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Sensitivity Analyses of a Hypothetical 6 Inch Break, LOCA in Ascó NPP using RELAP/MOD3.2

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ABSTRACT

A postulated 6 inch break Loss-of-Coolant Accident (LOCA) is analyzed for the Ascó I plant (a 3 loop PWR Westinghouse design) using RELAP5/Mod3.2. Different scenarios are calculated, including the total loss of the High Pressure Injection System (HPIS). In this case, the maximum cladding temperature rises above the steady state value at full power but, before the end of the transient, temperatures return to normal values by the effect of the accumulators injection.

Passive heat structures in the reactor pressure vessel have been incorporated to the model (i.e. the pressure vessel walls and some internals). Calculations with the model including passive heat structures lead to more severe scenarios (i.e. higher cladding temperatures) but still accumulators action is able to return the temperatures to normal values.

Finally, sensitivity to the trip time of the reactor coolant pumps is analyzed. Results show that lower cladding temperatures result when the pump trip is delayed.

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 code.

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.

UNESA
December 2009

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EXECUTIVE SUMMARY

The Department of Physics and Nuclear Engineering of the Technical University of Catalonia (UPC) holds a large background in the use of thermal-hydraulic codes for the Safety Analysis of Nuclear Power Plants (NPP).

A postulated 6 inch break Loss-of-Coolant Accident (LOCA) is analyzed for the Ascó I plant (a 3 loop PWR Westinghouse design). Different scenarios, including the total loss of the High Pressure Injection System (HPIS) are calculated.

A first set of calculations using the present RELAP5 model of the plant has been performed. In these calculations the effect of the HPIS actuation has been investigated. In addition to the Base Case (two out of two HPIS trains on duty), two alternative scenarios have been analyzed involving the failure of one and two HPIS trains. It is stated that in the Base Case the scenario does not represent any problem for the safety of the plant. In case of failure of one HPIS train, the cladding temperature of the upper part of the fuel elements may rise slightly but soon it is returned to the safety values by the action of the accumulators. In the case of malfunction of both HPIS trains, the cladding temperature rises above the steady state value but the injection from the accumulators returns the critical variables to normal values before the end of the transient.

Next step was a sensitivity analysis to assess the importance of including passive heat structures in the plant model. In order to do so, some vessel heat structures have been incorporated to the model (i.e. the pressure vessel walls and some internals). A new analysis was performed, comparing the worst case of the scenarios studied (complete loss of the HPIS trains) with and without passive heat structures. It is demonstrated that the inclusion of these passive heat structures leads to a more severe (i.e. higher cladding temperatures) but still accumulators action is able to return the temperatures to normal values before the end of the time span considered for the transient.

Finally, the sensitivity to the trip time of the reactor coolant pumps is analyzed. The worst case of the scenarios studied (complete loss of the HPIS trains) with passive heat structures is recalculated with varying reactor coolant pumps stop times. It is found that the more the pump trip is delayed the temperature excursions are milder and, eventually, disappear.

ABBREVIATIONS

AFW	Auxiliary Feed Water
ANAV	Asociación Nuclear Ascó-Vandellòs
AS1	Alternative Scenario 1
AS2	Alternative Scenario 2
BC	Base Case
EOP	Emergency Operation Procedure
HS	Heat Structures
HPIS	High Pressure Injection System
HRCT	Hot Rod Cladding Temperature
INL	Idaho National Laboratory
LOCA	Loss of Coolant Accident
MOL	Middle-of-Life
NPP	Nuclear Power Plant
PCT	Peak Cladding Temperature
RCP	Reactor Coolant Pumps
RELAP	Reactor Excursion and Leak Analysis Program
SG	Steam Generator
UPC	Universitat Politècnica de Catalunya (Technical University of Catalonia)
UNESA	Asociación Española de la Industria Eléctrica (Association of the Spanish Utilities)

ACKNOWLEDGEMENTS

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).

1 INTRODUCTION

The Department of Physics and Nuclear Engineering of the Technical University of Catalonia (UPC) holds a large background in the use of thermal-hydraulic codes for the Safety Analysis of Nuclear Power Plants (NPP). More precisely, the Department's Thermal-Hydraulics Studies Group has been cooperating for 15 years with the operators of the Catalan nuclear plants, Ascó (two units) and Vandellós II [1].

In this document, a postulated medium size break Loss-of-Coolant Accident (LOCA) is analyzed for the Ascó I plant (a three loop PWR Westinghouse design). The selected transient consists in a 6 inch break in the cold leg, in different scenarios, including the total loss of the High Pressure Injection System (HPIS) flow (Sections five and six).

There are several previous studies of LOCA of this size or similar for the Ascó NPP. Nevertheless, the most comprehensive compendium is quite old (prior to the change of steam generators).

2 PLANT 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 IBERDORLA (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 and 1028 MWe. The reactor vessel is cold head type. 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. The circulation ratio on the secondary side of the steam generators is 3.65 at rated power. The auxiliary feed water system is pumped by one turbo pump and two motor pumps. In the plant there are, among others, control systems for the reactivity (rods and boron), primary pressure, pressurizer level, steam dump and steam generator level. The reactor protection system includes safety valves in the pressurizer and the steam generator.

2.1 Reactor pressure vessel

The vessel component of Ascó NPP corresponds to PWR three-loop Westinghouse nuclear power plant design. It has a cylindrical shape and the bottom is ended with a semi-spherical shape. The vessel contains the core, its support structure, control rods and also the thermal shield and other items directly associated with the core.

The vessel is designed and manufactured in accordance with the requirements of ASME Code concerning nuclear equipment. The material of construction of the vessel is low carbon steel alloy. The internal surfaces in contact with the cooling water, in order to minimize corrosion, are coated with a layer of austenitic stainless steel, deposited by welding.

Table 1: 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

Table 1: Main features of Ascó NPP. (continued)

Pressurizer volume (m^3)	39.65
Pressurizer heaters power (kW)	1400

2.2 Core

Ascó NPP core consists of 157 fuel assemblies with varying degrees of enrichment and burn-up. Each fuel element contains 264 fuel rods, with a clad of Zircaloy-4 which contains in its internal UO2 pellets. The rod distribution matrix is 17x17. The number of control rods is 48. In every reload operation one third of the assemblies are changed.

2.3 Pressurizer

Pressurizer is standard Westinghouse design currently used in approximately 70 operating plants worldwide. The pressurizer volume is 1400 cubic feet, 39.64 m^3. The heaters are vertically mounted, extending up through penetrations in the bottom head of the pressurizer shell. They are also individually seal-welded to the penetrations providing the system pressure boundary. The pressurizer heaters are one of the many components that have achieved very good overall performance in operating plants, they have a power about 377 kW, also there are extra heaters with a power about 1023 kW. Normal operating pressure is 15.51 MPa and normal operating temperature is 618 K. The maximum allowable pressure is 17.13 MPa, and the maximum allowable temperature is 633 K.

2.4 Steam Generators

Original Westinghouse Steam Generators, were replaced by the present Siemens Steam Generators. The total primary side volume of the Steam Generators is 32.936 m^3. The secondary side of the steam generators consists of a casing that acts as pressure barrier around the bundle of tubes and an upper zone that houses separators of steam moisture. The heat transferred by the reactor cooling water, through U-tubes raises the temperature of secondary until saturation, producing steam with high humidity content. Later this steam passes through a series of separators and dryers that remove virtually all water in the flow of steam toward the turbine. The water separated from the steam by separators, is mixed with main feed water.

2.5 Reactor coolant pumps

There are three reactor coolant pumps, one for each loop. Pumps model is W-1 1011-A1 (93-DS). RCP is a vertical single-stage, centrifugal pump designed to move large volumes of reactor coolant water at elevated temperatures and pressures. Each pump consists of three general areas: the hydraulics, the seals and the motor. Pumps pressure is 8.6285 kg/cm^2.

2.6 Turbine system

Turbine system is composed of two parts. In one side there is a one-body high pressure turbine, Westinghouse design. On the other side there is the low pressure turbine system, composed by two turbine modules, also Westinghouse design. They have a rotation speed of 1500 rpm.

3 MODEL OF THE PLANT

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 [2].

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 prepared for RELAP5/MOD3.2 and has been subjected to a thoroughly validation and qualification process, which includes the simulation of transients occurred in the plant itself [3],[4].

Figure one shows a general view of the hydrodynamic part of the model. The nodalization diagram for the reactor pressure vessel (RPV) is sketched in Figure 2 two, whereas Figure three reproduces the nodalization scheme used for the steam generators.

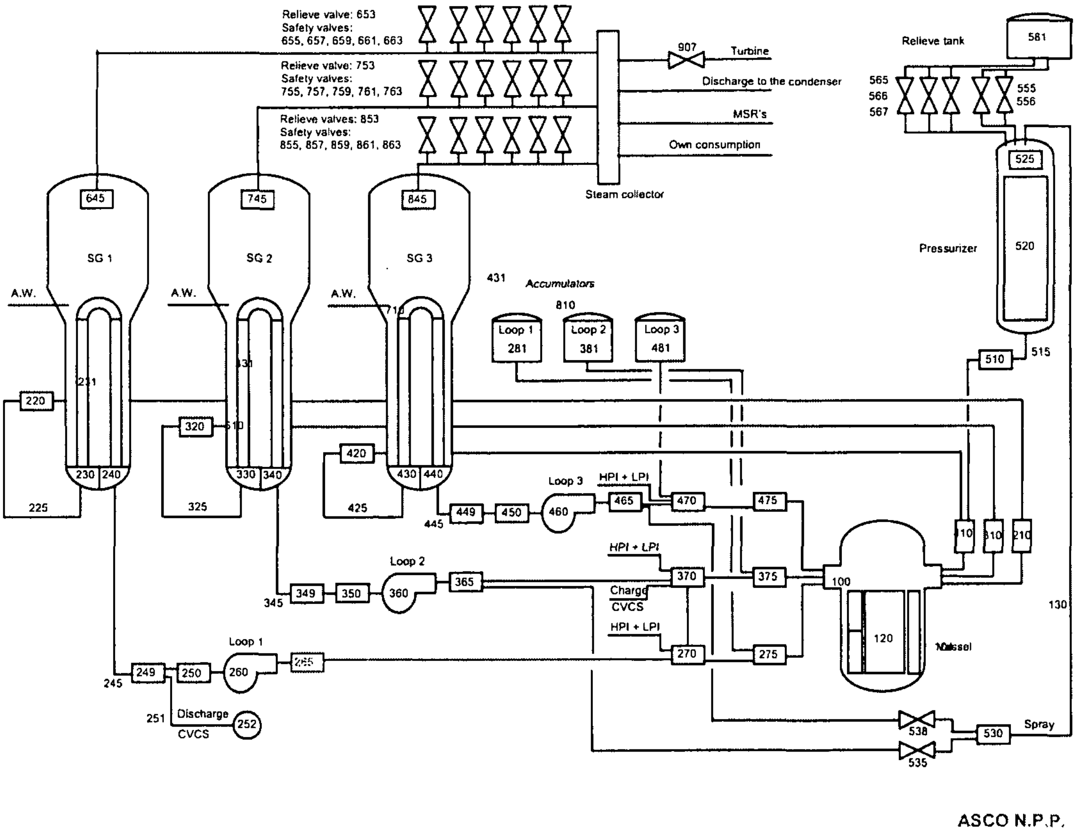


Figure 1. Diagram of the Ascó model for RELAP5.

As can be observed in figure one, High and Low Pressure Safety Injection Systems are connected to the primary system in the volumes numbered 270, 370 and 470. The modeling of the primary system safety injections is drawn in Figure 4, in which the two HIPS trains are indicated in red color.

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.

Table two, 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, besides, it incorporates some signals simulating operators' actions.

