ACCIDENT MANAGEMENT ACTIONS IN AN UPPER-HEAD SMALL-BREAK LOSS-OF-COOLANT ACCIDENT WITH HIGH-PRESSURE SAFETY INJECTION FAILED

KEYWORDS: *emergency operating procedures, SBLOCA, TRACE code*

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Since the Three Mile Island accident, an important focus of pressurized water reactor (PWR) transient analyses has been a small-break loss-of-coolant accident (SBLOCA). In 2002, the discovery of thinning of the vessel head wall at the Davis Besse nuclear power plant reactor indicated the possibility of an SBLOCA in the upper head of the reactor vessel as a result of circumferential cracking of a control rod drive mechanism penetration nozzle—which has cast even greater importance on the study of SBLOCAs. Several experimental tests have been performed at the Large Scale Test Facility to simulate the behavior of a PWR during an upper-head SBLOCA. The last of these tests, Organisation for Economic Co-operation and Development Nuclear Energy Agency Rig of Safety Assessment (OECD/NEA ROSA) Test 6.1, was performed in 2005. This test was simulated with the TRACE 5.0 code, and good agreement with the experimental results was obtained.

Additionally, a broad analysis of an upper-head SBLOCA with high-pressure safety injection failed in a Westinghouse PWR was performed taking into account different accident management actions and conditions in order to check their suitability. This issue has been analyzed also in the framework of the OECD/NEA ROSA project and the Code Applications and Maintenance Program (CAMP). The main conclusion is that the current emergency operating procedures for Westinghouse reactor design are adequate for these kinds of sequences, and they do not need to be modified.

I. INTRODUCTION

Vessel head wall thinning found in the reactor at the Davis Besse nuclear power plant (NPP) on February 16, 2002, raised a safety issue regarding vessel structural integrity; see Fig. 1 and Refs. 1 through 4. Circumferential cracking of the penetration nozzle of the control rod drive mechanism (CRDM) could cause a small-break loss-of-coolant accident (SBLOCA) at the pressure vessel upper head in a pressurized water reactor (PWR).

As part of participation in the Organisation for Economic Co-operation and Development Nuclear Energy Agency Rig of Safety Assessment (OECD/NEA ROSA) project and the Code Applications and Maintenance Program (CAMP), the Universidad Politécnica de Madrid has performed a broad analysis of an upper-head SBLOCA with high-pressure safety injection (HPSI) failed in a Westinghouse PWR:

1. In the first stage, simulation of OECD/NEA ROSA Test 6.1 was performed and compared extensively to the experimental results.

2. In the second stage, transients similar to those of OECD/NEA ROSA Test 6.1 were simulated with the TRACE model of the Almaraz NPP Unit 1 (Westinghouse three-loop design). This analysis took into account different accident management actions and conditions in order to check their suitability.

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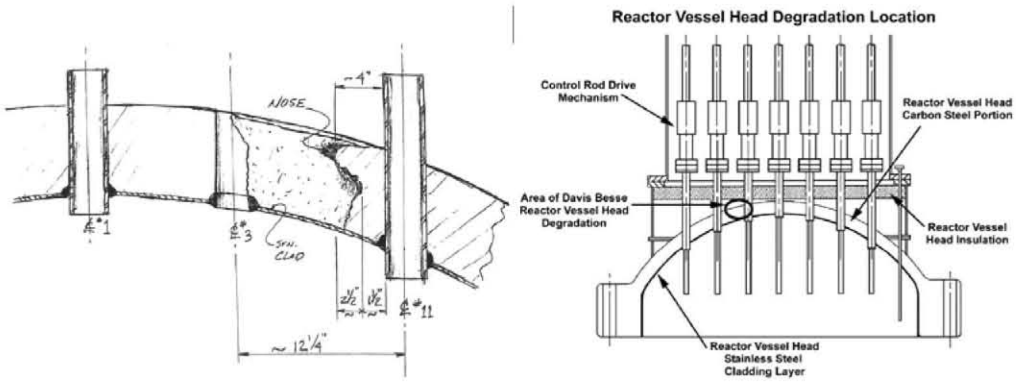

Fig. 1. Reactor vessel head degradation location; Davis Besse NPP (from http://www.nrc.gov).

