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International Agreement Report

EPR Medium Break LOCA Benchmarking Exercise Using RELAP5 and CATHARE

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ABSTRACT

Thermal-hydraulic analyses are a key part in support of regulatory work for new and existing nuclear power plant design and operation. This paper describes the approach to model the Loss of Coolant Accident (LOCA) in a Light Water Reactor as part of the “Safety Analysis Report in Warsaw University of Technology” (SARWUT) project and the framework of the “Familiarization with the calculation codes application” program.

The RELAP5 model of the European Pressurized Reactor (EPR) has been developed on the basis of an available CATHARE-2 input. Both thermal-hydraulic codes, RELAP5 and CATHARE-2, are used for the safety analysis of the NPP. The purpose of this report is to present the intermediate (6-inch) cold leg break calculations performed within the benchmark exercise using both RELAP5 and CATHARE-2 codes.

The results received are satisfactory, however as presented, the calculations performed with the use of both computer codes, at the early-stage give low cladding temperature but differ in the transient characteristics. The discrepancies in the values of chosen safety related parameters are analyzed in detail for understanding and future work.

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EXECUTIVE SUMMARY

There is a number of thermal-hydraulic computer codes used for safety analysis in nuclear installations. Such codes may vary within the scope of applicability, empirical correlations (e.g. for heat exchange under certain conditions or critical flow models) programmed in those codes and modeling approach. Warsaw University of Technology has been provided with several thermal-hydraulic codes. This benchmarking exercise has been carried out in order to better understand the performance of two of them: RELAP5 and CATHARE.

The Intermediate Break LOCA in the European Pressurized Reactor has been selected for modeling as a benchmarking exercise. The first step was to reach an acceptable steady-state. The second step was to perform calculations of the selected transient in order to find and compare predictions of both codes. A number of varying parameters have been plotted versus time. Finally, the differences between the results obtained by RELAP5 and CATHARE have been discussed.

In chapter 2 a brief description of the EPR design and main parameters is provided. It comprises general data with regard to nominal and design parameters, thermal power and electric output and core design data. The thermal-hydraulics data provided consists of nominal temperatures in cold and hot legs, coolant mass flow and heat fluxes. There is also a description of containment parameters, such as total volume, design pressure and temperature.

Chapters 4 and 5 cover a brief description of both CATHARE and RELAP5 codes. The main differences in applied correlations and modeling philosophy have been outlined. Nodalization schemes, together with detailed description of the core region modeling and safety systems have been also presented.

The analyzed scenario description has been provided in chapter 6. The steady-state results received in both RELAP5 and CATHARE have been compared to nominal values stated in the Pre-Construction Safety Report of the EPR [3]. The comparison between selected calculated parameters in both codes and those provided in the PCSR is satisfactory. Finally, transient calculations have been performed in accordance with the provided scenario.

A discussion of the obtained results, run statistics and conclusions are provided at the end of the report.

ACKNOWLEDGMENTS

The authors acknowledge the support from the Polish National Atomic Energy Agency (PAA) within the SARWUT project performed within the National Centre for Research and Development (NCBiR) strategic project “Technologies for the development of safe nuclear energy”, Research Task No. 9 entitled: “Development and implementation of safety analysis methods in nuclear reactors during disturbances in heat removal and severe accident conditions.”

ABBREVIATIONS

ASCII	American Standard Code for Information Interchange
CAMP	Code Application and Maintenance Program
CATHARE	The Code for Analysis of Thermalhydraulics during an Accident of Reactor and safety Evaluation
CEA	Commissariat à l'energie atomique et aux énergies alternatives (Atomic Energy and Alternative Energies Commission)
EDF	Électricité de France (Electricity of France)
EPR	European Pressurized Reactor
IET	Integral Effects Test
LHSI	Low Head Safety Injection
LOBI	Loop Blowdown Investigation
LOCA	Loss of Coolant Accident
LOFT	Loss of Fluid Test
LSTF	Large Scale Test Faciltiy
MHSI	Medium Head Safety Injection
MOX	Mixed Oxide Fuel
NCBiR	Narodowe Centrum Badań i Rozwoju (The National Centre for Research and Development)
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
PAA	Państwowa Agencja Atomistyki (Polish Atomic Agency)
PACTEL	Parallel Channel Test Loop
PCSR	Pre-Construction Safety Report
PKL	Primärkreislauf (German Large Scale Test Facility)
PMK	Scaled-down model of the Paks Nuclear Power Plant
PWR	Pressurized Water Reactor
RCS	Reactor Cooling System
RELAP	Reactor Excursion and Leak Analysis Program
RPV	Reactor Pressure Vessel
SET	Separate Effects Test
SNAP	Symbolic Nuclear Analysis Package
SPES	Simulatore Pressurizzato per Esperienze di Sicurezza (PWR Test Facility)
TRACE	TRAC/RELAP Advanced Computational Engine
US NRC	United States Nuclear Regulatory Commission

1. INTRODUCTION

There is a strong need to understand the characteristics of the Loss Of Coolant Accidents (LOCA) as such as the automatic countermeasures of the protection and safeguard systems designed to provide the withstand of the NPP for any break size and break location in the primary circuit. The NPP's supplier is obliged 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).

Warsaw University of Technology is in the process of becoming in the future a TSO has started developing knowledge related to accident simulation working in cooperation with the Polish National Atomic Energy Agency. This work is a representation of some of the efforts from learning and using thermal-hydraulic codes developed by the NRC and French supplied codes.

2. THE EPR GENERAL DESIGN

European Pressurized Reactor (EPR in Europe) or Evolutionary Pressurized Reactor (EPR in US) is a Generation III large PWR design by Framatome – currently owned by AREVA. It was developed on the basis of French and German experiences obtained during development of the N4 and Konvoi PWR reactors.

The reactor is characterized by a robust design based on the proven defense in-depth concepts. It has a high level of redundancy – with four safety divisions and independent emergency core cooling lines which provide proper protection against single failure and robust cooling capability. Most important systems of the four safeguard divisions contain a passive accumulator and two active systems with low and medium pressure head injections (LHSI & MHSI) per one loop. Plant design is characterized by a diversity of safety systems and emergency electric power and water supplies to strongly reduce the probability of a common cause failure. In the design there is a high level of complementarity in order to provide proper mix of both active and passive systems.

Among many safety related means, a crucial example is a large double walled containment with the outer shield building made of reinforced concrete and inner containment with steel liner and pre-stressed concrete. It forms the final barrier between the public, the environment and potential radioisotopes released during reactor or spent fuel pool accidents and severe accidents caused by both extreme internal and external hazards.

In order to withstand severe accidents, the plant utilizes an ex-vessel retention concept with a dedicated core-catcher to contain corium outside the vessel. The core catcher and the reactor cavity are initially dry to avoid a highly energetic interaction of corium with water and avoid a steam explosion. In order to prevent late containment failure due to the pressurization or the basemat melt-through, the core catcher system utilizes both passive and active water cooling systems with heat exchangers which are able to provide ultimate cooling and stabilization of the corium.

The containment is equipped with a spraying system with recirculation dedicated to reduce containment pressure in the case of containment pressurization. Moreover, the plant is equipped with a reliable RCS depressurization system which severely decreases the probability of high pressure melt ejection and potential early containment failure due to the Direct Containment Heating phenomena. Additionally, the containment is equipped with a set of passive autocatalytic re-combiners (PARs) forming part of a combustible gas control system designed to remove hydrogen gas and suppress potential deflagration or detonation [2], [3], [4], [5], [6].

Table 1 General Design Parameters for the EPR.