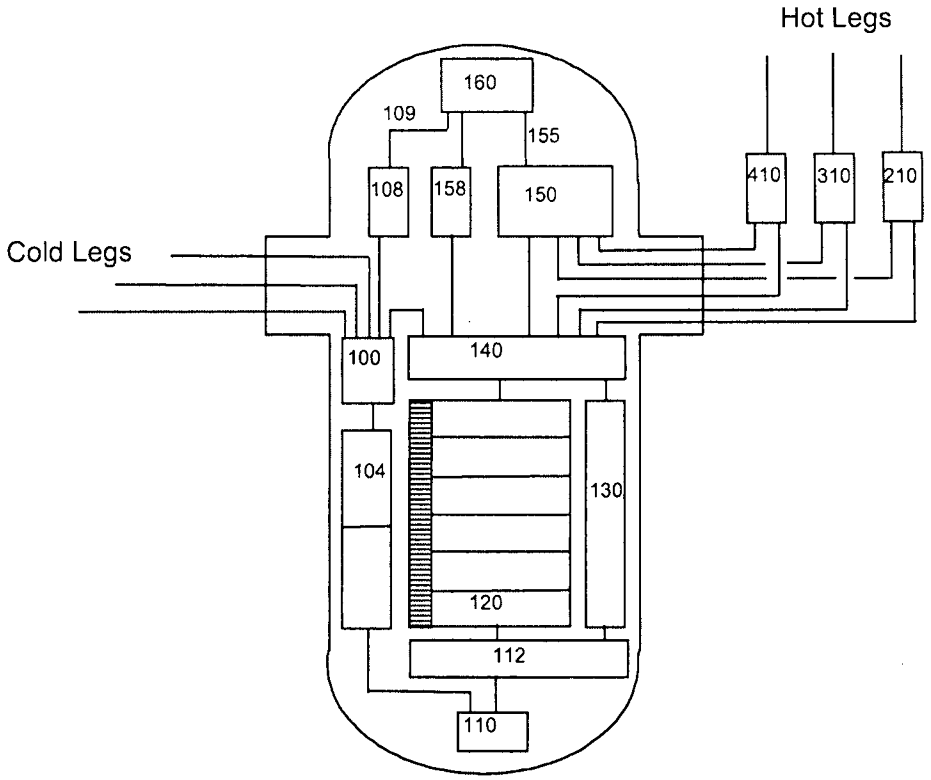
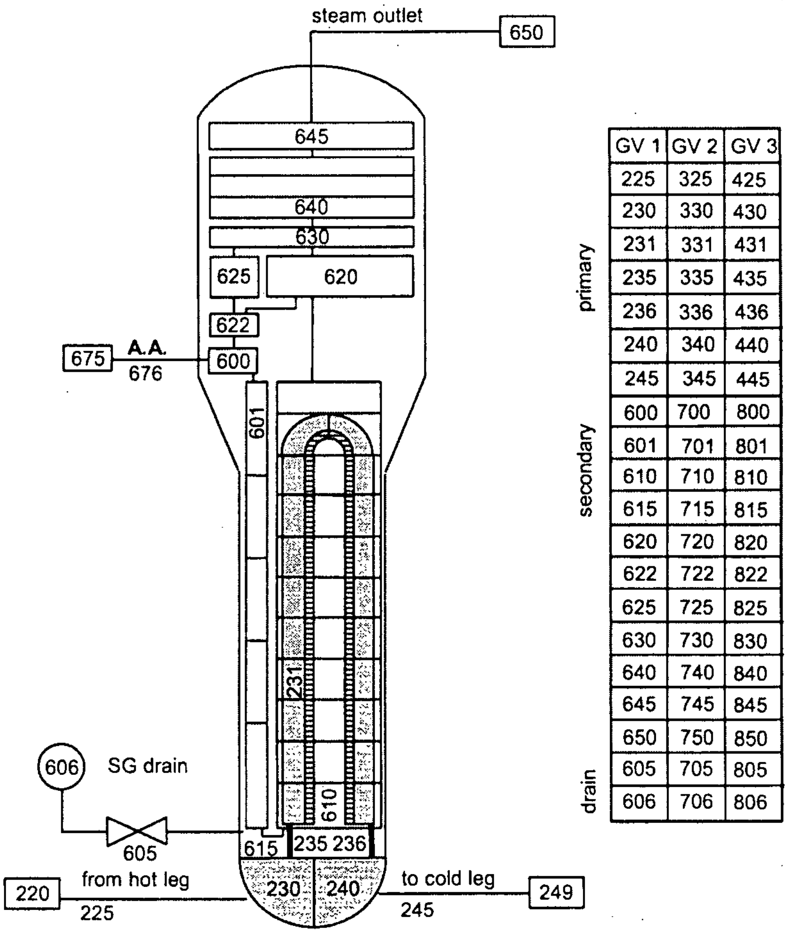


Figure 2. RPV nodalization scheme.

Table 2: Model Nodalization main statistics.

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 3. Steam Generators nodalization scheme.

The original model incorporated heat structures on the following components:

- Steam Generators.
- Core.
- Pressurizer.
- Feed water heaters.

It is known that in this kind of scenarios (i.e. medium sized LOCA) the heat stored in passive structures has a noticeable influence in the results. In order to assess this premise, the calculation for the worst case of the analyzed (failure of both trains of the HPIS) is repeated in section 7, where passive heat structures are included in the model, and the results are compared.

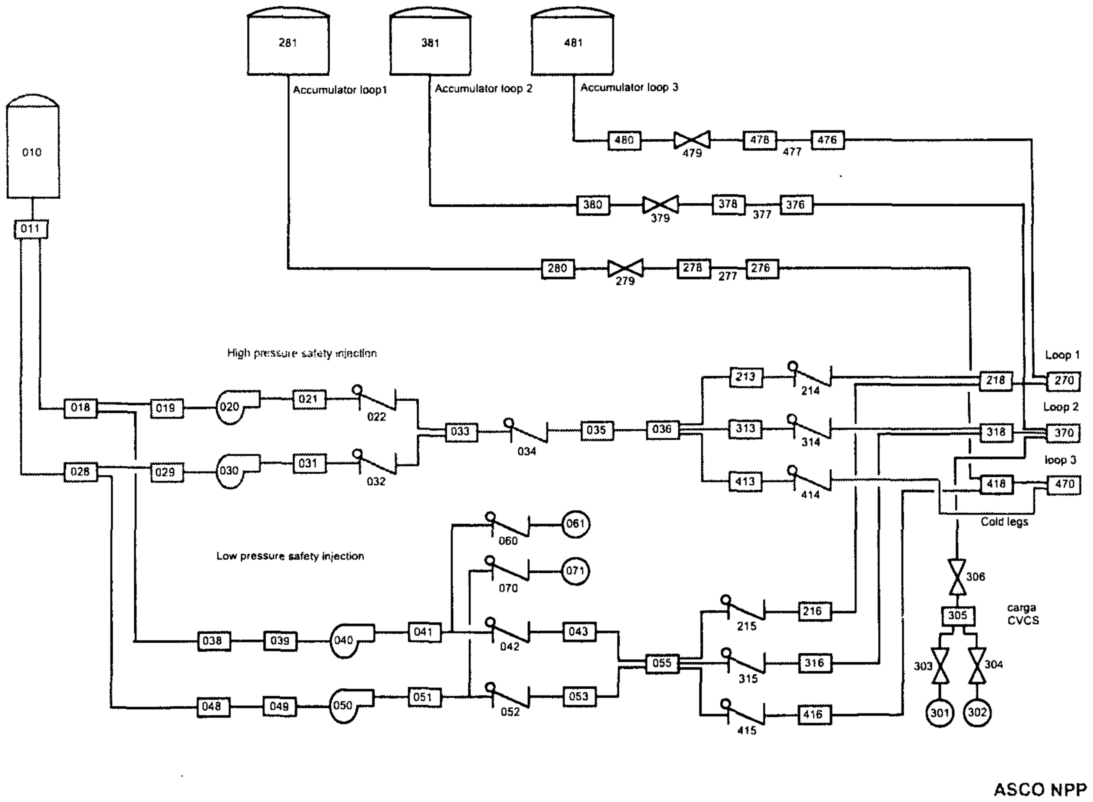


Figure 4. General safety injections systems nodalization.

4 SCENARIO

As it has been introduced above, the transient analyzed in this report is a postulated 6 inch break Loss-Of Coolant Accident (LOCA) in Ascó-I NPP. It is assumed that the plant is operating at full power when the break occurs. The simulation has been carried out with the middle-of-life (MOL) conditions of the reload cycle 17.

The break is simulated as a sudden rupture at 50 seconds into the transient calculation. Immediately, reactor, turbine and the reactor coolant pumps trip.

The LOCA is postulated to take place in the cold leg, down-flow the reactor coolant pump (RCP) in loop one (the red colored square in Figure one indicates the location of the break). In figure five, more details of the simulation of the break are shown. The arrow indicates the place where the break is located. Volume 265 in figure five corresponds to the first pipe segment down-flow the RCP and up-flow the Safety Injection point (all distances in millimeters). A new volume, numbered 291 and shown in figure five, has been added to the model to simulate the containment.

In order to run the transient, a previous steady state calculation has been necessary. It is worth to mention here that the only boundary conditions imposed on the steady state are, on the one hand, the characteristics of the fuel and the boron concentration (as said, Ascó I, Cycle 17, MOL), and in the other hand, the turbine admission pressure in the secondary. So, the model is allowed to run limited only by its own controls (which simulate the actual plant ones). In this case, the steady state calculation has been run for 1200 seconds.

For the transient, besides the break conditions (which is the only condition for the Base Case), some other restrictions have been imposed: for the Alternative Scenarios one and two one and two (of two) trains of the HPIS (depicted in red in figure four) have been considered non-available.

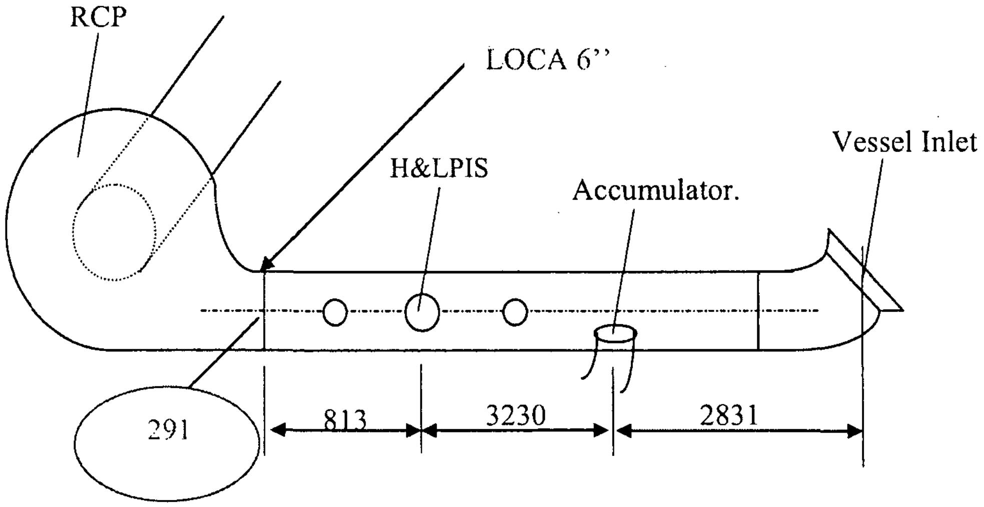


Figure 5. LOCA position scheme

5 BASE CASE RESULTS

In this section, the results for the Base Case simulation are presented (i.e. all the injection systems are in function). After the 6 inch break (at 50 s) takes place, a fast primary depressurization occurs and soon primary system decouples from the secondary one (figure six). Turbine and Reactor trip (figure seven), and also do RCPs (figure eight).

The Auxiliary Feed Water (AFW) enters almost immediately (5.1 seconds after the break) into function (figure nine) but there still exists an increase in the secondary pressure that forces the actuation of the Steam Dump (Figure 10). In figure nine it can be observed that AFW is far greater in SG2 than in the other two SG, this is due to the fact that AFW of this loop is driven by a turbopump while other loops use a motorpump.

The RCP coast down causes a decrease in the core mass flow to a value near to zero (Figure 11). The oscillations around 450 seconds may be due to accumulator injection. See Figure 14 for accumulator injection.

The system depressurization is driven by the break flow. Figure 12 shows the level in the pressurizer.

Soon after the break, when the pressure reaches the set point, the HPIS starts injecting cold water in the three loops (Figure 14 and Figure 15).

In Figure 13 it can be observed that in the Base Case no core uncovering occurs during the transient (collapsed level never gets lower than 3.7 m, which represents the Top of Active Fuel). Level recovery can be explained by comparing safety injection (accumulators plus HPIS) and discharge mass flows (Figure 14 and Figure 15).

Finally, the hot rod cladding temperature in the six core axial nodes is depicted in Figure 16. As can be seen in this plot, there is no rise of temperatures during the transient. This behavior will not be encountered in the alternative scenarios.

Table 3: Base Case chronology.

Event	**Time (s)**
Break	50
RCP stop	50
Power stop	50
Turbine stop	50
Accumulator injection	330
HPIS injection	62
Hot channel peak cladding temperature, Node 5	none
Hot channel peak cladding temperature, Node 6	none

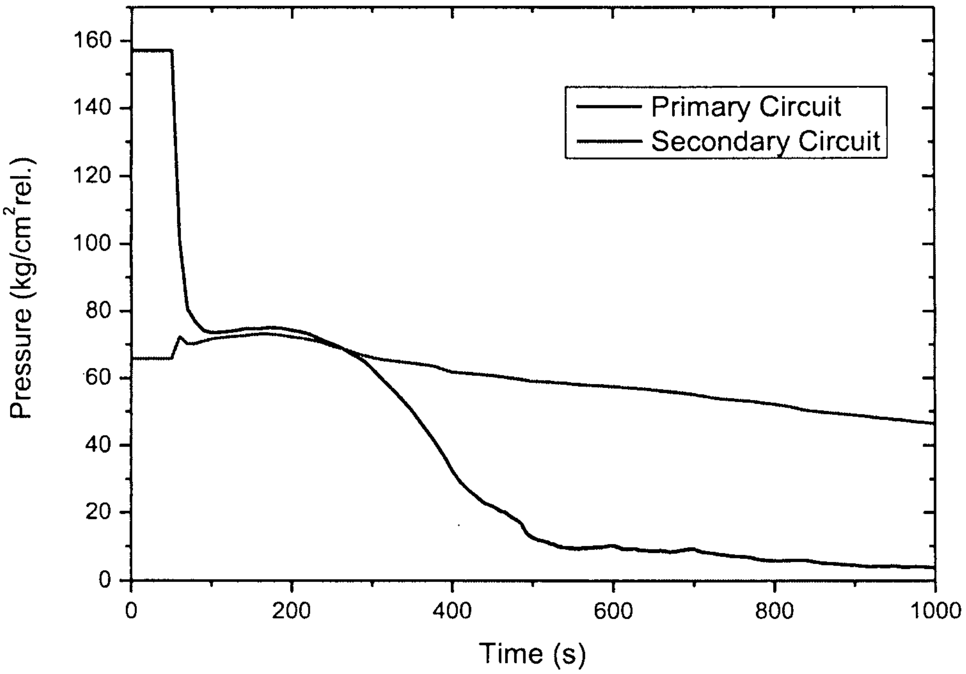


Figure 6. Base Case. Pressure.

