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# International Agreement Report

# Core Exit Temperature Response during an SBLOCA Event in the Ascó NPP

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ABSTRACT

Core exit temperature (CET) measurements play an important role in the sequence of actions during accidental conditions in pressurized water reactors (PWR). Given the difficulties in placing measurements in the core region, CET readings are used as criterion for the initiation of procedures because they can indicate a core heat up scenario. However, the CET response have some limitation in detecting inadequate core cooling, this is simply because the measurement is not taken in the position where the cladding excursion occurs and the superheated steam is generated. The Group of Thermal Hydraulics of the Technical University of Catalonia has conducted analytical studies to assess the performance of RELAP5 and the nodalization approaches for CET predictions through post-test analyses of the ROSA-2 Test 3 experiment. These studies have led to deriving a different nodalization approach for the core region and UP with a 3-dimensional representation.

The information learned with post-test analyses has been transferred to the NPP model through Kv scaling calculations. The scalability between the LSTF and the Ascó NPP has been analyzed in order to select the best scaling Kv factor for the specific scenario. The necessary changes in the nodalization in order to correctly reproduce the CET response, as indicated by the post-test calculations, have been added to the Ascó NPP model. The final step of the work presented here was to adapt the boundary conditions to a more realistic situation in the NPP in order to evaluate the relation between the CET and the PCT.

Finally, when the CET signal was activated in the Ascó NPP, the PCT measured was in the range of [777, 906] K depending on which CET measurement was considered as a reference. Due to the high temperatures at the time the set point is triggered, the effectiveness of the AM actions are at stake and therefore future studies should be focused on the analysis of the evolution of the scenarios after the CET signal is reached and the assessment of the CET set-point value.

# FOREWORD

This report represents one of the assessment/application calculations submitted in fulfillment of the bilateral agreement for cooperation in thermal hydraulic activities between the Consejo de Seguridad Nuclear (CSN) and the US Nuclear Regulatory Commission (USNRC) in the form of Spanish contribution to the Code Assessment and Management Program (CAMP) of the USNRC, whose main purpose is the validation of USNRC thermal hydraulic codes TRACE and RELAP5.

The CSN and UNESA (the association of the Spanish utilities), together with some relevant universities, have set up a coordinated framework (CAMP-Spain), whose main objectives are the fulfillment of the formal CAMP requirements and the improvement of the quality of the technical support groups that provide services to the Spanish utilities, the CSN, the research centers and the engineering companies

This report is one of the Spanish utilities contributions to the above mentioned CAMP-Spain program and has been reviewed by the AP-28 Project Coordination Committee for the submission to the CSN.

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# EXECUTIVE SUMMARY

Core exit temperature (CET) measurements play an important role in the sequence of actions during accidental conditions in pressurized water reactors (PWR). Given the difficulties in placing measurements in the core region, CET readings are used as criterion for the initiation of procedures because they can indicate a core heat up scenario. Within the OECD countries, the CET readings are used in: Emergency Operation Procedures (EOP) as a prevention of AM, the transition from EOP to Sever Accident Management Guidelines (SAMG), in SAMG (mitigation AM) and, in some cases, in emergency planning. However, the CET response have some limitation in detecting inadequate core cooling, this is simply because the measurement is not taken in the position where the cladding excursion occurs and the superheated steam is generated. Therefore, differences between the CET and the peak cladding temperature (PCT) are expected. Therefore, core uncovery will be unnoticed during a certain period of time. Assessing capabilities of system code to simulate the relation between the CET and the PCT is of main importance in the field of nuclear safety for PWR power plants.

In 2008, the Committee on the Safety of Nuclear Installations (CSNI) launched activities to review the background knowledge on the CET performance and related AM procedures. The CSNI concluded that computer codes used to simulate this kind of scenario may not be fully validated and recommended to verify to what extent state-of-the-art system codes are able to reproduce the delay and differences between rod surface temperatures and CET readings. Following the recommendations of the CSNI report, further experiments on this issue were carried out in both the OECD/NEA ROSA-2 and PKL-2 projects by making use of the LSTF and the PKL test facilities. Through the participation to these projects, the Group of Thermal Hydraulics of the Technical University of Catalonia has conducted analytical studies to assess the performance of RELAP5 and the nodalization approaches for CET predictions in order to carry out safety evaluations of NPPs.

The simulation of the experiments has allowed the group to understand the physical mechanisms that govern the differences between the CET and the PCT. These studies have led to deriving a different nodalization approach for the core region and UP with a 3-dimensional representation. In this way, the different radial core zones and different steam velocities are taken into account. Results of the post-test calculation of the ROSA-2 Test 3 have shown a good performance of the nodalization and that the CET response can be predicted with sufficient confidence by the RELAP5 code.

The information learnt with post-test analysis has been transferred to the NPP model through Kv scaling calculations. The scalability between the LSTF and the Ascó NPP has been analyzed in order to select the best scaling Kv factor for the specific scenario. The necessary changes in the nodalization in order to correctly reproduce the CET response, as indicated by the post-test calculations, have been added to the Ascó NPP model. The final step of the work presented here was to adapt the boundary conditions to a more realistic situation in the NPP in order to evaluate the relation between the CET and the PCT.

Finally, when the CET signal was activated in the Ascó NPP, the PCT measured was in the range of [777, 906] K depending on which CET measurement was considered as a reference. Due to the high temperatures at the time the set point is triggered, the effectiveness of the AM actions are at stake and therefore future studies should be focused on the analysis of the evolution of the scenarios after the CET signal is reached and the assessment of the CET set-point value.

# ACKNOWLEDGMENTS

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# ABBREVIATIONS

| | |
|---|---|
| Acc | Accumulator |
| AFW | Auxiliary Feed Water |
| AM | Accident Management |
| ANAV | Asociación Nuclear Ascó-Vandellòs |
| BOL | Begining-of-Life |
| CET | Core Exit Temperature |
| CSNI | Committee on the Safety of Nuclear Installations |
| DC | Downcomer |
| EOP | Emergency Operation Procedure |
| HPSI | High Pressure Safety Injection |
| HS | Heat Structures |
| ITF | Integral Test Facility |
| JAEA | Japan Atomic Energy Agency |
| LOCA | LOss of Coolant Accident |
| LPSI | Low Pressure Safety Injection |
| LSTF | Large Scale Test Facility |
| MSIV | Main Steam Isolation Valve |
| NPP | Nuclear Power Plant |
| PCT | Peak Cladding Temperature |
| PV | Pressure Vessel |
| RCP | Reactor Coolant Pumps |
| SAMG | Severe Accident Management Guideline/Guidance |
| SBLOCA | Small Break LOCA |
| SG | Steam Generator |
| UNESA | Asociación Española de la Industria Eléctrica (Association of the Spanish Utilities) |
| UP | Upper Plenum |
| UPC | Universitat Politècnica de Catalunya (Technical University of Catalonia) |

# 1 INTRODUCTION

Core exit temperature (CET) measurements play an important role in the sequence of actions during accidental conditions in pressurized water reactors (PWR). Given the difficulties in placing measurements in the core region, CET readings are used as criterion for the initiation of procedures because they can indicate a core heat up scenario. Within the OECD countries, the CET readings are used in: Emergency Operation Procedures (EOP) as a prevention of AM, the transition from EOP to Sever Accident Management Guidelines (SAMG), in SAMG (mitigation AM) and, in some cases, in emergency planning [1]. However, the CET response have some limitation in detecting inadequate core cooling and core uncovery, this is simply because the measurement is not taken in the position where the cladding excursion occurs and the superheated steam is generated. Therefore, differences between the CET and the peak cladding temperature (PCT) are expected. In fact, if CET measurements indicate the presence of superheated steam, it is in all cases with certain delay from its formation and the steam temperature will be always lower than the actual maximum cladding temperature taking place in the core. Therefore, core uncovery will be unnoticed during a certain period of time. Assessing capabilities of system code to simulate the relation between the CET and the PCT is of main importance in the field of nuclear safety for PWR power plants.

Experimental results obtained at the Large Scale Test Facility (LSTF) within the OECD/NEA ROSA-1 project [2] suggested that the response of the CET thermocouples could be inadequate to initiate the relevant AM actions. In particular, during Test 6-1 [3], a small break loss-of-coolant-accident (SBLOCA), it was observed that core uncovery started well before CET thermocouples reported sufficient high temperatures. In order to address this issue, the Committee on the Safety of Nuclear Installations (CSNI) launched activities to review the background knowledge on the CET performance and related AM procedures. As a result, the CSNI delivered a report in 2010 with conclusions and recommendations on the issue [1]. The CSNI concluded that computer codes used to simulate this kind of scenario may not be fully validated and recommended to verify to what extent state-of-the-art system codes are able to reproduce the delay and differences between rod surface temperatures and CET readings. The CSNI report concluded that further research should be dedicated, among others, at the following activities:

- Assessment of physical models to predict heat transfer modes affecting CET behavior
- Development of a “best practice guideline” for the nodalization approach of the uncovered core section up to the point of CET location
- Based on comparison results, assessment of the possible impact of 3D effects not modelled in these codes
- Investigate the problem of CETs issue “scaling” (methods of extrapolating) from experimental facilities size, like LSTF, to commercial PWR reactors

Therefore it is important to evaluate, with the use of experimental work at integral test facilities (ITF) and the use of system codes, the relation of the evolution of the CET and the PCT. However, scaling and geometrical effects are thought to have a strong impact on the CET measurements. Following the recommendations of the CSNI report, further experiments on this issue were carried out in both the OECD/NEA ROSA-2 and PKL-2 projects by making use of the

LSTF and the PKL test facilities. In particular, Test 3 of the OECD/NEA ROSA-2 project, an SBLOCA at the hot leg, was designed with the intention to study the evolution of the CET in comparison to the PCT.