General data	Core thermal power	4500	MWth
	Electric power (net)	1600-1650	MWe
	Number of cooling loops	4	-
	Nominal primary system pressure	155	bar
	RPV design pressure	176	bar
	Nominal secondary system pressure	78	bar
	Secondary side design pressure	100	bar
	Fuel array	17x17	-
	Number of fuel assemblies	241	-
	Fuel rods per assembly	265	-
	Number of control rod clusters	89	-
	Basic fuel	UO2 (up to 5% %wt) or MOX (core up to 30%)	-
	Plant lifetime	60	yr
	Average discharge burn-up	55-65	GWd/MTU
Basic thermal-hydraulics	Thermal design flow rate per one loop	27185	m^3/h
	Core bypass flow rate	5.5	%
	Nominal core inlet temperature	295.6	°C
	Nominal core outlet temperature	331.6	°C
	Core heat transfer surface	8005	m^2
	Average core heat flux	0.547	MW/m^2
	Maximum core heat flux during normal operation	1.573	MW/m^2
	Average linear heat flux	16.34	kW/m
	Maximum linear heat flux during normal operation	47.0	kW/m
	Power density	94.6	MW/m^3
Containment	Overall form	Spherical cap and cylindrical	-
	Containment volume	80 000	m^3
	Design pressure	5.5	Bar
	Design temperature	170	°C
	Design leak rate	0.3	% volume per day

3. THE BENCHMARK EXERCISE PROCEDURE

A benchmark calculation between two thermal-hydraulic codes requires maximum understanding in the phenomenology, methodology, code's specifications and experience in the creation of a plant model. The analytical procedure should involve the following steps:

- Step 1: Preparation of equivalent inputs
  Remove the inconsistencies in the nodalization scheme, initial conditions and model's options as far as possible
- Step 2: 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 options, local pressure losses, main coolant pump characteristic, decay power history)
- Step 3: Definition of a new basic input deck
  Checking full consistencies of the new inputs what has not been done in the Step 1. Performing new calculations
- Step 4: Comparison of results and qualifying the capability of the code based on the simulation of accident
  Comparison and assessment of results accordingly to the code's governing equations, specific models and features.

4. THE RELAP5 THERMAL-HYDRAULIC CODE MODEL

The RELAP5/MOD3.3 code has been developed for best-estimate transient simulations of light water reactor 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 that 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 [7], [8],[9].

4.1 <u>RELAP5 Nodalization</u>

Nodalization of the reactor pressure vessel is presented in Figure 1. The reactor core is modelled by two pipe components no. 535 &536, with 9 axial meshes, where the meshes 1 through 9 represent the heated length of the core. The division was based on core enrichment differences and differentiating between the inner and outer zones. Pipe 536 has two heat structures modeling the purple and blue region and pipe 535 has three heat structures modeling the green, yellow and red regions in Figure 2. These regions represent zones of different heat generation rates in the reactor core.

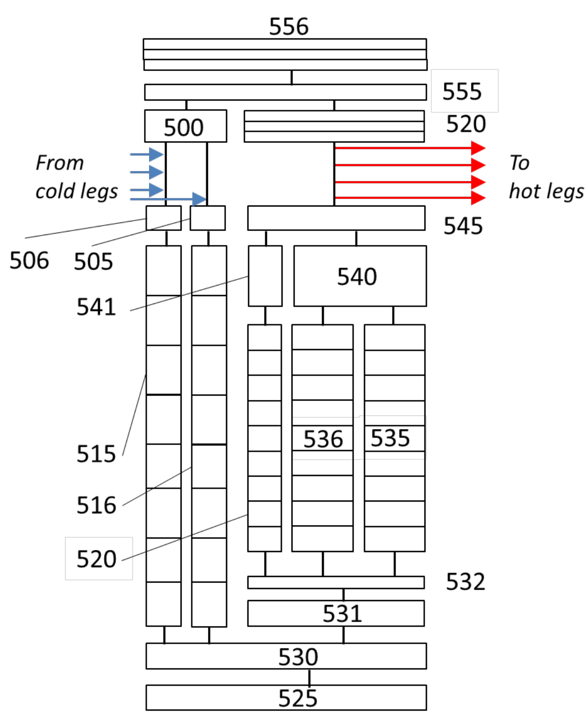

Figure 1 RPV Nodalization Scheme.

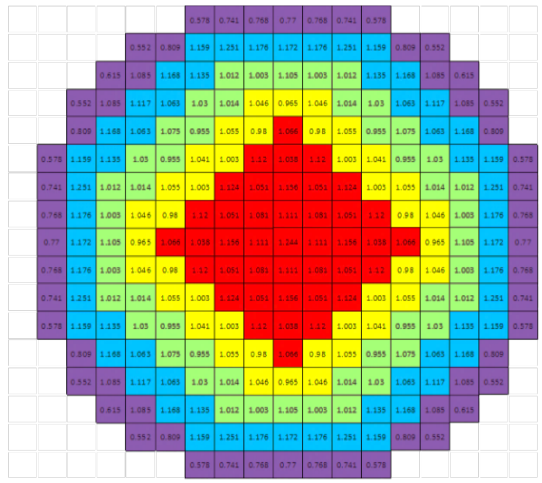

Figure 2 Core Heating Regions With Zone Coloring.

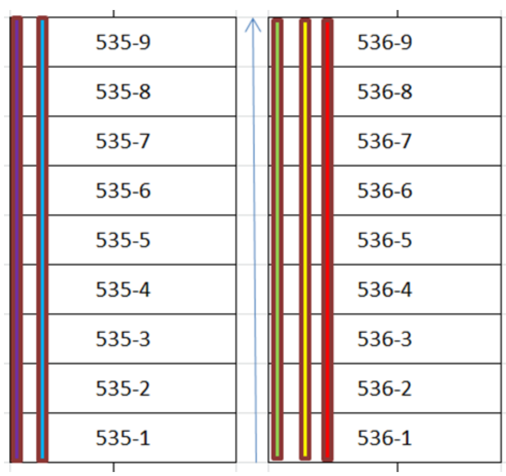

Figure 3 Core Nodalization Scheme With Zone Coloring.

The downcomer is modeled using two annulus components: 515, 516 , branches 505 and 506 and multiple junctions 207,208. Core bypass is modeled by a pipe component 520 and branch 541. Lower plenum of the vessel is represented in the model by the two branches 525, 530. Guide tubes are represented by a one branch 500. Upper plenum of the vessel is modeled by two pipes 550, 556 and branches 545, 555. Each component used for core modelling is connected thermally via a heat structure. The lower plenum is modeled by branches 525, 530.

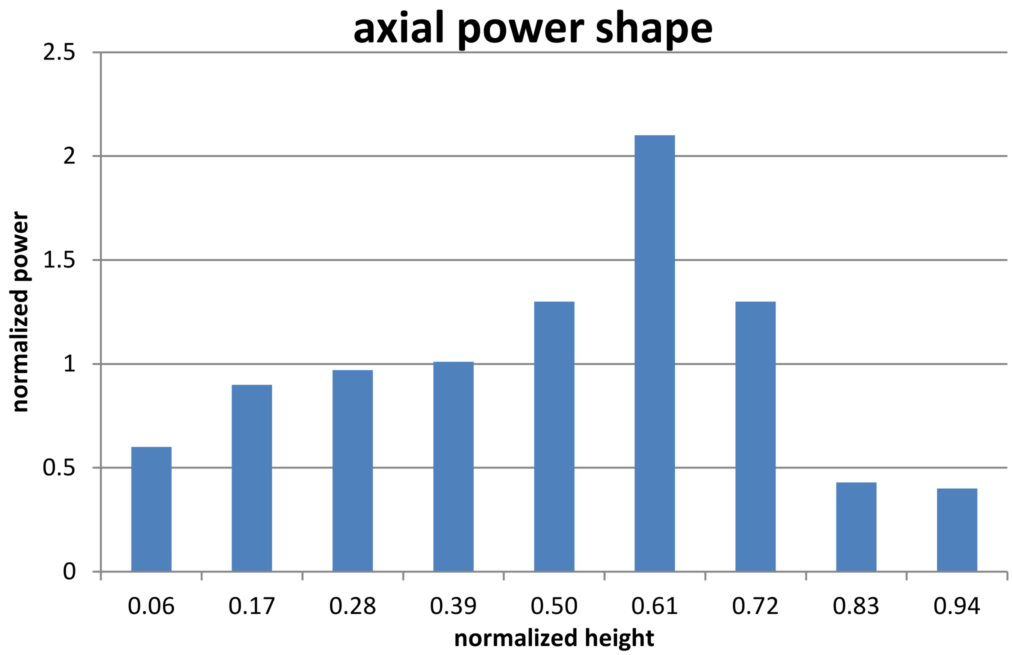

Figure 4   Axial Power Shape As a Function of the Normalized Core Height.

The power axial distribution shown in Figure 4 is a version used for calculations in RELAP5. It is a normalized linear power average versus normalized core height.