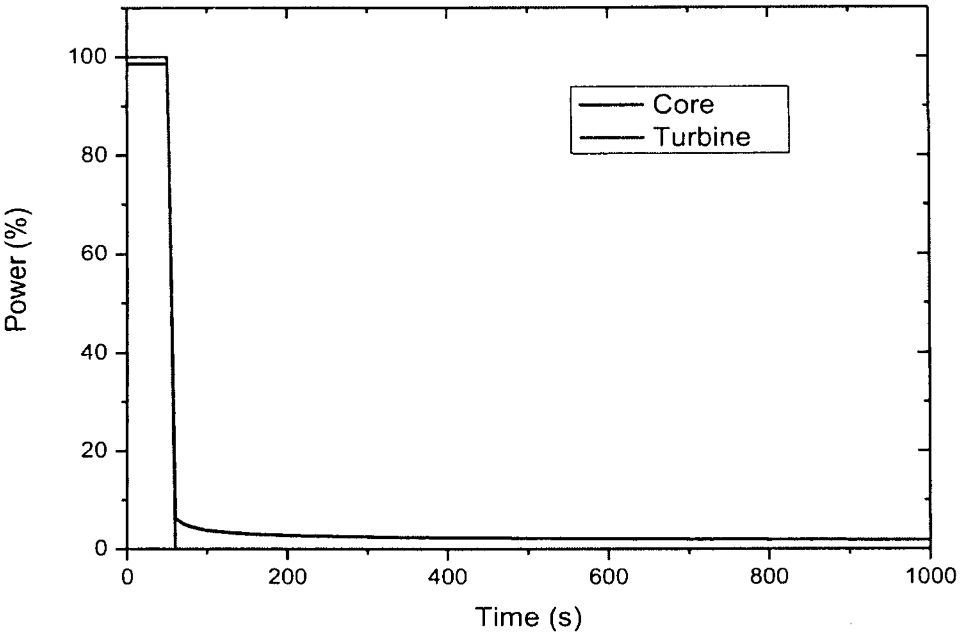


Figure 7. Base Case. Nuclear and Turbine Power.

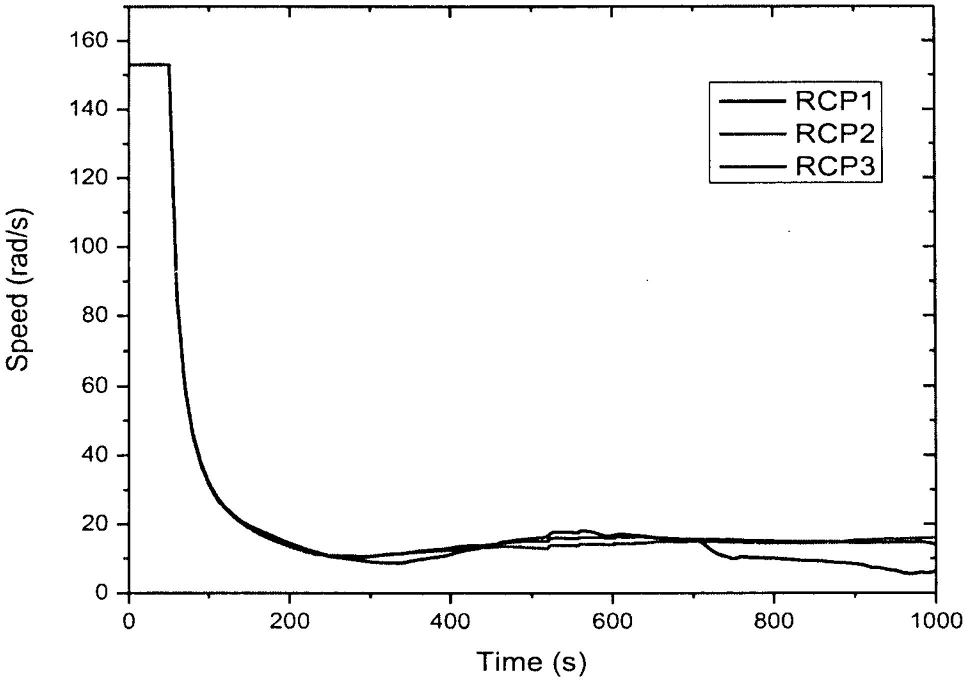


Figure 8. Base Case. Reactor Coolant Pumps Speed.

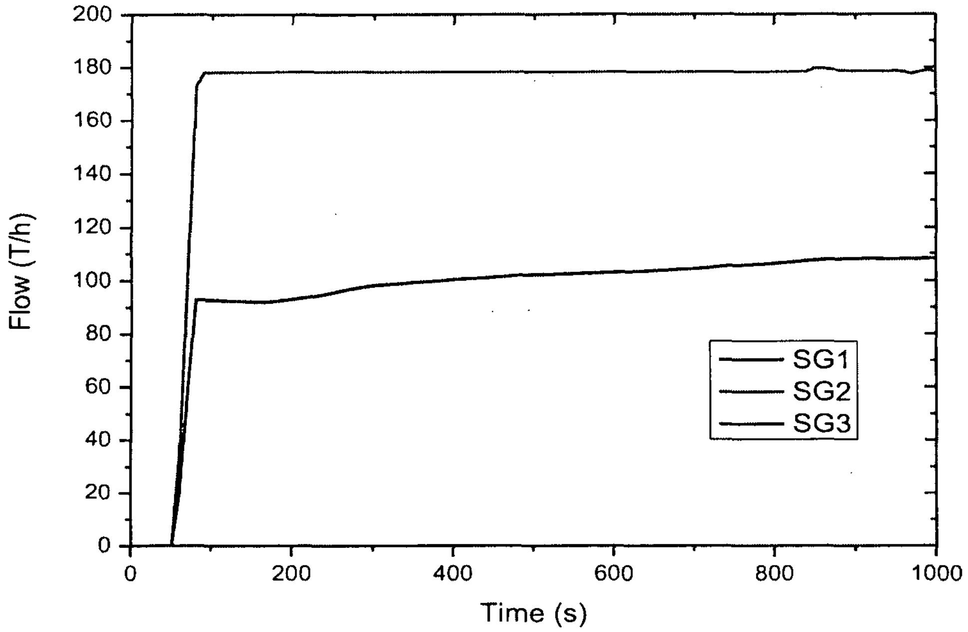


Figure 9. Base Case. Steam Generators AFW Flow.

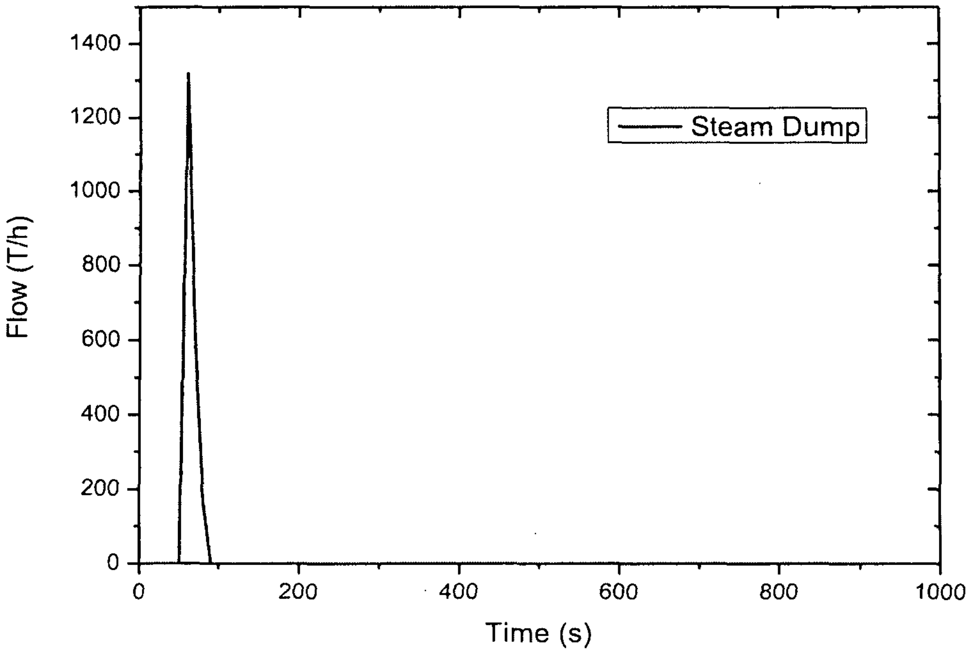


Figure 10. Base Case. Steam Dump Mass Flow.

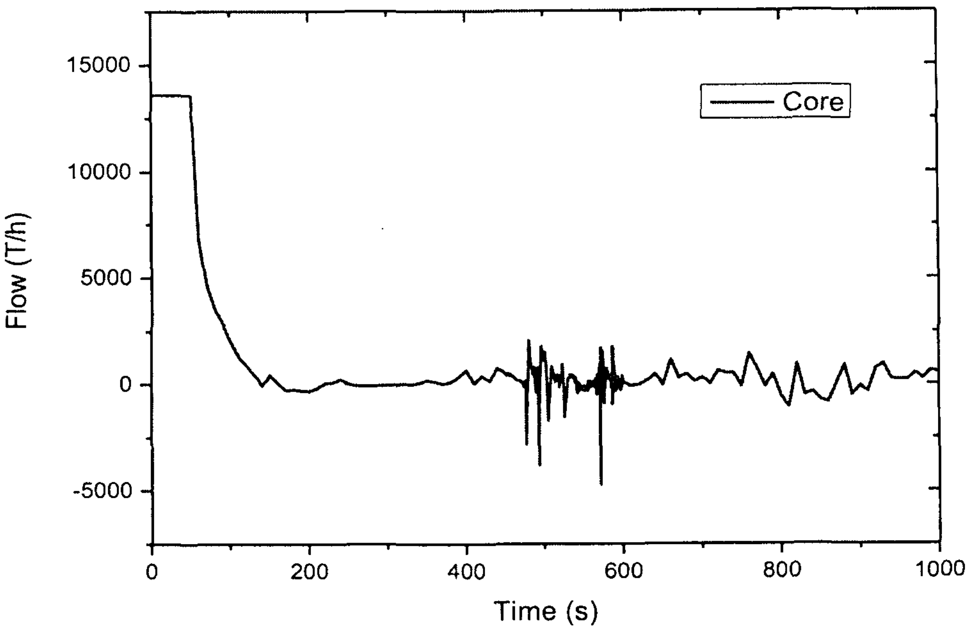


Figure 11. Base Case. Core Mass Flow.

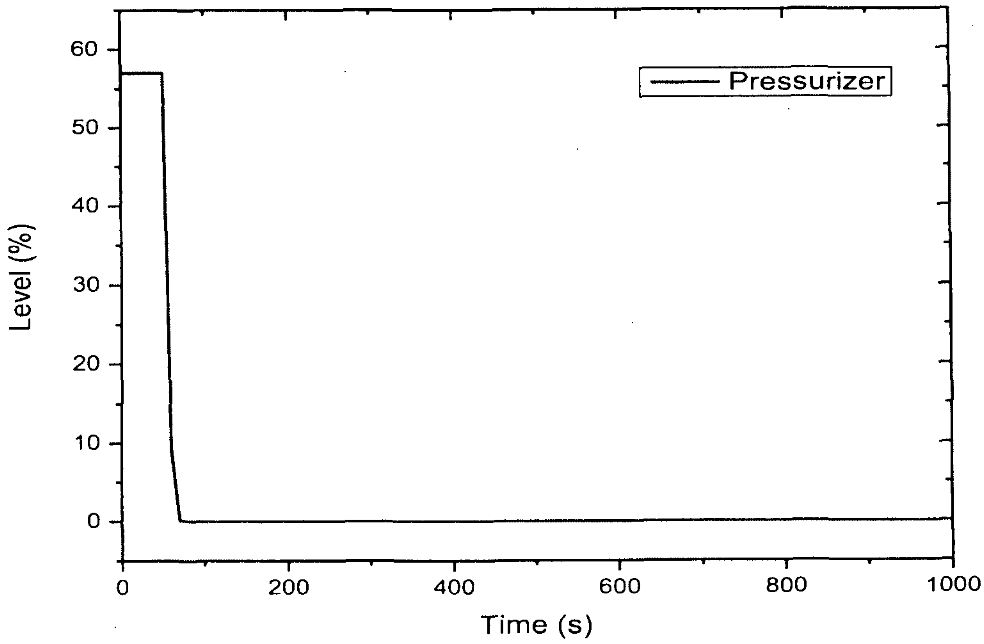


Figure 12. Base Case. Pressurizer Level.

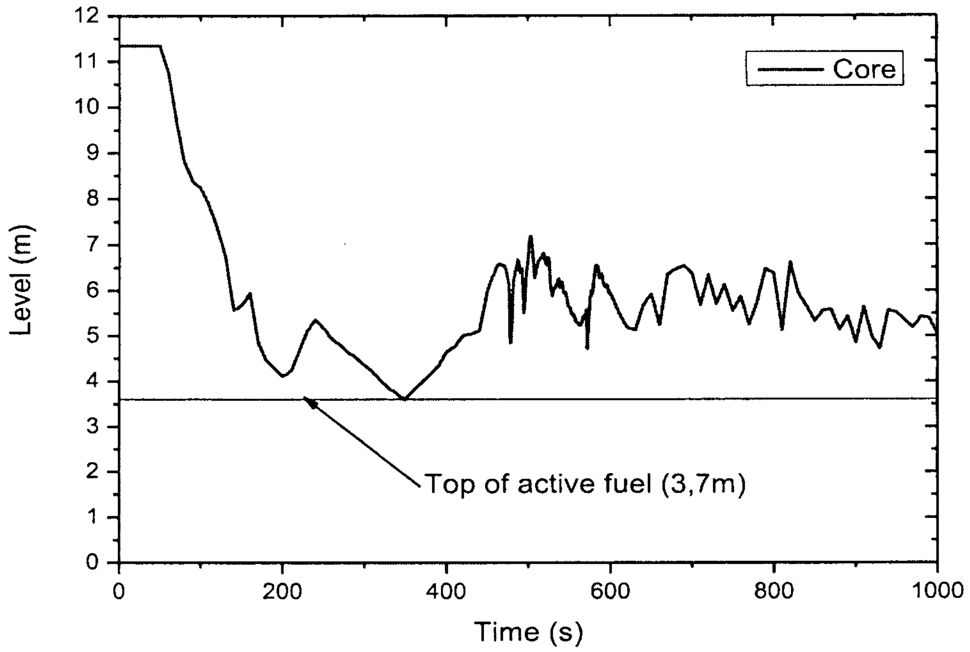


Figure 13. Base Case. Vessel Collapsed Level.

