NUREG/IA-0497

International Agreement Report

IBLOCA Analysis for Vandellòs-NPP Using RELAP5/MOD3.3. Sensitivity Calculations to EOP Actions

Prepared by:
M. Pérez-Ferrgut, J. Freixa, F. Reventós and I. Sol

Department of Physics
Universitat Politècnica de Catalunya
ETSEIB, Av. Diagonal 647, Pav. C
08028 Barcelona
Spain

K. Tien, Project Manager

Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Manuscript Completed: January 2018
Date Published: February 2019

Prepared as part of
The Agreement on Research Participation and Technical Exchange
Under the Thermal-Hydraulic Code Applications and Maintenance Program (CAMP)

Published by
U.S. Nuclear Regulatory Commission

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at NRC's Library at www.nrc.gov/reading-rm.html. Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and Title 10, "Energy," in the *Code of Federal Regulations* may also be purchased from one of these two sources.

1. **The Superintendent of Documents**
   U.S. Government Publishing Office
   Mail Stop IDCC
   Washington, DC 20402-0001
   Internet: bookstore.gpo.gov
   Telephone: (202) 512-1800
   Fax: (202) 512-2104

2. **The National Technical Information Service**
   5301 Shawnee Road
   Alexandria, VA 22312-0002
   www.ntis.gov
   1-800-553-6847 or, locally, (703) 605-6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:

Address: **U.S. Nuclear Regulatory Commission**
Office of Administration
Multimedia, Graphics, and Storage & Distribution Branch
Washington, DC 20555-0001
E-mail: distribution.resource@nrc.gov
Facsimile: (301) 415-2289

Some publications in the NUREG series that are posted at NRC's Web site address www.nrc.gov/reading-rm/doc-collections/nuregs are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—

The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852-2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—

American National Standards Institute
11 West 42nd Street
New York, NY 10036-8002
www.ansi.org
(212) 642-4900

Legally binding regulatory requirements are stated only in laws; NRC regulations; licenses, including technical specifications; or orders, not in NUREG-series publications. The views expressed in contractor prepared publications in this series are not necessarily those of the NRC.

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG–XXXX) or agency contractors (NUREG/CR–XXXX), (2) proceedings of conferences (NUREG/CP–XXXX), (3) reports resulting from international agreements (NUREG/IA–XXXX), (4) brochures (NUREG/BR–XXXX), and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of NRC's regulations (NUREG–0750).

NUREG/IA-0497

International Agreement Report

IBLOCA Analysis for Vandellòs-NPP Using RELAP5/MOD3.3. Sensitivity Calculations to EOP Actions

Prepared by:
M. Pérez-Ferrgut, J. Freixa, F. Reventós and I. Sol

Department of Physics
Universitat Politècnica de Catalunya
ETSEIB, Av. Diagonal 647, Pav. C
08028 Barcelona
Spain

K. Tien, Project Manager

Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Manuscript Completed: January 2018
Date Published: February 2019

Prepared as part of
The Agreement on Research Participation and Technical Exchange
Under the Thermal-Hydraulic Code Applications and Maintenance Program (CAMP)

Published by
U.S. Nuclear Regulatory Commission

ABSTRACT

This report analyzes the effect of the plant specific EOPs for the RCPs during a hypothetical IBLOCA in Vandellòs II NPP, a 3 loop Westinghouse design reactor.

The scenarios simulated are:

- Cold leg IBLOCA simulating the rupture of an accumulators connection to the RCS;
- Hot leg IBLOCA simulating the rupture of the pressurizer surge line connection to the RCS.

The selection of an intermediate break size conforms to the risk-informed approach for the assessment of the ECCS performance.

The RELAP5/MOD3.3 model of Vandellòs II NPP was developed by the Advanced Nuclear Technologies (ANT) group at the Universitat Politècnica de Catalunya (UPC).

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 TRACE and RELAP5 system codes.

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.

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

EXECUTIVE SUMMARY

The Department of Physics of the Universitat Politècnica de Catalunya (UPC) holds a large background in the use of thermal-hydraulic codes for the Safety Analysis of Nuclear Power Plants (NPP).

This report analyzes the effect of the Spanish Emergency Operating Procedures (EOPs) for the Reactor Coolant Pumps (RCPs) during a hypothetical IBLOCA in Vandellòs NPP, a 3 loop Westinghouse design reactor. The Spanish EOPs to stop the RCPs require that at least one charging pump is injecting to the RCS, simultaneously with a certain degree of RCS subcooling.

The selected scenarios are Intermediate Break Loss of Coolant Accidents (IBLOCAs):

- Rupture of the accumulators' connection to the RCS (Cold Leg IBLOCA);
- Rupture of the pressurizer surge line connection to the RCS (Hot Leg IBLOCA).

The selection of an intermediate break size conforms to the risk-informed approach for the assessment of the ECCS performance.

The boundary conditions for the IBLOCAs are chosen from the OECD/NEA ROSA-2 project. These conditions impose unavailability of HPIS and AFW pumps. As a consequence, and according to the Spanish NPPs, the RCP should not be stopped.

For each IBLOCA scenario, two cases have been analyzed:

(1) RCPs are always on (accordingly to Spanish EOPs).
(2) Disconnection of RCPs after safety injection signal.

The results for the Cold Leg (CL) scenario clearly show that for case (1), which conforms to the Spanish EOPs for the RCPs, leads to safe conditions faster than case (2).On the other hand, the results for the Hot Leg (HL) case show that for case (1) core uncovery takes place for a short time whereas for case (2) the fuel rods remain wetted throughout the whole transient. In this sense, case (2) leads to a safer situation for the HL break. However, the increase of cladding temperature is very low and the safety of the plant is guaranteed All in all, it can be concluded that the safety margins are larger by maintaining the RCPs on, in accordance to the Spanish EOPs.

The calculations are performed using RELAP5/MOD3.3 code, and the RELAP5 NPP model for Vandellòs II NPP developed by the Advanced Nuclear Technologies (ANT) group at UPC.

ACKNOWLEDGMENTS

Authors are grateful to Association of the Spanish Utilities (UNESA) for funding the work presented in this report and to “Asociación Nuclear Ascó-Vandellós” (ANAV) for their continuous help and cooperation. Specially, authors recognize the valuable advices from Mr. Ismael Sol (ANAV).

ABBREVIATIONS AND ACRONYMS

AFW	Auxiliary Feed Water
ANAV	Asociación Nuclear Ascó-Vandellòs
BE	Best Estimate
CL	Cold Leg
DEGB	Double Ended Guillotine Break
DWR	Downcomer
ECCS	Emergency Core Cooling System
EOP	Emergency Operation Procedure
GET	Thermal-Hydraulics Studies Group (UPC)
HL	Hot Leg
HPIS	High Pressure Injection System
IBLOCA	Intermediate Break Loss-Of-Coolant-Accident
LOCA	Loss of Coolant Accident
LPIS	Low Pressure Injection System
LS	Loop Seal
NPP	Nuclear Power Plant
PCT	Peak Cladding Temperature
PWR	Pressurizer Water Reactor
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RELAP	Reactor Excursion and Leak Analysis Program
RPV	Reactor Pressure Vessel
SBLOCA	Small Break Loss-Of-Coolant-Accident
SG	Steam Generator
TBS	Transition Break Size
UP	Upper Plenum
UPC	Universitat Politècnica de Catalunya
UNESA	Asociación Española de la Industria Eléctrica (Association of the Spanish Utilities)

1 INTRODUCTION

The “Asociación Nuclear Ascó-Vandellòs-II” (ANAV) is a utility that runs three operating reactors. The Ascó reactors started up in 1983 (unit 1) and 1985 (unit 2), while Vandellòs-II started up in 1987. All of them are Westinghouse-design 3-loop PWRs, with an approximate electrical power of 1000MW. In the past ANAV prepared an Integral Plant Model for each plant using Relap5. The model is used for supporting plant operation and control from the point of view of safety and operation. The Universitat Politècnica de Catalunya (UPC) has been working with ANAV since 1991 in order to establish, qualify and use these Best Estimate (BE) models.

The models of Ascó and Vandellòs-II plants have been extensively used by the UPC-ANAV team. Most of the calculations performed are devoted to safety issues [1, 2], engineering issues such as Probabilistic Safety Assessment (PSA) and Emergency Operating Procedure (EOP) development [3, 4], others to qualification procedures [5] and, finally, others to operational issues, mainly to improve the understanding of actual operating events. Those belonging to this latter group have been usually performed with both aims in mind: to clarify the events that occur and also to provide an answer to requests from the person responsible for operation.

The Advanced Nuclear Technologies (ANT) group of the UPC holds a large background in the use of thermal-hydraulic codes for the Safety Analysis of Nuclear Power Plants (NPP). More precisely, ANT has been cooperating for 15 years with the operators of the Catalan nuclear plants, Ascó (two units) and Vandellós II [4].

This report analyzes the Peak Cladding Temperature (PCT) response to the Reactor Coolant Pumps (RCPs) behavior during a postulated Intermediate size break Loss-of-Coolant Accident (IBLOCA) in Vandellòs NPP (a three loop PWR Westinghouse design).

The selected scenarios are the following IBLOCAs:

- Cold leg 9.75 in break simulating the rupture of the accumulators’ connection to the Reactor Coolant System (RCS);
- Hot leg 11.18 in break simulating the rupture of the pressurizer surge line connection to the RCS.

The selection of these transients is based on the results of a previous study [6] carried out by ANT aimed to support risk-informed decisions on Emergency Core Cooling System (ECCS) performance. The proposed ECCS rule [7] divides the spectrum of LOCA break sizes into two regions defined by the Transition Break Size (TBS): the first region includes SBLOCAs up to and including the TBS, the second region includes breaks larger than the TBS up to and including the double ended guillotine break (DEGB) for the largest RCS pipe. The work presented in ref. [6] analyzed potential TBS scenarios from intermediate primary pipe ruptures. Therefore, the phenomenological analysis of the two scenarios is not detailed in the present document.

The aim of the present work is to analyze the PCT response to the behavior of the RCPs. To do so, in the first place the plant specific EOPs [8] are considered: the criterion to stop the RCPs during a LOCA scenario requires that the two following conditions are met simultaneously:

(1) Charging pumps – At least one is injecting to the RCS.
(2) RCS subcooling at the core outlet according to tables in Appendix C and D of ref. [8].

2 PLANT DESCRIPTION

Vandellòs II is a three loop PWR NPP of Westinghouse design owned by Endesa (72%) and Iberdrola (28%), and operated by ANAV. It is located near Tarragona, in the northeast of Spain, and uses the Mediterranean Sea water as heat sink. It started its commercial operation in March 1988. Its nominal power is currently of 1,087 MWe (2,940.6 MWt).

The reactor vessel is Westinghouse designed. The plant has three Westinghouse (model F) steam generators of U-tubes kind without pre-heaters. Feed water enters directly the upper part of the downcomer through J-shaped nozzles.

The main features of the plant are shown in **Table 1**.

Table 1 Main Features of Vandellòs II NPP

Reactor thermal power (MWt)	2940.6
Electrical power (MWe)	1087
Fuel	UO_2
Number of fuel bundles	157
Number of cooling loops	3
Reactor operating pressure (MPa)	15,4
Mean coolant temperature (K):	
Hot zero power	564,8
Full power	582,3
Steam generator (SG)	Westinghouse type F
Number of tubes in one SG	5626
Total tubes length in one SG (m)	98759
Tubes inner diameter (m)	0.0156
Tubes wall material	INCONEL
Coolant Recirculation Pumps	Westinghouse D 100
Volume of the primary (m^3)	106.19
Volume of pressurizer (PZR) (m^3)	39.65
PZR heaters power (kW)	1400

Some of these general features depend on the plant configuration. The table refers to the current one. The specific plant conditions for the analyzed transient will be presented in Section 4. The most relevant features of main equipment of the Nuclear Steam Supply System (NSSS) are presented in [9-13].