The present report, intends to partly address the issues raised in the CSNI report by post-test calculations of the LSTF Test 3 of OECD/ROSA-2 and scaled calculations for the Ascó nuclear power plant (NPP). The main objective of this report is to test the validity of RELAP5 to evaluate the relationship between the CET and the PCT, and then apply the lessons learnt to study the evolution of the CET in a similar scenario for the Ascó NPP.

## 1.1 Use of Scaling Calculations

Experimental results obtained at ITFs are used by the scientific community to understand the behavior of the system in its full complexity. In addition, they can be used to validate the performance of thermal-hydraulic system codes under conditions similar to those expected in accidental situations in actual NPPs. These experiments are intended to reproduce as accurately as possible the conditions in the reference NPP through a series of scaling considerations.

Historically, the power to volume scaling theory has been employed in most ITFs and has proven to be the most adequate approach to face the scaling of complicated geometries. However, a “perfect” scaling of an intricate system is rather difficult if not impossible. In nuclear systems, most of the problems in scaling emerge when gravitational forces are of the same order of magnitude as inertial forces which may occur in transient or accidental situations. In this situation the scaling of horizontal and vertical pipes can be influenced by gravitational forces in different manners and the perfect scaling of both directions with a single scaling approach is not possible. A compromise must be taken that can lead to distortions in the outcome. It can be then affirmed that these distortions will be reduced as we increase the size of the facility. On the other hand, considering that a power to volume approach is applied, the power of the facility must be increased if we increase its volume, and this can be done only up to a certain extend. This leads again to a compromise. Most facilities have been designed with a large volume but are only able to operate and initiate the experiments at a reduced power (around 10% of the scaled initial power). Considering these drawbacks and the additional geometrical differences, it becomes evident that the results obtained in the ITF cannot be directly applied to the NPP scale. Therefore, plant-scaled calculations at the NPP level are needed to close the loop (NPP design, ITF experiment, ITF simulation, NPP simulation).

Plant-scaled calculations (called Kv-scaled analyses following reference [4]) are strongly involved in the qualification process of nodalizations. They consist of adjusting the transient conditions of an NPP nodalization to the test conditions of an ITF experiment. It allows the behavior of NPP and ITF nodalizations to be compared under the same conditions in order to check the capabilities of an NPP nodalization and to improve it if required. Several plant-scaled calculations have been done during recent years [5,6,7,8,9].

# 2 TEST 3 OF THE OECD/NEA ROSA-2 PROJECT

The OECD/NEA ROSA-2 project aimed to resolve key light water reactor thermal-hydraulics safety issues by using the LSTF facility at the Japan Atomic Energy Agency (JAEA). LSTF is a full-height and 1/48 volumetrically scaled test facility for system integral experiments simulating the thermal-hydraulic responses at full pressure conditions of a 1100 MWe-class. The reference plant is Unit-2 of Turuga NPP of the Japan Atomic Power Company, a Westinghouse design [10].

Test 3 [11] simulated a PWR hot leg SBLOCA as a counterpart test to a PKL experiment [12]. The main objective of the experiment was to analyze the reliability of core exit thermocouples which are utilized worldwide as an important indicator of core heat-up and to start an accident management operator action.

## 2.1 Test Rig Description

LSTF is an experimental facility operated by JAEA, it is designed to simulate a Westinghouse-type 4-loop 3,420 MWth PWR under accidental conditions. It is a full-height and 1/48 volumetrically-scaled two-loop system with a maximum core power of 10 MW (14 % of the scaled PWR nominal core power) and pressures scaled 1:1. Loops are sized to conserve volumetric factor (2/48) and to simulate the same flow regime transitions in the horizontal legs (preserving $L/\sqrt{D}$ factor).

There is one steam generator (SG) for each loop respecting the same scaling factors. They have 141 full-size U-tubes, inlet and outlet plena, steam separator, steam dome, steam dryer, main steam line, four downcomers (DC) and other internals.

All emergency systems are represented and have a big versatility referred to their functions and positions. Many break locations (20) are available too.

LSTF test facility has about 1,760 measurement points that allow an exhaustive analysis of the tests. There are two types of data or measurements of interest: directly measured quantities (temperature, pressure, differential pressure), and derived quantities (from the combination of two or more direct measured quantities –coolant density, mass flow rate…).

## 2.2 Test Description

This test was divided in three phases, a high pressure phase, reproducing the NPP scenario at full pressure; a low pressure phase, reproducing the same scenario at counterpart conditions with PKL; and finally, an intermediate phase, with the purpose of conditioning the LSTF conditions at the end of the high pressure phase to the PKL counterpart test conditions. Table 2 shows the list of the imposed conditions for each phase. The present report is focused on the high pressure phase only because the conditions are similar to the ones occurring in an NPP.

Test 3 is initiated by opening a valve located at the upper side of the hot leg with a throat opening of 1.5% of the cold leg area. At the same time, loss of offsite power is assumed to take place leading to the shut down of the primary pumps and the unavailability of the high pressure safety injection (HPSI) and the main feedwater system. Due to the loss of coolant, a steep depressurization of the primary system takes place and, hence the SCRAM signal is reached. As a consequence, the main steam isolation valves are closed causing an increase of the secondary pressure. The set-points for the opening of the secondary side relief valves are soon reached and the secondary pressure oscillates around this pressure following the successive openings and closings of the valve. At this stage, reflux condensation conditions are reached and the primary pressure remains slightly above the secondary pressure. As coolant is being depleted continuously and no injection is available, the reflux condensation conditions are finally broken, primary pressure becomes lower than the secondary pressure and the core level begins to fall. The core rods are exposed and the cladding temperature increases abruptly. With some time delay, the CET also increases but with a lower increase rate. The high pressure phase is ended when the PCT reaches 750 K.

**Table 1 Boundary Conditions of LSTF Test 3 of the OECD/NEA ROSA-2 Project**

| **Event** | **Condition** |
|---|---|
| High pressure transient | |
| Break valve opened | t = 0 s |
| Low pressure scram signal | $P_{prim}$ < 12.96 MPa |
| Secondary system isolation | scram signal |
| Initiation of primary coolant pump coastdown | scram signal |
| Initiation of core power decay curve simulation | scram signal |
| Conditioning phase | |
| Initiation of HPI coolant injection into PV UP | $T_{PCT}$ > 750 K |
| Break valve closed | $P_{prim}$ < 5 MPa |
| Power constant | $P_{prim}$ < 5 MPa |
| Termination of HPI coolant injection into PV UP | $H_{HL} \cong ½ \cdot H_{HL}$ |
| SG's RV depressuritzation | $H_{HL} \cong ½ \cdot H_{HL}$ |
| Secondary system isolation | $P_{prim}$ < 3.9 MPa |
| Low pressure transient | |
| Break valve re-opened | $P_{prim}$ > 4.5 MPa |
| SG depressurization as AM action | $T_{CET}$ > 623 K |
| Initiation of AFW in both loops | AM action signal |
| Initiation of ACC system in both loops | $P_{prim}$ < 2.6 MPa |
| Termination of ACC system in both loops | $P_{prim}$ < 1.2 MPa |
| Initiation of LPI system in both loops | $P_{prim}$ < 1 MPa |

# 3 LSTF MODEL AND RESULTS OF TEST 3

A base case calculation was performed using the UPC LSTF RELAP5mod3.3 nodalization that had been qualified previously for the LSTF test 3.1 and test 3.2 of the OECD/ROSA-1 project (references [13] and [14]). The major features of the vessel nodalization were:

- core and fuel modeled with one channel and one heat structure respectively
- fuel axial core power calculated as an average of the Low, Medium and High experimental core power profiles
- characterization of passive heat structures simulating control rods, core barrel, upper core support plate, instrumentation and environment heat losses.

Even though the calculation showed a quite good agreement for reproducing the main events of the transient, a considerably different slope in the almost linear relation between the CET and the PCT was obtained [15]. The UPC LSTF nodalization was then improved with a Pseudo 3D modeling in order to represent the radial temperature profiles in the core.

**Table 2 Chronology of the Most Relevant Events for the ROSA Test-3, and the RELAP5 Calculations by Using the LSTF Model and the Ascó NPP Model**

| Event | Experiment (s) | LSTF RELAP (s) | Ascó NPP RELAP (s) |
|---|---|---|---|
| Break valve opened | 0 | 0 | 0 |
| Scram signal (primary pressure = 12.97 MPa) | 29 | 33 | 28 |
| Turbine trip and closure of SG MSIVs | 30 | 33 | 28 |
| Initiation of coastdown of primary coolant pumps | 33 | 33 | 28 |
| Termination of SG main feedwater | 34 | 33 | 28 |
| Primary pressure became lower than SG secondary-side pressure | About 1310 | 1253 | 1152 |
| Start of increase in fuel rod surface temperature | 1595 | 1521 | 1501 |
| Maximum fuel rod surface temperature = 750 K (end of the high pressure phase) | 1840 | 1743 | 1932 |

In that sense, a UPC LSTF Pseudo 3D nodalization was implemented splitting the core in 13 channels with 18 axial levels (see Figure 1). The low, medium and high core power axial profiles were simulated, arranging them in each channel as in the experimental radial power distribution. Cartesians crossflows were used for organizing them radially and transversal momentum equations were activated in order to take into account the possible radial ΔP's. Passive heat structures (HS) were split according to the geometries. Finally, the upper plenum (UP) was modified simulating it with two channels, one hot channel, connected to the outlet of the hottest core channel, and another one simulating the rest of the plenum.