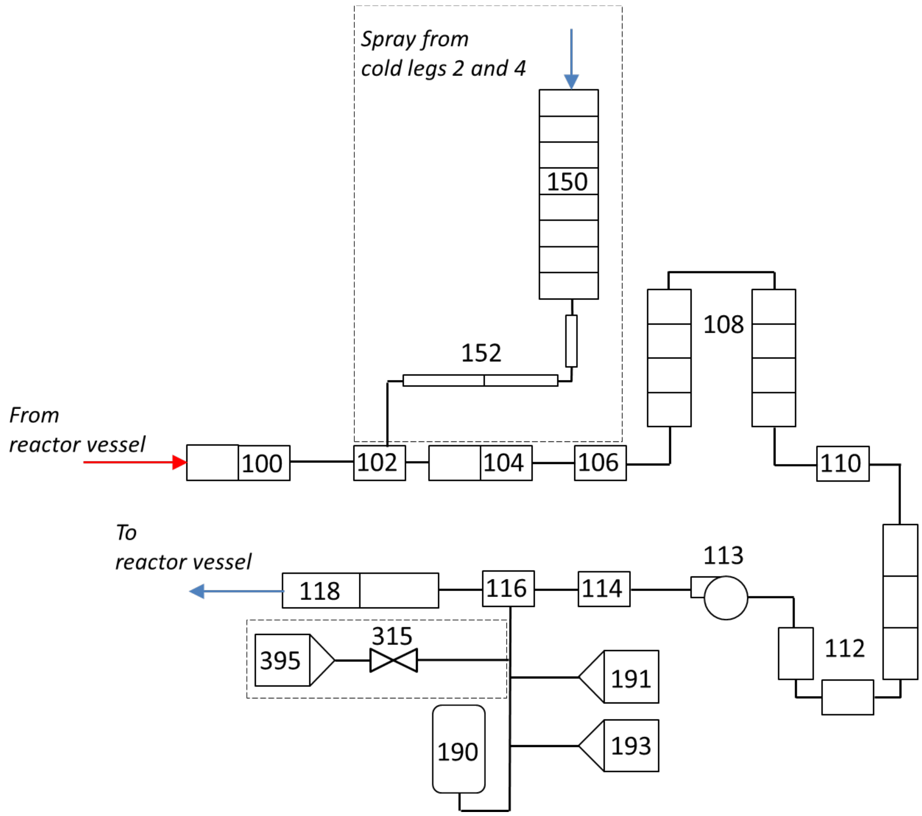

Figure 5 Primary Side Single Loop Nodalization Scheme.

Nodalization of the primary side is presented in Figure 5. The model consists of 4 separate loops. Loop no. 1 is presented in Figure 3. Loops 2 and 3 and 4 are modelled identically with the sole difference that the pressurizer is connected to loop no. 1. Finally, loop no. 3 holds the trip valve (element no. 600 in Figure 1.), which has an area of 0.184m^2, in order to simulate the postulated accident.

The horizontal part of the hot leg connected is modeled by two elements - 100, 102. The pressurizer is modeled by pipe 150 consisting of 8 nodes.

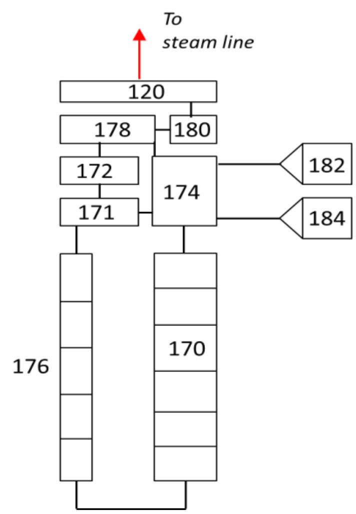

Figure 6 Steam Generator Secondary Side Nodalization Scheme.

The secondary side nodalization scheme is presented in Figure 6. The inlet plenum for the steam generator is modeled with a single volume 106. U-tube bundles are modeled with pipe 108 also having 8 nodes, the outlet plenum – single volume 110.

The water supply for the steam generator secondary side is flowing through the time dependent volume 182 and junction 181. Branch 174 mixes water returning from the steam separator 171 and is also connected to the downcomer 176 modeled with an annulus. The downcomer has 5 nodes. The mixture is then transported to the riser pipe 170. The riser has a heat structure 4 out of 6 nodes connecting the primary and secondary sides.

The most outer part of the steam separator is a single volume 172 and branches 178, 180 and single volume 120 is used to model the dryer and steam dome.

The rest of the secondary side is a steam line modeled with pipe 122. Valve 185 simulates the cutoff valve to the steam collector. The steam collector pipe 902 as well as the time dependent volumes for the turbine 904 and the condenser 906 close out the steam circuit. Additionally the model also has an isolation valve for the turbine 901 and steam bypass valve 903 from which steam goes directly to the condenser.

The break is modeled with two valves which are located on the second loop, the first one just after the reactor coolant pump and the second one at the cold leg just before the reactor vessel. The containment is modeled by a single volume component.

4.2 <u>Safety Systems</u>

The primary and secondary side are equipped in 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 broken loop and pressurizer loop)
- Medium head safety injection (MHSI) – 2 injection systems (on broken loop and pressurizer loop)
- Accumulators injection – four acculumators each having 31.74 m^3 of liquid opening at pressure 45 bar.

On the secondary side:

- Emergency steam generator feedwater system.

The injection schemes are shown in Figure 7 for the medium and low head injection systems. The mass flow for the steam generator injection is constant and is equal to 25 kg/s.

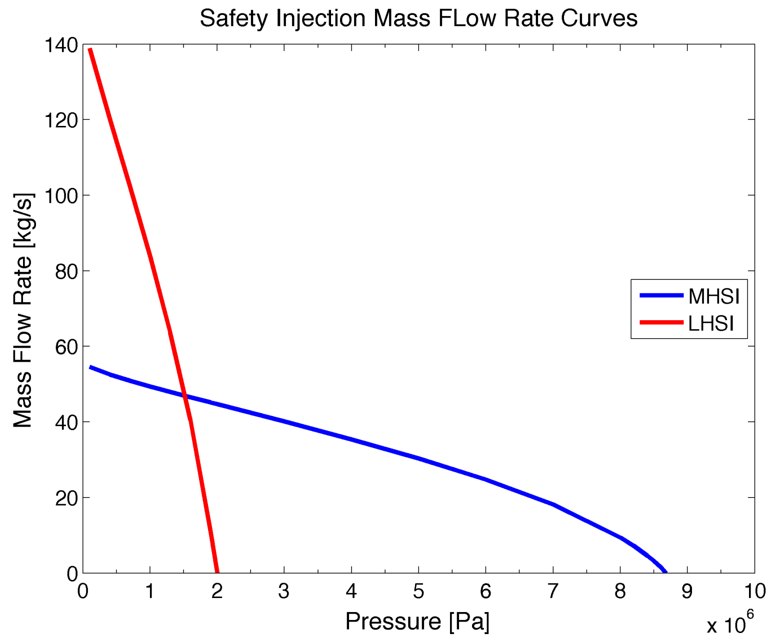

Figure 7 MHSI and LHSI Mass Flow Rate Curves.

5. CATHARE ANALYTICAL METHOD

5.1 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 mock-ups as well as entire Pressurized Water Reactors.

The approach adopted for the physical validation of CATHARE can be broken down into two tasks, which are:

1. Qualification in analytical tests or separate effects tests (SETs)

2. Verification in global experiments or integral effects tests (IETs).

The matrix of SETs, which is used for qualification, brings together about 300 tests chosen from experiments regarding critical flow, determination of flow diagrams, depressurization of adiabatic or hot test sections in various geometries, reflooding, filling the downcomer, phase separation at Tee junctions, counter-current for complex geometries, the response of steam generators and reactor coolant pumps and the thermo mechanics of the fuel rod.

The matrix of IETs, which is used for verification, is made up of 27 tests carried out on BETHSY, LOBI, LSTF, PACTEL, PMK, LOFT, PKL and SPES mock-ups [10-12].

5.2 CATHARE Nodalization

The EPR model in CATHARE has been developed by experts from AREVA and has been provided to the SARWUT project as a reference model. The model consists of four loops which are modeled separately. Safety systems, the medium head injection system and the low head injection system are modeled with the use of gadget components, however during the transient they operate only in the broken loop and pressurizer. Four accumulators are also included. All safety systems are connected to the cold legs at the distance of around 5 m from the reactor pressure vessel (RPV). The break is modeled with the use of boundary conditions which are blind during steady state. The break is located 5 m from the RPV. The pressurizer is a single volume element which is attached to the hot leg by the surge line (one axial element) at the distance of 5.5 m from the RPV.

There are three boundary conditions associated with the pressurizer in order to appropriately model the safety valves functioning. The pressurizer can be approximated by one element, due to the fact that it gets empty at the early stages of transient calculations and therefore it has no impact during the latter stage of transient.