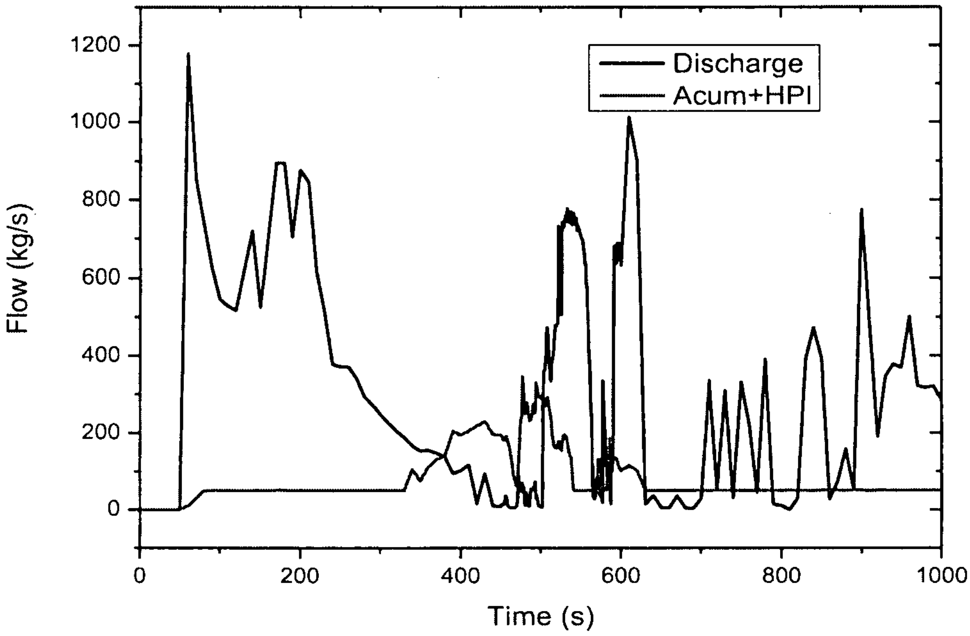


Figure 14. Base Case. Discharge and Injected Mass Flows.

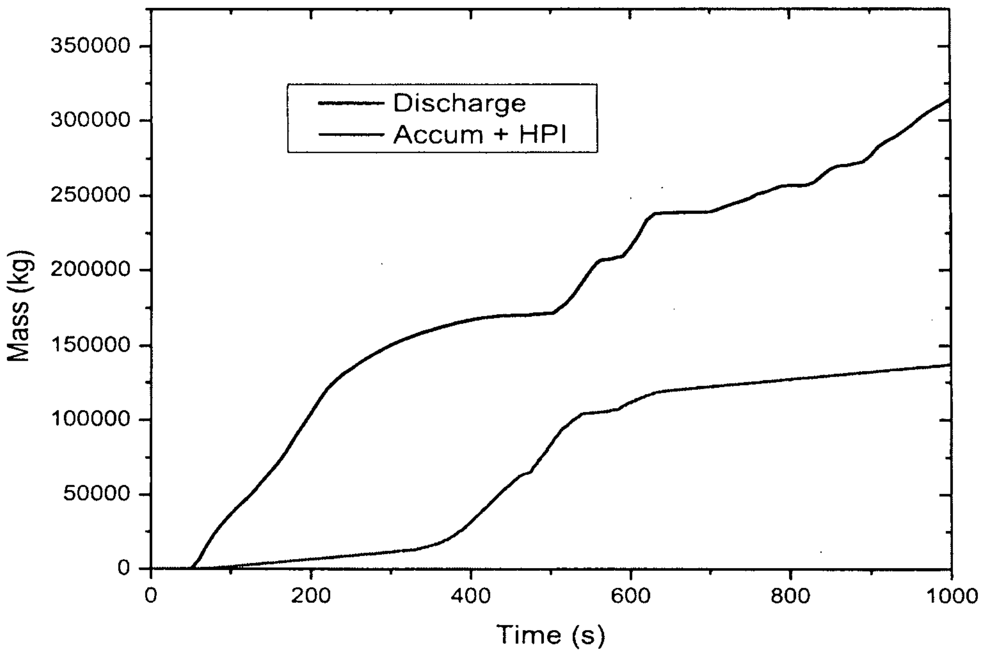


Figure 15. Base Case. Integrated Discharge and Injected Mass Flows.

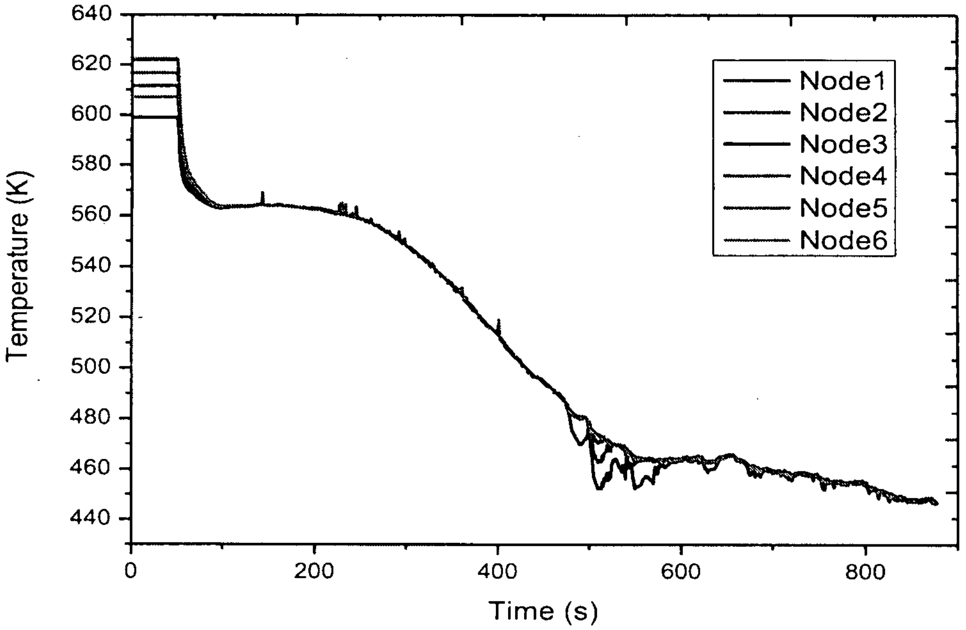


Figure 16. Base Case. HRCT (axial nodes from 1 to 6).

6 ALTERNATIVE SCENARIOS

This section deals with the simulation of two alternative scenarios. In both cases the break is postulated to be the same that in Base Case. The difference among scenarios is that in the first alternative scenario (AS1) a malfunction of one HPIS train is postulated, whereas in the second alternative scenario (AS2), there is a total loss of the HPIS availability (Figure 17). In the Base Case (BC) HPIS is fully available.

In this section it is analyzed how partial or total unavailability of the HPIS modifies the progression of the scenario. Figures commented in this section use blue color for plots representing the worst of the scenarios (AS2), red color for the middle one (one train available) whereas black corresponds to the Base Case.

Figure 18 compares the primary circuit pressure for the three cases. In AS2 the pressure is the highest. Break flow (Figure 19 and Figure 20) is a slightly lower in the worst scenario because there is no HPIS injection on that scenario.

Collapsed levels plotted in Figure 21 indicate that a core uncover can occur in both alternative scenarios, a fact that is corroborated by the void fraction value in the volume above the core (Figure 22). The plot corresponding to AS2 remains at its maximum longer than those of the other scenarios. When vapor fraction in the core nodes reaches values close to 1, cladding temperatures also start rising. Figure 27 and Figure 28 show the hot rod cladding temperatures in the scenarios AS1 and AS2, respectively. In both cases the temperatures decrease again when the accumulators start injecting. In Figure 23 to Figure 26, the flow from the three accumulators is plotted for each of the alternative scenarios.

Table 4: Alternative Scenarios chronology.

Event	**Alternative Scenario I**	**Alternative Scenario II**
Discharge (s)	50	50
RCP stop (s)	50	50
Power stop (s)	50	50
Turbine stop (s)	50	50
Accumulators injection (s)	340	360
HPIS injection (s)	62	none
Hot channel peak cladding temperature, Node 5 (s)	360	380
Hot channel peak cladding temperature, Node 6 (s)	370	390

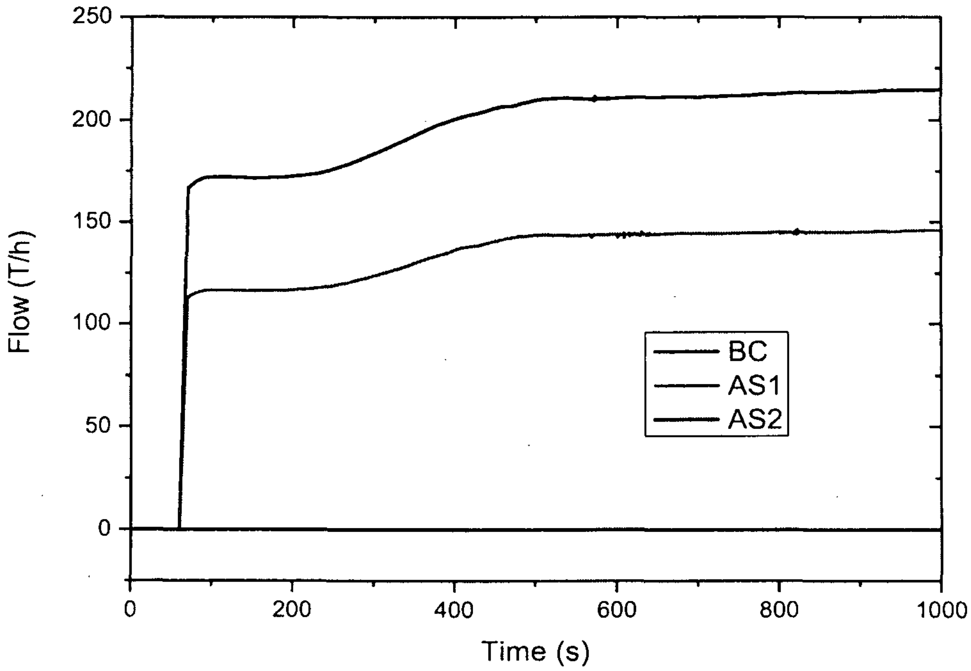


Figure 17. HPIS Injected Mass Flow Rate for the three scenarios.

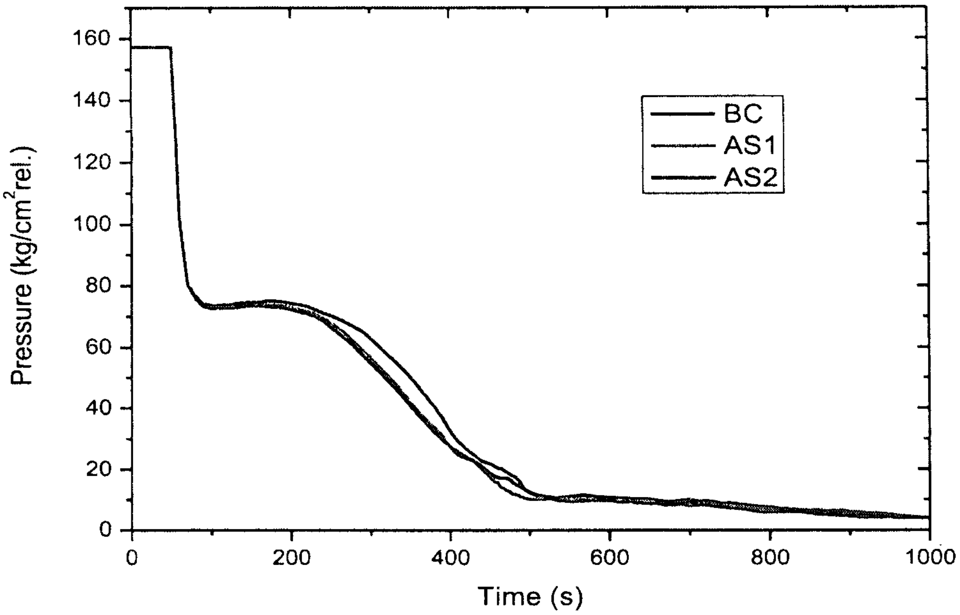


Figure 18. Primary Circuit Pressure for the three scenarios.

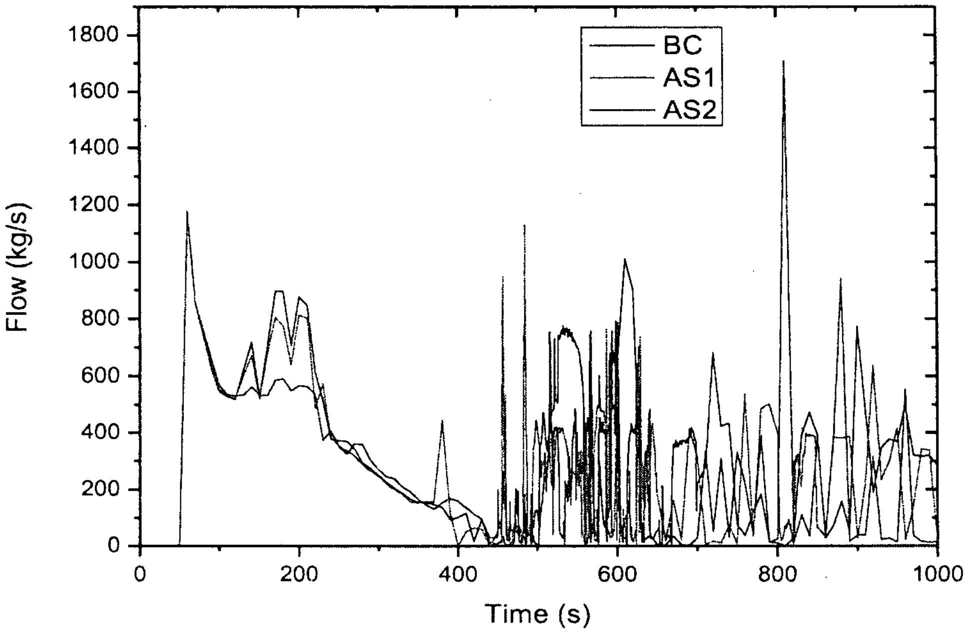


Figure 19. Break Mass Flow Rate for the three scenarios.