3 MODEL OF THE PLANT

The RELAP5 model for Vandellòs NPP has been developed by ANT at the UPC. A detailed description of the model can be found in [14], while the present section provides the main features of the model. The model has been prepared for RELAP5/MOD3.3 [15] and has been subjected to a thoroughly validation and qualification process, which includes the simulation of transients occurred in the plant itself [5], [16].

The RELAP5 code has been developed for best-estimate transient simulation of light water reactor coolant systems during postulated accidents. The code simulates the coupled behavior of the reactor coolant system and the core for loss-of-coolant accidents and operational transients such as anticipated transient without scram, loss of offsite power, loss of feed water, and loss of 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 permit modeling of plant controls, turbines, condensers, and secondary feed water systems.

Figure 1 shows the main nodalization diagram of the Vandellòs- II plant. The model is quite complete and includes a number of important systems like: safety injection systems, steam lines, main and auxiliary feed water and detailed diagrams of vessel, pressurizer and steam generators. Following the corresponding logic diagrams, a full model for control and protection systems was implemented. Control systems with a certain degree of complexity number approximately 30 in the Vandellòs-II model. The model conforms to the current NPP detailed in Table 1.

Regarding the core, the total number of fuel assemblies is 157. A fuel bundle consists of a 17x17 matrix of fuel pins, with 25 inactive positions for instrumentation and control rods. Active core is 3,654 m high and it has a volume of water of 2,609 m^3. In the model, the core is divided into six axial nodes, each one 0,609 m high. The Reactor Pressure Vessel model has been recently modified to a so-called pseudo-3D nodalization [6], i.e. it includes parallel channels for the downcomer and active core connected through crossflow junctions. These modifications are described in Section 3.1 and in more detail in reference [6].

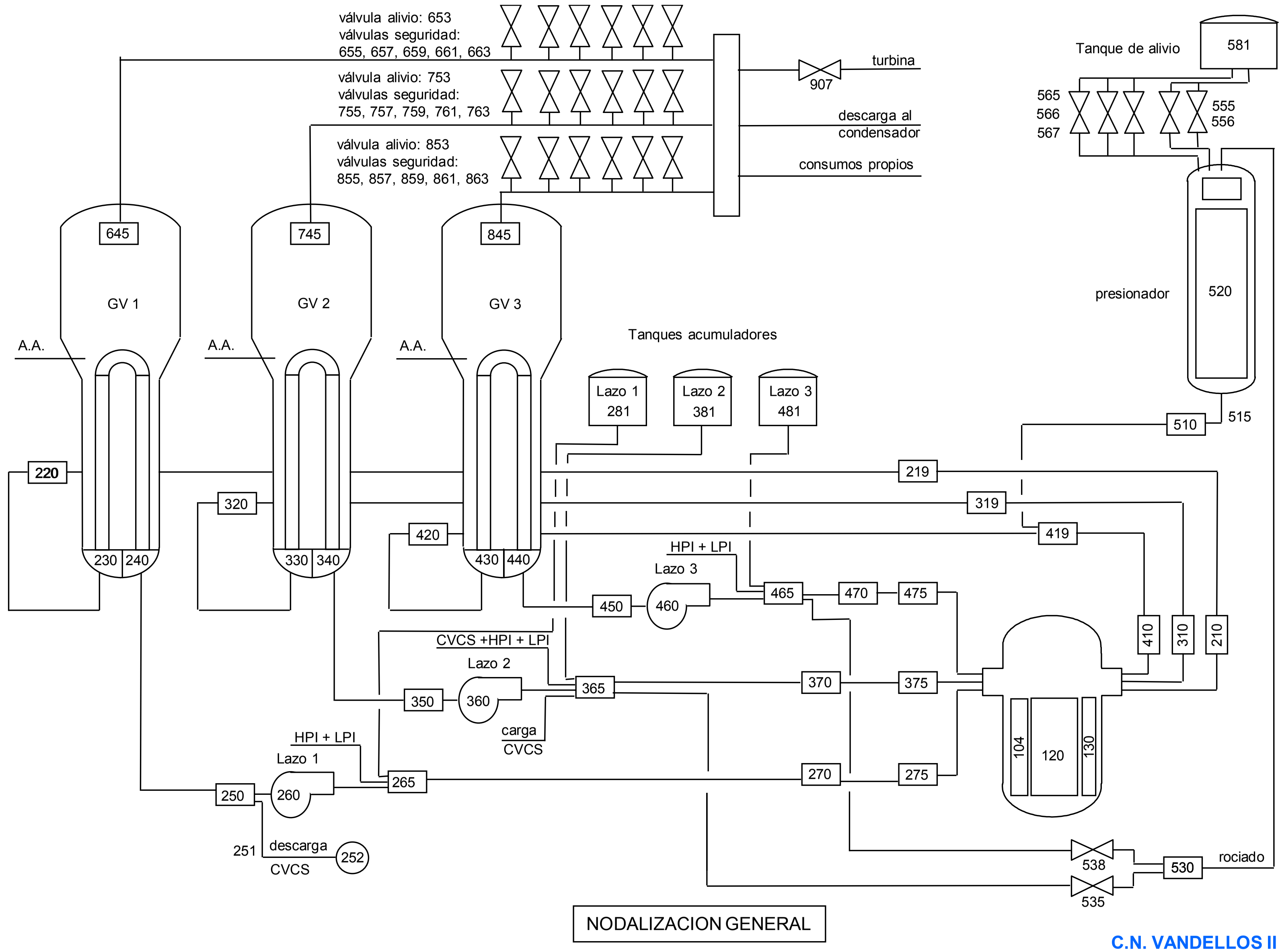

Figure 1 Main Nodalization Diagram of the Vandellòs-II NPP

Table 2 summarizes the model's degree of detail. During the preparation of the model, a great effort was devoted to the control and protection systems. Vandellòs II model is able to reproduce the automatic response of the plant systems in practically all the circumstances and, besides, it incorporates some signals simulating operators' actions.

Table 2 Model Nodalization Main Statistics

Component type	Number of elements
Hydrodynamic volumes	410
Hydrodynamic junctions	487
Heat structures	169
Heat structure mesh points	800
Control variables	1166
Variable trips	113
Logical trips	223

The original model incorporated heat structures on the following components:

- Steam Generators;
- Core;
- RPV;
- Pressurizer; and
- Feed water heaters.

3.1 Plant Model Modifications

The modifications introduced in the Vandellòs NPP model (Relap5 system code version RBIC/3.3) are aimed at improving the simulation of asymmetrical effects by means of a pseudo 3-dimensional nodalization of the vessel: the RELAP5 component "crossflow junction" combined with geometrical data of the control volumes in the crossflow direction (y-direction or z-direction), allows for the pseudo 3-dimensional modeling (see [15]). This nodalization approach was tested with the simulation of IBLOCA experiments at the ROSA/LSTF facility within the OECD/NEA ROSA and provided results in good agreement with the experimental data (see ref [17] for details)

In addition to these changes, the core pipe has been modeled with more detail, the hot leg connections to the vessel have been modified, and the effect of the CCFL model at SG inlet and upper tie plate was tested (steady-state case 3). The nodalization diagram of the new vessel model is shown in Figure 2. The changes included in the model are listed below; the effect of each of these changes was analyzed in ref [6]:

- Active core: changes related to hydrodynamic volumes and fuel heat structures, specifically: 18 axial nodes; two core channels with cross flow junctions; and required changes to passive heat structures (HS) to match hydrodynamic nodalization.
- Fuel HS: 3 representative HSs for peripheral, average and hot rods.
- Downcomer (DWR) nodalization: pseudo 3D nodalization of downcomer by means of three pipes connected with cross flow junctions at each axial level and required changes to passive heat structures to match hydrodynamic nodalization.

- Counter current flow limitation (CCFL) model: activation of the CCFL model at SG inlet volumes and upper core tie plate
- Vessel to HL connections: junctions from vessel to hot legs split into two junctions from volume 150 and junctions from volume 140

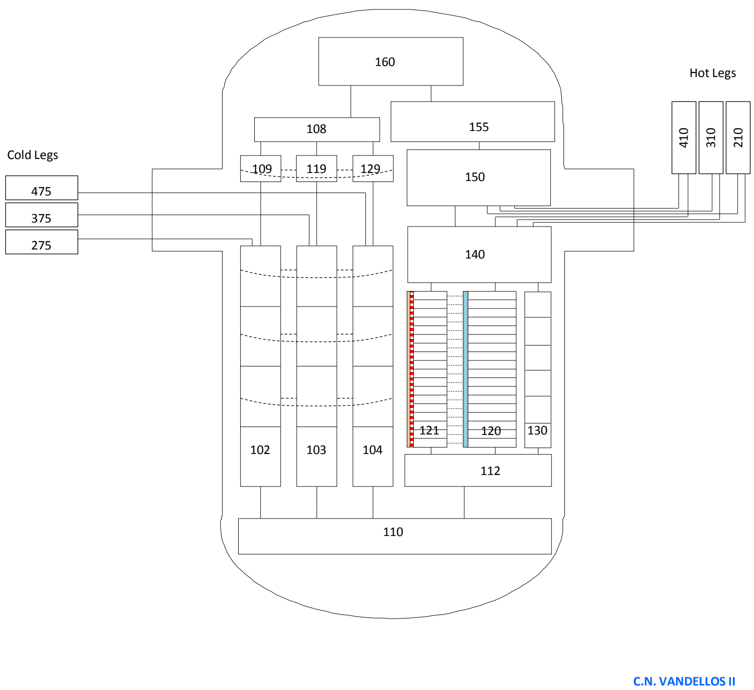

Figure 2 RPV Nodalization

4 SCENARIO

The transients analyzed in this report are two Intermediate Break LOCAs, in the first one the break is located in the cold leg and the second one has the break located at the hot leg.

The boundary conditions of the two scenarios have been established following a risk-informed approach in order to define the size and location of the break. In ref. [6], the DEGB of connecting pipes were analyzed and the two most limiting cases were selected (one for the hot leg and another one for the cold leg). Further boundary conditions were adopted from experiments at the ROSA/LSTF facility within the OECD/NEA ROSA-2 project. In particular, the Japan Atomic Energy Agency (JAEA) conducted three [18, 19, 20] IBLOCA experiments at the Large Scale Test Facility (LSTF), 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 PWR [21]. ANT has carried out post-test calculations of these experiments and reported them in ref [6].

4.1 <u>Cold Leg Case</u>

The Cold Leg IBLOCA simulates the rupture on the accumulator pipe connected to the RCS, this means that only 2 out of 3 accumulators are available. The boundary conditions imposed suppose the unavailability of HPIS and AFW pumps as a result of a Loss-of-offsite Power concurrent with the scram signal. Two cases are analyzed, one considering the RCPs trip and a second one where the RCPs are kept on.

The general evolution of the scenario for the case with the trip of the pumps is as follows. The rather large size of break caused a fast primary depressurization. Break flow turned from single-phase liquid to two-phase flow in a very short time after the break. The primary pressure soon became far lower than the secondary pressure. Core dryout took place due to rapid liquid level drop in the core that occurred simultaneously with LS clearing. Liquid was accumulated in the Upper Plenum (UP), Steam Generator U-tube upflow-side and SG inlet plenum before the LS clearing due to CCFL by high velocity vapor flow, causing further decrease in the core liquid level. The core reflooding and bottom-up quench started after the incipience of accumulator coolant injection. Further details are given in Section 5

.