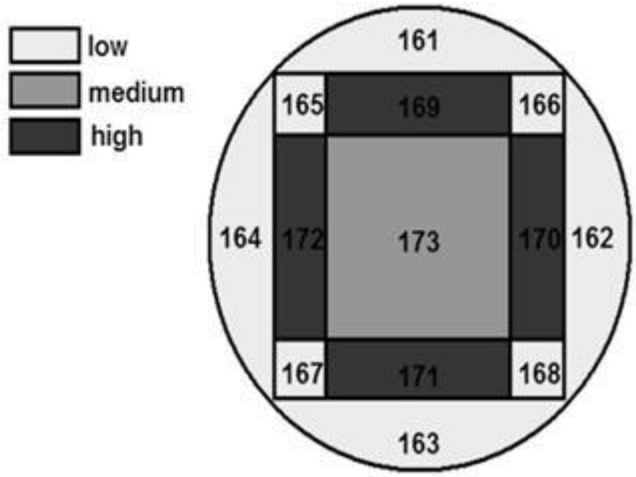



**Figure 1 Detail of the Core Region Nodalization for the LSTF Facility**

Results of the UPC LSTF Pseudo 3D nodalization are shown in Figure 2. Overall, a good agreement was obtained, being the only discrepancy the starting time of the core uncovery. The hot leg at the break location was emptied earlier leading to an earlier depressurization and decrease of the RPV levels. As regards to the evolution of the PCT and CET, both time trends showed a similar slope after core uncovery (Figure 3). The rather good agreement between the calculation and the experiment is a good starting point to perform a scaling calculation for the Ascó NPP.

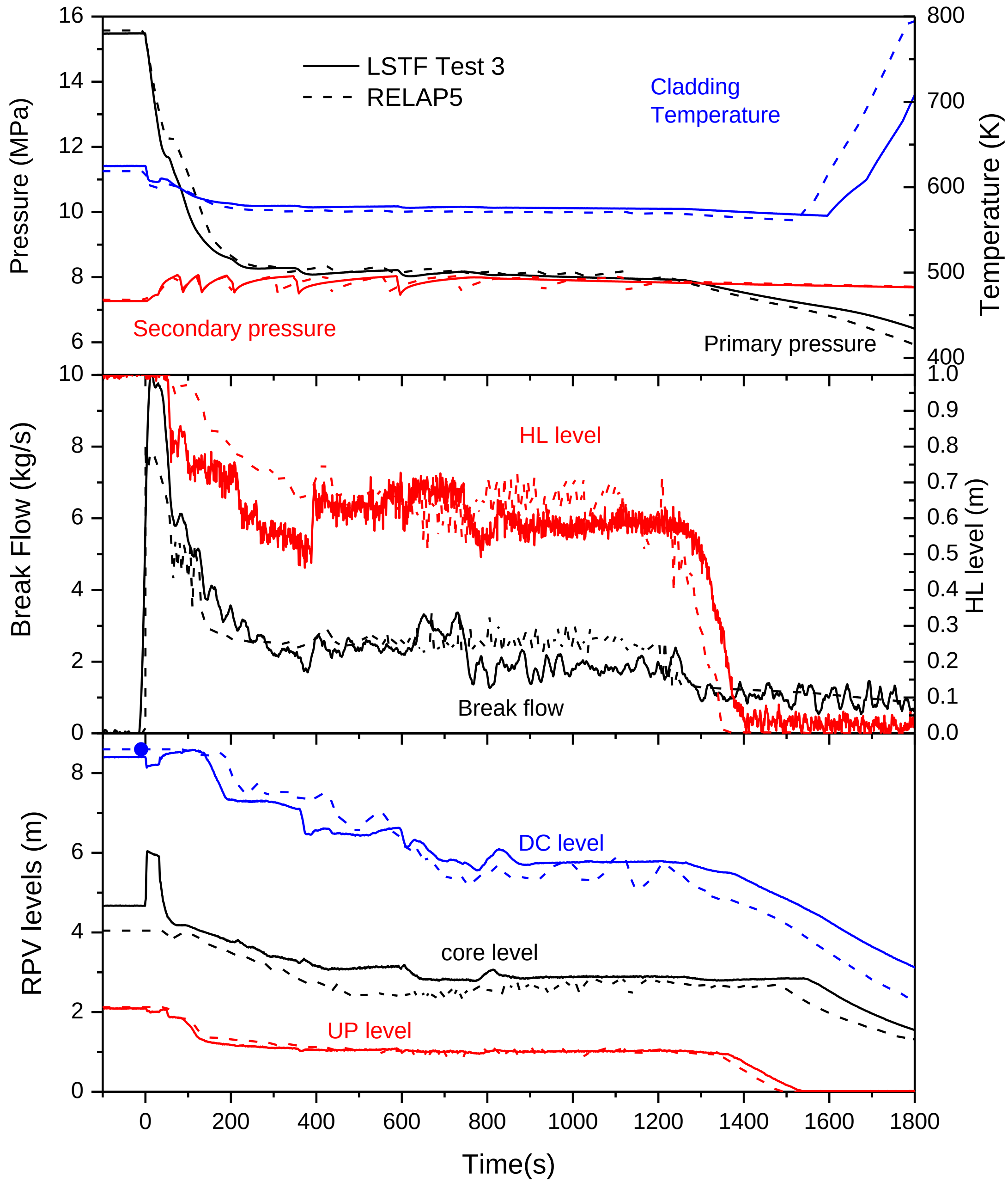



**Figure 2** **RELAP5 Results for Test 3. From top to bottom: (1) primary and secondary pressure along with the maximum cladding temperature, (2) break flow and hot leg level close to the break location (3) RPV water levels**

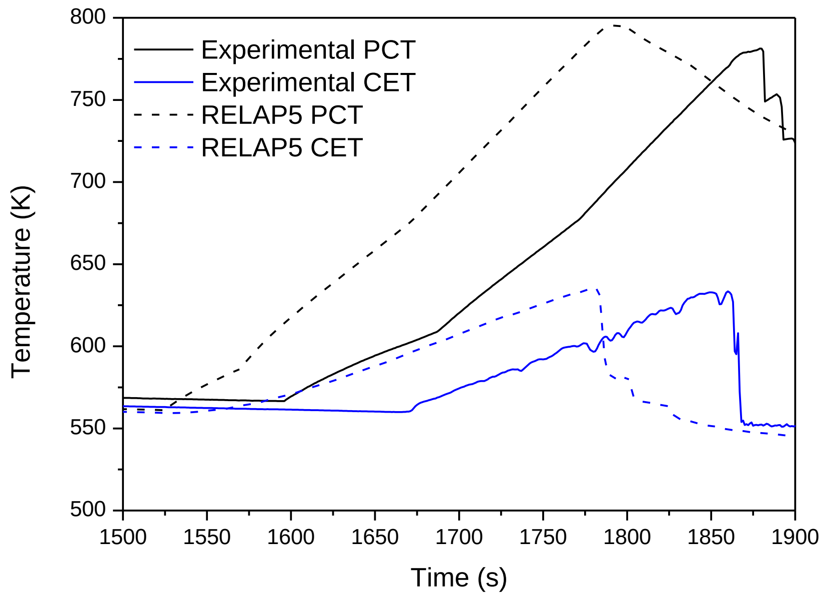



**Figure 3** **RELAP5 Results for the PCT and CET Compared to the Experimental Values**

# 4 ASCO NPP PLANT MODEL DESCRIPTION

Ascó NPP has two units; each of them is a three-loop PWR of Westinghouse design. The first unit is owned by ENDESA (100%). Second unit is owned by ENDESA (85%) and Iberdrola (15%). The units are located close to Tarragona, in the north east of Spain, and they use the Ebro river as a final heat sink. The commercial operation of the plant started on December 1984. The actual nominal power of each unit is 2952.3 MWt equivalent to 1028 MWe. The reactor vessel is cold head type; the main characteristics of the reactor are summarized in Table 3. The plant is equipped with the three Siemens (type SG 61 W/D3) steam generators. The feed water is fed directly to the upper part of the downcomer via J-tubes.

**Table 3 Main Features of Ascó NPP**

| | |
|---|---|
| Thermal reactor power (MWt) | 2952.3 |
| Electric power (MWe) | 1028 |
| Fuel | $UO_2$ |
| Number of fuel elements | 157 |
| Loops | 3 |
| Reactor operation pressure (MPa) | 15,51 |
| Average coolant temperature(K): | |
| Hot Zero Power | 564,8 |
| Hot Full Power | 582,3 |
| Steam generator | Siemens SG 61W/D3 |
| Number of U-tubes in SG | 5130 |
| Total tube length (m) | 98759 |
| Inside tubes diameter (m) | 0.0156 |
| Tubes material | INCONEL |
| Pumps type | Westinghouse D 100 |
| Primary Circuit volume ($m^3$) | 106.19 |

The Ascó NPP RELAP5 model is prepared to simulate both units of the plant. Only slight changes are needed, concerning mainly to the fuel load, to switch from one to another. When at full power, each plant produces, in the actual configuration, 2952.3 MW thermal (1028 MW electric). Although most of the main components of the plant are Westinghouse design, the present steam generators were designed by Siemens.

The model of the plant includes hydrodynamic elements (primary, secondary, safety systems and auxiliary systems), heat structures, and control and protection systems. The model has been subjected to a thoroughly validation and qualification process, which includes the simulation of transients occurred in the plant itself [16], [17].

Figure 4 shows a general view of the hydrodynamic part of the model. The nodalization diagram for the reactor pressure vessel (RPV) is sketched in Figure 5, whereas Figure 6 reproduces the nodalization scheme used for the steam generators.