II. EMERGENCY OPERATING PROCEDURES RELATED TO SBLOCA SEQUENCES

In this kind of sequence, i.e., SBLOCA with HPSI failed, the operators must follow several emergency operating procedures (EOPs). The main tasks of the EOPs corresponding to the Westinghouse reactor design are described in Fig. 2 and Ref. 5:

1. Whenever there is a reactor SCRAM, EOP E-0 (i.e., reactor TRIP or safety injection) must be started. In step 22, the reactor coolant system (RCS) integrity is checked, and if it is not intact, there is a transition to EOP E-1 (loss of reactor or secondary coolant).

2. In EOP E-1, step 1, the operator checks if the reactor coolant pumps (RCPs) should be stopped (they will be stopped by the operator if there is at least one HPSI pump running and loss of subcooling).

3. Following EOP E-1, step 11, the operator checks the primary pressure, and if it is not below ~15 bars (the exact value depends on the NPP), there is a transition to EOP ES-1.2 [post–loss-of-coolant-accident (LOCA) cooldown and depressurization].

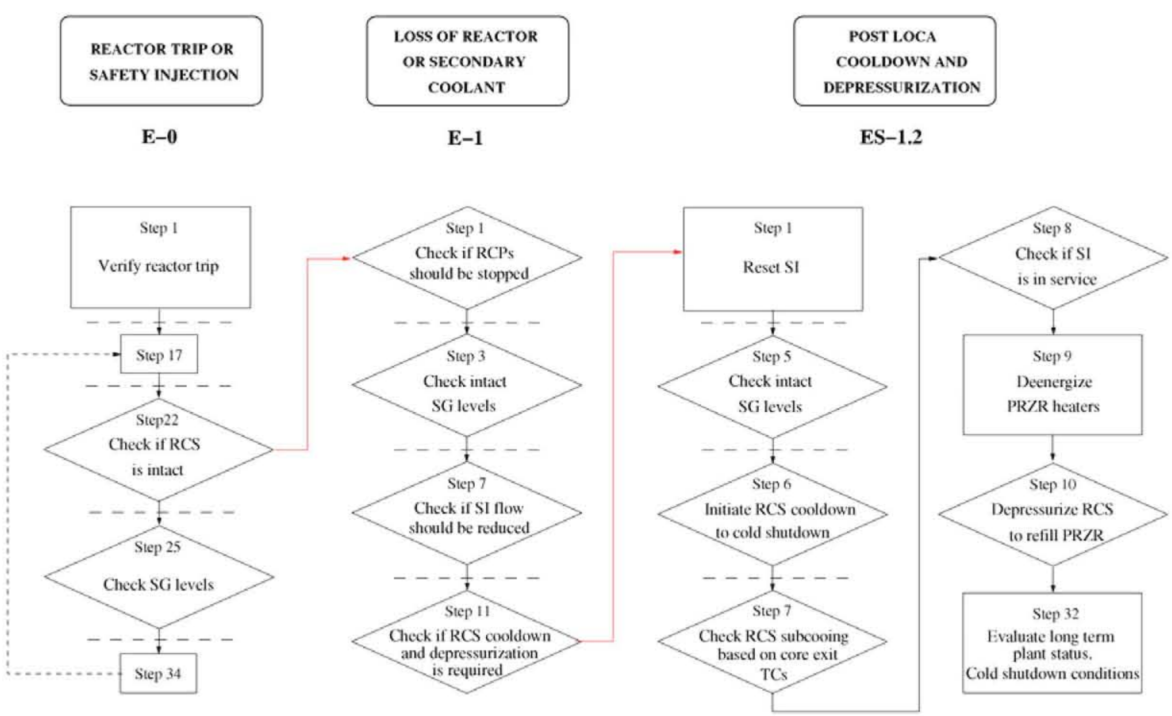

Fig. 2. Main steps of the EOPs related to an upper-head SBLOCA.