4.2 <u>Hot Leg Case</u>

The Hot Leg IBLOCA simulates the rupture on the pressurizer surge line connected to the RCS, this means that the coolant available in the pressurizer will not be available. Again, the boundary conditions imposed suppose the unavailability of HPIS and AFW pumps as a result of a Loss-of-offsite Power concurrent with the scram signal. The boundary conditions are similar to those from ROSA-2 project Test-1.

The general evolution of the scenario for the case with the trip of the pumps is as follows. The primary system was rapidly depressurized after the break. The primary system presented a quasi-stationary period when the primary pressure reached values similar to the secondary pressure. However, the mass of coolant in the primary system kept decreasing and at some point the break flow conditions switched from two phase flow to single vapor flow. This caused the core water to boil off and a further depressurization of the system which led to the actuation of the accumulators. Even though the core level decreased, the fuel rods remained wetted until the initiation of the accumulator injection and the occurrence of loop seal clearing. Further details are provided in Section 6.

5 COLD LEG CASE RESULTS

This section presents a comparison between the cold leg IBLOCAs, simulating the rupture of an accumulator line connection to the RCS, defined as follows:

- CL1 case: RCPs are not stopped.
- CL1-rcpstop case: RCPs are stopped with safety injection signal (HPIS) signal.

As described in Section 4, the boundary conditions are no HPIS and no AFW. And, since the scenario simulates the break at the connection line of accumulator 1, then the number of available accumulators are 2/3 accumulators.

CL1-rcpstop case produces a PCT of roughly 700K at 129s after break initiation; CL1 case does not produce any PCT (Figure 2).

Having the RCPs on in CL1 case (Figure 3) maintains a forced circulation in the RCS that causes a larger amount of coolant moving to the downcomer (and later to the core) than in CL1-rcpstop case. The effect of having the RCPs on can be clearly seen in Figure 4, which shows the mass flows in the cold leg connections to the downcomer for the broken loop and intact loop 2. The two cases have the same behavior until CL1-rcpstop case stops the RCP (at roughly 53s) in which case the mass flow rate rapidly decreases (intact loops). Figure 5 shows that delta pressure between the RCP inlet and the downcomer inlet in the broken loop is larger for the case with forced circulation (CL1 case).

In addition to that, the forced circulation in CL1 does not allow for liquid accumulation in the LSs (Figure 6) and thus vapor from the SGs can flow to the break more easily than in CL1-rcpstop case as shown in Figure 7. As a consequence CL1 case looses a larger amount of vapor (Figure 8).

As a result of the phenomenon explained above, the amount of coolant lost through the break is larger when the RCPs are stopped. Specifically, as shown in Figure 8, the amount of liquid lost in CL1-rcpstop is larger (while the amount of vapor is smaller) and so is the total amount of coolant lost through the break. As it can be seen in Figure 8, at roughly 90s CL1 case vapor mass flow is already the same as liquid mass flow and that increases the depressurization rate Figure 10), while for the CL1-rcpstop case, that does not occur until 115s. Figure 9 shows a comparison of the amount of mass lost through the break.

Figure 11 shows the effect of the forced circulation on the downcomer level: in CL1 case the downcomer empties, and thus feed the downcomer volumes (see Figure 12 for core collapsed level) until water from the accumulator reaches the downcomer volumes, while in the CL1-rcpstop case the downcomer level keeps at a relative constant value until quasi-equilibrium conditions are lost. Therefore, up to that point there is no net flow to the core for the CL1-rcpstop case.
Afterwards, the behavior of the collapsed core level (Figure 12), and the cladding temperature, can be explained as follows:

- Core level starts decreasing as a result of blowdown following the break initiation.
- At roughly 85 seconds quasi-equilibrium conditions are established in both cases, and as a result the core level stops decreasing.
- At roughly 115 seconds the quasi-equilibrium state is no longer maintained as the depressurization rate is too strong.

- In the CL1 case, the core level remains high because most of the fluid through the break is vapor (at 90s the vapor break mass flow starts being larger than the liquid break mass flow) and thus, there is no cladding temperature excursion.
- In the CL1-rcpstop case, at roughly 110-120s (see Figure 6 for intact loop 2) loop seal clearing occurs in the intact loops, and that sharpens the depressurization rate (Figure 10). As a result, the core collapsed level decreases. When the collapsed core level falls below 20% the heat transfer across the cladding deteriorates and a cladding temperature excursion starts. Core level cannot recover until the water from the accumulators reaches the core bottom. The accumulators' injection occurs at roughly the same time in both cases.

Figure 13 shows the heat transfer correlations used by RELAP5 (see page 111 in Volume IV of ref. [15]) at a mid position of the hot rod:

- Heat transfer modes 3 and 4 correspond to nucleate boiling in subcooled and saturated conditions, respectively.
- Modes 5 and 6 correspond to transition boiling in subcooled and saturated conditions, respectively.
- Modes 7 and 8 correspond to film boiling in subcooled and saturated conditions, respectively.
- Mode 9 corresponds to single phase vapor.

In CL1-rcpstop case, as the core collapsed level is below 20% the cladding heat transfer degrades (heat transfer modes for saturated film boiling and single phase vapor) and the temperature heats up, while for the CL1 case the heat transfer mode for the hot rod always remains at saturated nucleate boiling conditions and thus there is no cladding temperature excursion. Table 3 summarizes the timing of events.

Table 3 CL IBLOCA - Timing of Events

Event	Time (s) CL1 case	Time (s) CL1-rcpstop case
Break initiation	50.0	
SCRAM signal on PZR low pressure (and turbine trip signal, MFW stop)	50.4	50.4
Subcooling at core outlet ≤ 4º (volume 140) [8]	51	51
RCPs trip on safety injection signal, given subcooling ≤ 4º at core outlet	-	53.2
2-phase break flow	57-58	57-58
Steam dump closes	71	72
RCS (HLs) mass flow < 10%	108	85
LS clearing	-	110-120
Cladding temperature excursion	-	142
Accumulators injection (2/3)	165	161
PCT	-	179
Complete core quenching ($T_{clad} \leq T_{sat}+30$) **	-	204
LPIS injection (3/3)	282	240