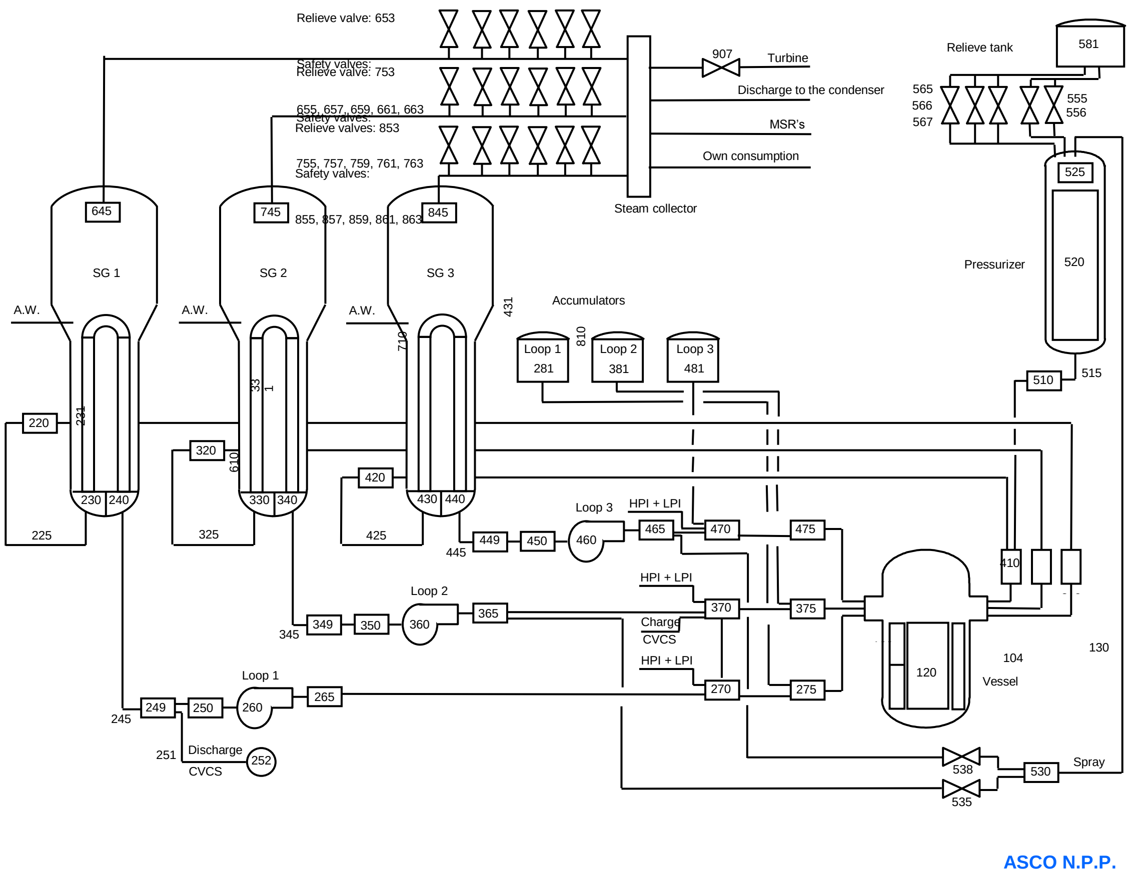



**Figure 4 Diagram of the Ascó Nodalization for RELAP5**

Table 4 summarizes the model's degree of detail. During the preparation of the model, a great effort was devoted to the control and protection systems. Ascó model is able to reproduce the automatic response of the plant systems in practically all the circumstances and, in addition, it incorporates some signals simulating operators' actions.

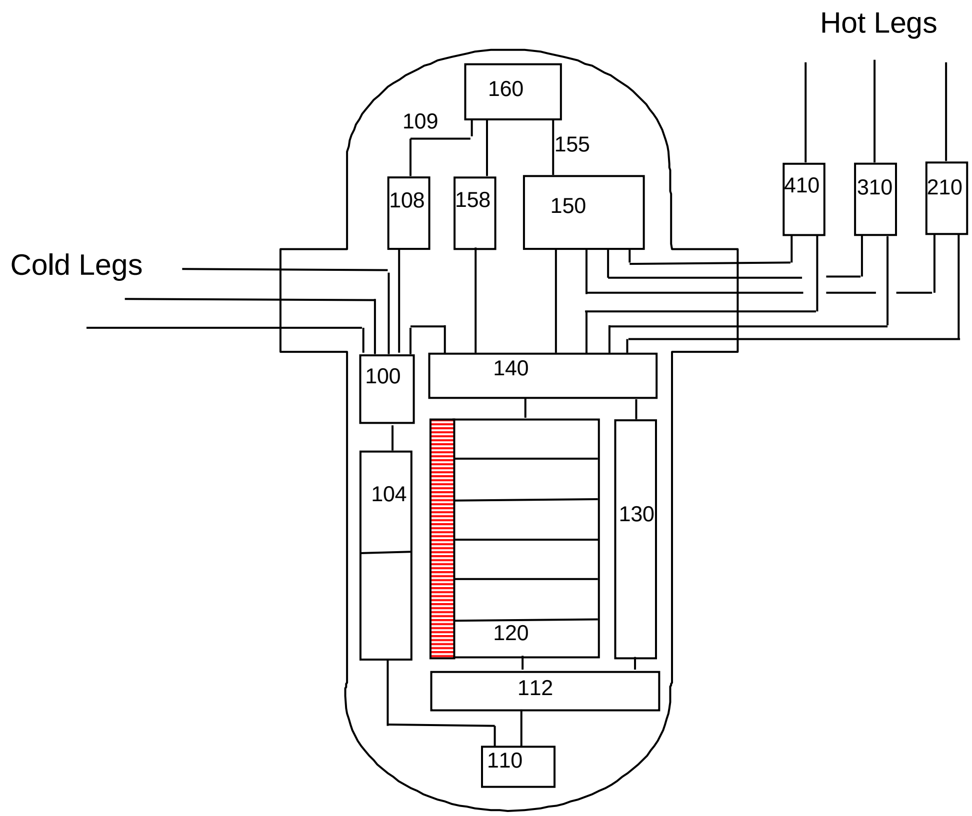
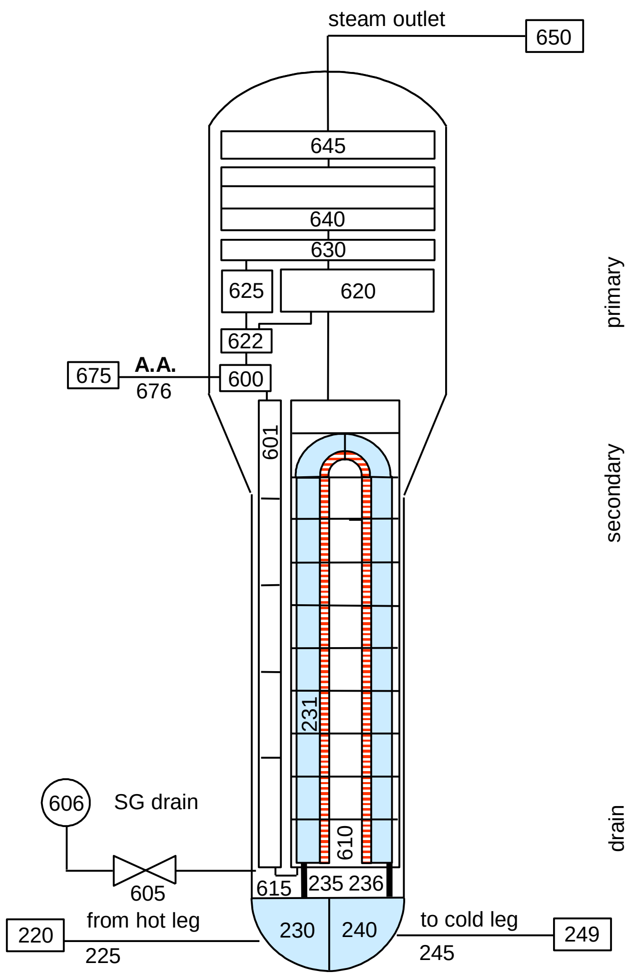



**Figure 5** **Detail of the RPV Nodalization for the Ascó NPP**

**Table 4** **Number of Elements of the RELAP5 Input Deck for the Ascó NPP**

| Component type | Number of elements |
|---|---|
| Hydrodynamic volumes | 549 |
| Heat slabs | 138 |
| Heat structure nodes | 559 |
| Control variables | 1454 |
| Variable trips | 219 |
| Logical trips | 431 |
| Tables | 241 |




| | GV 1 | GV 2 | GV 3 |
|---|---|---|---|
| primary | 225 | 325 | 425 |
| | 230 | 330 | 430 |
| | 231 | 331 | 431 |
| | 235 | 335 | 435 |
| | 236 | 336 | 436 |
| | 240 | 340 | 440 |
| | 245 | 345 | 445 |
| secondary | 600 | 700 | 800 |
| | 601 | 701 | 801 |
| | 610 | 710 | 810 |
| | 615 | 715 | 815 |
| | 620 | 720 | 820 |
| | 622 | 722 | 822 |
| | 625 | 725 | 825 |
| | 630 | 730 | 830 |
| | 640 | 740 | 840 |
| | 645 | 745 | 845 |
| | 650 | 750 | 850 |
| drain | 605 | 705 | 805 |
| | 606 | 706 | 806 |

**Figure 6 Detail of the SG Nodalization for the Ascó NPP**

# 5 SCALING CONSIDERATIONS

In order to perform a Kv-scaled analysis for an NPP model of an experiment test carried out at an ITF, scaling considerations should be made. The scaling factors between the ITF and NPP must be evaluated to assess the viability of the scaling analyses. In addition, the analyst should define a scaling factor between the two designs that should be employed to define the boundary conditions of the Kv-scaled calculation. For the current scenario, the scaling factor will be only used to define the break size, the core power, the pump speed, and the area of the secondary relief valves. Table 5 displays the scaling factors used in the design of LSTF and the ones between LSTF and the Ascó NPP. It can be noticed that even for the scaling of the reference plant there is a variability of the factors depending on the parameter, this is a direct consequence of the fact that a perfect scaling of a complex system is not possible. Most values are around 48 which is the design scaling factor, but some are considerably lower like the volume of the loops. This means that the volume of the loops in LSTF is proportionally larger than the one from the reference plant. In the case of the Ascó NPP, one can observe that two different scaling values prevail. The scaling factor observed for the parameters related to the RPV are around 39.0 and the values linked to the primary loops are around 36. In order to define one single scaling factor, expert judgment is needed, and there is not just one correct answer.