4. In EOP ES-1.2, the operator cools and depressurizes the primary system, opening the steam dump valves, or if that is not possible, the operator opens the steam generator relief valves at the secondary side, making sure that the cooling of the RCS is close to 55 K/h.

It is important to comment on some considerations with respect to the two main operator actions: RCP trip and cooling and depressurizing the primary system by means of steam generators:

1. *RCP trip*: The need to review the manual RCP trip conditions during an SBLOCA was an issue that arose as a result of the Three Mile Island (TMI) accident. Westinghouse analyzed this issue for SBLOCA sequences with and without HPSI (see Refs. 6, 7, and 8), and the main conclusions were the following:

   a. If the HPSI is available, the RCP must be tripped at the beginning of SBLOCA sequences in order to avoid worse consequences following a delayed RCP trip.

   b. If the HPSI is not available, it is better to not trip the RCPs in order to cool the core with a high mass flow.

In other designs like the current Siemens reactors, EPR and AP1000, there is an automatic trip coincident with the safety injection system (SIS) demand, regardless of HPSI availability.[9–11] In the French reactor design there is also a manual trip in the A1.2 procedure (corresponding to the LOCA sequence).[12]

2. *Primary-side cooling*: If the RCS is in saturation conditions, it is possible to obtain the equivalence of a 55 K/h cooling rate (following EOP ES-1.2, as mentioned earlier) in bars per hour, from Figs. 3 and 4 (it must be noted that this equivalence is valid only in saturation conditions). In other designs like the current Siemens reactors and EPR, the cooling rate during an SBLOCA is 100 K/h, and cooling is performed automatically by the protection system.[9,10] In the AP1000 there is no secondary-side depressurization, and the primary-side depressurization is performed using an automatic depressurization system (ADS) with four stages, which is necessary when the core makeup tank is below 70% (Refs. 11 and 13). In the AP1000 design, EOPs direct the operator to actuate the normal residual heat removal system (RHRS) in order to avoid the actuation of the fourth ADS stage.[14] For the French reactor design, the operators must follow procedure A1.1 (small primary system break) during an SBLOCA. The objective of this EOP is to cool the RCS with the steam generator to conditions that enable implementing the RHRS, which it is similar to EOP ES-1.2.

If the accident management actions included in EOP E-0, EOP E-1, and EOP ES-1.2 are not enough to avoid core damage or if there is an error or delay in operator actions, then it is possible to get inadequate core cooling (ICC) conditions; see Refs. 15 through 20 for more details. In this case the operators must follow Status Tree F.0.2 (core cooling) and EOP FR C.1 (response to ICC) and EOP FR C.2 (response to degraded core cooling). The status tree that is related to the critical function of core cooling is F.0.2 (Westinghouse design); see Fig. 5. This status tree directs the operators to the function recovery guideline (FRG) that must be used depending on the values of several parameters. In this case Status Tree