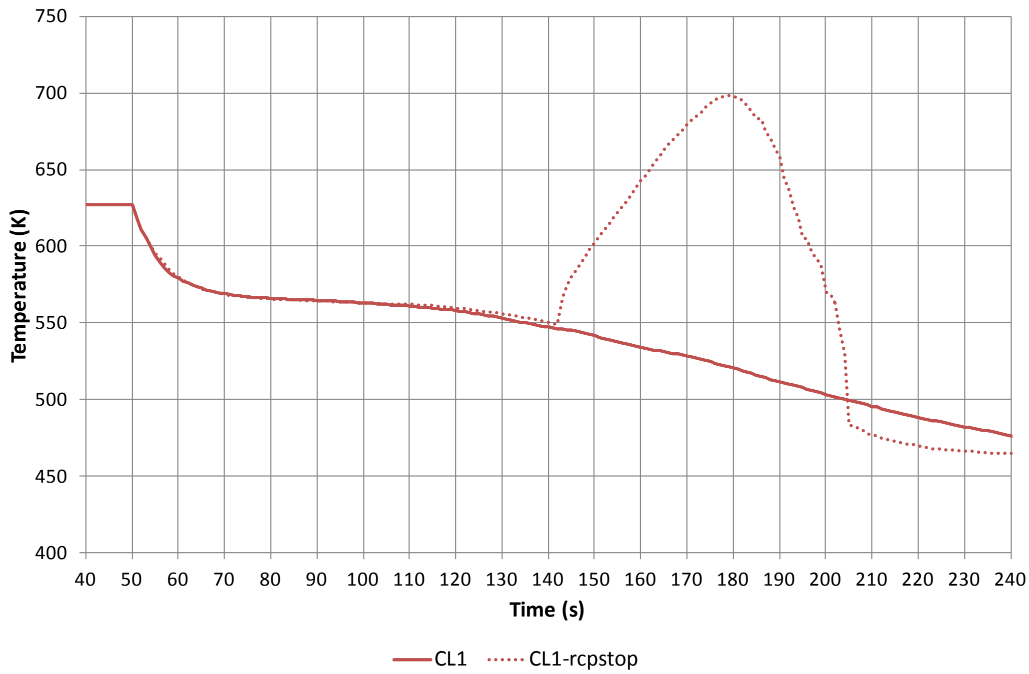

Figure 2 CL IBLOCA - Maximum Cladding Temperature

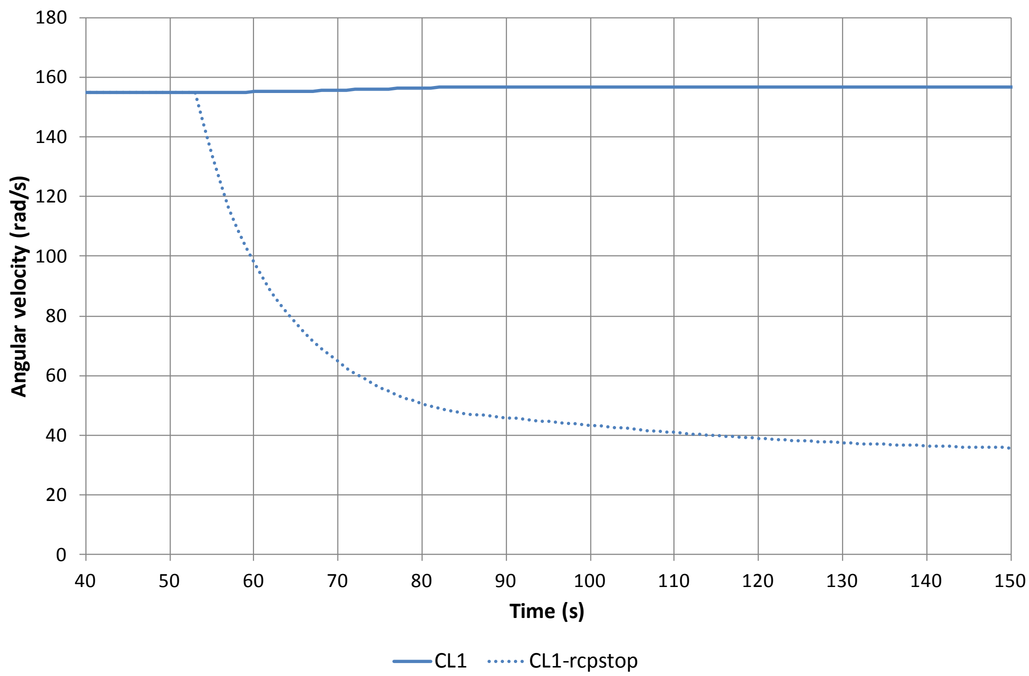

Figure 3 CL IBLOCA - RCP Loop 2 Velocity

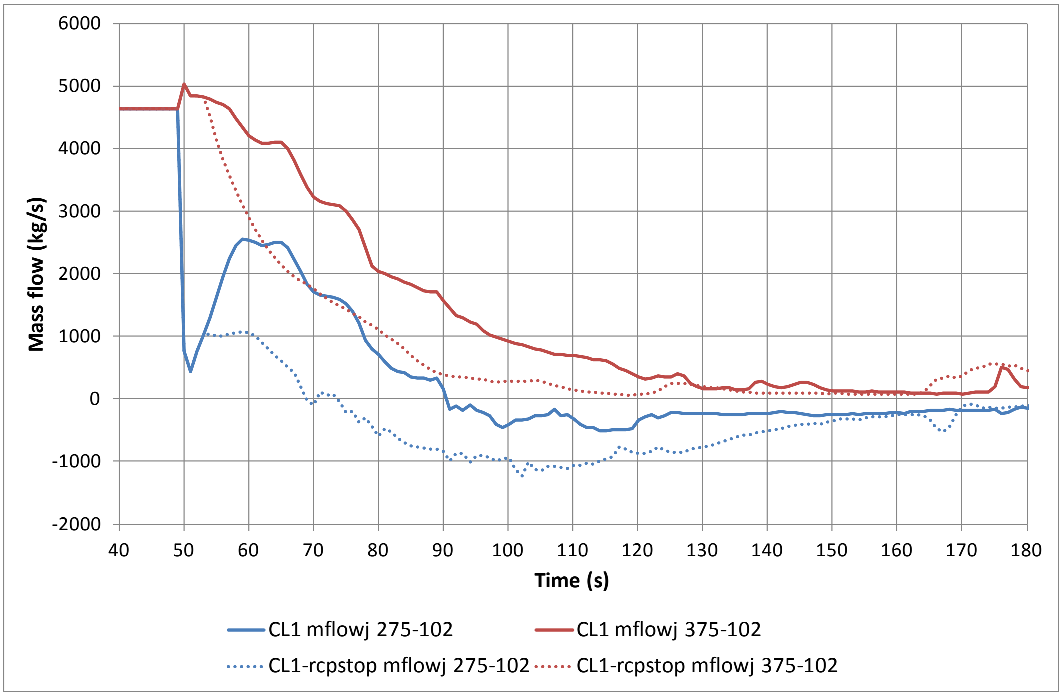

Figure 4 CL IBLOCA - Mass Flow in CLs to DWR

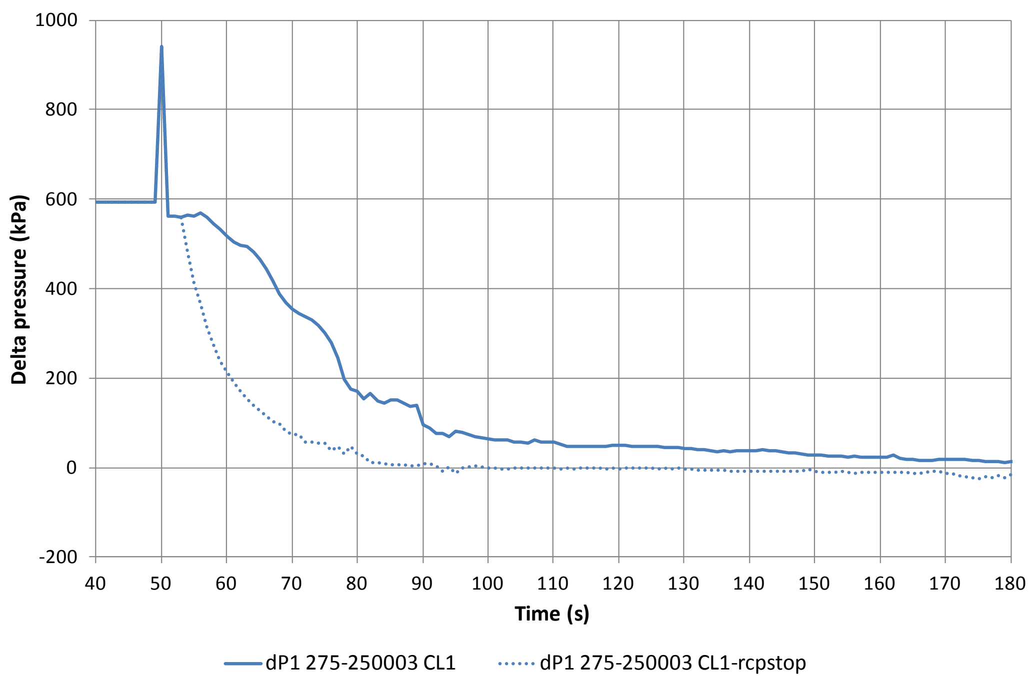

Figure 5 CL IBLOCA - Delta Pressure (DWR inlet - RCP inlet) in Broken Loop

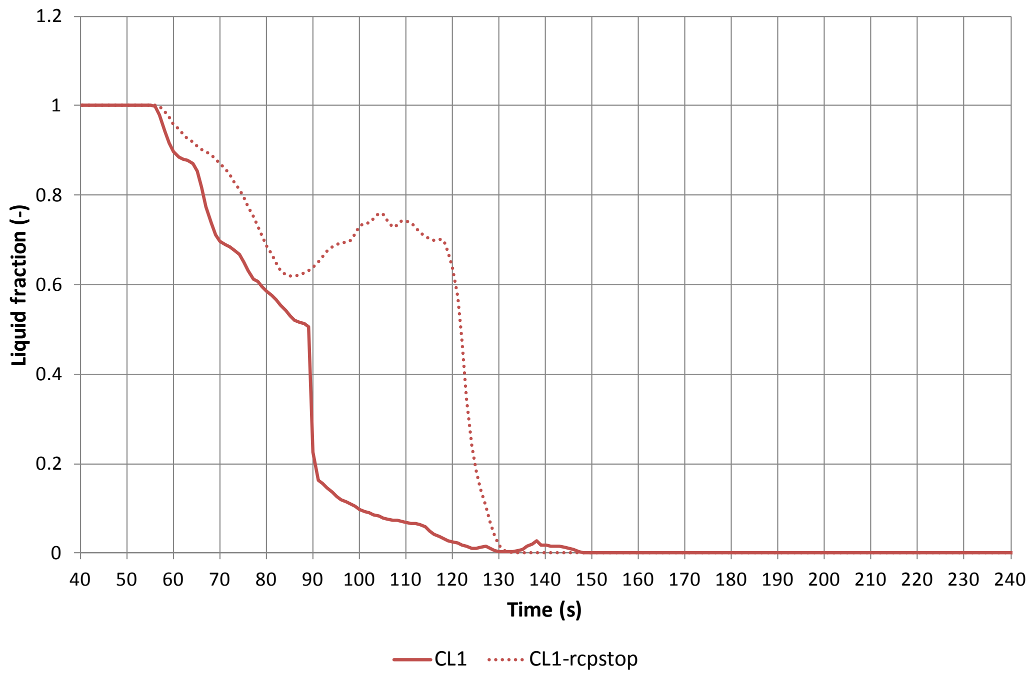

Figure 6 CL IBLOCA - Liquid Fraction in RCP Loop 2 Inlet

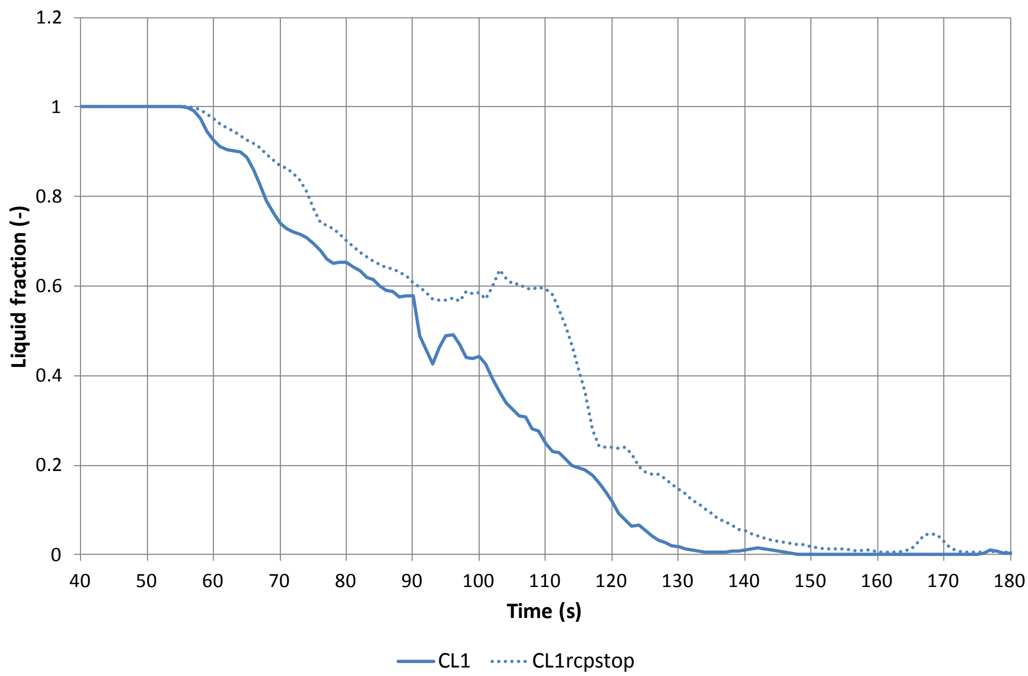

Figure 7 CL IBLOCA - Liquid Fraction of Break Mass Flow

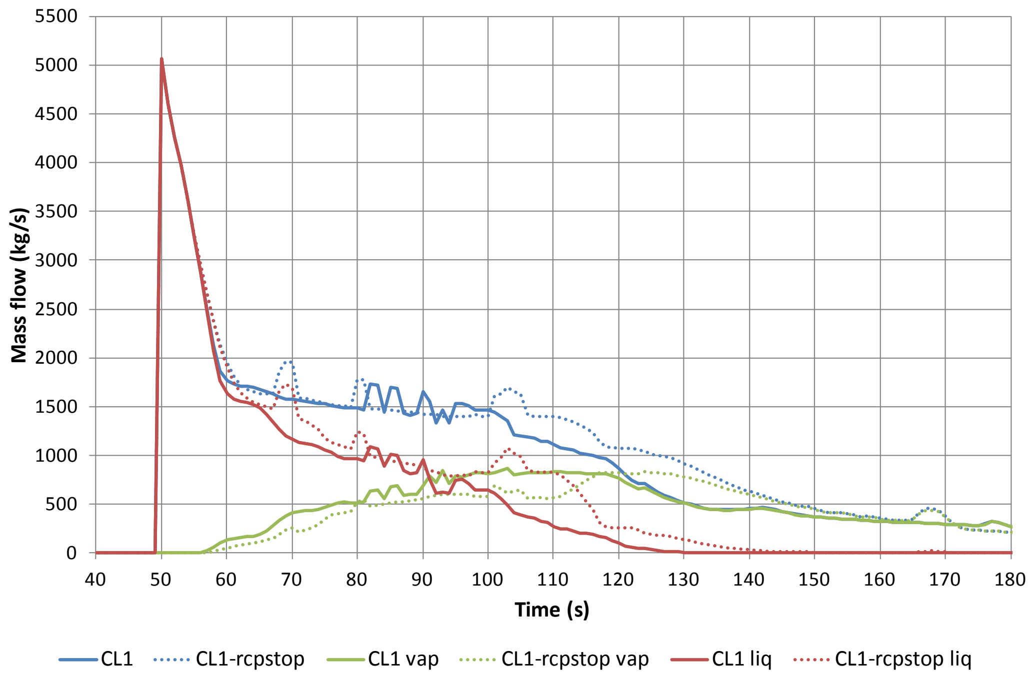

Figure 8 CL IBLOCA - Break Mass Flow

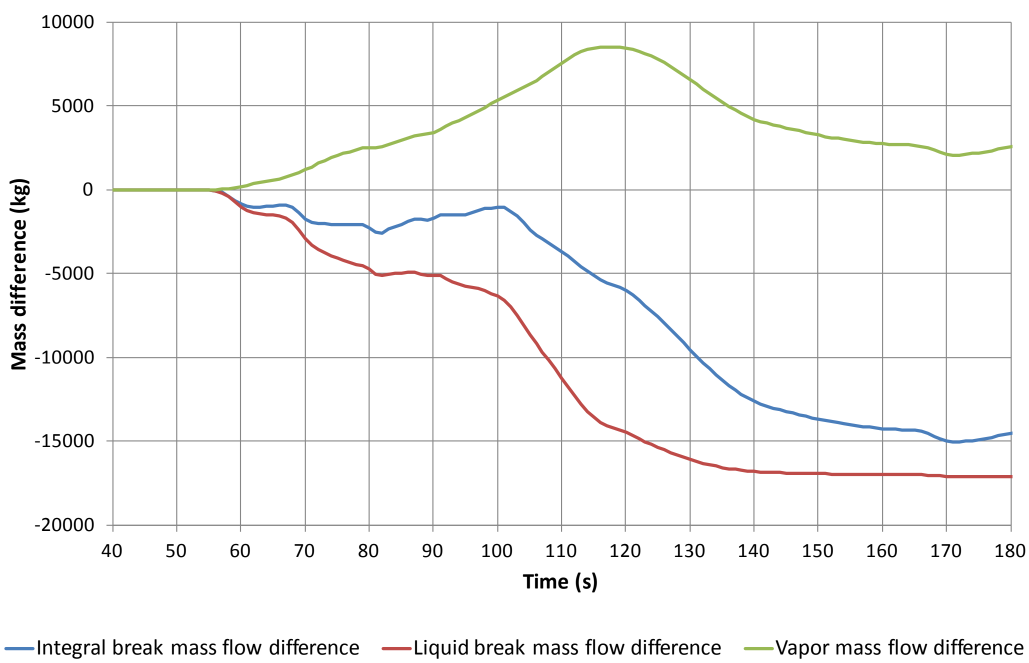

Figure 9 CL IBLOCA - Difference Break Mass (CL1 - CL1-rcpstop)

