupled behavior of the reactor coolant system and the core region during loss-of-coolant accidents and operational transients such as anticipated transients without SCRAM, loss of offsite power and loss of feedwater flow. A generic modeling approach is used which permits simulating a variety of thermal hydraulic systems. Control system and secondary system components are included to allow modeling of plant controls, turbines, condensers, and secondary feedwater systems. The RELAP5 default critical flow model (described as the Henry-Fauske model) represents a first-principle approach to the calculation of subcooled, two-phase mixtures and a vapor critical discharge. The model is based on a one-dimensional flow assumption, and discharge coefficients are generally necessary to account for geometry-specific, two-dimensional effects. For the subcooled flow, an empirical correlation is employed to calculate pressure undershoot (liquid superheat) at the choke point to estimate the choke plane pressure. The thermal equilibrium assumptions were provided in the developing of an analytic choking criterion for two-phase flow. The model has been tested against a wide variety of data from experimental facilities and against tabulated critical flow models, such as Henry-Fauske [8][11].

<u>*RELAP5 model nodalization and assumptions:*</u> The nodalization scheme of the reactor pressure vessel is presented in Figure 1(a). The reactor core is modelled by a pipe component no. 170, with 14 axial meshes, where the meshes 2 through 13 represent the heated length of the core. Downcomer is modeled using two annulus components: 130, 220 and branches from 100 to 120. Core bypass is modeled by a pipe component 180. Lower plenum of the vessel is represented in the model by the single volume 150 and branch 140. Guide tubes of the reactor vessel are modeled using pipe 250. Upper plenum of the vessel is modeled by a branch 200 and single volume 210. Single volumes 230 and 260, together with branch 240 represent the upper head.

Nodalization of the primary side is presented in Figure 1(b). The model consists of 3 separate loops. In the figure, loop no. 3 is presented. Loops 2 and 3 are modelled identically with the only difference being that the pressurizer is connected to loop no. 3. Finally, loop no. 1 holds also the trip valve (element no. 908 in Figure 2), which is 6 inches in diameter, in order to simulate the postulated accident.

Hot leg is modeled by pipe 300, branch 310 and single volume 320. The steam generator inlet plenum is modeled by the single volume 330. The steam generator U-tubes are represented by the pipe 340. The 10-mesh nodalization applied for the U-tube pipe has been demonstrated to be sufficient to simulate the phenomena such as reflux condenser or reduction of steam generator water level on the secondary side [3]. Single volume 350 represents the steam generator outlet plenum. Pipe 360 represents the cross-over leg. The pump component 366 is used for modeling the primary coolant pump. RELAP5 Westinghouse single phase pump indicator tables are used with two-phase multipliers. The cross-over leg is modeled by the single volume 370, branch 380 and pipe 390. For each loop the high and low pressure safety injection pumps are modeled by the time dependent volumes and junctions (elements 374 and 376 respectively). The fluid temperature is defined in the time dependent volume, while the injected flow rates are given in the time-dependent junction as function of the cold leg pressure. An accumulator component 372 is used to simulate the injection of high borated water from the nitrogen-charged accumulator.