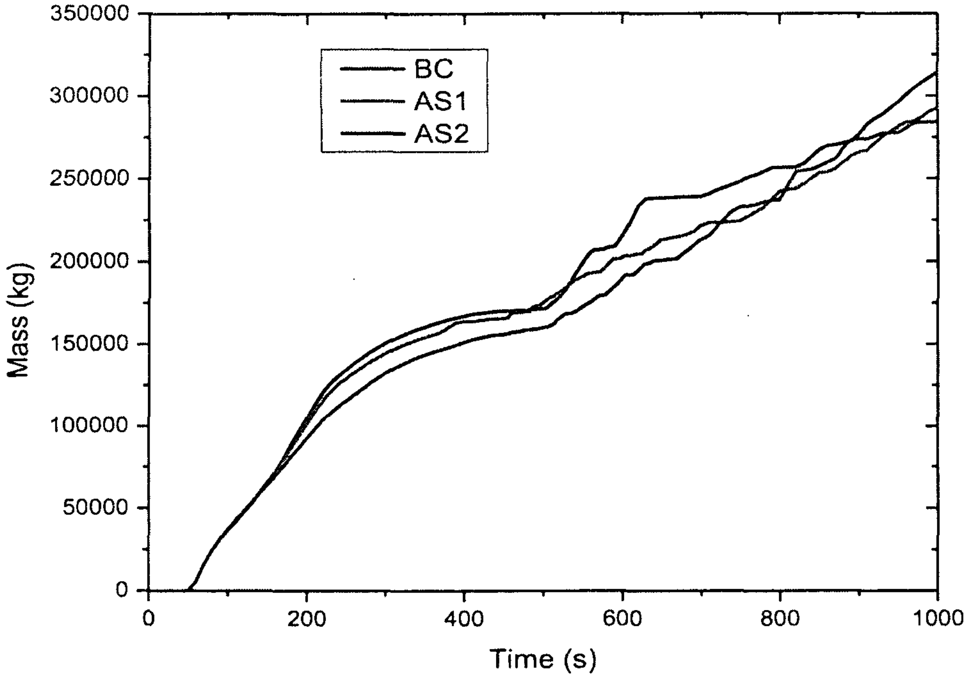


Figure 20. Integrated Break Mass Flow Rate for the three scenarios.

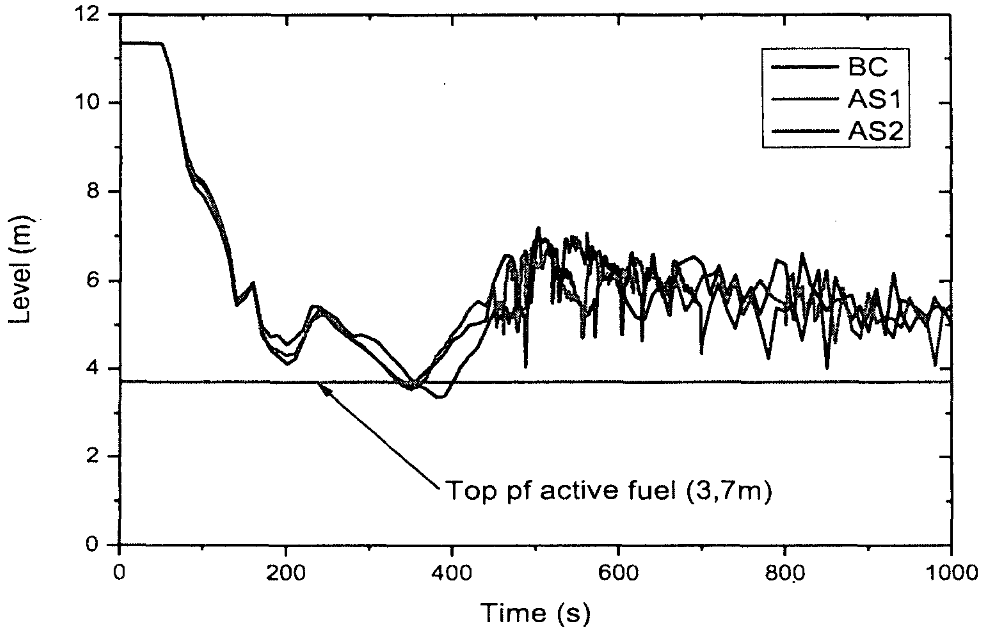


Figure 21. Vessel Collapsed Level in the three scenarios.

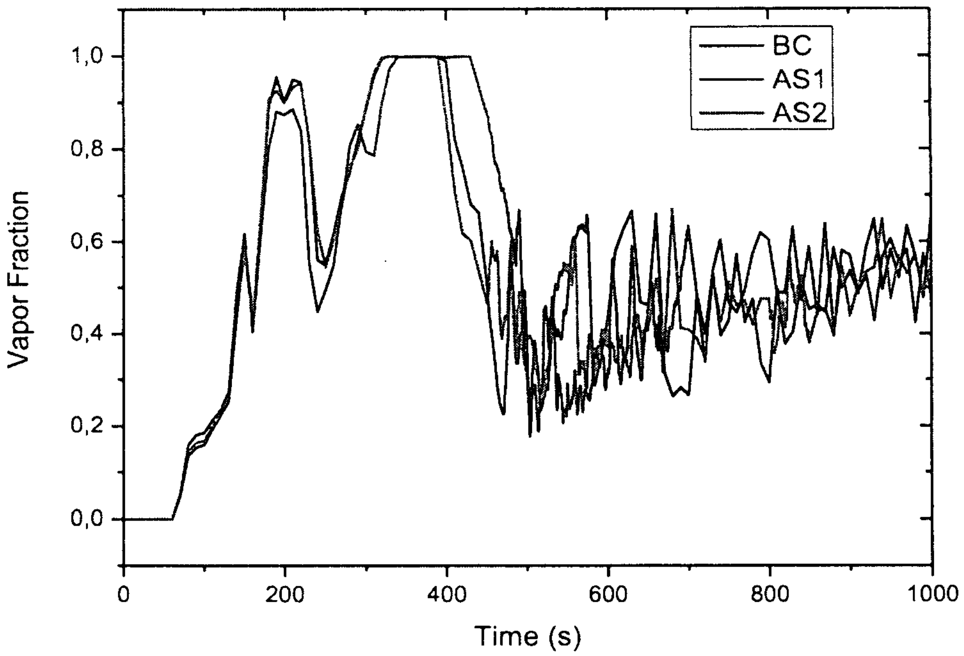


Figure 22. Void Fraction in the volume above the core (for the three scenarios).

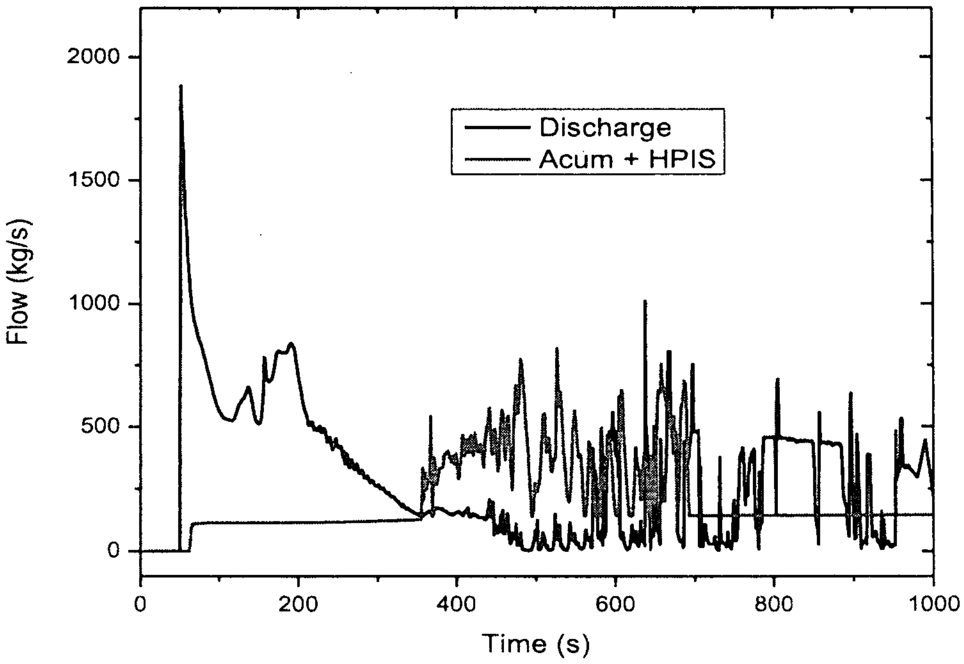


Figure 23. Alternative Scenario I. Discharge and Injected Mass Flows.

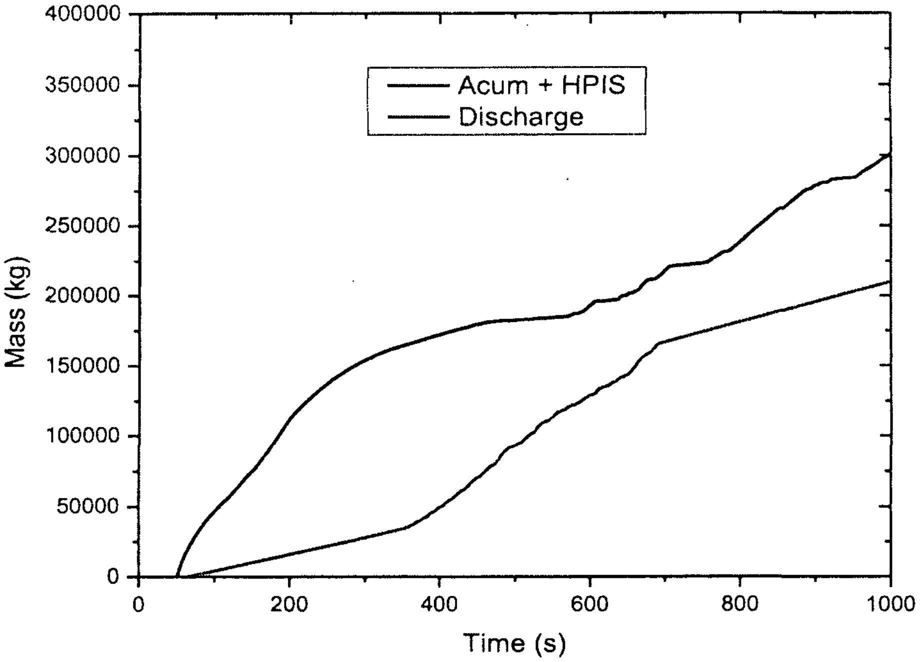


Figure 24.. Alternative Scenario I. Integrated Discharge and Injected Mass Flows.

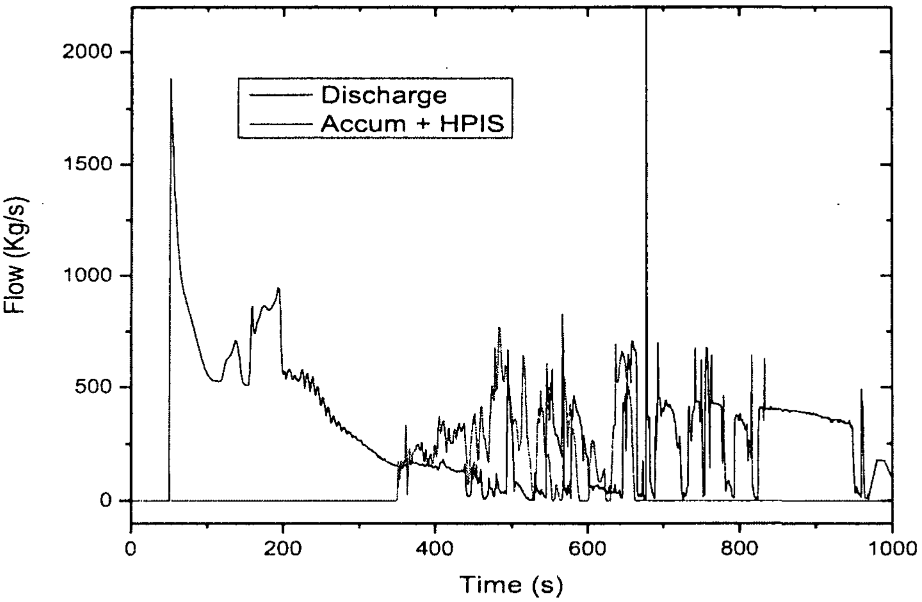


Figure 25. Alternative Scenario II. Discharge and Injected Mass Flows.