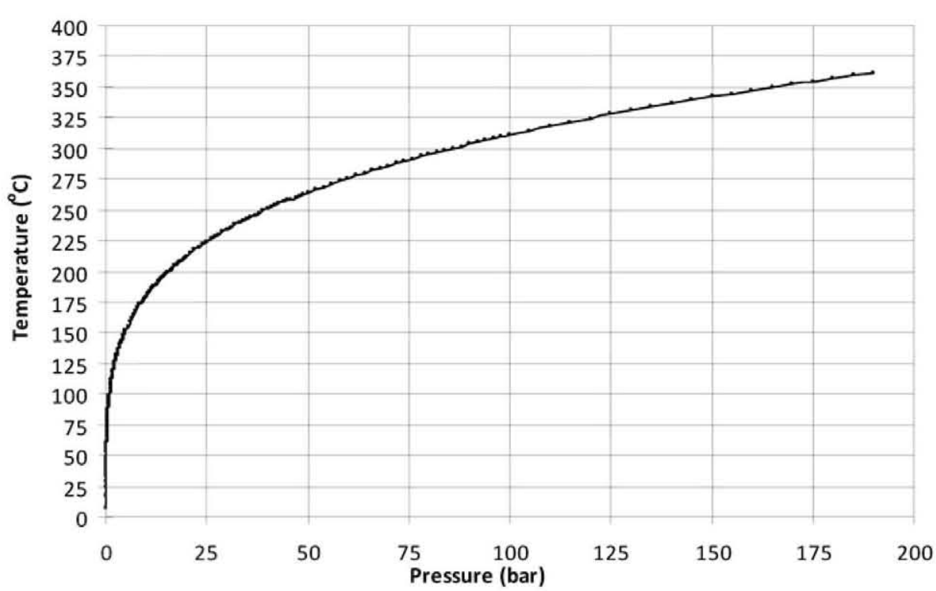

Fig. 3. Saturation temperature.

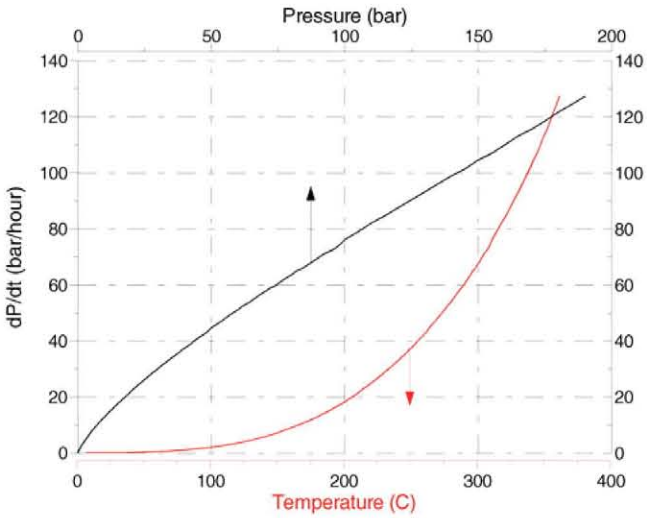

Fig. 4. Primary depressurization rate in saturated conditions for a cooling rate of 55 K/h.

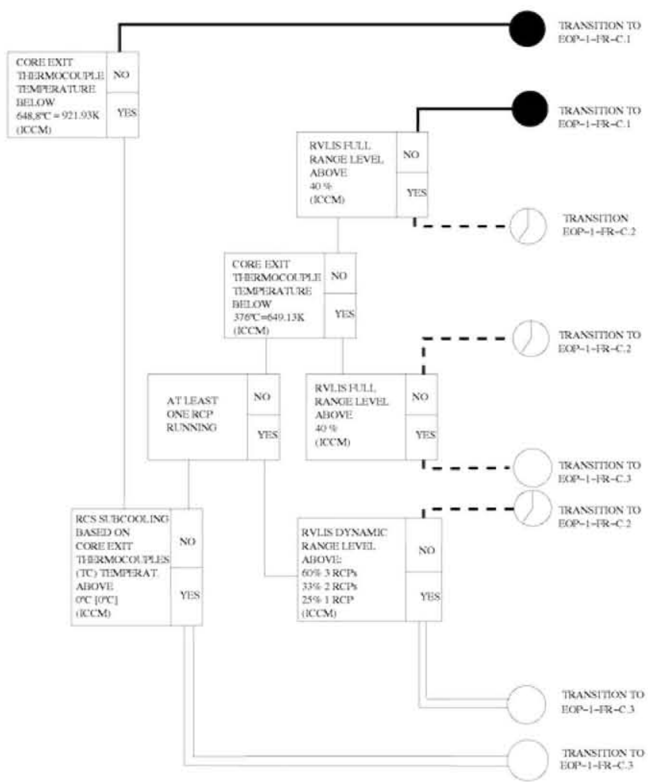

Fig. 5. Status tree related to the FRG of core cooling (i.e., F.0.2).