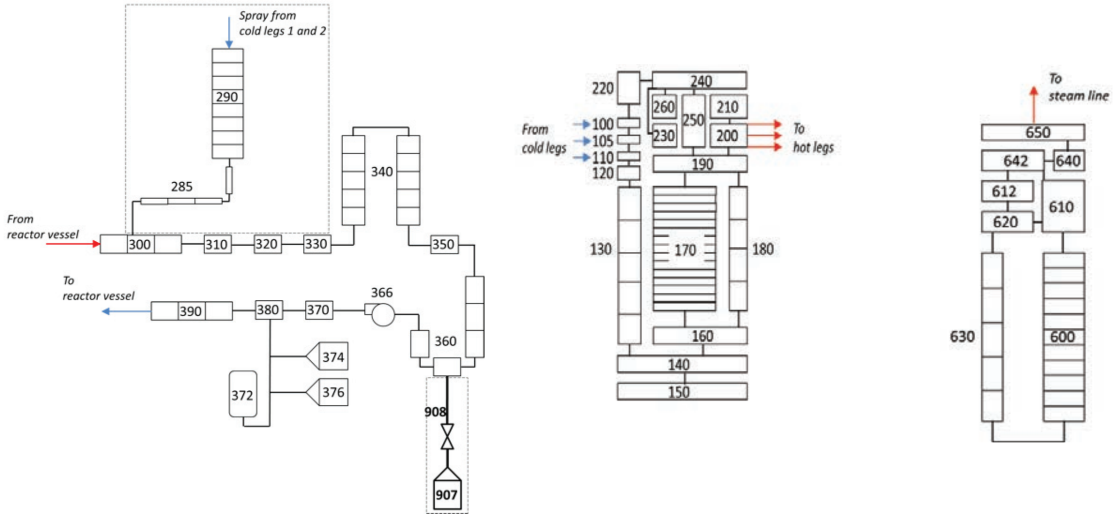

Figure 1 (a)Nodalization of the reactor pressure vessel in RELAP5 (b)Nodalization of the primary side in RELAP5 (c) Nodalization of steam generator secondary side in RELAP5

Nodalization of the steam generator secondary side is presented in Figure 1(c). Main feedwater enters the downcomer annulus at a branch 620 and mixes with water backflow from the separator component 610. This combined flow moves downwards the downcomer annulus 630 and upwards enters the riser 600. The riser is subdivided into 11 axial meshes, where the first 8 meshes are thermally connected to the primary side U-tubes and the remaining 3 meshes simulate the channels feeding the separator swirls. The separator cylinder is modeled by the separator component 610. The volume between the outermost separator cylinder and the inside of the SG shell is modeled by a single volume 612. The top of the steam generator, including the dryer and the steam dome regions, is modeled by the branches 642, 640 and single volume 650. Steam from steam generators is then collected in main steam line and supplied to the turbine, modeled by time-dependent volume, which are not presented in the figure.

CATHARE 2 thermal-hydraulic code: The CATHARE second generation thermal hydraulic code has been developed jointly by the CEA, EDF and Areva NP to carry out safety analyses. CATHARE is a modular, two fluids code, capable of modeling test facilities as well as Pressurized Water Reactors. Two types of elements: axial and volume are used in order to model the systems.

The implemented flow models in CATHARE are able to calculate precisely the two-phase flow situations such as stratification flow, counter current flow and the critical flow. The equation to determine the critical mass flow rate consists of several parameters such as the mixture density, the void fraction, the ratio between length and the hydraulic diameter, the pressure losses and the difference between the pressure of the liquid and the saturation pressure at given temperature. This difference can be approximated by the correlations which depend on the liquid temperature. If the liquid temperature is lower than the saturated temperature at given pressure (single phase liquid), then the difference is taken from the equation base on the given pressure and the temperature in equilibrium condition [4,5,6].

CATHARE 2 model nodalization and assumptions: The RPV is composed of four volumes and 3 axial components. Coolant flow is divided into two streams. First flows up into the upper head (UPHEAD) via the downcomer upper part (VOLDOWN) and the second flows down into the lower plenum (LWPLEN) via the downcomer lower part (LDOWNCO). From the lower plenum coolant flow is divided in two: the core inlet and the core bypass flows. The components lower plenum, downcomer upper part and the upper head are presented by the volume elements. The downcomer lower part, the core bypass (BYPASS) and the core (MIDCORE) are presented by the axial elements. The hydraulic part of the core is divided into 12 volume components with rod-bundle geometry. The fuel in the core is irradiated and the core power is calculated by the point kinetic model.