The RPV is modeled with the use of 9 components, five of which are volume components and 4 of them are axial components. The coolant flows into the inlet plenum and is divided into 2 streams. During steady state calculations it is assumed that less than 1 % of the mass flow enters the upper head of the RPV. The water from the downcomer, which is modeled as an axial component, enters the lower volume. The lower volume models the lower plenum of the RPV and the free volume of the lower core support structure. About 95% of the total flow of the downcomer flow enters the reactor core during the steady state calculations and 5 % bypasses the core through the two axial components.

The core is divided into 59 segments but only 55 of them model the active part of the core. The reactor power is set as a function of time. The characteristics of the mentioned 55 fuel segments are the same except for the axial peaking factor which differs at each elevation, simulating in this way the proper power distribution in the core.

The coolant accumulates in the outlet plenum modelled by one volume. The free volume of the guide tubes is also modelled by one volume located above. The connection with the steam dome is possible only via the guide tubes and a connection with the inlet plenum. The steam generator is modeled using 6 components. The downcomer and the riser part below the U-tube bend is divided into 2 parallel axial components. One simulates the co-current part of the U-tube heat exchanger and the other simulates the countercurrent part of the U-tube heat exchanger. This simulates the economizer in the steam generator. The mixture is accumulated in a very small volume and is distributed to the axial element which simulates a U-tube bending and a riser above bending.

The steam separator model is set at the junction which connects the riser with a volume simulating the steam dome and the free volume of the separator. The steam flows through an axial pipe and is accumulated in the volume with boundary conditions set to simulate the turbine. The nodalization scheme of the RPV and the primary loop are shown on Figures 6. and 7. respectively. The nodalization of the secondary side is presented in Figure 8.

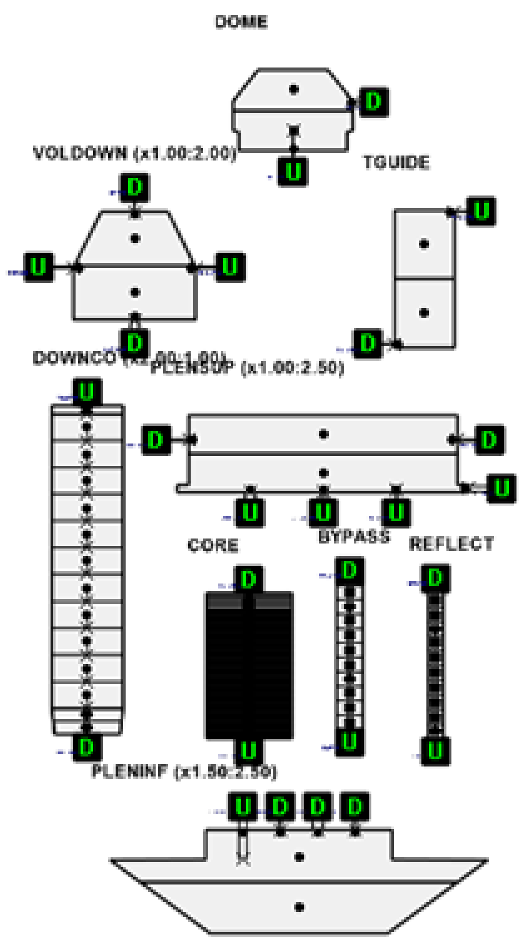

Figure 8 Reactor Pressure Vessel Nodalization Scheme.

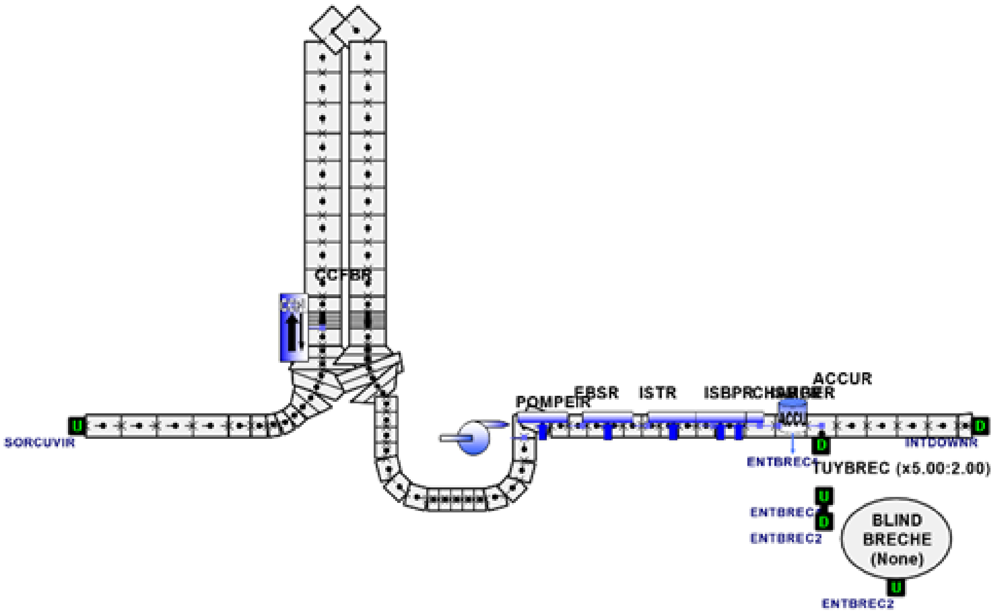

Figure 9 Primary Loop Nodalization Scheme.

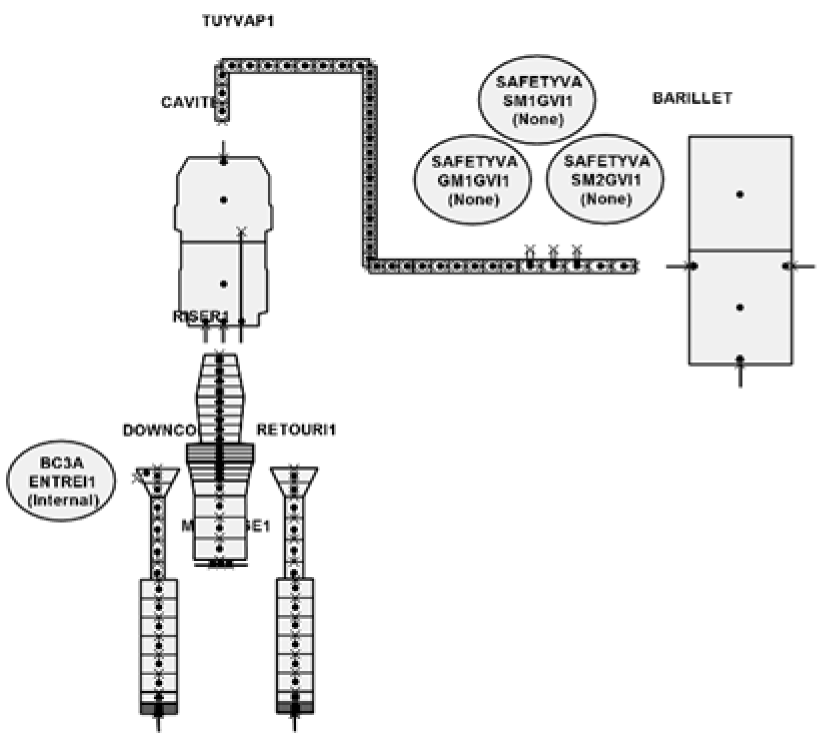

Figure 10 Nodalization Scheme of the Secondary Side.

5.3 Critical Flow Model in CATHARE

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.

The Critical model implemented in CATHARE has been validated against several experiments like Super Moby Dick, Bethsy, Marviken and Rebeca, in a wide range of pressure and different ratios between length and hydraulic diameter of a discharge pipe for subcooled and saturated fluid and mixture of water, steam and air [11], [12].

6. CALCULATIONS

6.1 Scenario Description

The scenario taken into consideration assumes a 6-inch in diameter break on the cold leg at the distance of about 5 m from the RPV.

Only two MHSI and two LHSI are available and work without delays. All available accumulators are available. Decay heat tables are used for core power calculation after reactor trip and constant power is assumed beforehand. Pump coastdown is delayed based on a function comparing primary side pressure as obtained from the CATHARE input deck. On the secondary side a partial cooldown procedure is modeled which simulates cooling down at a rate of 250K/h.

6.2 Steady-State Results

A summary of nominal conditions, taken from [2] is presented in Table 2. It includes also the results obtained in the steady-state simulations. Both RELAP5 and CATHARE predictions of steady-state working conditions are similar and compared to nominal conditions give good agreement. The pressurizer pressure and water level are slightly overestimated in CATHARE. On the other hand, the average temperature values in the primary side as well as the secondary pressure calculated by RELAP5 are a little higher than nominal. These results were used as initial conditions for LOCA simulations.