F.0.2 selects the FRG EOP FR C.1 or EOP FR C.2 depending on the core exit thermocouple (CET) temperatures (see Refs. 21 and 22) and the vessel level, measured by the reactor vessel level indicator system (RVLIS) in the Westinghouse reactor design, which is a differential pressure measuring system for determining the collapsed water level in the reactor vessel; see Refs. 23 and 24 for more details. In other reactor designs electrical resistance detectors at different vessel levels are used instead of the RVLIS (Refs. 25 and 26). Both instrumentation systems, CET and RVLIS, are part of the ICC instrumentation system, which has been required since the TMI accident in 1979 (Ref. 15).

In EOP FR C.2, the operator will cool down the primary side with a maximum cooling rate of 55 K/h (as in EOP ES-1.2), and in EOP FR C.1, the operator will fully open all secondary-side relief valves. In several simulations it has been observed that the cooling rate with full opening is near 300 K/h.

The generic probabilistic risk analysis of the French reactor design (see Ref. 12) mentions that in the event of failure of HPSI during LOCA sequences, the operator will trigger an accelerated cooling by the steam generator (task included in procedure U1), making it possible to attain low-pressure safety injection (LPSI) operation conditions (similar to EOP FR C.1). The time available for this operation is estimated to be 1 h.

III. LARGE SCALE TEST FACILITY: OECD/NEA ROSA TEST 6.1

The Large Scale Test Facility (LSTF) is a full-height, full-pressure, 1/48 volumetrically scaled simulator for a Westinghouse-type four-loop [3423-MW(thermal)] PWR with primary and secondary coolant systems including an electrically heated simulated core, emergency core cooling systems (ECCSs), and control systems for accident management actions; see Ref. 27 and Fig. 6 for more details. The maximum core power

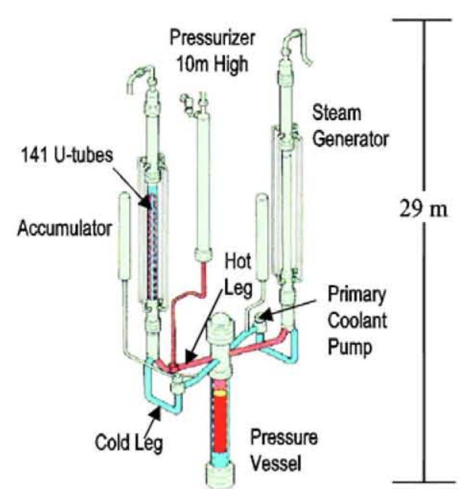

Fig. 6. Large Scale Test Facility.

of 10 MW is equivalent to 14% of the 1/48-scaled PWR rated power covering the scaled PWR decay heat after the scram.

The Universidad Politecnica de Madrid has been working with the ROSA/LSTF TRACE model since February 2006 (Refs. 28, 29, and 30). The TRACE model is based on the TRAC-PF1 model presented by the Japan Atomic Energy Research Institute (JAERI) to the participants of the OECD/NEA ROSA project. The main tasks performed in translating and modifying the model are the following:

1. The old STGEN component was translated to the TRACE model as a set of components (TEEs and PIPEs), conserving volumes and lengths. The steam generator recirculation ratio was adjusted. Later, a new steam generator model with nine different heights of tubes was developed.

2. The old VESSEL component was translated to the TRACE model, and the temperature in the upper head of the vessel was adjusted to the measured one (~586 K). The REFLOOD model was activated.

3. The total mass flow was adjusted in the primary loops using friction coefficient (FRIC) parameters and the rated head in the RCP. The mass flow rate from the downcomer to the upper head of the vessel was adjusted to the specified one (0.3% of the downcomer vessel total mass flow).