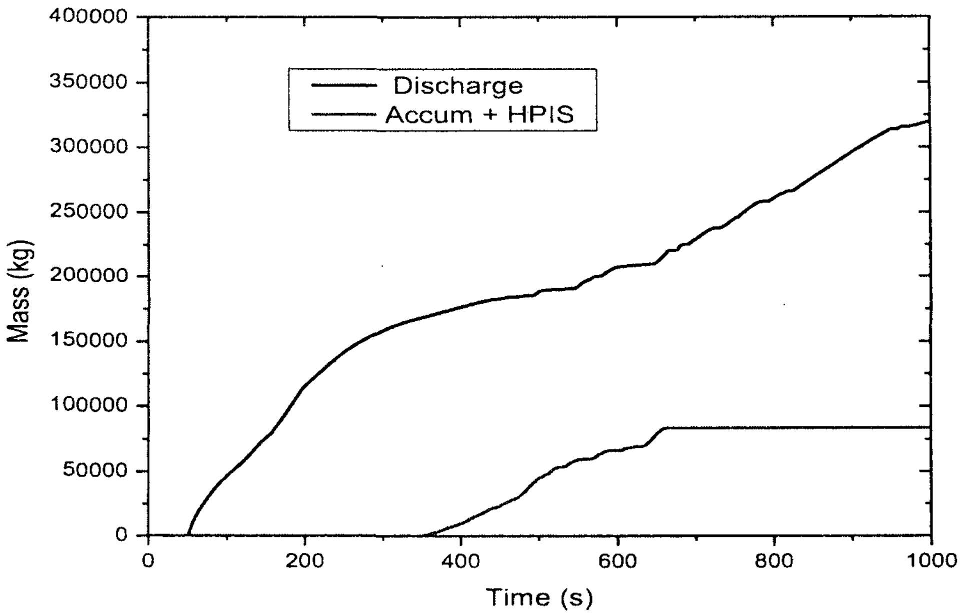


Figure 26. Alternative Scenario II. Integrated Discharge and Injected Mass Flows.

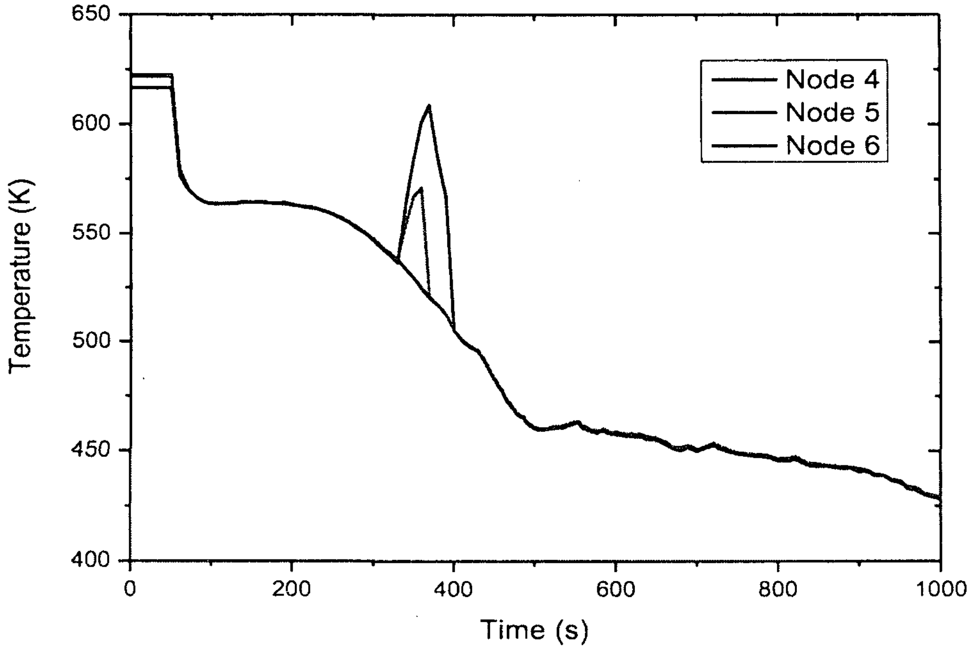


Figure 27. Alternative Scenario I. HRCT, axial nodes 4 to 6.

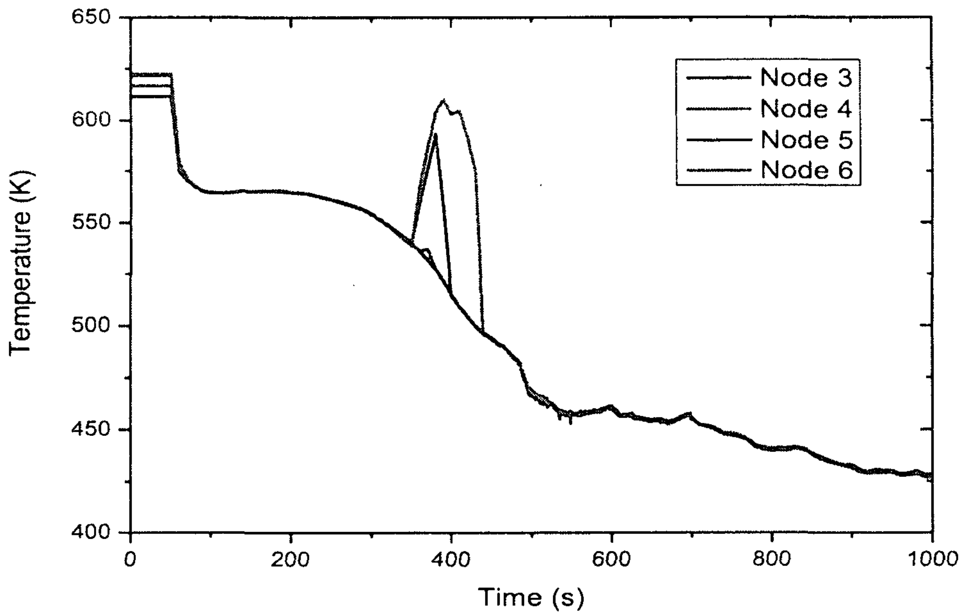


Figure 28. Alternative Scenario II HRCT, axial nodes 3 to 6.

7 SENSITIVITY TO THE PASSIVE HEAT STRUCTURES

As has been mentioned in Section three, the model used so far does not have passive heat structures (HS) in the vessel volumes. In this section a comparison is done in order to test how sensitive the calculation results are to the incorporation of these heat structures. So, passive heat structures have been added to the model in order to better capture the phenomena of the analyzed scenario

The new added heat structures include: the three vessel inlet nozzles (three), the downcomer walls (two), the lower and upper plenum walls (four), the core walls and the bottom and top vessel walls (eight). At the end, 17 new heat structures have been included in the model, with 85 new heat structure nodes.

A new analysis has been performed, comparing the worst case of the scenarios studied (complete loss of the HPIS trains, AS2) with and without passive heat structures. To do so, a calculation has been run with the new model.

From Figure 29 it can be observed that the case with passive HS yields a slightly higher pressure inside the primary circuit. This difference is caused by the heat accumulated by the passive HS being released as the primary pressure decreases, providing an extra steam generation that reduces the depressurization rate (Figure 30).

As can be observed in Figure 31, the break flow rate is quite similar in both cases. The time of accumulator injection, nevertheless, is different (Figure 32). Without the passive HS the accumulators start injecting slightly before (due to difference in primary pressure), but this difference is enough to cause a noticeable advancement in the level recovery (Figure 33).

Consequently, as can be stated from Figure 34 and Figure 35, the lack of the passive heat structures in the model leads to a quite non-conservative result.

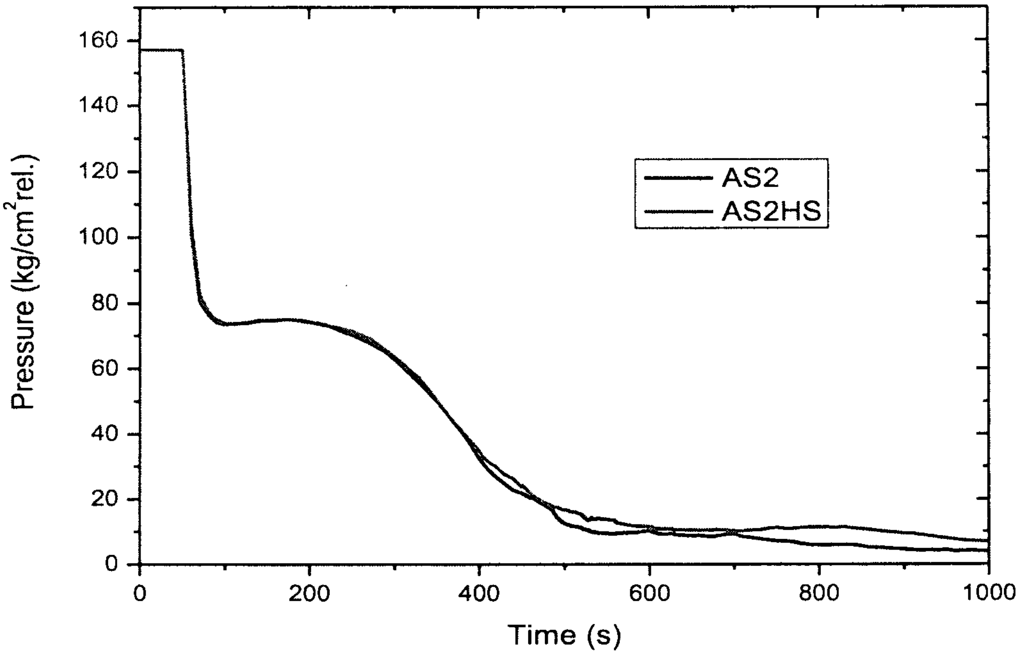


Figure 29. Sensitivity to passive HS. Primary Circuit Pressure.

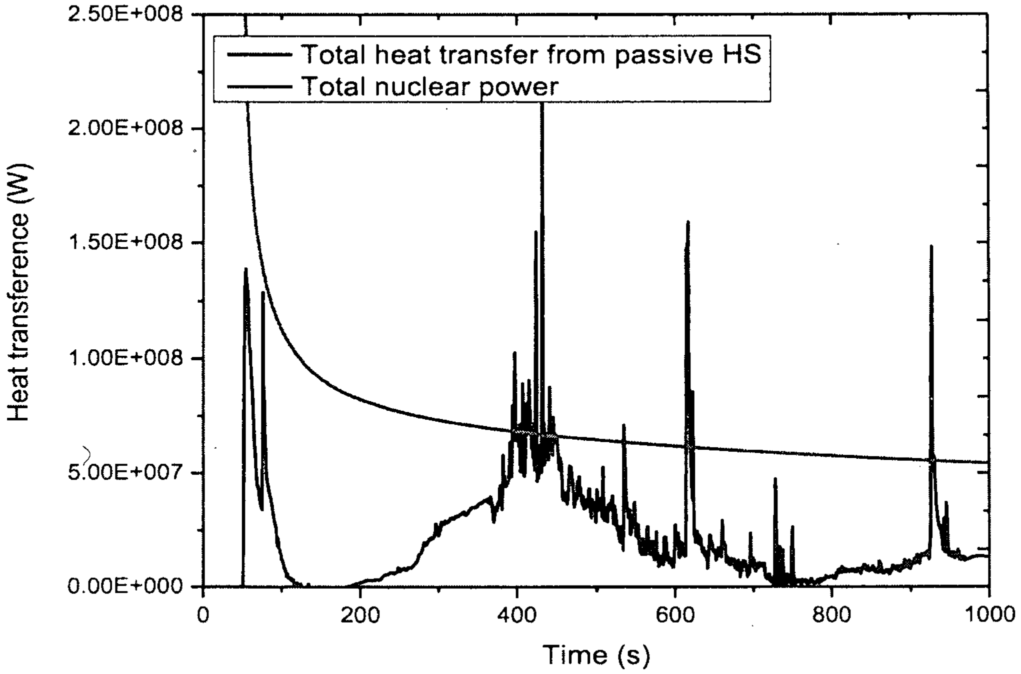


Figure 30. Alternative Scenario II. Heat tansfer from passive HS to coolant.

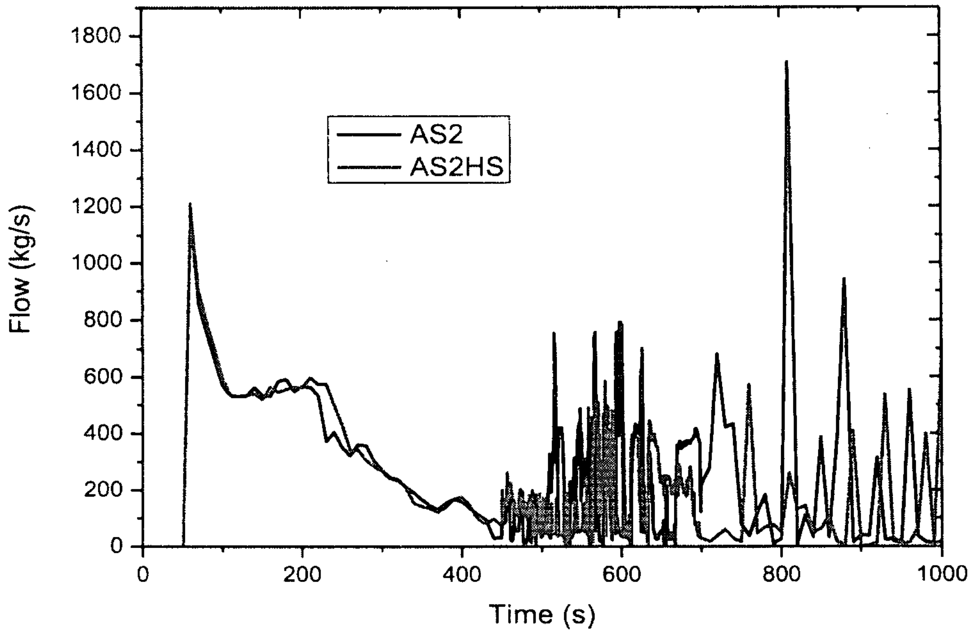


Figure 31. Sensitivity to the modeling of passive HS. Break Mass Flow Rate.