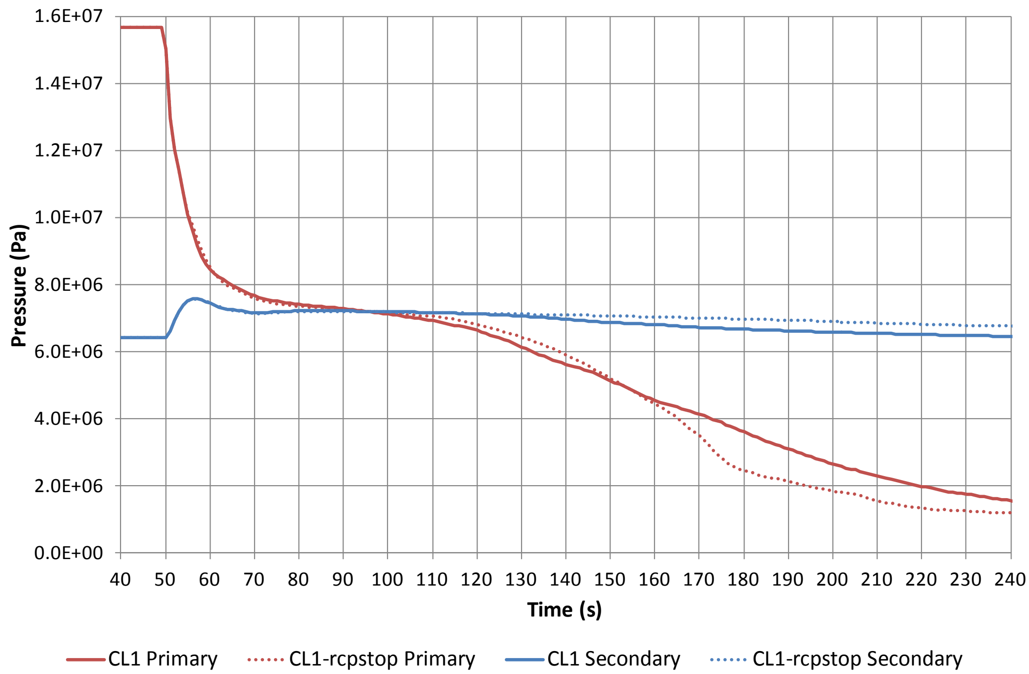

Figure 10 CL IBLOCA - Primary and Secondary Pressure

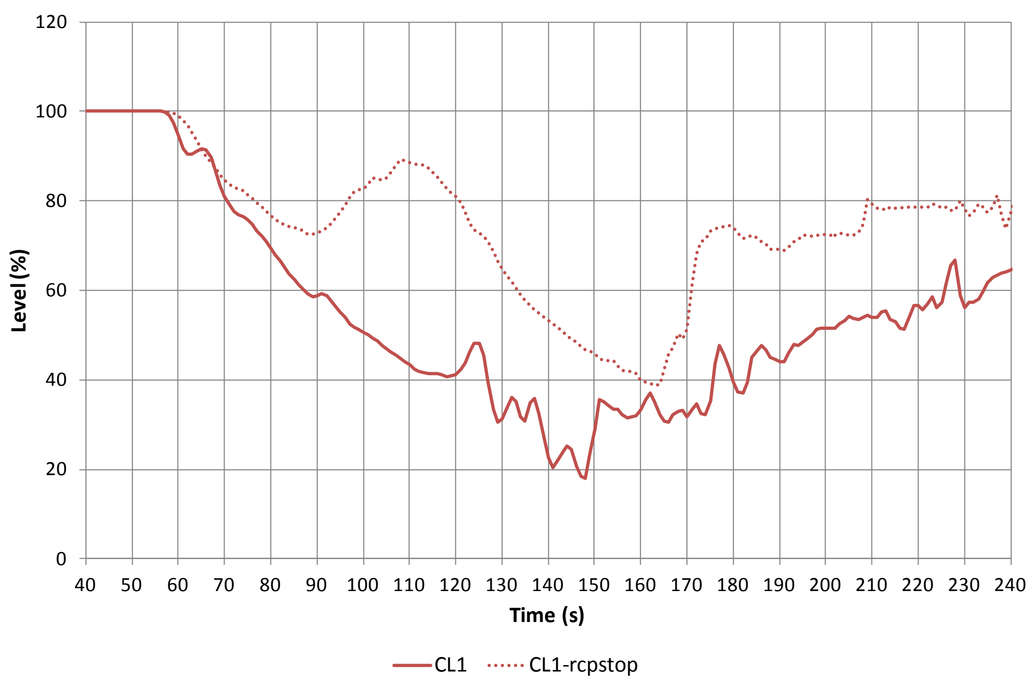

Figure 11 CL IBLOCA – Downcomer Collapsed Level

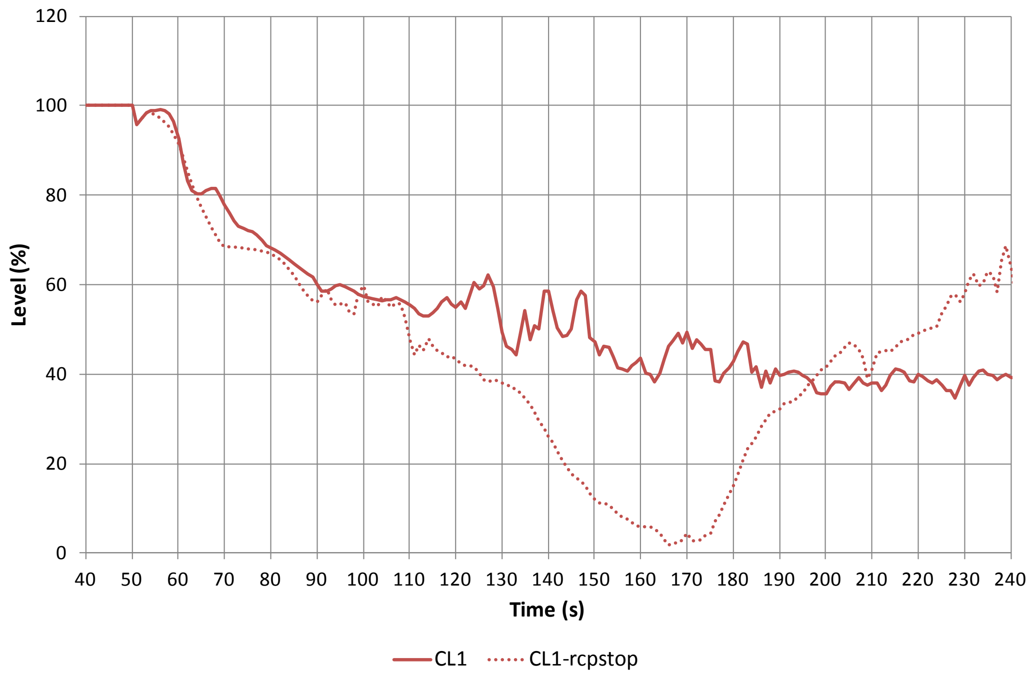

Figure 12 CL IBLOCA - Core Collapsed Level

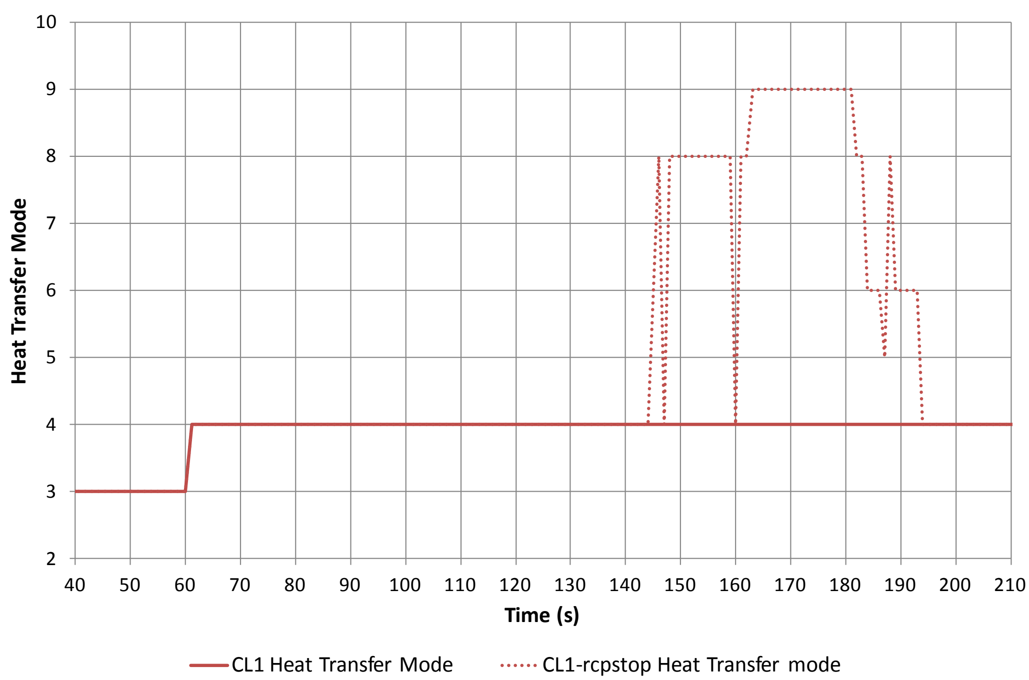

Figure 13 CL IBLOCA - Core Conditions at Mid Position of Hot Rod

6 HOT LEG CASE RESULTS

This section presents a comparison between the hot leg IBLOCAs simulating the rupture of the pressurizer surge line, defined as follows:

- HL3: RCPs are not stopped.
- HL3-rcpstop: RCPs are stopped with safety injection signal (HPIS signal)

As described in Section 0, the boundary conditions are no HPIS and no AFW.