4. Volume-versus-height plots were checked with respect to the facility data, and all the volume and height discrepancies were corrected.

5. A new two-dimensional model of the pressurizer was created to avoid excessive cooling in the upper cells of the model during long quasi-steady-state transients, which was a problem with the earlier model. Also, stabilization of the pressurizer level and pressure control systems was included to adequately fix the steady state. Finally, new, more detailed proportional and base heaters were also added.

6. Heat losses and pressure drops of the whole model were adjusted.

7. The OFFTAKE model was activated in the connections of the valves that simulate breaks in different localizations of the LSTF.

8. An animation mask was created with the SNAP application; see Fig. 7. This mask allows videos of the simulations to be performed, which allows the transient behavior to be easily interpreted.

The ROSA/LSTF TRACE model has 178 thermal-hydraulic components (2 VESSEL, 45 PIPE, 8 TEE, 2

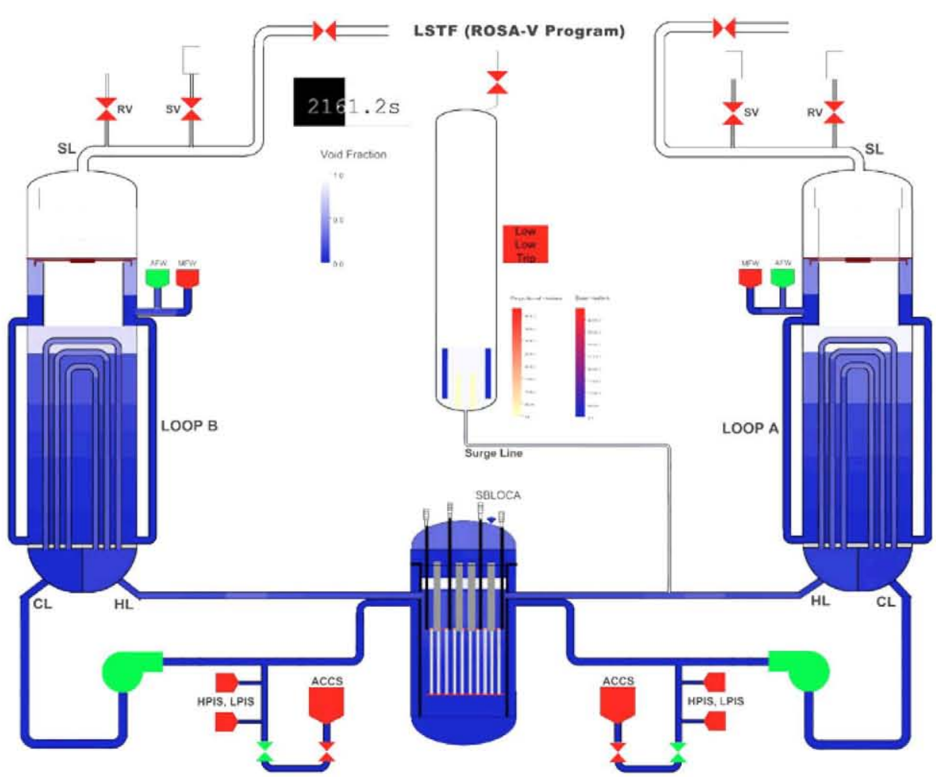

Fig. 7. SNAP mask of ROSA/LSTF. Void fraction in primary and secondary sides during an upper-head SBLOCA.