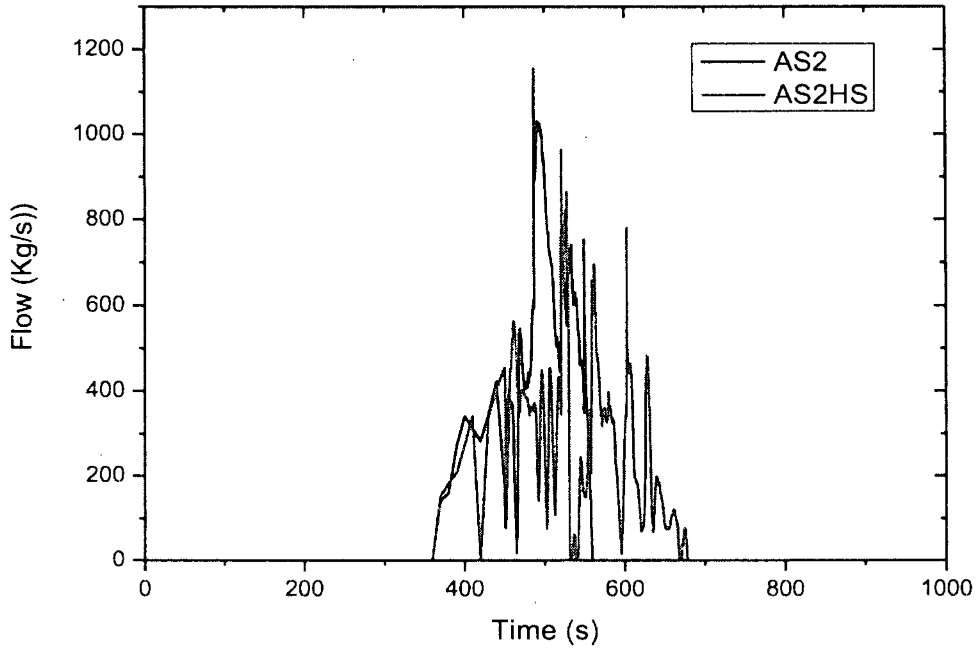


Figure 32. Sensitivity to the modeling of passive HS. Accumulators Water Flow.

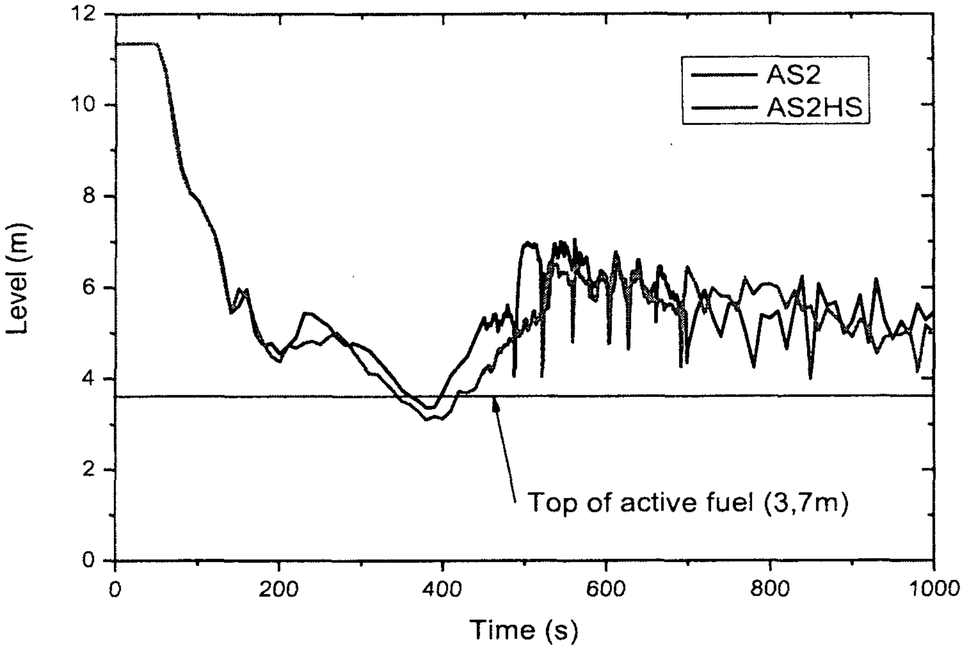


Figure 33. Sensitivity to the modeling of passive HS. Core levels.

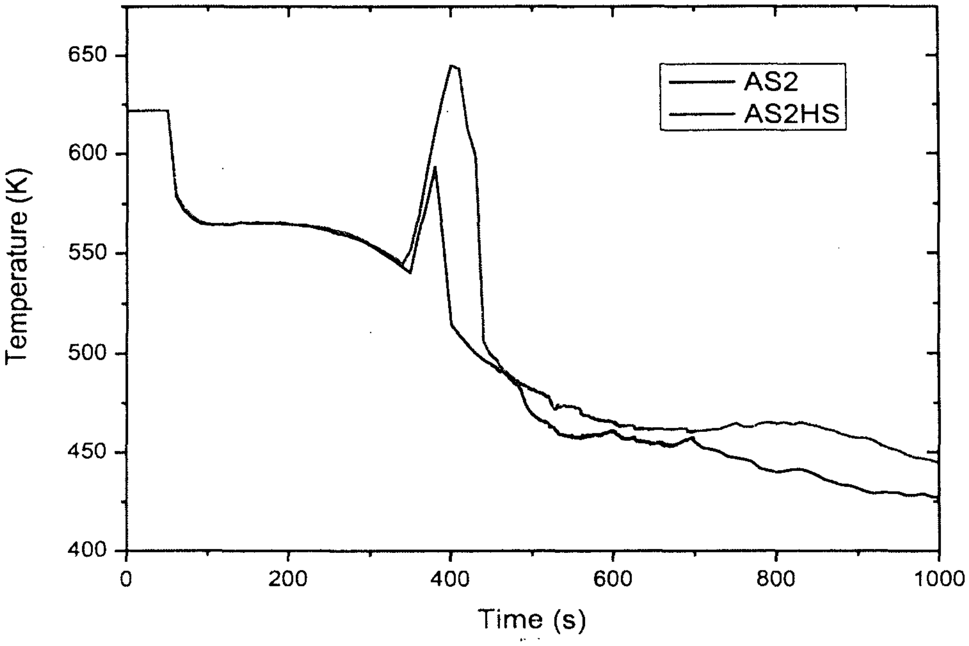


Figure 34. Sensitivity to the modeling of passive HS. HRCT, axial node 5

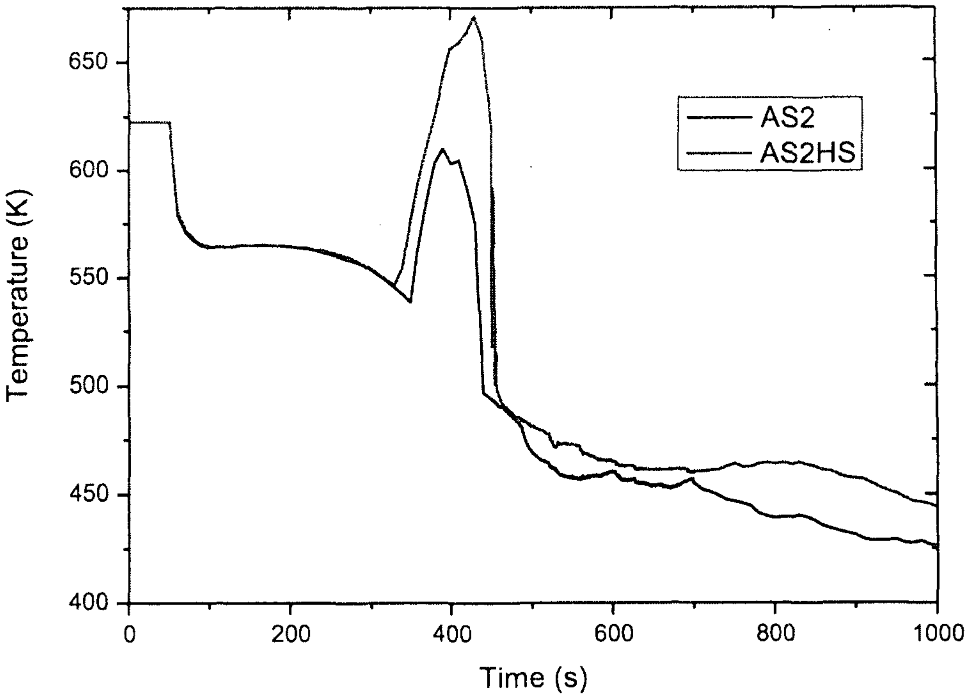


Figure 35. Sensitivity to the modeling of passive HS. HRCT, axial node 6

8 SENSITIVITY TO RCP STOP TIME

In this section a comparison is done in order to test how sensitive the results are to the trip time of the RCPs. A new calculation has been performed, comparing the worst case of the scenarios studied (complete loss of the HPIS trains) with passive heat structures when the reactor coolant pumps stop at different times.

Several data are plotted in each graph. *RCP no action* means that pumps do not trip.

RCPstop means that reactor coolant pumps trip at same time that LOCA occurs.

RCP30 means that pumps are stopped 30 seconds after the LOCA. This selection has been chosen under *"IOE-E-1 perdida del refrigerante del reactor o secundario"* recommendations. IOE-E-1 is an EOP for Ascó NPP [5]. There, the instruction is to stop RCP pumps, when subcooling is lower than 0 °C. From the RCP no action scenario calculation, this condition happens at a time about 20 seconds into the transient. 10 more seconds have been added to account for operators actions delay.

RCP70 means that pumps are stopped 70 seconds right after the LOCA.

In Figure 36 it can be observed a comparison of the core mass flow for the different cases studied. Figure 37 shows how the break flow rate is quite similar in all cases. Accumulators start to inject in different times (Figure 38), and this fact is enough to cause noticeable advancement in the level recovery (Figure 39 and Figure 40). Peak cladding temperature gets a quite non-conservative scenario for the RCP no action. The more the pump trip is delayed, the temperature excursions are milder and, eventually, disappear (Figure 41 and Figure 42).

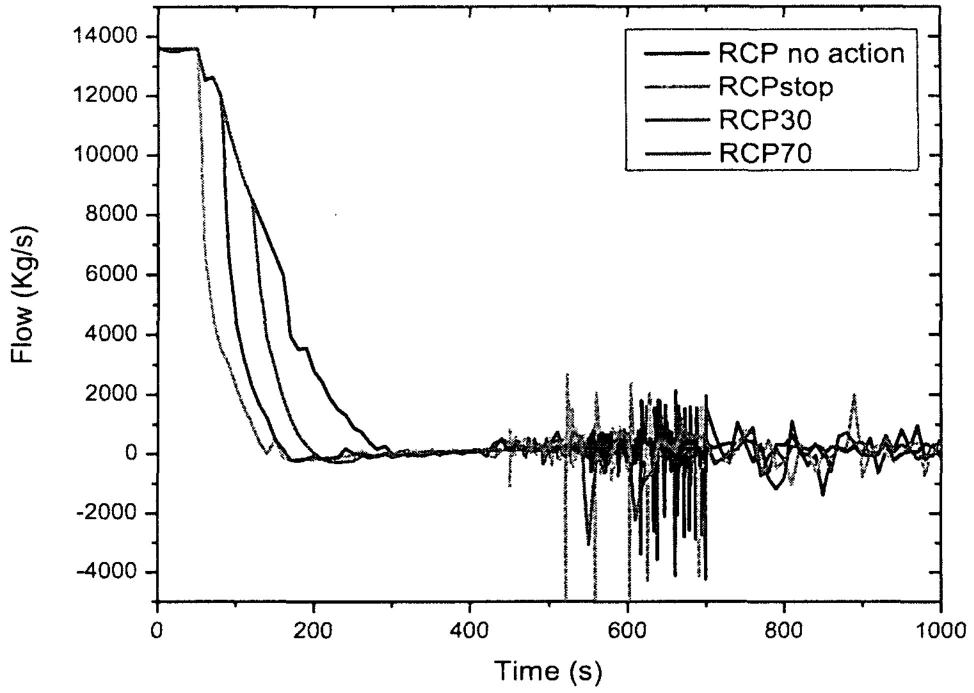


Figure 36. RCP pumps sensitivity. Core water flow.

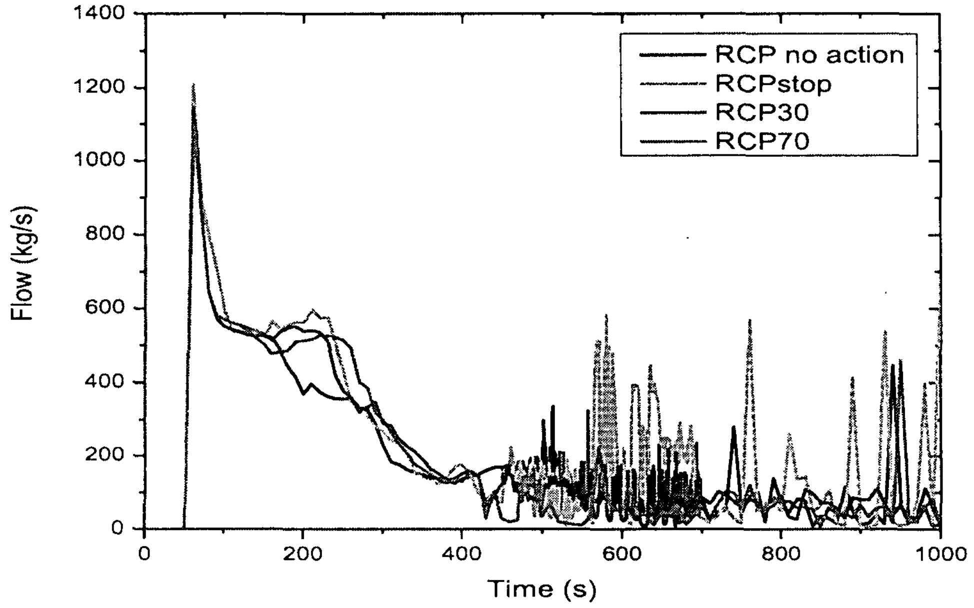


Figure 37. RCP pumps sensitivity. Discharge water flow.