HL3 case produces a PCT of 559K (lower than steady-state value) at 160s after break initiation; HL3-rcpstop case does not produce any PCT Figure 14).

As seen in Figure 15 both cases show a similar blowdown phase and a short quasi-equilibrium period. During the quasi-equilibrium period the downcomer and core collapsed levels (Figure 16 and Figure 17, respectively) stabilize.

In the HL3 case the downcomer level stabilizes for a short period of time and soon continues decreasing due to the forced convection. At roughly 110s, the mass flow at the outlet of the UP connected to the broken hot leg reverses (see Figure 18, and that produces a short recovery in the core level because the broken SG empties to the cold leg (as the liquid is pulled by the pump) and also to the break. After that, the core level decreases sharply as the downcomer and the cold legs are empty.

On the other hand, for the HL3-rcpstop case the LS clearing (see Figure 19) at 110s allows the liquid seal and the vapor to flow to the break, then the core collapsed level starts decreasing at a relatively smooth rate.

In addition to what has been explained above, in the HL3 case the forced circulation makes the delta pressure between the break and the core outlet larger than in the case with no RCPs (see Figure 22), and therefore the amount of coolant lost in the HL3 case is larger (Figure 20). Specifically, the amount of liquid mass and total mass lost through the break is larger in the HL3 case but the amount of vapor mass lost is larger in HL3-rcpstop (Figure 21). As a consequence in the HL3 case, the core practically empties before the water from the accumulators reaches the bottom of the core.

Figure 23 shows the heat transfer correlations used by RELAP5 (see page 111 in Volume IV of [15]) at a mid-position of the hot rod:

- Heat transfer mode 2 corresponds to single phase liquid
- Modes 3 and 4 correspond to nucleate boiling in subcooled and saturated conditions, respectively.
- Modes 5 and 6 correspond to transition boiling in subcooled and saturated conditions, respectively.
- Modes 7 and 8 correspond to film boiling in subcooled and saturated conditions, respectively.

In the HL3 case, as the core collapsed level is below 20% the cladding heat transfer degrades (heat transfer modes for saturated transition and film boiling) and the temperature slightly increases instead of cooling down, while for the HL3-rcpstop case the heat transfer mode for the hot rod remains in saturated nucleate boiling and thus there is no cladding temperature excursion. Table 4 summarizes timing of events.

Table 4 HL ILBOCA - Timing of Events

Event	Time (s) HL3 case	Time (s) HL3-rcpstop case
Break initiation	50.0	
SCRAM signal on PZR low pressure (and turbine trip signal, MFW stop)		50.0 / (50.0, 50.1)
Subcooling at core outlet ≤ 4° (volume 140) [8]	51	51
RCPs trip on safety injection signal, given subcooling ≤ 4° at core outlet		52.6
2-phase break flow	51	51
Steam dump closes	61	61
RCS (CLs) mass flow < 10%	103	84
LS clearing	-	120
Cladding temperature excursion	139	-
Accumulators injection (3/3)	146	138
PCT	160	-
Complete core quenching ($T_{clad} \leq T_{sat}+30$)	187/195	-
LPIS injection (3/3)	189	187