Table 2 Steady-State Results Calculated in RELAP5 and CATHARE.

			Nominal	RELAP5	CATHARE
Primary side	Pressurizer pressure	MPa	15.5	15.49	15.77
	Saturation temperature in pressurizer	^0C	345	344.77	346
	Pressurizer water level	%	56	55.5	61.8
	RPV inlet temperature	^0C	295.6	301.6	297
	RPV outlet temperature	^0C	329.8	334	330.7
	Average temperature	^0C	312.7	317.8	313.9
	Total coolant flow rate	kg/s	22 235	22 177	21 931
	Bypass flow rate	%	5.5	5.6	5.1
Secondary side	Feedwater temperature	^0C	230	230	230
	Steam generator narrow water level	m	49	47.54	48.7
	Steam pressure	MPa	7.71	7.77	7.72
	Total main steam flow rate	kg/s	2 552.4	2 531.22	2 555

6.3 Transient Results

Table 3. presents a sequence of events occurring during the simulation. There is a good comparison between RELAP5 and CATHARE predictions. The reactor is tripped after 16 s from the beginning of the simulated transient. Immediately after that, the turbine is tripped and the main steam bypass valves are opened. Approximately after 30 s, the pressure in the steam generator's steam dome reaches 9.6 MPa, the partial cooldown is initiated and lasts for about 450 s. Next, the pressurizer becomes empty and the medium head injection (MHSI) pump starts working. Injection from the accumulator tanks starts at 670 s in RELAP5 and 595 s in CATHARE. This is the most significant difference in the transient calculations. The main coolant pumps are tripped after 79 s. Calculations are finished at 1 000 s.

Table 3 Sequence of Events.

RELAP5	CATHARE	Event
0	0	Leak opening in cold leg
16.5	16.72	Reactor trip
18.7	18.82	Turbine trip
18.7	18.82	Opening of main steam bypass valves
27	31.41	SG 9.6 MPa, Safety Valve open, Partial CoolDown Start, 250 ^{0}C/h
74	45	Pressurizer is empty
78	79.6	Main coolant pumps trip
188	200	Beginning of MHSI injection
485	500	End of partial cooldown
668	595	Beginning of accumulator injection
1 000	1 000	End of calculation

Figures 11. to 27. represent the results of the benchmark calculation between the two codes. The CATHARE results are illustrated with a blue color and RELAP5 with a red color.

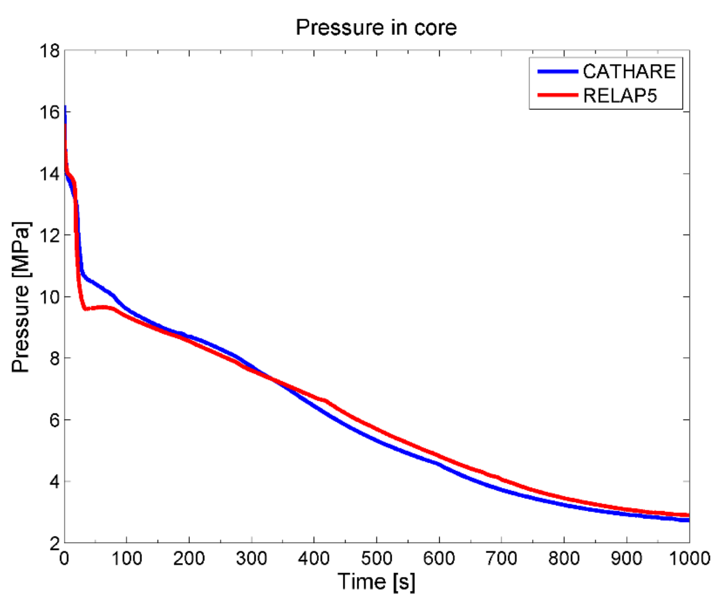

Figure 11 Pressure in Core.

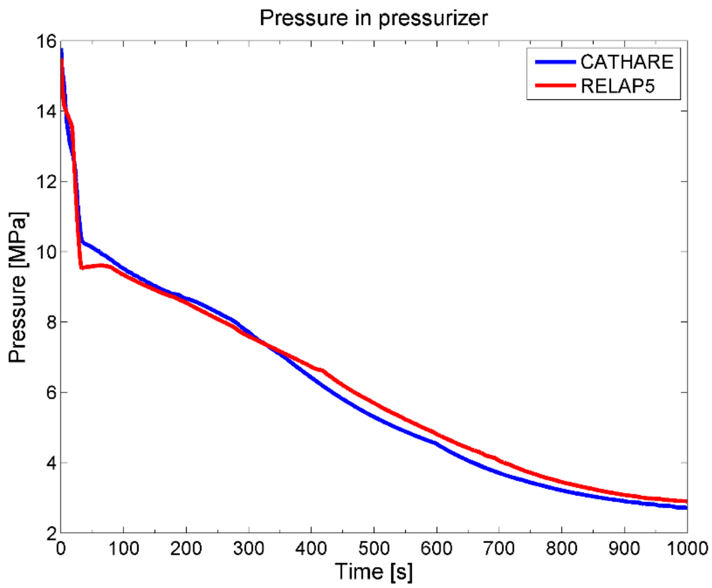

Figure 12 Pressurizer Pressure.

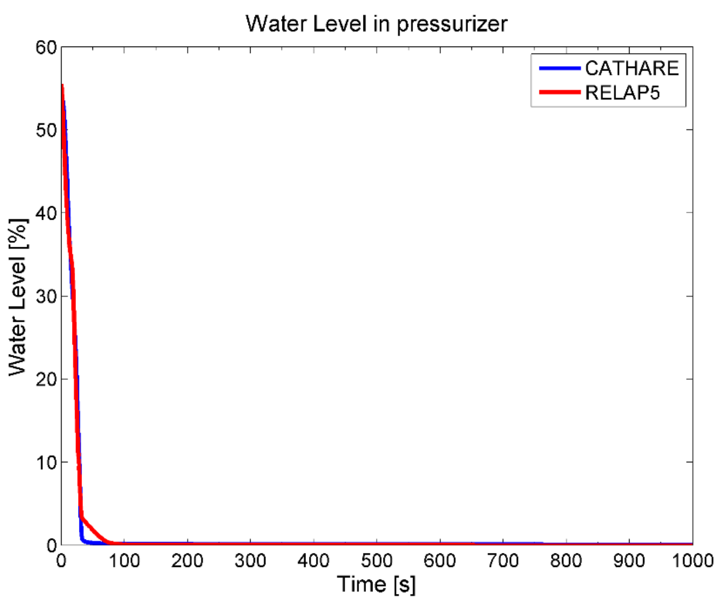

Figure 13 Water Level in the Pressurizer.

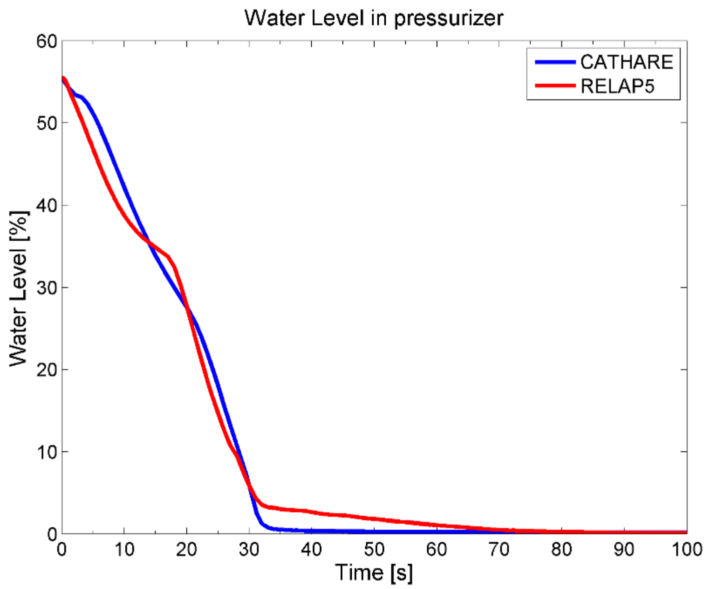

Figure 14   Water Level in the Pressurizer (First 100 s).