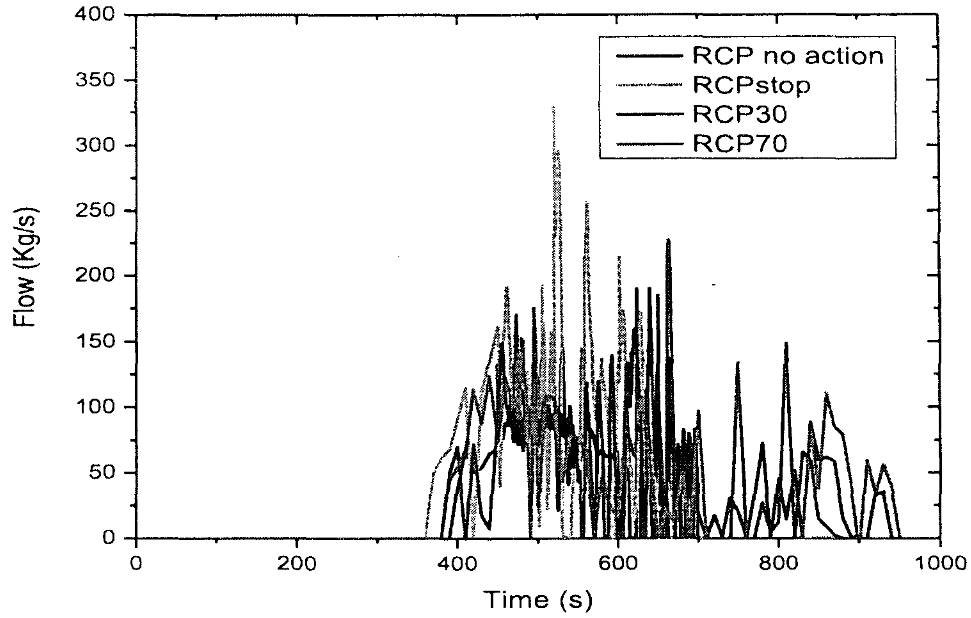


Figure 38. RCP pumps sensitivity. Accumulators water flow.

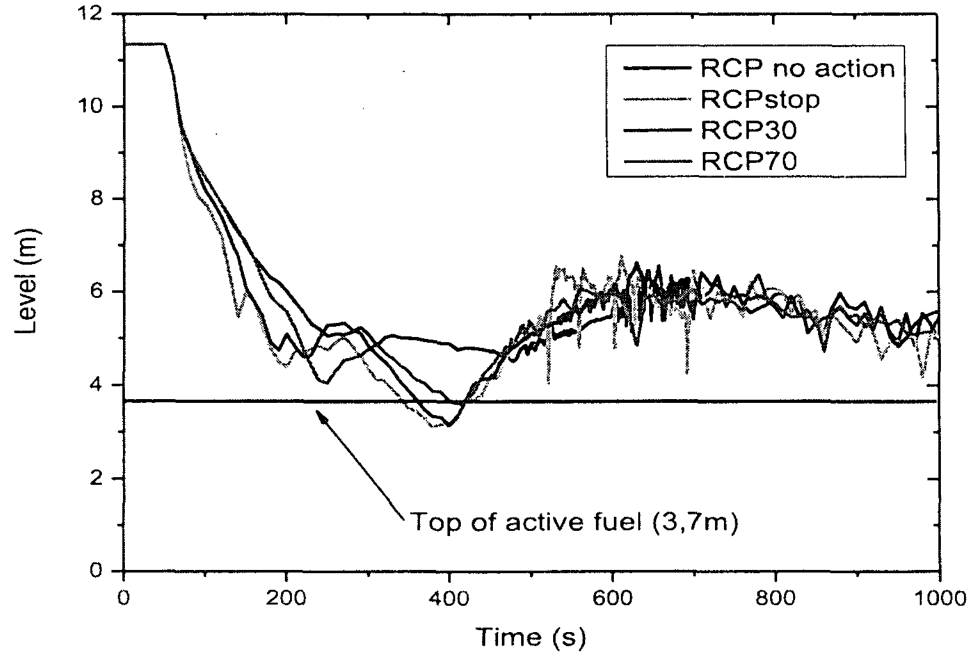


Figure 39. RCP pumps sensitivity. Core levels.

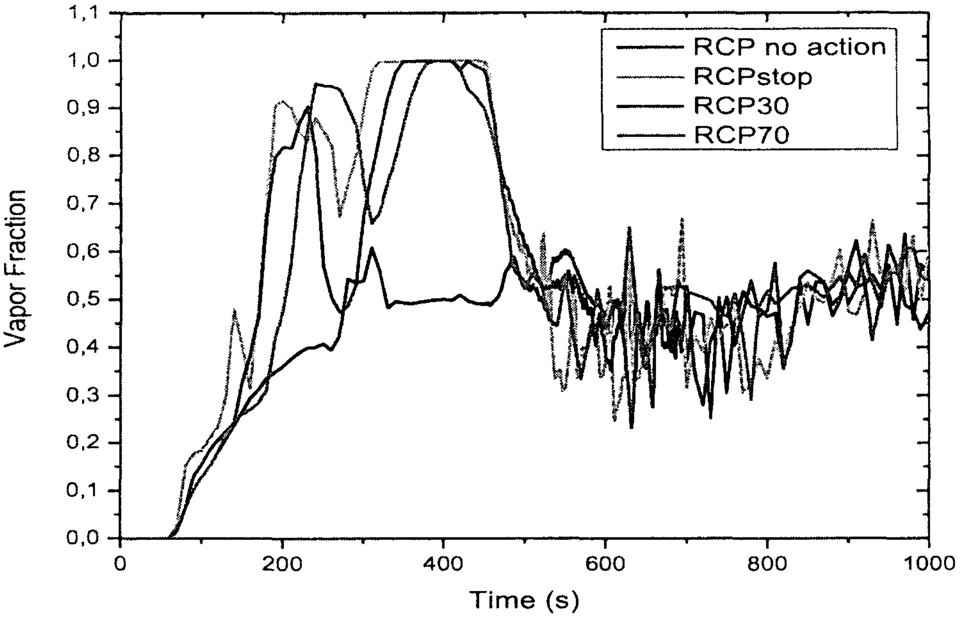


Figure 40. RCP pumps sensitivity. Vapor fraction in the volume above the core.

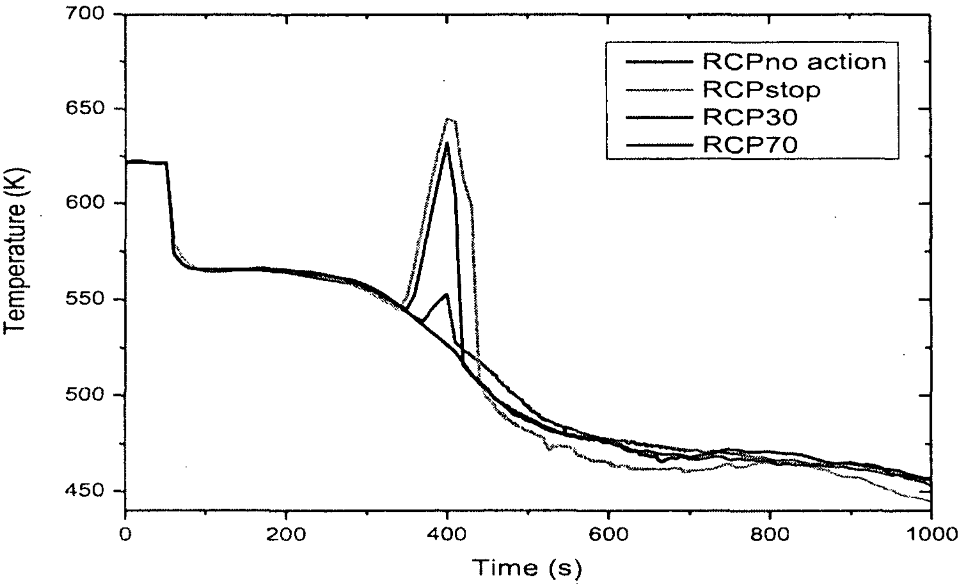


Figure 41. RCP pumps sensitivity. HRCT, Axial Node 5.

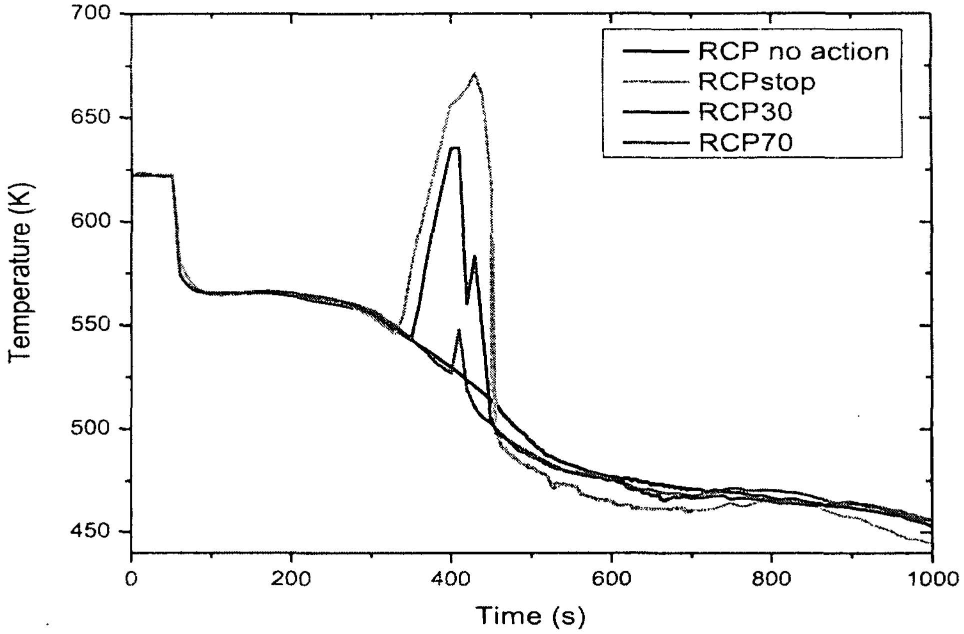


Figure 42. RCP pumps sensitivity. HRCT, Axial Node 6.

9 CONCLUSIONS

A 6 inch cold leg loss of coolant accident has been simulated for the Ascó 1 NPP at the conditions of Middle of Life for the load cycle 17. The model of the plant, prepared for RELAP5/Mod3.2, has been thoroughly tested and validated (see [3] and [4]) and is considered fully qualified for this purpose.

From the calculations it is stated that, in this kind of accidents, if all the HPIS are on duty the scenario does not represent any problem for the safety of the plant. In case of failure of one HPIS train, the cladding temperature of the upper part of the fuel elements may rise slightly but soon it is returned to the safety values by the action of the accumulators. Only in the case of malfunction of both HPIS trains, the cladding temperature can rise above the steady state value. Even in this case, the accumulators injection returns the critical variables to normal values before the end of the transient.

After a first set of calculations, passive heat structures have been added to the model to simulate the heat accumulated in the vessel materials. After the sensitivity calculation performed (in which the total failure of HPIS scenario is repeated) the importance of simulating the passive heat structures has become clear. The heat accumulated by this material is released during the transient (at a rate that is the same order of magnitude than residual power during some hundreds of seconds), slowing the depressurization rate and causing a larger temperature excursion in the core than that calculated without those structures. Even in this case, however, accumulator injection is able to return the temperatures to normal values before the end of the transient.

Finally, the worst case of the scenarios studied (complete loss of the HPIS trains with passive heat structures) was recalculated with varying reactor coolant pumps stop times. This sensitivity analysis shows that imposing a pump trip simultaneously with the break is a conservative option because the more the trip is delayed the milder are the cladding temperature peaks.

10 REFERENCES

1. F. Reventós, L. Batet, C. Llopis, C. Pretel and I. Sol, *Thermal-Hydraulic Analysis Tasks for ANAV NPPs in Support of Plant Operation and Control*, Science and Technology of Nuclear Installations, Volume 2008.

2. The RELAP5 Code Development Team, *RELAP5/Mod3 Code Manuals,* NUREG/CR-5535, 1995.

3. 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.

4. 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.

5. Central Nuclear de Ascó. IOE-E-1 *Pérdida de refrigerante del reactor o secundario.*

6. C.N. Vandellòs II, (in Spanish) *Nota de cálculo del modelo de C.N. Vandellòs II con RELAP5/Mod3.2*

7. Westinghouse; Model F steam generator termal-hydraulic data report (WTP-ENG-TN-81-001)

8. Westinghouse Nuclear Española, (in Spanish) *Descripción del SNGV de Westinghouse,* 1983.

9. Westinghouse Energy Systems International; Precautions, limitations and setpoints. C.N. Vandellòs II

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10. SUPPLEMENTARY NOTES

A. Calvo, NRC CAMP Program Manager

11. ABSTRACT *(200 words or less)*

A postulated 6 inch break Loss-of-Coolant Accident (LOCA) is analyzed for the Ascó I plant (a 3 loop PWR Westinghouse design) using RELAP5/Mod3.2. Different scenarios are calculated, including the total loss of the High Pressure Injection System (HPIS). In this case, the maximum cladding temperature rises above the steady state value at full power but, before the end of the transient, temperatures return to normal values by the effect of the accumulators injection. Passive heat structures in the reactor pressure vessel have been incorporated to the model (i.e. the pressure vessel walls and some internals). Calculations with the model including passive heat structures lead to more severe scenarios (i.e. higher cladding temperatures) but still accumulators action is able to return the temperatures to normal values. Finally, sensitivity to the trip time of the reactor coolant pumps is analyzed. Results show that lower cladding temperatures result when the pump trip is delayed.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)* Loss-of-Coolant Accident (LOCA) RELAP5/Mod3.2 High Pressure Injection System (HPIS) Technical University of Catalonia Consejo de Seguridad Nuclear (CSN) Code Assessment and Management Program (CAMP) Ascó I plant (a 3 loop PWR Westinghouse design) Vandellós II High Pressure Injection System (HPIS) Thermal-hydraulic codes	13. AVAILABILITY STATEMENT unlimited 14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified 15. NUMBER OF PAGES 16. PRICE

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