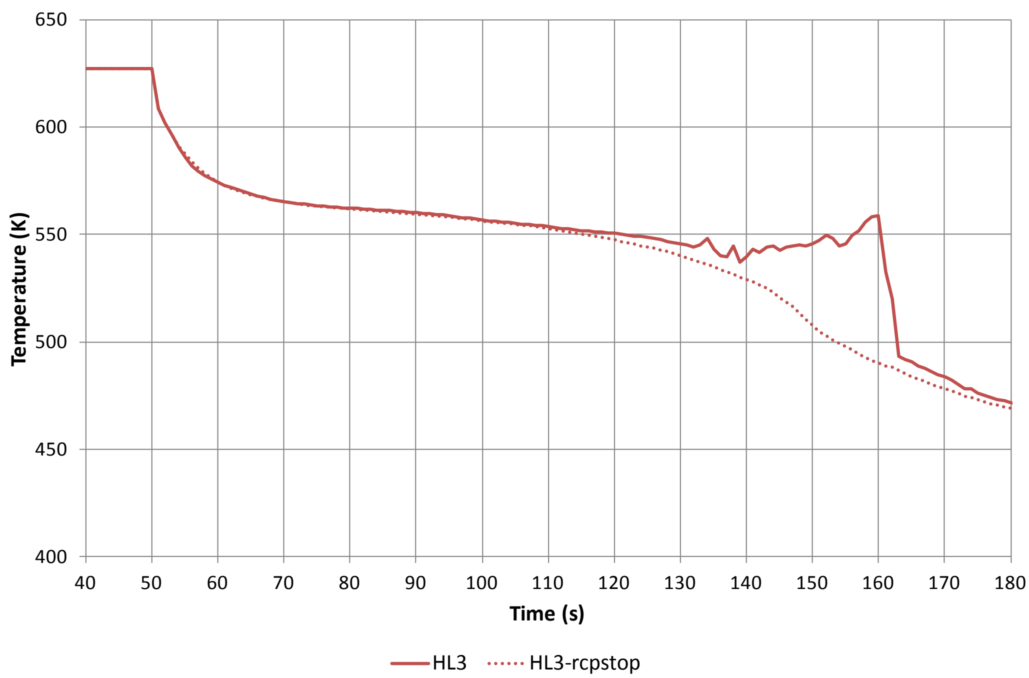

Figure 14 HL IBLOCA - Maximum Cladding Temperature

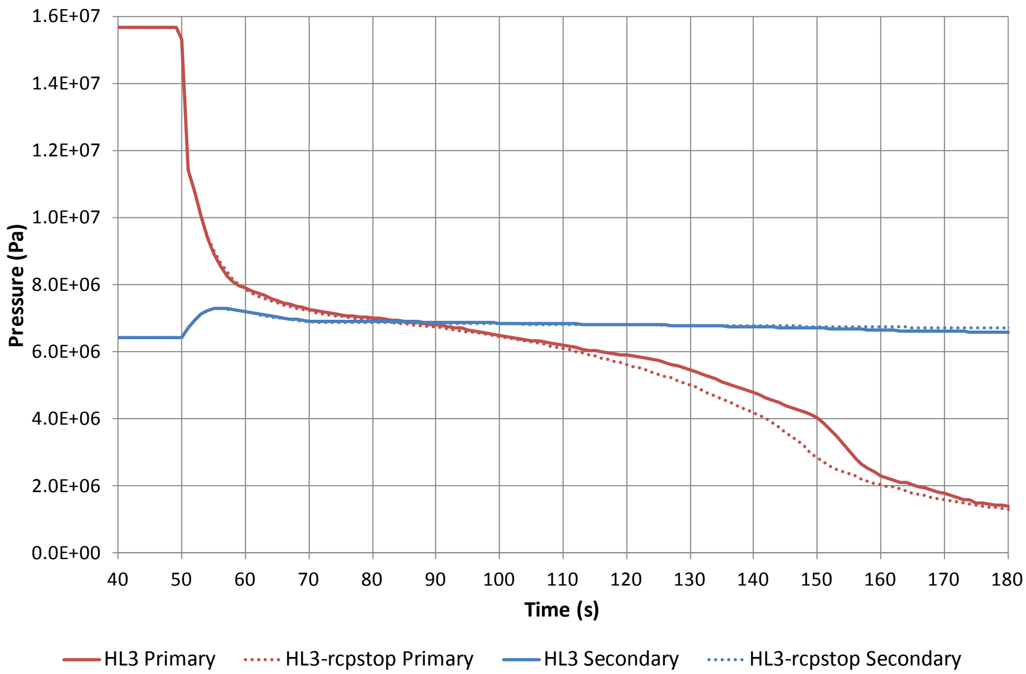

Figure 15 HL IBLOCA - Primary and Secondary Pressure

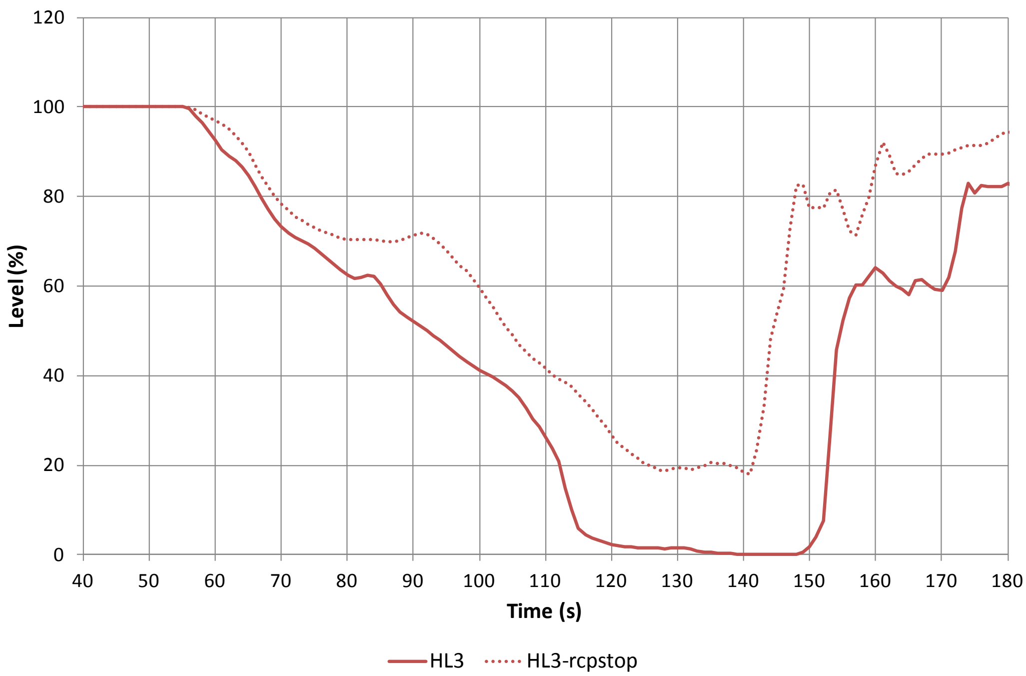

Figure 16 HL IBLOCA - Downcomer Collapsed Level

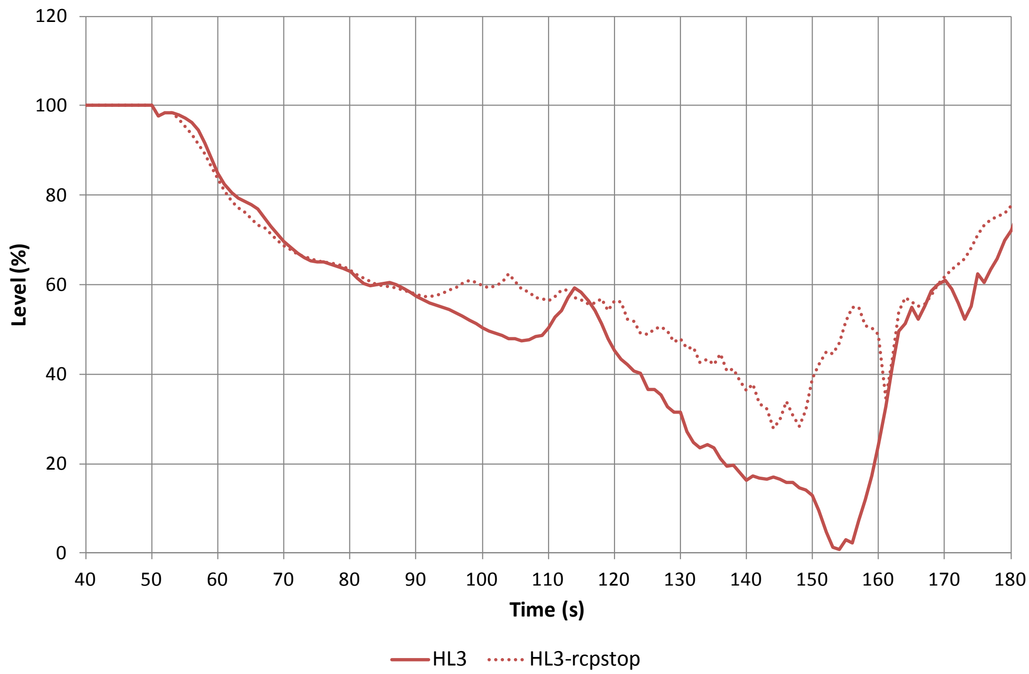

Figure 17 HL IBLOCA - Core Collapsed Level

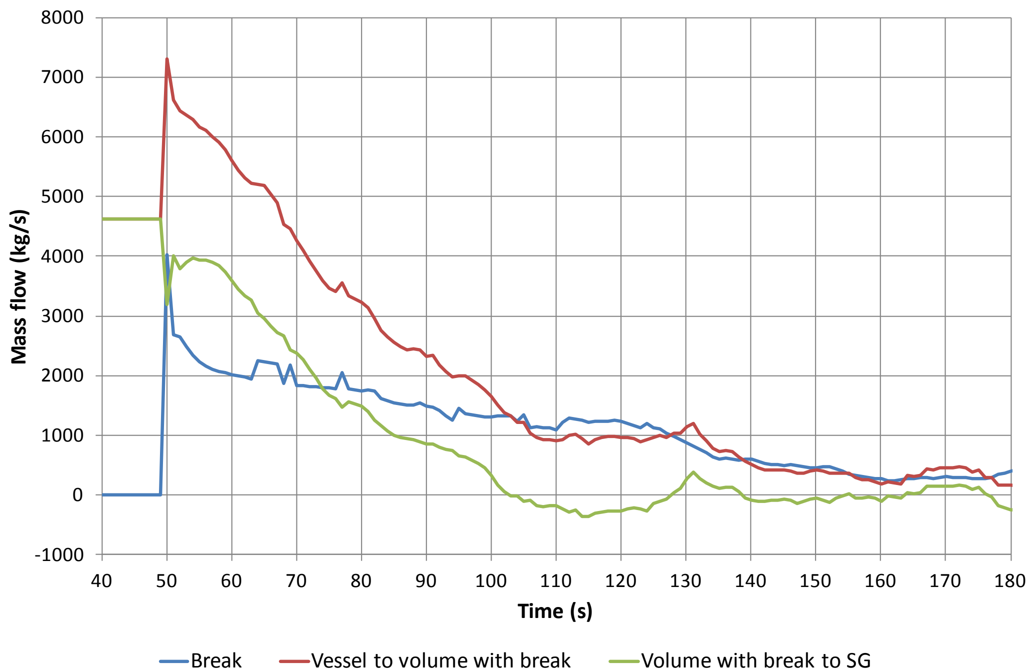

Figure 18 HL IBLOCA - HL3 Mass Flow in Broken Hot Leg

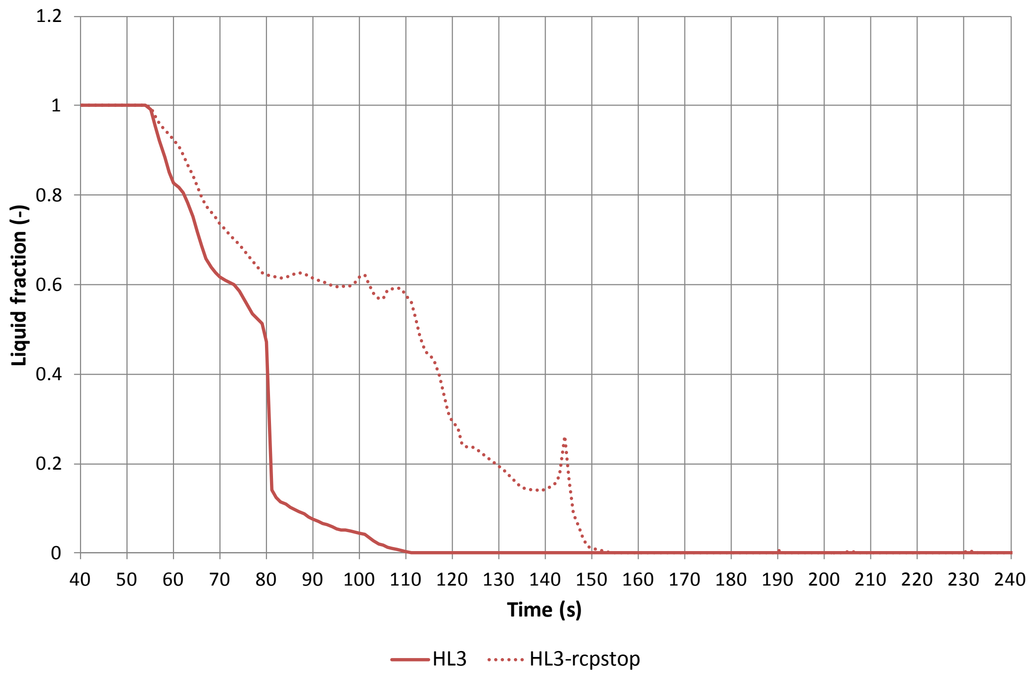

Figure 19 HL IBLOCA - Liquid Fraction in RCP Loop 2 Inlet

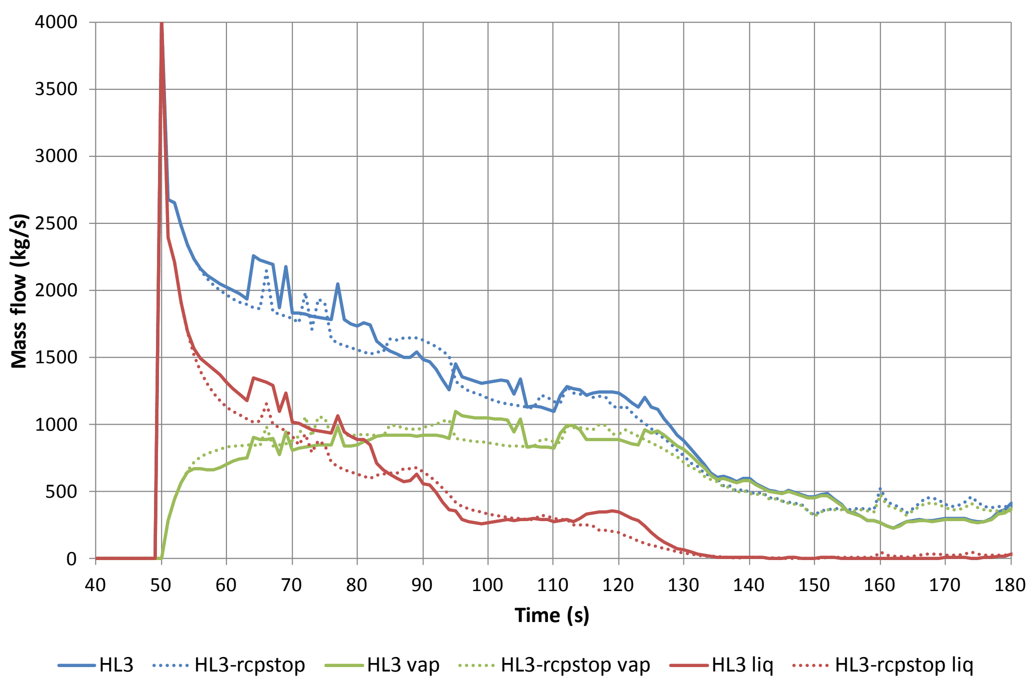

Figure 20 HL IBLOCA - Break Mass Flow

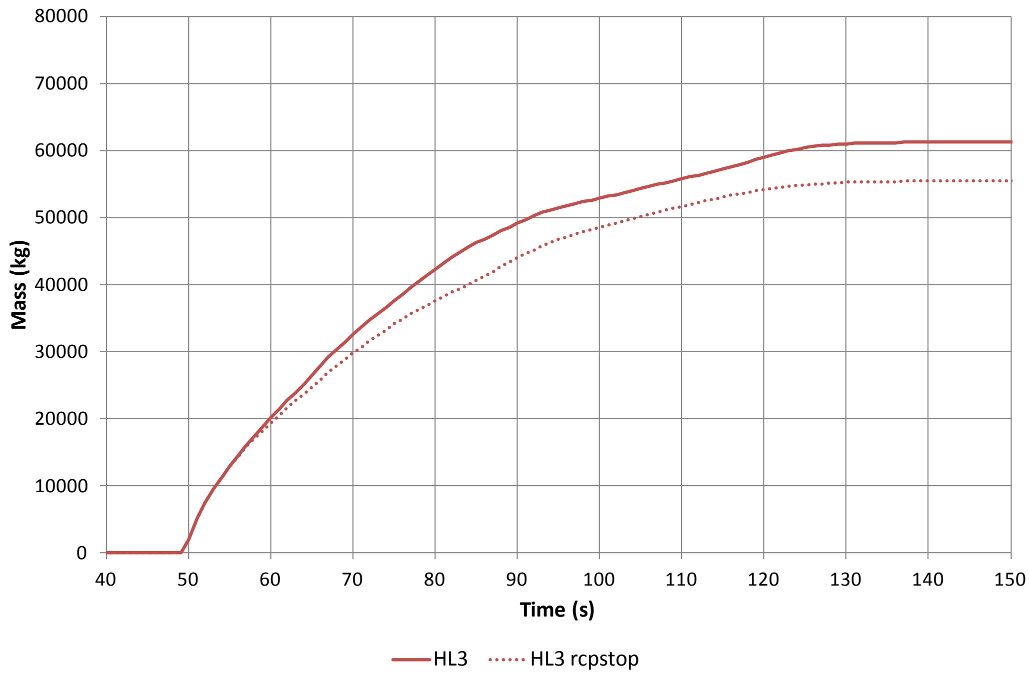

Figure 21 HL IBLOCA - Integral Liquid Mass Flow

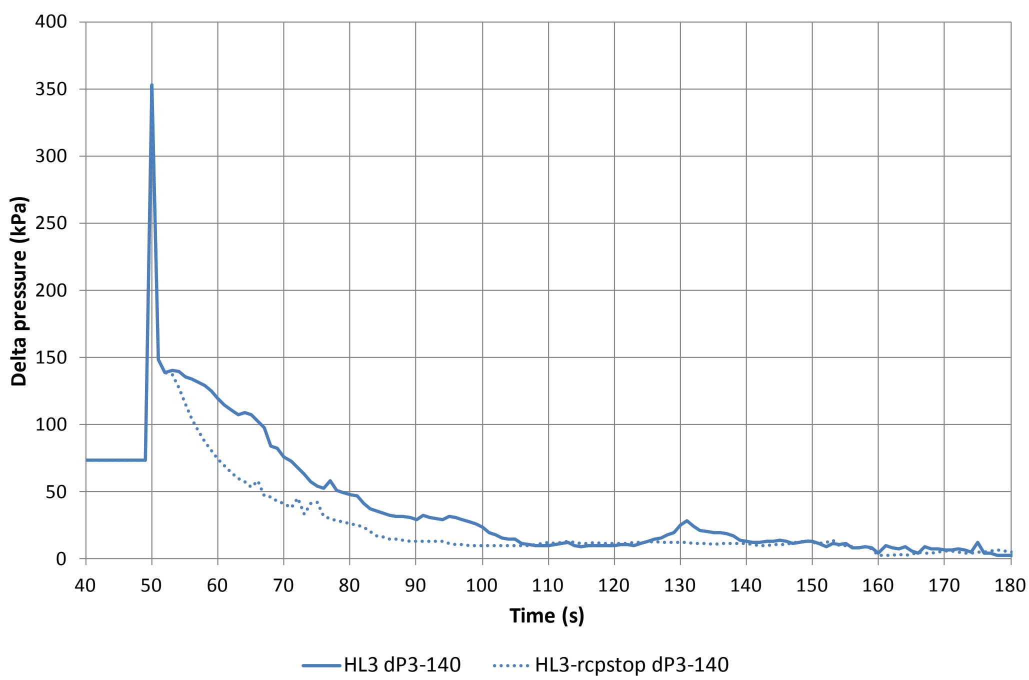

Figure 22 HL IBLOCA - Delta Pressure (Hot Leg - Upper Plenum)