In this scenario, if a scaling factor of around 40 is selected, the coolant in the loops will be depleted faster due to the proportionally smaller volume of the loops in the Ascó NPP. Therefore, the break flow will transit earlier to single phase vapor causing the boil off of the core coolant and an earlier core uncovery. These events would take place at a lower power compared to the LSTF experiment, and a significant distortion on the CET-PCT correlation would be expected. On the other hand, if a scaling factor close to 35 is chosen, the break flow and the voiding of the U-tubes and hot leg region will be similar as it occurs in the experiment. However, if the scenario evolves and a further depletion of the primary side takes place, distortions will appear because the volume of the RPV will be larger in comparison to the break flow. Because, the purpose of the study is to focus on the early stage of the core uncovery and the evolution of the CET temperature as a function of the PCT, a scaling factor of 35 is selected. This means that, the timing of the phenomena and also the power in the core at this time will be correctly scaled.

The changes introduced in the Ascó NPP model in order to perform the scaling calculation are as follows:

- Break nodalization. The same break nodalization as employed in the LSTF RELAP5 nodalization is used with scaled areas by a factor of 35.

- The core power is defined as 35 times the core power in the LSTF Test 3

- The pumps coastdown is the same as in the experiment

- The initial conditions are adjusted to be the same as in the experiment. The initial PZR level is adjusted so that the volume of liquid is 35 times larger than the initial volume of liquid of the LSTF Test 3

- The secondary relief valves setpoints are modified to be the same as in the experiment. The area of the valves is scaled to be 35x2/3 times the area of the valves in LSTF.

**Table 5 Scaling Factors Between LSTF and Its Reference Plant Compared to the Scaling Factors Between LSTF and Ascó NPP**

| Parameter | Scaling factor, reference plant | Scaling factor, Ascó NPP |
|---|---|---|
| Core power | 47.9 | 41.2 |
| Total volume RPV | 47.8 | 39.8 |
| Core volume | 39.1 | 38.2 |
| Core flow area | 41.9 | 37.8 |
| Number of fuel rods | 50.5 | 39.0 |
| PZR volume | 42.5 | 36.2 |
| Hot leg $\boldsymbol{L/\sqrt{D}}$ | 1.0 | 0.944 |
| Volume of the loops | 39.2 | 36.0 |
| U-tube outer surface | 43.0 | 36.8 |
| Number of U-tubes | 48.0 | 36.38 |
| Volume of SG primary side | unknown | 32.8 |
| Volume of U-tubes | unknown | 35.0 |

# 6 RESULTS

The most relevant results obtained with the RELAP5 Ascó NPP nodalization with the scaled boundary conditions are shown in Figure 7. It is important to notice that the results are compared to the RELAP5 calculation of Test 3 and not with the experimental data. Here, the intention is to see the differences between the nodalizations due to scaling and not the performance of the physical models in the code. By doing a code to code comparison, we assure that the differences are due to either scaling or user choices. The occurrence of the important events is summarized in Table 2. All the events occurred similarly in both calculations. The observed differences are summarized below:

- Increase of peak cladding temperature during core uncovery. Again, due to scailing differences, a slower increase of the peak cladding temperature is expected. The number of fuel rods and thus the volume and the heat capacity of the fuel rods have a scaling value of 39 and in the analysis a factor of 35 is used, thus the power density is smaller in the Ascó NPP calculation.

- Voiding of the broken hot leg. This is related to the distribution of the volumes in the primary system. The volume of the SG primary side has a scaling value of 32.8 while the boundary conditions scaling value was 35. Therefore, the volume above the hot leg break is proportionally smaller in the Ascó NPP model, leading to an earlier voiding of this region.

- Break flow during the transition from subcooled to two-phase flow at the break location (50-200 seconds). Even though the break nodalization, choked flow model employed and model coefficients were the same, the results obtained during the transition from subcooled to two-phase flow was slower in the Ascó NPP model. This is related to the HL level decrease during this phase, it might be related to the different Froude number in the horizontal section of the HL. A deeper analysis by using the UPC-scaling methodology [15,18] should be carried out to determine the source of this discrepancy, however this analysis is beyond the scope of the present publication.

- Initiation of DC level decrease. The initiatiation of the DC level decrease is correlated with the transition from two phase flow to single phase flow at the break and the reversing of the heat transfer between the primary and the secondary side. A further analysis with the UPC-Scaling methodology [15,18] would be required to correctly describe this difference.

Despite the differences described above, the evolution of both systems is rather similar which indicates a good performance of the Ascó NPP model in the reproduction of the case of study. One can thence study in detail the evolution of the CET and the PCT. Figure 8 shows the evolution of the CET and The PCT, here again a different slope of the PCT and the CET is observed. The shadowed area indicates the time region where different actions (conditioning phase) were taken in the LSTF experiment, this actions are not performed in the Ascó NPP model because the focus of the present study is in the evolution of the CET and PCT and not on the actions taken afterwards to prevent core damage.

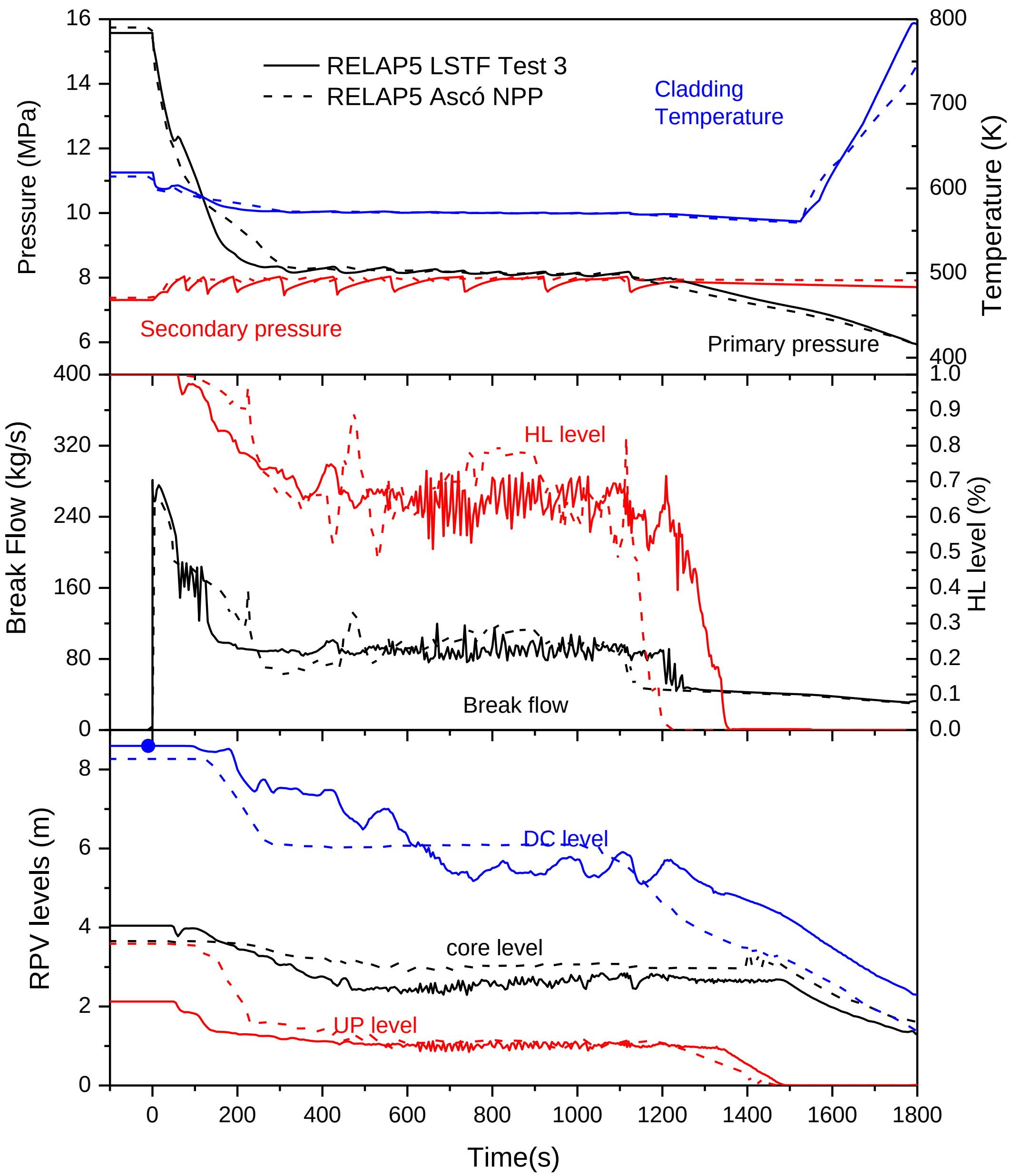



**Figure 7** **RELAP5 Results for Both the Post-Test Calculation of Test 3 and the Scaling Calculation With the Ascó NPP Model. From top to bottom: (1) primary and secondary pressure along with the maximum cladding temperature, (2) break flow and hot leg level close to the break location (3) RPV water levels**