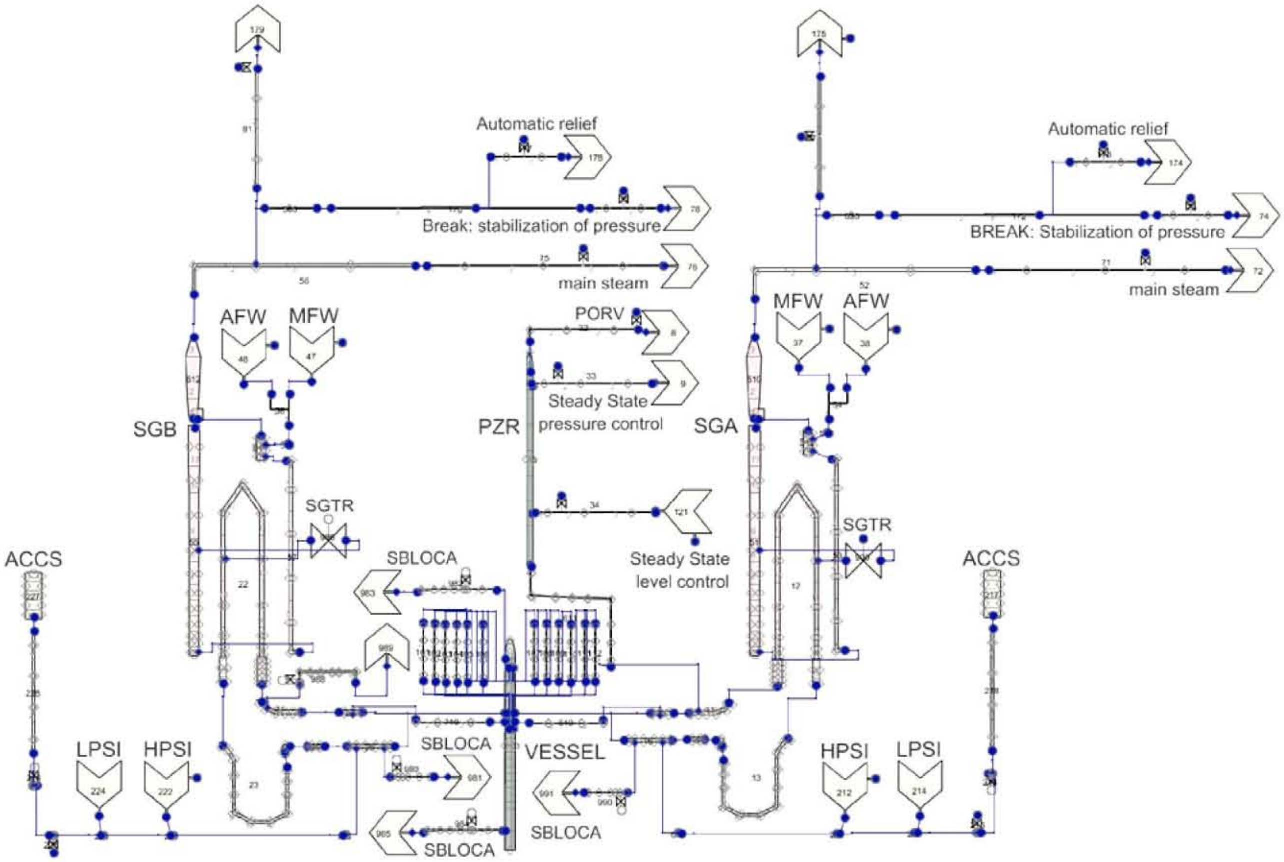

Fig. 8. TRACE/SNAP model of ROSA/LSTF.

SEPARATOR, 22 VALVE, 2 PUMP, 9 FILL, 15 BREAK, 70 HEAT STRUCTURE, and 3 POWER components), 1013 SIGNAL VARIABLES, 167 CONTROL BLOCKS, and 20 TRIPS; see Fig. 8.

The OECD/NEA ROSA project, which started in 2005 by agreement among the Japan Atomic Energy Agency (JAEA), OECD/NEA, and 13 member countries, has conducted an SBLOCA test (Test 6-1, SB-PV-09 at JAEA). This test simulates a PWR vessel top-break SBLOCA assuming a total failure of the HPSI with a break size equivalent to a 1.9% cold-leg break; see Table I for more details on the sequence. The objective of the test is to study the effect of accident management action and to provide integral test data for assessment and development of advanced analytical codes.