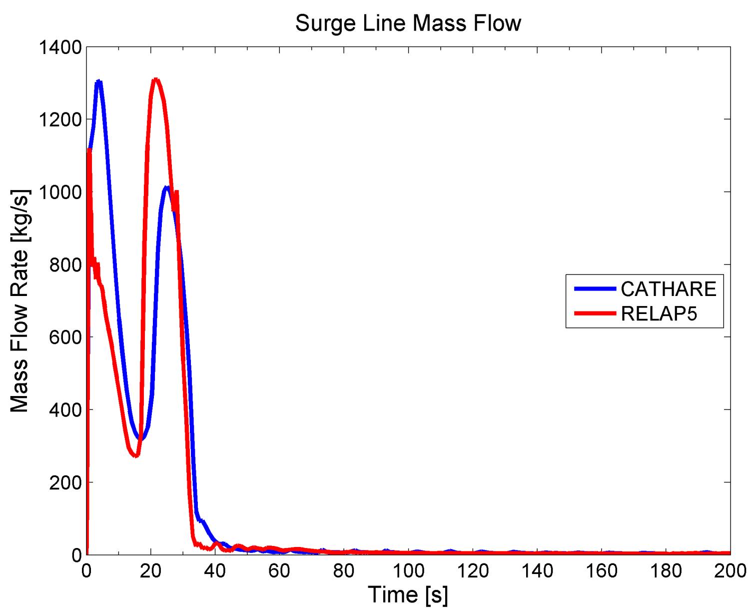

Figure 15   Surge Line Mass Flow.

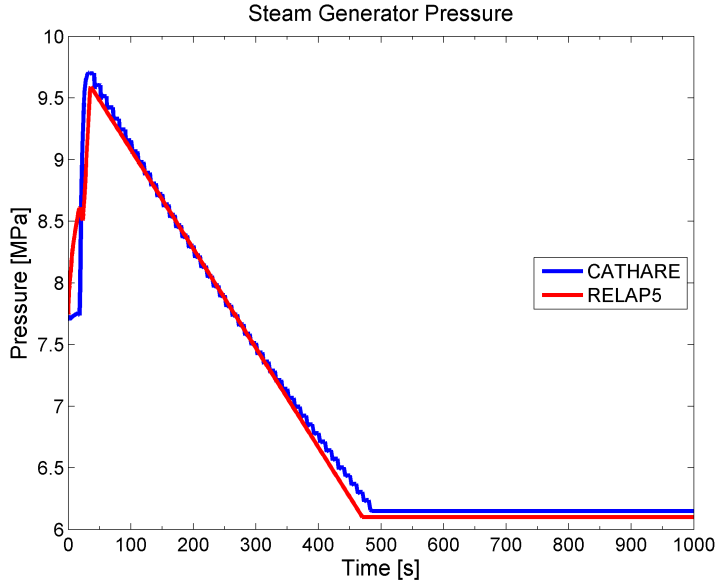

Figure 16 Steam Generator Pressure.

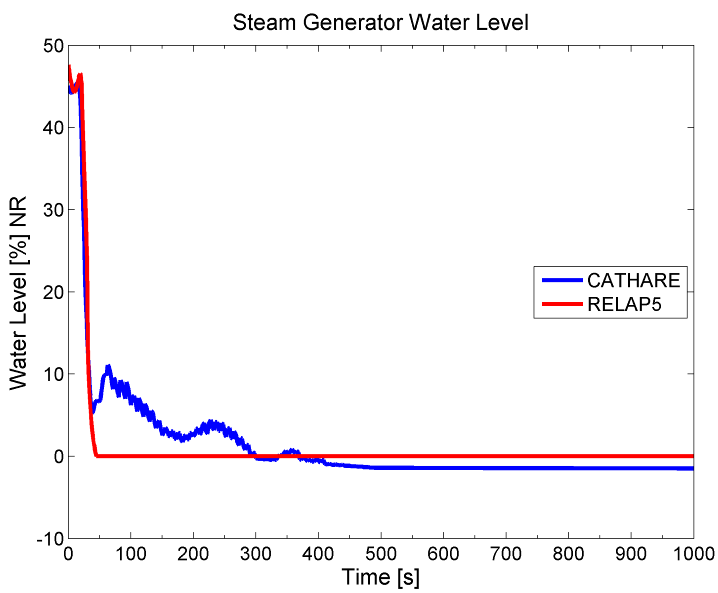

Figure 17 Steam Generator Water Level.

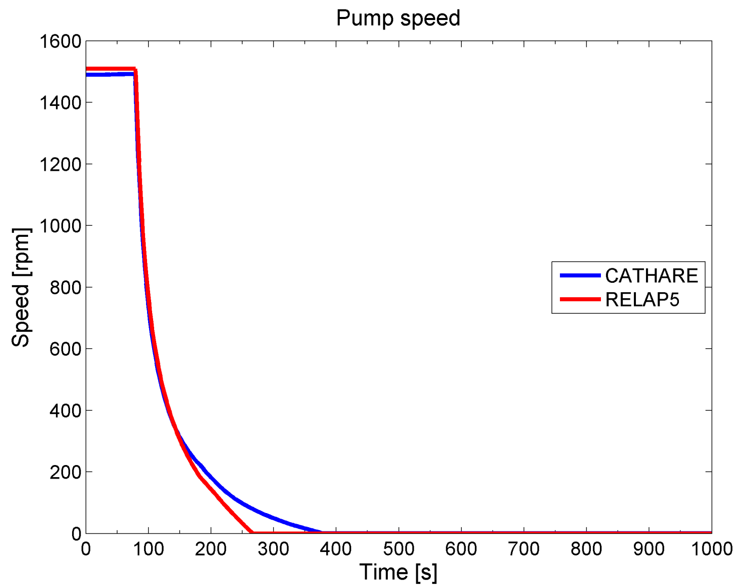

Figure 18 Pump Speed.

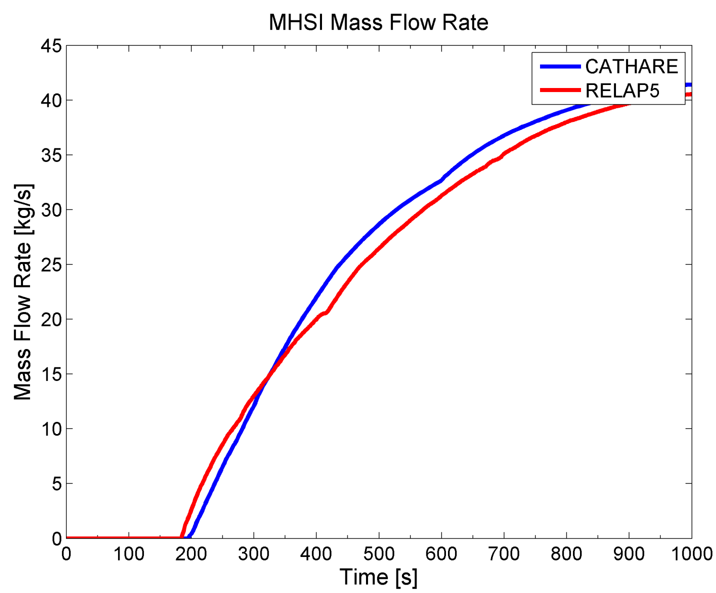

Figure 19 MHSI Mass Flow Rate.

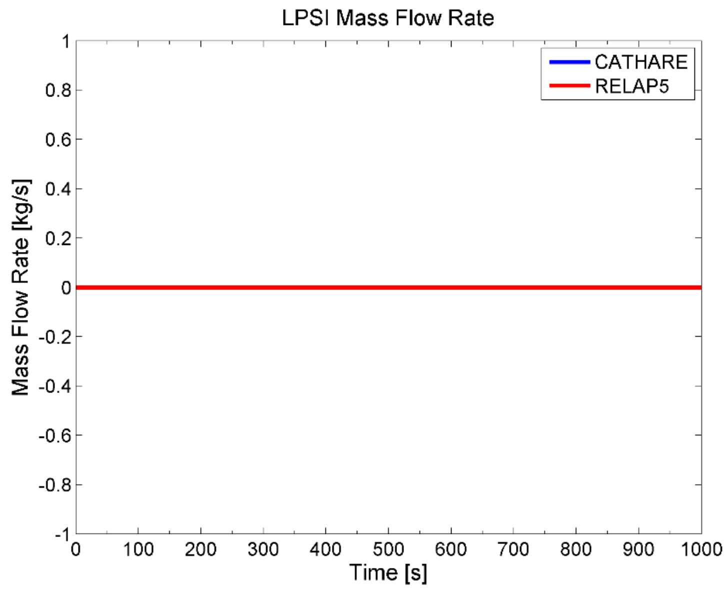

Figure 20 LPSI Mass Flow Rate.