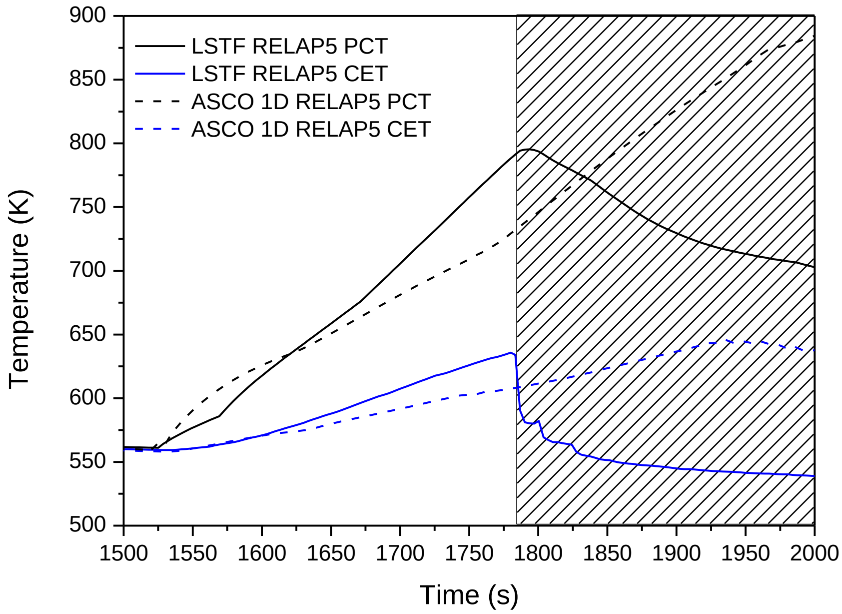



**Figure 8 PCT and CET Results Obtained by the Post-Test Calculation of Test 3 and the Scaling Calculation With the Ascó NPP Model**

Figure 9 displays the maximum PCT in the core as a function of the CET for the ROSA experiment, the LSTF RELAP calculation and the Ascó NPP RELAP model. This figure is of main importance because it compares the information seen by the operator represented by CET measurements and the maximum temperature found in the core. In this sense, the CET value is given in as many points as possible (depending on the number of cells available in the core outlet). For the experiment, two sets of points are plotted: the first one corresponds to the thermocouple that detected the highest temperatures (located above the hottest core zone) and the second set corresponds to the core exit thermocouple that detected the lowest temperatures (located at the periphery of the core outlet).

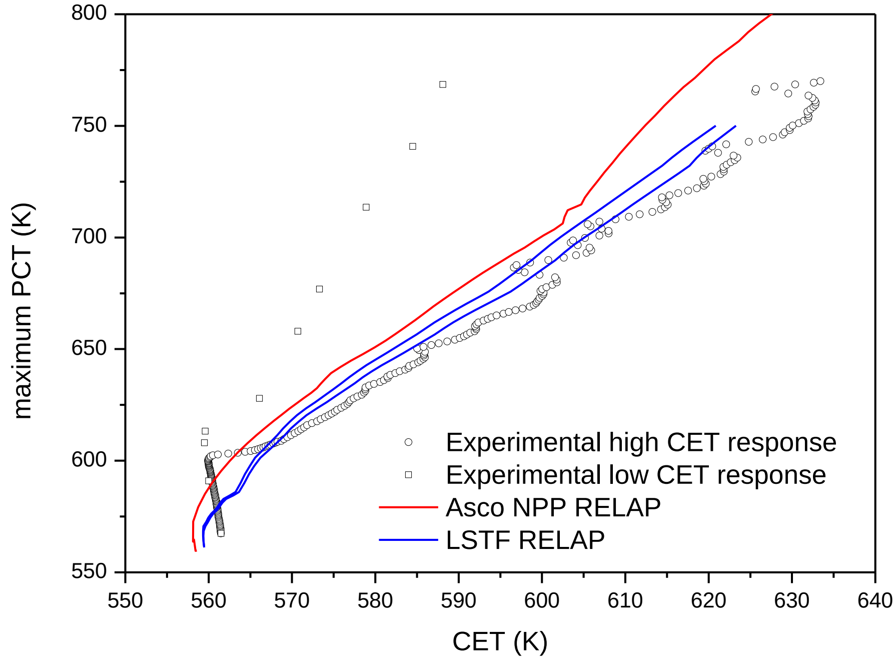



**Figure 9 PCT as a Function of the CET for Test 3 (Experiment and Calculation) and the Scaling Calculation Performed With the Ascó NPP Model**

## 6.1 <u>3D Representation of the Core and Upper Plenum Region</u>

As it was mentioned in section 3, in order to correctly represent the relation between PCT and CET, a detailed nodalization of the core region was needed. The reason is because the CET temperature will strongly depend on the steam velocities and the heat transfer processes with the passive heat structures in the core and core outlet regions. A 3D nodalization permits the correct representation of the core power and the location of the passive heat structures. In this sense, one can conclude that a 3D representation of the core region and the core outlet is needed for the Ascó NPP model.

The core region and core outlet of the Ascó NPP model were renodalized following a similar approach as in the LSTF model. The 6 axial nodes were renodalized into 18 nodes. The single channel was split in 4 pipes (see Figure 10). Crossflow junctions were added between the zones. The criteria used to distribute the proportion of total area and volume of each pipe was carried out by dividing the core in power zones. The fuel assemblies were sorted according to their linear heat generation rates (LHGRs), and 4 zones were defined by grouping similar LHGR fuel assemblies. After that, the area proportion of each pipe was calculated by comparing the number of fuel rods in each zone to the total number of fuel rods. One additional heat structure was included to represent the hot rod; this structure was added to the hot zone 1 hydraulic channel. The proportion of each zone and their peaking factors are displayed in Table 6. The first node of the UP was also re-nodalized in order to observe different CET at the exit of the core depending on the zone as shown in Figure 10.

**Table 6 Power Zones in the Ascó NPP 3D RELAP5 Nodalization**

| Zone | Area of the core (%) | Peaking factor |
|---|---|---|
| Hot rod | - | 1.583 |
| Hot zone 1 | 13 | 1.324 |
| Hot zone 2 | 28 | 1.242 |
| Average zone | 36 | 1.098 |
| Periphery zone | 23 | 0.424 |

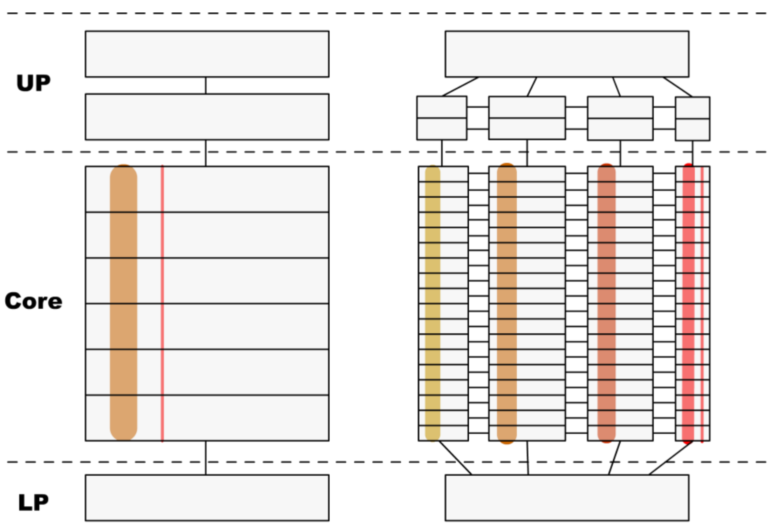



**Figure 10 Renodalization of the Core Region for the Ascó NPP Model**

The scaling calculation was performed one more time with the new core nodalization. The main results of the system behavior are shown in Figure 11 (primary pressure and PCT). The overall performance of the model did not present any significant difference in terms of system behavior. The differences between the two models appear when a close look to the details is given.

In Figure 12, the CET and PCT evolution for the coarse and fine nodalizations are plotted. For the fine nodalization there are now 4 CET measurements corresponding to the four channels. The figure shows that even though the PCT is the same in both calculations, the core exit temperature presents different values. The lowest of the CET temperatures in the fine nodalization is equal to the CET found in the coarse nodalization. Therefore, one can say that the 1D approach is more conservative for this scenario. In addition, it can be noticed that the 3D approach provides a spectrum of CETs providing some uncertainty depending on the position of the measurement in respect to the radial power distribution. In terms of safety, with the 1D nodalization, the CET never reaches the set point (653 K) to activate the required AM to mitigate the core heat up. Therefore, in this scenario, the 1D nodalization would most probably lead to core damage.

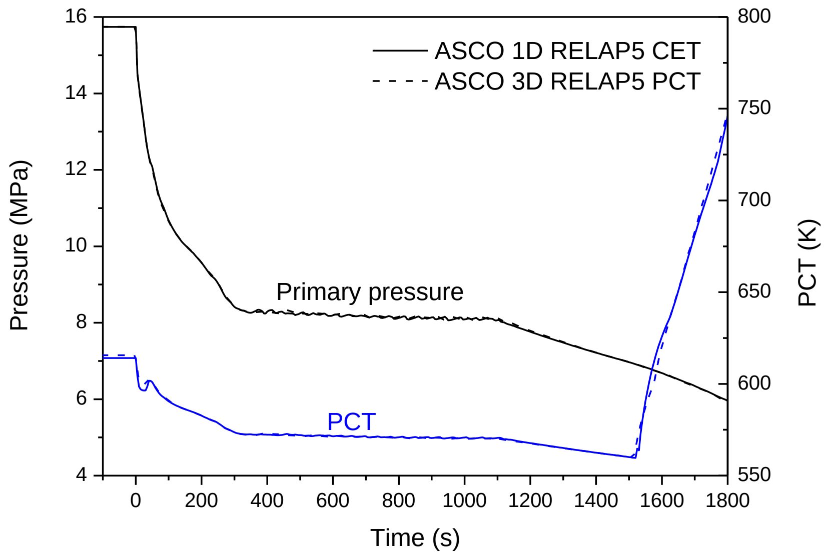



**Figure 11 Main Results of the Scaling Calculation With the Ascó NPP Model by Using a Coarse and Fine Nodalization of the Core**