Hot legs, primary side of steam generators, seal loops and cold legs are modeled with use one component which is an axial component. Main coolant pumps are modeled using homologous degradation curves. The pressurizer is attached to the hot leg of the broken loop no. 3 by a surge line. The Pressurizer is presented by a volume component. The simplified model is appropriate for the calculation of primary system pressure due to the fact that a homogenous liquid drain establishes in the pressurizer during the depressurization (no stratification of liquid temperature). The break pipe connects the cold leg loop 3 with the containment which is represented with the

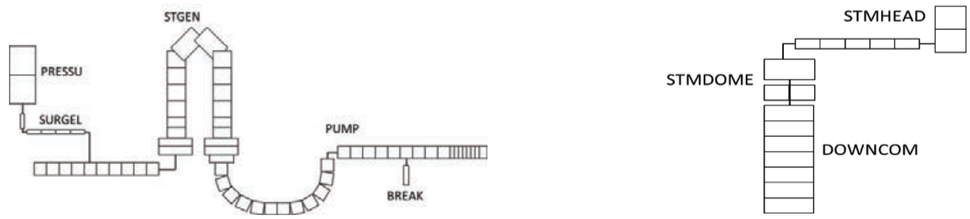

Figure 2 (a) Nodalization of the primary side in CATHARE 2. (b) Nodalization of steam generator secondary side in CATHARE2.

boundary condition operator (BCONDIT). There is no flow in the break pipe during the steady state (BLIND mode). Safety systems are modeled by using the SOURCE elements which inject cooling water into the intact cold legs. Injected water from the safety pumps is given as function of pressure. Figure 2(a) presents the nodalization scheme used in CATHARE.

Figure 2(b) illustrates the secondary system nodalization. The downcomer of SG and the riser are connected by a short horizontal pipe which increases its area from the downcomer area to the riser area. All three components mentioned above are modeled by one axial element. At the top of the riser the separation of the two phase flow is set at the junction between the riser and the volume which simulates the separator and steam dome. The steam generator of the intact loop has a weighting factor 2, that means 2 identical steam generators are present. The main steam pipes, equipped with the safety valves, connect the steam generators with the turbine via the common header.

5. Results and discussion

5.1. Scenario description:

General assumptions:

- Both codes used point kinetic model for the calculation of core power during the steady state
- The reactor is shut down at a setpoint of pressure in the pressurizer going below 129.5 bar
- Pumps are shut down in the moment of the reactor SCRAM and follow the same degradation curves
- No operator action is performed throughout the accident
- For the considered 6-inches in diameter break accident, only the safety injections into the intact loops are taken into account
- For the LOCA studies a conservative, chopped cosine axial power distribution is used. During the steady state and transient phase the core power is calculated by means of the point reactor kinetics model. The decay heat curve is defined by a general table.

<u>Steady state</u>**:** The first step before the performing of the transient calculations is to achieve a proper steady state. In the scenario, the initial and boundary conditions calculated by CATHARE are used as referent data for RELAP steady state calculation.

Table 1 presents the plant initial parameters calculated by both codes.

As it has been shown in table 1, both codes calculate nearly comparable initial values. RELAP5 calculates lower upper head pressure of 156.9 bar (-2 bar), higher cold leg temperature of 284.1°C (+5 °C) and slightly higher SG levels but same pressurizer level.

Table 1 Comparison between Cathare 2 and Relap5 steady state

Parameter	*CATHARE 2*	*RELAP5*

Mass flow/loop (kg/s)	*4961*	*5032*
Core ΔT	*37*	*37*
Reactor coolant pump delivery head ΔP (MPa)	*0.61*	*0.63*
Hot leg and cold leg pressure (MPa)	*15.77*	*15.57*
	16.08	*15.91*
Upper head pressure (MPa)	*15.98*	*15.72*
Vessel pressure loss (MPa)	*0.31*	*0.27*
Loop pressure loss (MPa)	*0.3*	*0.36*
Hot leg and cold leg temperature (°C)	*315.6/279.9*	*318.6/284.1*
Pressurizer Liquid level (m)	*7.30*	*7.30*
Pressurizer Liquid temperature (°C)	*346.0*	*345.1*
Pressurizer pressure (MPa)	*15.7*	*15.73*
Steam Gen. Sec .Side Downcommer level (m)	*11.44*	*12.75*
Steam Gen. Sec .Side Steam dome pressure (MPa)	*5.75*	*5.74*
Feedwater temperature (°C)	*220*	*219.41*
Feed water mass flow rate (kg/s)	*515*	*524*
Downcommer liquid temp (°C)	*260*	*261.13*
Recirculation flow (kg/s)	*1476*	*1481*
Pressure in Steam Header (MPa)	*5.66*	*5.65*
Temperature in Steam Header (°C)	*272*	*271.75*
Accu total volume (m^3)	*41.06*	*41.06*
Accu liquid volume (m^3)	*27.75*	*27.75*
Initial pressure in emergency cooling system (MPa)	*4.187*	*4.187*