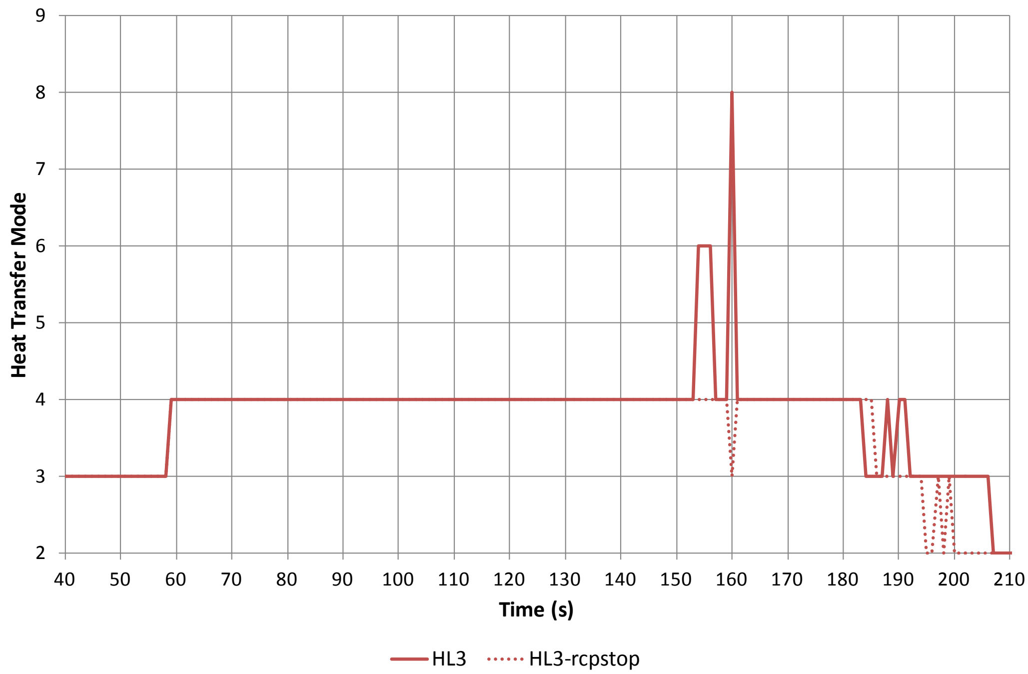

Figure 23 HL IBLOCA - Core Conditions at Mid Position of Hot Rod

7 CONCLUSIONS

This report summarizes the studies performed on the effect of the Spanish Emergency Operating Procedures (EOPs) for the Reactor Coolant Pumps (RCPs) during a hypothetical IBLOCA in Vandellòs NPP, a 3 loop Westinghouse design reactor. The Spanish EOPs to stop the RCPs require that at least one charging pump is injecting to the RCS, simultaneously with a certain degree of RCS subcooling. The first step of the work was focused on a literature review on IBLOCA experiments and simulations. The RELAP5 nodalization of the Vandellòs-II NPP has been optimized for IBLOCA scenarios following the lessons learned through the OECD/NEA ROSA-2 project and from other literature sources including internal expert judgment. The most relevant modification has been a renodalization of the core and DC regions by splitting them in parallel channels, crossflows have been included.

The transition break scenario conditions have been established by studying the pipes connected to the primary side loops. This work has been done in cooperation with engineers from ANAV and has been focused on pipe sizes, location and function. The function of the broken pipe is important because it will determine the boundary conditions of the scenario (i.e. ECC injections, PZR coolant availability). Two scenarios have been analysed and compared: full rupture of the surge line and full rupture of the accumulator line. For each IBLOCA scenario, two cases have been analyzed: one with the RCPs always running (accordingly to Spanish EOPs) and a second one where the RCPs are disconnected after the safety injection signal is triggered.

The analysis presented in the present work shows that, for the IBLOCA scenarios selected, the EOPs for the RCPs leads to safe conditions. The specific conclusion for each of the break locations are:

- In CL IBLOCA scenario, having the RCPs running in accordance with I/IOE-E-1 procedure prevents a cladding temperature excursion: the forced convection maintains core cooling from intact loops and allows vapor to exit the RCS through the break. On the other hand, if the RCPs are tripped, the cladding temperature increased up to 700 K.
- In HL IBLOCA scenario, having the RCPs running in accordance with I/IOE-E-1 procedure produces a small PCT increase: the forced convection increases the dP between the core outlet and the break, and more liquid is lost through the break. However the PCT is small (its value is below steady-state values) and the accumulator's injection is enough to mitigate the temperature increase.

All in all, it is concluded that the I/IOE-E-1 is appropriate for this particular scenario. Having the RCPs running prevents a core uncovery in the CL break location which is potentially the worst case scenario.

8 REFERENCES

1. A. Cuadra, J.L. Gago, F. Reventós; Analysis of a Main-Steam-Line Break in Ascó NPP, Nuclear Technology 146, 41-48, 2004.

2. J. Freixa, F. Reventós, C. Pretel, I. Sol; SBLOC*A w*ith Boron Dilution in Pressurized Water Reactors. Impact on Operation and Safety. Nuclear Engineering and Design 239, 749-760, 2009.

3. F. Reventós, L. Batet, C. Pretel, M. Ríos, I. Sol; Analysis of the Feed & Bleed Procedure for the Ascó NPP. First Approach Study for Operation Support. Nuclear Engineering and Design 237, 2006-2013.

4. F. Reventós, L. Batet, C. Llopis, C. Pretel, I. Sol; Thermal-hydraulic Analysis Tasks for ANAV Npps in Support of Plant Operation and Control. Science and Technology of Nuclear Installations 13, doi: 10.1155/2008/153858, Article ID 153858, 2008.

5. F. Reventós, L. Batet, C. Llopis, C. Pretel, M. Salvat, I. Sol; Advanced qualification Process of ANAV NPP Integral Dynamic Models for Supporting Plant Operation and Control. Nuclear Engineering and Design 237, 54-63, 2007.

6. M. Pérez, J. Freixa, F. Reventós. Thermal-Hydraulic Analysis of IBLOCA Scenarios at the Vandellòs NPP. ANT internal report, UPC 2013

7. NRC 10CFR Parts 50 and 52, Risk-Informed Changes to Loss-of-Coolant Accident Technical Requirements https://www.federalregister.gov/articles/2009/08/10/E9-18547/risk-informed-changes-to-loss-of-coolant-accident-technical-requirements.

8. Central Nuclear Ascó I/IOE-E-1 (in spanish); Pérdida de Refrigerante Del Reactor o Secundario.

9. Westinghouse Nuclear Española (in spanish); Descripción del SNGV de Westinghouse, ISBN: 978-84-300-8592-7,1983

10. Westinghouse; Model F Steam Generator Thermal-Hydraulic Data Report. WTP-ENG-TN-81-001

11. C.N. Vandellòs II, Logic Diagrams

12. C.N. Vandellòs II, Loop Diagrams

13. Westinghouse Energy Systems International; Precautions, Limitations and Setpoints. C.N. Vandellòs II.

14. C.N. Vandellòs II, (in Spanish) Nota de Cálculo del Modelo de C.N. Vandellòs II con RELAP5/Mod3.2

15. The RELAP5 Code Development Team, RELAP5/Mod3.3 Code Manuals, NUREG/CR-5535, 2001.

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. J. Freixa, V. Martínez-Quiroga, F. Reventós. Modelling Guidelines for CCFL Representation during IBLOCA Scenarios of PWR Ractors. NURETH-17, Xi'an, Shaanxi, China, September 3-8, 2017

18. T. Takeda, M. Suzuki, H. Asaka, and H Nakamura. Quick-look Data Report of ROSA-2/LSTF Test 2 (Cold Leg Intermediate Break LOCA IB-CL-03 in JAEA. Technical Report JAEA-Research 2012-, Japan Atomic Energy Agency, 2012.

19. T. Takeda, M. Suzuki, H. Asaka, and H Nakamura. Quick-look Data Report of Test 1, Test for Hot Leg Intermediate Break Loss of Coolant Accident with Break Size Equivalent to 17Flow Area. Technical Report JAEA-Research 2012-, Japan Atomic Energy Agency, 2012.

20. T. Takeda, M. Suzuki, H. Asaka, and H Nakamura. Quick-look Data Report of ROSA-2/LSTF Test 7 (Cold Leg Intermediate Break LOCA IB-CL-05 in JAEA). Technical Report JAEA-Research 2013-, Japan Atomic Energy Agency, 2013.

21. 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.

NRC FORM 335 (12-2010) NRCMD 3.7 U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) **NUREG/IA-0497**
2. TITLE AND SUBTITLE IBLOCA Analysis for Vandellòs-NPP Using RELAP5/MOD3.3. Sensitivity Calculations to EOP Actions	3. DATE REPORT PUBLISHED MONTH: **February** YEAR: **2019** 4. FIN OR GRANT NUMBER
5. AUTHOR(S) M. Pérez-Ferrgut, J. Freixa, F. Reventós and I. Sol	6. TYPE OF REPORT Technical 7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Department of Physics
Universitat Politècnica de Catalunya
ETSEIB, Av. Diagonal 647, Pav. C
08028 Barcelona, Spain

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)

Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

10. SUPPLEMENTARY NOTES

K. Tien, Project Manager

11. ABSTRACT (200 words or less)

This report analyzes the effect of the plant specific EOPs for the RCPs during a hypothetical IBLOCA in Vandellòs II NPP, a 3 loop Westinghouse design reactor.

The scenarios simulated are:
- Cold leg IBLOCA simulating the rupture of an accumulators connection to the RCS;
- Hot leg IBLOCA simulating the rupture of the pressurizer surge line connection to the RCS.

The selection of an intermediate break size conforms to the risk-informed approach for the assessment of the ECCS performance.

The RELAP5/MOD3.3 model of Vandellòs II NPP was developed by the Advanced Nuclear Technologies (ANT) group at the Universitat Politècnica de Catalunya (UPC).

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT
Asociación Nuclear Ascó-Vandellòs (ANAV) Committee on the Safety of nuclear Installations (CSNI) Consejo de Seguridad Nuclear (Spanish Nuclear Regulatory Commission CSN) Thermal-Hydraulics Studies Group UPC (GET) Severe Accident Management Guideline/Guidance (SAMG) Asociación Española de la Industria Eléctrica (UNESA) Intermediate Break Loss-Of-Coolant-Accident (IBLOCA)	unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE

NUREG/IA-0497 IBLOCA Analysis for Vandellòs-NPP Using RELAP5/MOD3.3. Sensitivity Calculations to EOP Actions February 2019