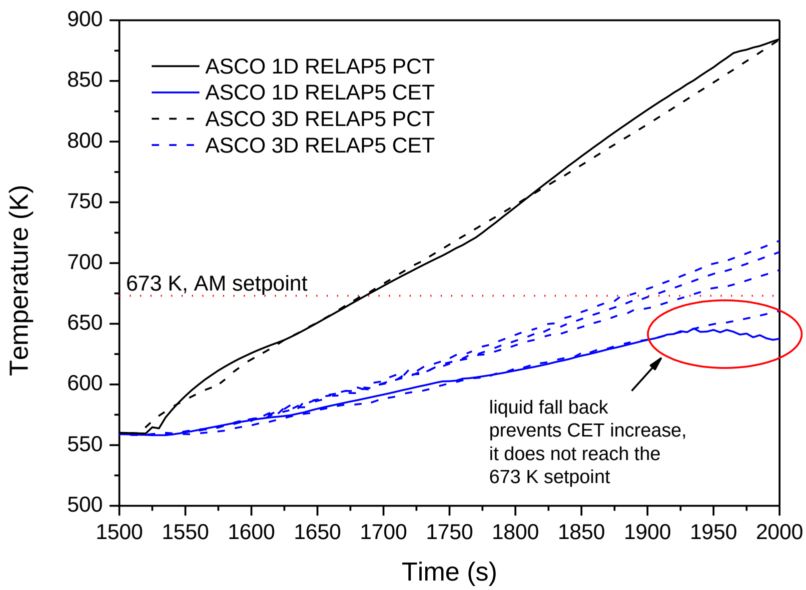



**Figure 12 PCT and CET Results of the Scaling Calculation With the Ascó NPP Model by Using a Coarse and Fine Nodalization of the Core**

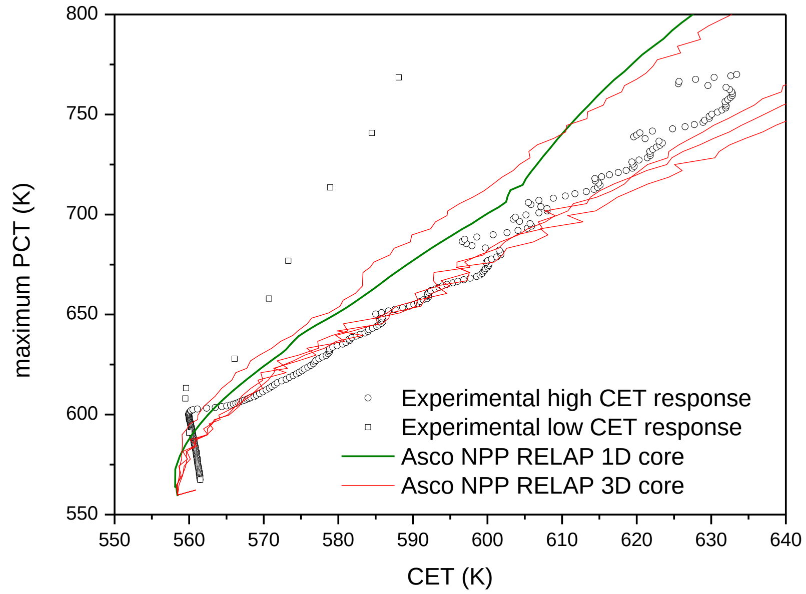



**Figure 13 PCT as a Function of the CET for the Scaling Calculation With the Ascó NPP Model by Using a Coarse and Fine Nodalization of the Core**

# 7 REALISTIC SCENARIO

The scaling calculation was performed with boundary conditions equivalent to the ROSA Test 3. Some of these changes might have an important effect on the evolution of the transient and were used in order to avoid unnecessary distortions when comparing the RELAP5 calculations done by the LSTF and the Ascó NPP nodalization. Since some of these conditions might have an effect on the relation between CET and PCT, it is interesting to perform a calculation of the same scenario with more realistic boundary conditions. The changes performed are summarized below:

- Point kinetics are used instead of a predefined power table
- The coast down of the reactor coolant pumps (RCP) is based on the homologous curves of the Ascó NPP
- The secondary relief valves set points are set to the original Ascó NPP set points
- Initial conditions are set to the original Ascó NPP initial conditions

In Figure 14, the main results of the realistic case are compared to the results of the scaling calculation (both cases are carried out with the detailed core nodalization). The differences between the two calculations are minor and are mostly related to a different core power decrease at the time of scram. In addition, the secondary pressure is slightly higher in the realistic case since the set points for the SG relief valves are higher in Ascó than in LSTF. Therefore, the primary pressure remains slightly higher during the reflux-condensation phase (400-1200 seconds) and thus the break flow during this phase was also higher. The consequence is that the coolant in the loops was depleted earlier and hence core uncovery occurred about 100 seconds earlier.

The PCT as a function of the CET is shown in Figure 15 and compared with the results obtained with the scaled up BC. The correlation of both temperature changes very little with the new boundary conditions. The maximum PCT when the CET reaches 653 K is in the range of [777, 906] K depending on which CET measurement is taken as a reference. This means that when AM actions are taken to mitigate the core heat up, the PCT might be as high as 906 K. The question remains on whether the fast secondary depressurization as AM action will be sufficient or in time to avoid core damage. According to ITF experiments at PKL and LSTF the secondary depressurization produces a fast replenishment of the core, however the PCT in the experiments at the time the AM actions were taken were of about 725 K. Therefore, further analyses should be performed in order to assess the effectiveness of the AM actions taken at the specified set point (CET=653K).

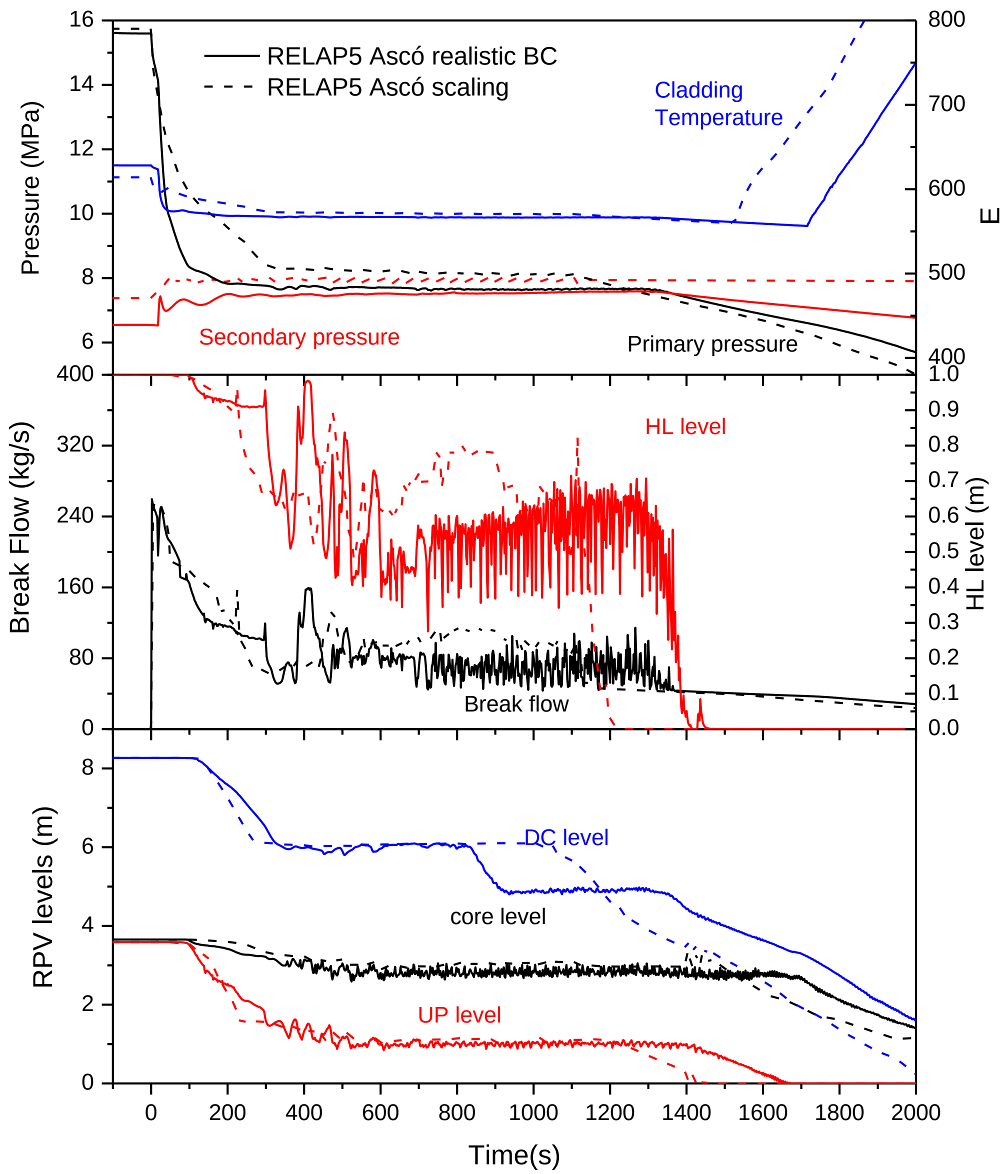



**Figure 14** **RELAP5 Results of the Ascó NPP Model With Scaled and Realistic Boundary Conditions. From top to bottom: (1) primary and secondary pressure along with the maximum cladding temperature, (2) break flow and hot leg level close**