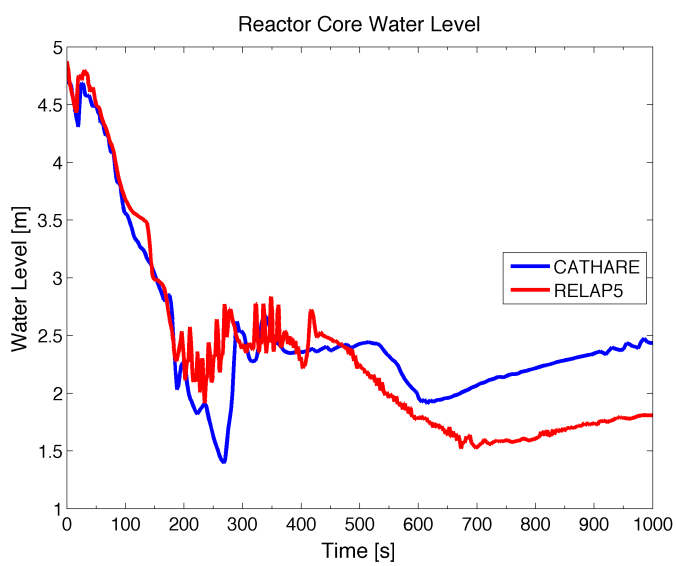

Figure 21 Reactor Core Water Level.

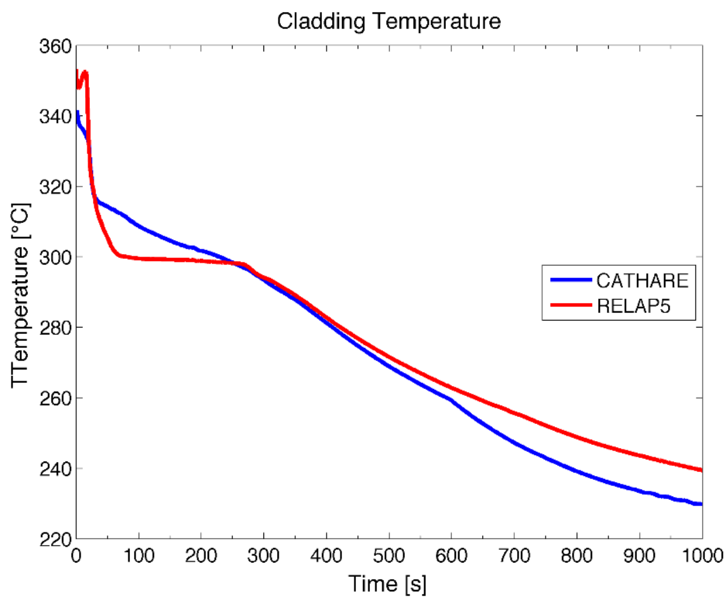

Figure 22 Cladding Temperature.

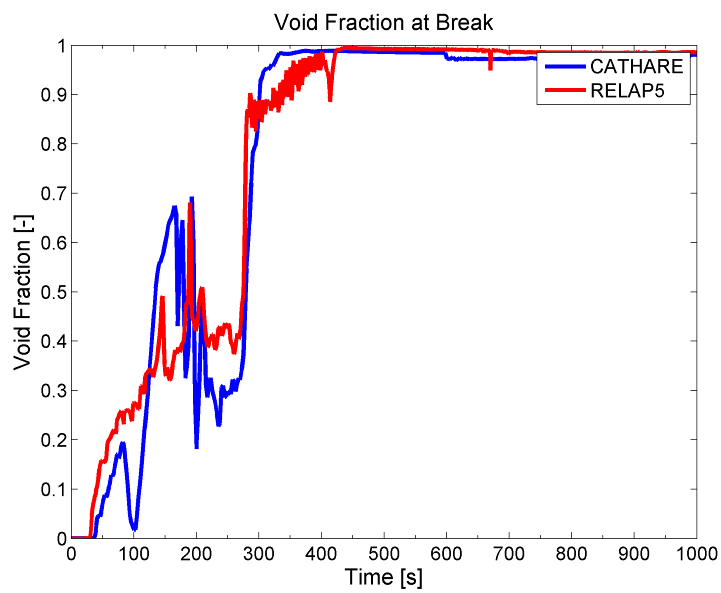

Figure 23 Void Fraction at the Break.

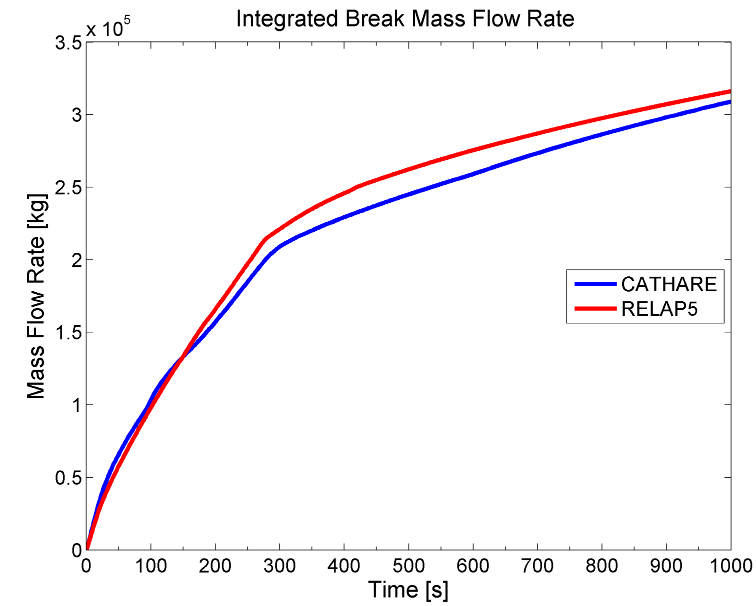

Figure 24 Integrated Break Mass Flow Rate.

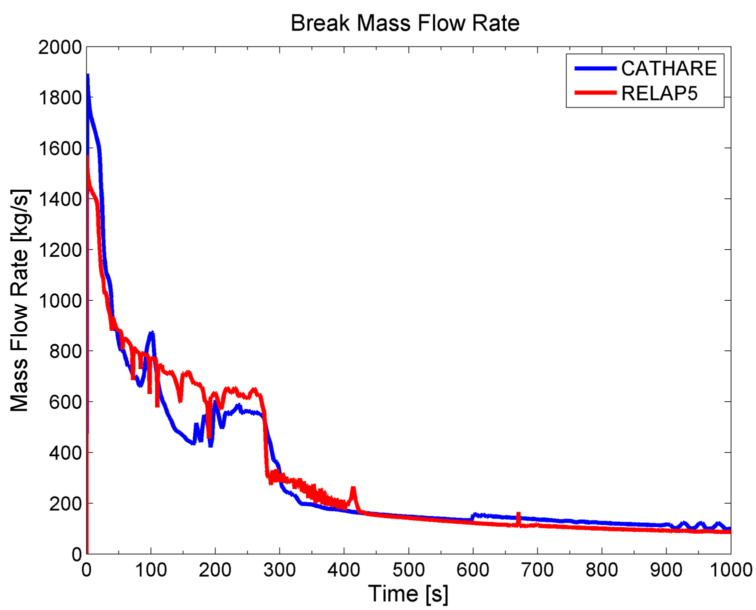

Figure 25 Break Mass Flow Rate.

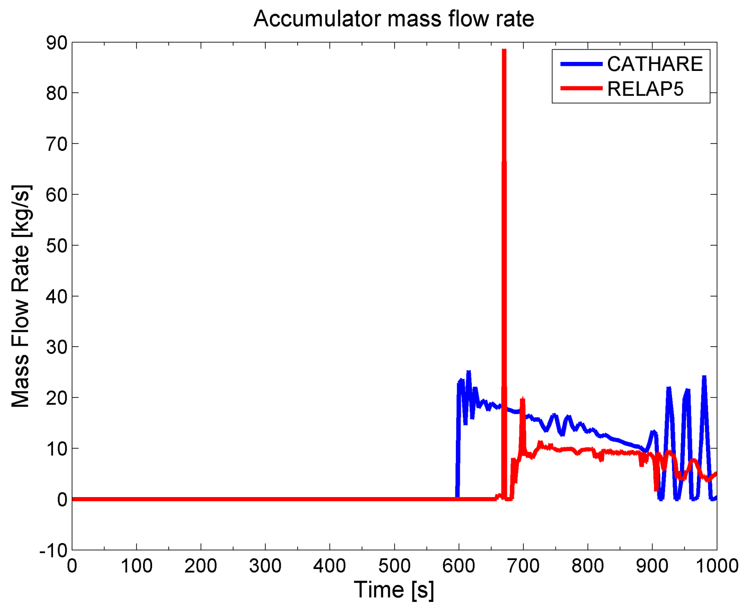

Figure 26 Accumulator Mass Flow Rate.