Transient**:** The considered scenario assumed a break with a diameter of ~ 15.24 centimeters (6 inches) in the cold leg 1 of the loop without the pressurizer. This break size is classified as intermediate break LOCA. The time zero corresponds to the time when the break occurred. The sequence of events is presented in the Table 2.

As it can be seen (Figure 3(a)), the pressure in the case of RELAP is decreasing a bit faster than in CATHARE, which is caused by a bigger mass flow through the break. The same trend can be observed until 150 seconds.

Table 2 Time sequence of events

RELAP5	***CATHARE***	***Sequence***
0.0s	*0.0s*	*Leak initiation*
2.9s	*5.4s*	*Primary pressure < 129.5 bar: Reactor trip*
6.2s	*7.1s*	*Primary pressure < 117.5bar Safety injection actuation*
16s	*15s*	*Pressurizer empty*
150s	*30s*	*Core upper part voided*
21s	*37.4s*	*Start of safety injection pumps*
61s	*65.4s*	*Start of auxiliary feedwater injection*
300s	*100s*	*Downcomer voided, vessel head start to drain*
154s	*200s*	*Core upper part uncovered*
200s	*250s*	*Complete steam discharge at the break*
247s	*370s*	*Primary pressure <42bar, start of accumulator injection*
600s	*600s*	*End of calculation*

The difference in the behavior of the calculated systems can be seen in Figure 3(b,c), which shows the steam mass flow rate through the break. At the beginning of the calculation in case of CATHARE -immediately there is steam at the break and its void fraction is around 0.2, and 20 seconds later its value is more than 0.85. Therefore, for 120 seconds the mass flow in the break is higher in case of RELAP5. It can be seen clearly in Figure 3(d), which shows the integrated mass flow rate.

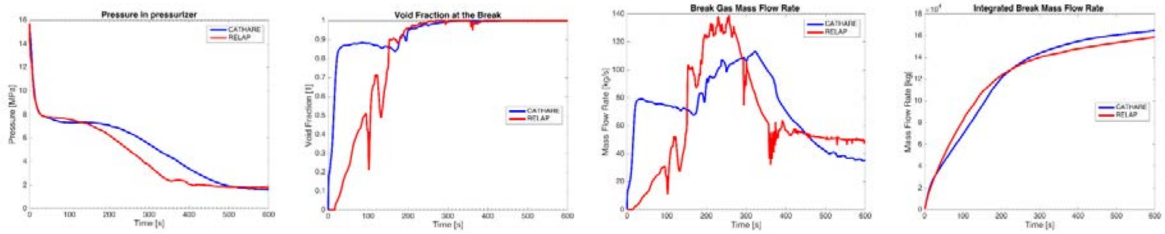

Figure 3 (a)System pressure during transient (b) Steam discharge through the break. (c)Void fraction at the break (d) Integrated leak discharge through the break

Figure 4(a) shows the changes in water level during the loss of coolant accident. The water levels in the reactor core are almost the same, with one difference around 100 seconds. In case of CATHARE there is about 1 meter less in water height.

In the upper part of the core distribution of the temperature of the cladding in time is very similar until 200 seconds elapse (Figure 4(b)). After this time, the temperature in RELAP decreased faster and there was a high peak, which can be explained by low water level (steaming) in the core which was not observed in CATHARE though having a lower minimal level. The decay power curves have been adequately simulated for both codes (Figure 4(c)).