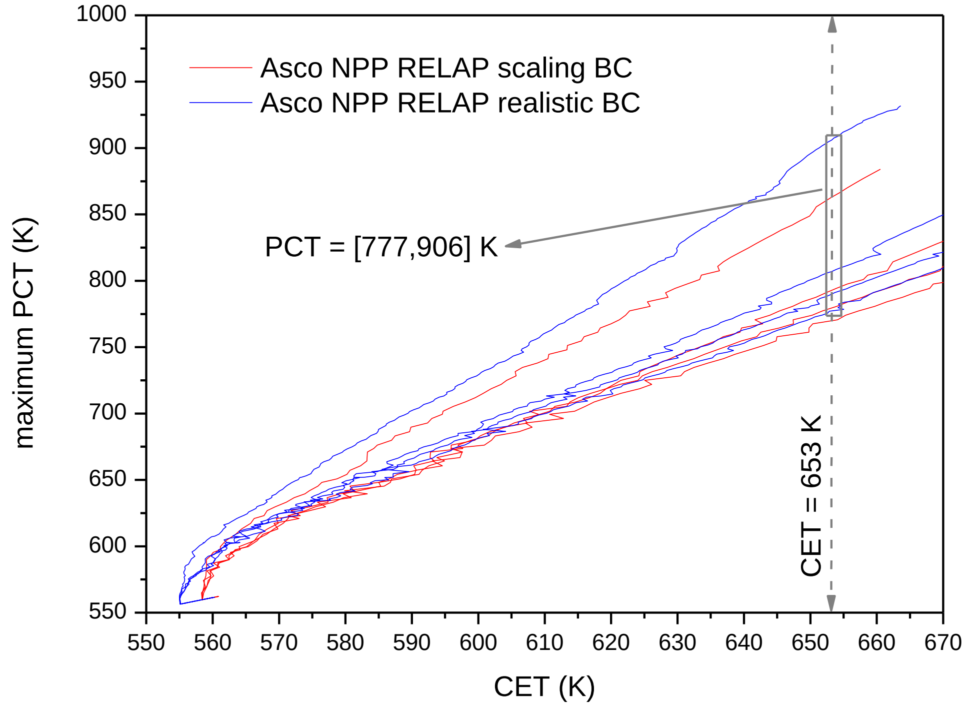



**Figure 15** **PCT as a Function of the CET With the Ascó NPP Model With the Scaled Up and the Realistic Boundary Conditions**

# 8 CONCLUSIONS

Following the recommendations from the CSNI suggesting the need for further research on the effectiveness of the CET measurements in accident management of nuclear power reactors, the Group of Thermal Hydraulics of the Technical University of Catalonia has conducted analytical studies to assess the performance of RELAP5 and the nodalization approaches for CET predictions. In particular, the analytical work has been possible through the participation on both international projects OECD/NEA-ROSA-2 and OECD/NEA-PKL-2 that featured ITF experiments reproducing a hot leg SBLOCA scenario where the CET response is crucial.

The simulation of the experiments has allowed the group to understand the physical mechanisms that govern the differences between the CET and the PCT. These studies have led to deriving a different nodalization approach for the core region and UP with a 3-dimensional representation. In this way, the different radial core zones and different steam velocities are taken into account. Results of the post-test calculation of the ROSA-2 Test 3 have shown a good performance of the nodalization and that the CET response can be predicted with sufficient confidence by RELAP5.

The scalability between the LSTF and the Ascó NPP has been analyzed in order to select the best scaling Kv factor for the specific scenario. Scaled boundary conditions for the Ascó NPP have been then defined accordingly. The necessary changes in the nodalization in order to correctly reproduce the CET response, as indicated by the post-test calculations, have been added to the Ascó NPP model. The scaled calculation showed a very similar response between the LSTF model and the Ascó NPP model. Only a few scaling issues were detected.

The final step of the work presented here was to adapt the boundary conditions to a more realistic situation in the NPP. This was done by mainly adding the nominal initial conditions, applying the point kinetics model in order to simulate the core power and the use of the homologous pump curves to define the RCP coast down.

The final conclusions in terms of reactor safety are:

- The three calculations (Test 3 post-test, scaling calculation and the realistic scenario) and the experimental results provided a very similar correlation between the PCT and the CET. However, the difference in temperature between the low and high CET measurements was larger in the experiment.

- The set point for the CET measurement to activate AM actions in the Ascó NPP is set at 653 K. For this set point, the PCT measured was in the range of [777, 906] K depending on which CET measurement is taken as a reference.

- The use of a 3D approach brought forward the differences of the outlet core steam temperatures depending on the radial location. This shows that having several CET thermocouples in the NPP is crucial.

- The results showed that 1D results might be conservative, in this case the CET did not even reach the 653 K set-point, therefore, the scenario proposed would lead to a most probable severe accident situation.

- Due to the high temperatures at the time the set point is triggered. Future studies should be focused on the analysis of the effectiveness of the AM actions, for instance full secondary depressurization.

- In the case that the effectiveness of the AM actions cannot be proven, the validity of the CET set point should be addressed

# 9 REFERENCES


[1] CSNI, “Core Exit Temperature (CET) Effectiveness in Accident Management of Nuclear Power Reactor” *Committee on the Safety of Nuclear Installations OECD Nuclear Energy Agency,* November 2010

[2] M Suzuki, T. Takeda and H. Nakamura, “Performance of Core Exit Thermocouple for PWR Accident Management Action in Vessel Top Break LOCA Simulation Experiment at OECD/NEA ROSA Project”. Journal of Power and Energy Systems, Vol. 3, No. 1, p. 146-157, 2009.

[3] OECD/NEA ROSA Project Supplemental Report for Test 6-1 (SB-PV-09 in JAEA) – Performance of Core Exit Temperatures for Accident Management Action in LSTF 1.9% Top Break LOCA Test, JAEA-Research 2007-9001.

[4] F. D'Auria and G. M. Galassi, "Scaling in Nuclear Reactor System Thermal Hydraulics", Nuclear Engineering and Design, 240, pp. 3266-3293 (2010)

[5] Martínez, V.; Reventós, F.; Pretel, C.; Sol, I.; Code Validation and Scaling of the ROSA/LSTF Test 3-1 experiment; International Topical Meeting on Safety of Nuclear Installations TopSafe 2008; Octubre 2008; Croacia (pendent de revisió)

[6] Pla P., Reventós F., Pretel C., Giannotti W., D’Auria F., Annunziato A.; “Code Validation and Scaling of the LOBI BL-30 experiment”; Proceedings of ICCAP 2007, Paper 7492; Nice, May 2007

[7] S. Petelin, B. Mavko, B. Koncar, Y. Hassan; “Scaling of the small-scale termal-hydraulic transient to the real nuclear power plant; Thermal Hydraulics; 2006, July 31.

[8] J. Freixa, [et al.]. SBLOCA with boron dilution in pressurized water reactors. Impact on operation and safety. "Nuclear engineering and design", Abril 2009, vol. 293, núm. 4, p. 749-760.

[9] J. Freixa, A. Manera, "Verification of a TRACE EPR model on the basis of a scaling calculation of an SBLOCA ROSA test", Nucl. Eng. Des., 241 (3), pp. 888-896 (2011).

[10] The ROSA-V Group. “ROSA-V Large Scale Test Facility (LSTF) system description for the third and fourth simulated fuel assemblies”. Technical Report JAERI-Tech 2003-**037**, Japan Atomic Energy Agency, 2003.

[1] T. Takeda, M. Suzuki, H. Asaka, and H Nakamura. Quick-look Data Report of ROSA-2/LSTF Test3 (Counterpart Test to PKL SB-HL-18 in JAEA). Technical Report JAEA-Research 2012, Japan Atomic Energy Agency, 2012.

[12] B. Schoen, S. P. Schollenberger and K. Umminger, “Test PKL III G7.1: SB-LOCA with Total Failure of HPSI (Counterpart Testing with ROSA/LSTF) – Quick Look Report”, Technical Report PTCTP-G/2011/en/0008, ARECA, 2012

[13] Martínez, V; Reventós, F.; Pretel, C.; Post-Test Calculation of the ROSA/LSTF Test 3-1 using RELAP5/mod3.3; NUREG/IA-409, 2012

[14] Martínez, V; Reventós, F.; Pretel, C.; Post-Test Calculation of the ROSA/LSTF Test 3-2 using RELAP5/mod3.3; NUREG/IA-410, 2012

[15] Martínez, V; Freixa J., Reventós, F.; Applying UPC scaling-up methodology to the LSTF-PKL Counterpart Test; Science and Technology of Nuclear Installations. vol. 2014, Article ID 292916, 18 pages, 2014. doi:10.1155/2014/292916.

[16] Pretel C., Batet L., Cuadra A., Machado A., San José G. de, Sol I., Reventós F., April 2000. Qualifying, validating and documenting a thermal-hydraulic code input deck. Workshop proceedings. Advanced thermal-hydraulic and neutronic codes: current and future applications, NEA/CSNI/R (2001) 2, Vol 2 pp.239-250.

[17] F. Reventós, L. Batet, C. Pretel, M. Salvat, and I. Sol, Advanced qualification process for ANAV integral plant models, Nuclear Engineering and Design, vol. 237, no. 1, pp. 54–63, 2007.

[18] V. Martínez-Quiroga, F. Reventós, “The use of system codes in scaling studies. Relevant techniques for qualifying NPP nodalizations for particular scenarios” Science and Technology of Nuclear Installations. vol. 2014, Article ID 138745, 13 pages, 2014. doi:10.1155/2014/138745

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K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

Core exit temperature (CET) readings are used as criterion for the initiation of procedures because they can indicate a core heat up scenario. However, some limitation exist simply because the measurement is not taken at the position where the cladding excursion occurs and the superheated steam is generated. The Group of Thermal Hydraulics of the Technical University of Catalonia has conducted analytical studies to assess the performance of RELAP5 and the nodalization approaches for CET predictions through post-test analyses of the ROSA-2 Test 3 experiment. These studies have led to deriving a different nodalization approach for the core region and UP with a 3-dimensional representation. The information learned has been transferred to the NPP model through Kv scaling calculations. The scalability between the LSTF and the Ascó NPP has been analyzed in order to select the best scaling Kv factor for the specific scenario. The necessary changes in the nodalization in order to correctly reproduce the CET response, as indicated by the post-test calculations, have been added to the Ascó NPP model.

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Consejo de Seguridad Nuclear (Spanish nuclear regulatory commission, CSN)
Core Exit Temperature (CET)
Severe Accident Management Guideline/Guidance(SAMG)
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