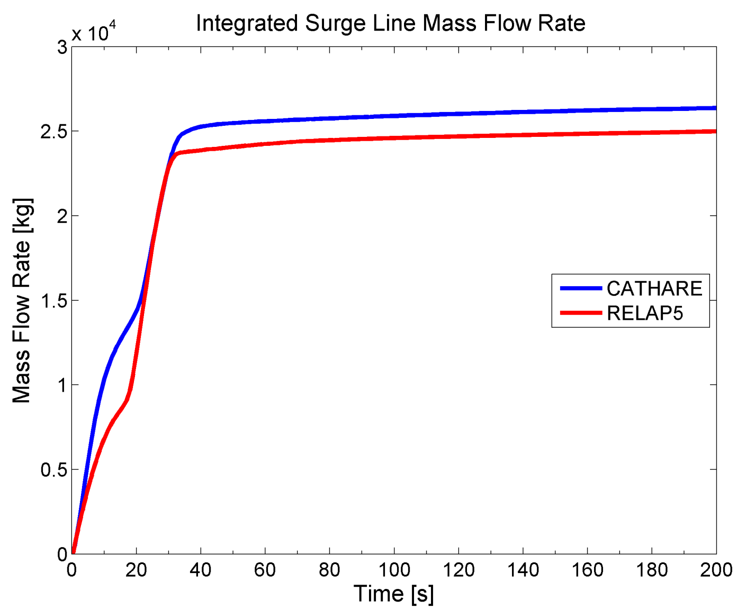

Figure 27 Integrated Surge Line Mass Flow Rate.

7. DISCUSSION

The results shown are generally in good comparison. Most differences can be assigned to a different model for critical two-phase flow in both codes as well as different pump degradation curves and multipliers. As for the critical two-phase flow, CATHARE uses a six equation model where in RELAP5 the standard Henry-Fauske model is used.

Figure 11. shows that pressure in the core has a sharper downward spike in RELAP5 about 50 seconds into the transient but both pressures align after the 100th second and RELAP5 predicts higher pressure after 400 seconds till the end of the calculation. Figure 23. showing void fraction shows that until the 200th second RELAP5 predicts a constant rise of void fraction at the break where in CATHARE between the 50th and 100th second steam content is dropping even though, at that time, no additional water source (e.g. safety injection) is active. Additionally, CATHARE predicts a slightly higher integrated break mass flow rate than RELAP5.

The pressurizer empties 30 seconds later in RELAP5 mainly due to the last 5% of inventory being cleared slowly. Figure 15. shows an interesting behavior of the mass flow rate through the surge line where though the flows are comparable, the two peaks occurring in the flow have a reversed order (high-to-low in CATHARE, low-to-high in RELAP5). Figure 16. shows the steam generator pressure at the secondary side which is controlled directly in CATHARE by a routine that sets steam generator pressure (after breaching 9.6 MPa) using a function representing a partial cooldown procedure of 250 ^{0}C/h.

An additional time-dependent-volume (per each loop) connected to the steam generator has been added in RELAP5 to mimic the same behavior as in CATHARE. The difference visible in Figure 17. concerning the steam generator water level can be attributed to the difference in the representation of geometry in both codes. The methodology for steam generator modelling is also different and therefore this result might not be comparable.

The pump speeds shown in Figure 18. are in good alignment up to the 200th second where the slope in RELAP5 is much steeper. The MHSI in Figure 19. starts at almost the same time but because of being driven by pressure the flow rate in RELAP5 is a bit smaller; LHSI does not start in this transient calculation. Figure 21. presenting the reactor core water level is very closely predicted up to the 200th second and though there is a difference of about 0.5 meter, the trend shows increasing temperature in RELAP in the last part of the transient.

Pump data was an issue due to an inability to access CATHARE's built in curves and two-phase multipliers. An issue was also observed with RELAP5 accumulator discharge flow where a short, sharp spike of mass flow can be seen in Figure 26. Figure 16. shows an implementation of the partial cooldown which is slightly different for RELAP5 and CATHARE due to a difference in logic controllers in CATHARE resulting from direct forcing of secondary pressure behavior.

8. RUN STATISTICS

The calculations were performed using Intel® Core™ i5 M 560 @ 2.67 GHz processor. The operating system was Windows 7 Professional.

Table 4. shows the run statistics for the codes RELAP5/MOD3.3 Patch 0.4 and CATHARE 2.5 calculations. Both times are comparable.

Table 4 Run Statistics

Code	Transient Time (s)	CPU Time (s)	CPU/Transient Time	Number of Time Steps
CATHARE	1 000.00	986.40	0.9864	Time steps:6006/ Iterations:22380
RELAP5/MOD3.3 Patch 4	1 000.01	1 062.74	1.063	109364

9. CONCLUSIONS

This benchmark activity is very valuable in terms of user experience in system modeling as both codes, though very similar, enforce different modelling of some critical components – e.g. the steam generator or safety system logic – e.g. partial cooldown. Achieving comparable results is an iterative process which demands in-depth understanding of both codes: RELAP5 and CATHARE.

Though there are differences in the results obtained from CATHARE and RELAP5, they can be attributed to factors related to lack of precise data such as pump data. Another factor can be related to differences in physical phenomena modelling such as critical two-phase flow in the codes. Finally, different representation of geometry of some components (e.g. the pressurizer in CATHARE has a trapezoid base compared to a rectangular RELAP5 shape), might also lead to discrepancies in the results of both codes.

10. REFERENCES

1. Pressurized water reactor (PWR) systems, Reactor Concepts Manual, NRC.
2. Areva and EdF, UK EPR Pre-construction Safety Analysis Report, 2013
3. http://www.epr-reactor.co.uk/
4. Areva, US EPR Final Safety Analysis Report, 2013
5. http://www.nrc.gov/reactors/new-reactors/design-cert/epr.html
6. Computational validation of the EPR combustible gas control system, H. Dimmelmeier, et. al, Nuclear Engineering and Design, 249 (2012), 118-124.
7. M. Fischer, The severe accident mitigation concept and the design measures for core melt retention of the European Pressurized Reactor (EPR), Nuclear Engineering and Design, 230 (2004), 169-180.
8. EPR safety in post-Fukushima context, Areva, March 2012, Presentation.
9. H. Ransom, et.al., RELAP5/MOD3 Code manual volume 1: Code structure, system models, and solution method.,1994.
10. RELAP5 MOD3.3 Code manual volume 4: models and correlations, October 2010, Information Systems Laboratories, Inc..
11. S. Shieh V. H. Ransom, RELAP5/MOD3 Code manual volume 6: Validation of numerical techniques in RELAP5/MOD3.0,1994.
12. Eymard, C. (2005). CATHARE2 V2. 5 User manual No. SSTH/LDAS/EM/2004-040). Grenoble, France: CEA.
13. Lavialle, G. L. (2005.). cathare2 v2.5_1:User guidelines No. SSTH/LDAS/EM/2004-067). Grenoble, France, CEA.
14. Darona, J. (2006). cathare2v2.5_2mod8.1: Dictionary of directives and operators, CSSI.

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10. SUPPLEMENTARY NOTES
K.Tien, NRC Project Manager

11. ABSTRACT (200 words or less)
Thermal-hydraulic analyses are a key part in support of regulatory work for new and existing nuclear power plant design and operation. This paper describes the approach to model the Loss of Coolant Accident (LOCA) in a Light Water Reactor as part of the "Safety Analysis Report in Warsaw University of Technology" (SARWUT) project and the framework of the "Familiarization with the calculation codes application" program.
The RELAP5 model of the European Pressurized Reactor (EPR) has been developed on the basis of an available CATHARE-2 input. Both thermal-hydraulic codes, RELAP5 and CATHARE-2, are used for the safety analysis of the NPP. The purpose of this report is to present the intermediate (6-inch) cold leg break calculations performed within the benchmark exercise using both RELAP5 and CATHARE-2 codes.
The results received are satisfactory, however as presented, the calculations performed with the use of both computer codes, at the early-stage give low cladding temperature but differ in the transient characteristics. The discrepancies in the values of chosen safety related parameters are analyzed in detail for understanding and future work.

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