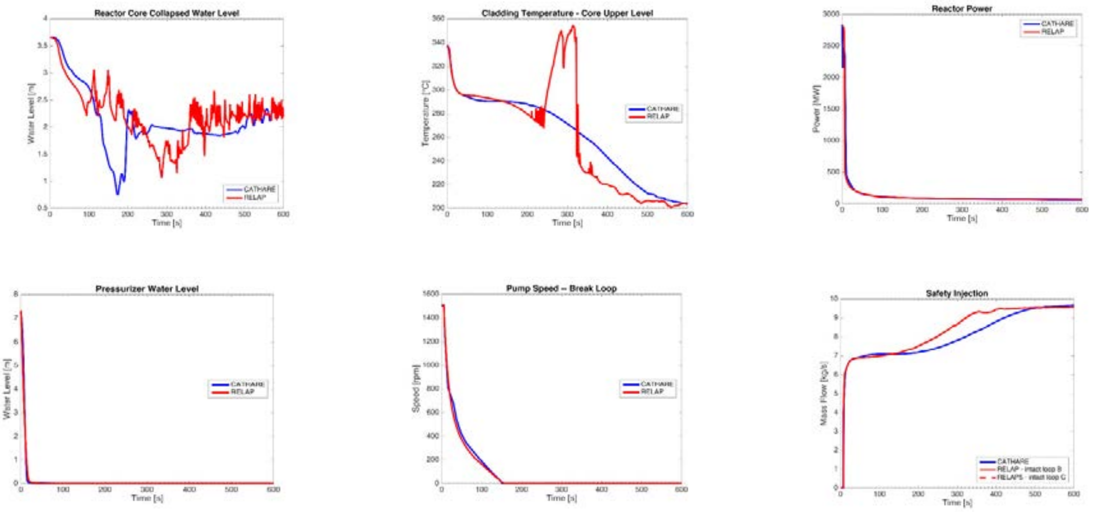

Figure 4 Water level in the reactor core. (b)Cladding temperature in the core upper part (c) Total reactor power [MW] (d)Safety injection mass flow (e)Pressurizer water level (f)Pump speed break loop

It can be also observed that the safety injection which adds more water according to the decreasing pressure although following the same pressure vs. flow dependence yields different flows (Figure 4(f)).

The amount of water in both pressurizers is the same and follows at an almost identical curve (Figure 4(d)). Additionally, both pumps have been shut down at the same moment and follow the same coastdown curve (Figure 4(e)).

5.2. Results summary:

It is not an easy task to achieve the same results because of the following:

Suggested modifications in the new RELAP5 input is for the 3rd step in the benchmark procedure the following should be installed in the new RELAP5 input:

1. Henry Fauske critical flow is generally suggested for the small break with a short nozzle. This model is designed to be used in the high pressure range (ROSA Test AP-CL-03) and albeit the discharge factor has to be adjusted to compensate for the difference at starting time void fraction results.

In the steady state calculation, the relevant system parameters such as primary pressure (-2 bar) and cold leg coolant temperatures (+4 °C) may be corrected to get more agreement with CATHARE results.

As for the CATHARE input the following issues should be addressed:

1. As shown in Figure 3(a) the loop seal clearing effect is well simulated (120-180 sec) in RELAP but not cleary indicated in CATHARE calculation. This IB-LOCA phenomenon can be evaluated after performing new calculations.[10]
2. Cathare neglects heating the fuel element when reactor water level drops. The averaging of temperatures for the rod bundle should be revisited.

6. Conclusions

The total break flow and pressure drop in the reactor during the IB-LOCA are modelled consistently. The course over time, however, differs mainly due to a different approach to modelling critical flow. RELAP5 employs a specific model (Henry-Fauske), whreas CATHARE uses only solving of six equations over time with no particular model for critical conditions. In both codes, the cladding temperature is far below the melting temperature defined by the NRC under CFR10.50 Appendix K. [9]

Loop seal clearing effect is not visible in CATHARE which is under investigation; lack of temperature increase over the uncovered rod length in CATHARE will be further investigated as well.

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