OECD/NEA ROSA Test 6-1 was conducted on November 17, 2005, using the LSTF at JAEA. At the beginning of the test, a rather large break and core uncovery caused fast primary depressurization, which resulted in the primary pressure being far lower than the steam generator secondary-side pressure when an accident management action was initiated by fully opening of the steam generator relief valves following the detection of high CET temperature ($T > 623$ K). The peak CET temperature appeared at the center. The accident management action was ineffective in the early stage until the steam generator secondary-side pressure decreased to the primary pressure. The LSTF core protection system automatically decreased the core power to 10% of

TABLE I

Description of OECD/NEA ROSA Test 6.1—
Chronology of Major Events and Procedures

Event	Time (s)
Break valve open	0
SCRAM signal (low primary pressure)	26
Core uncovery	≈800
Beginning of secondary-side depressurization due to high CET temperature	1090
Initiation of core protection system due to high cladding temperature	1200
Accumulators' injection	1300
LPSI starts	2900
End of experiment (valve closed)	3266

the decay power level as the maximum fuel rod surface temperature exceeded the core protection limit ($T >$ 958 K).

As can be seen in Figs. 9 through 13, the test was correctly simulated with the TRACE model. The primary and secondary pressures match the experimental result fairly well. The core uncover behavior and the CET temperature evolve the same as in the test. There was only a little delay in primary pressure compared to the test results. The data are shown normalized because they are proprietary until April 2012. Other groups participating in the OECD/NEA ROSA project also simulated this test and, in general, obtained good results.[31,32]

The results of OECD/NEA ROSA Test 6.1 showed that the accident management action of manual depressurization in the secondary system was effective, but it was late, because the temperatures at the core increased

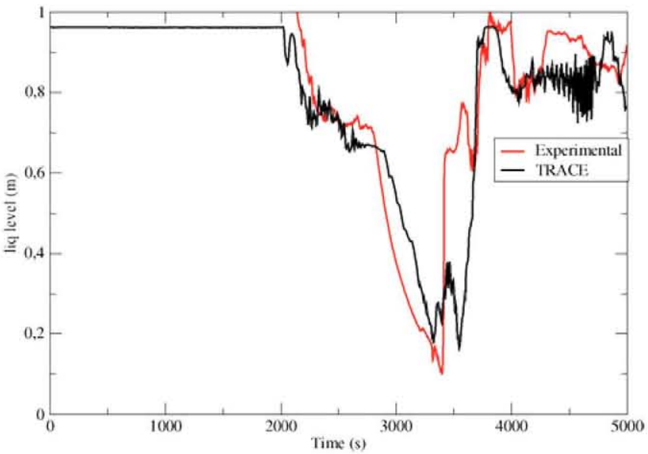

Fig. 11. Core level. Simulation of OECD/NEA ROSA Test 6.1.

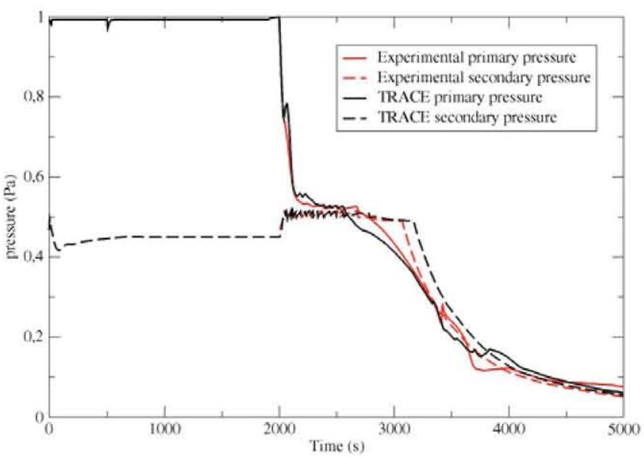

Fig. 9. Primary and secondary pressures. Simulation of OECD/NEA ROSA Test 6.1.

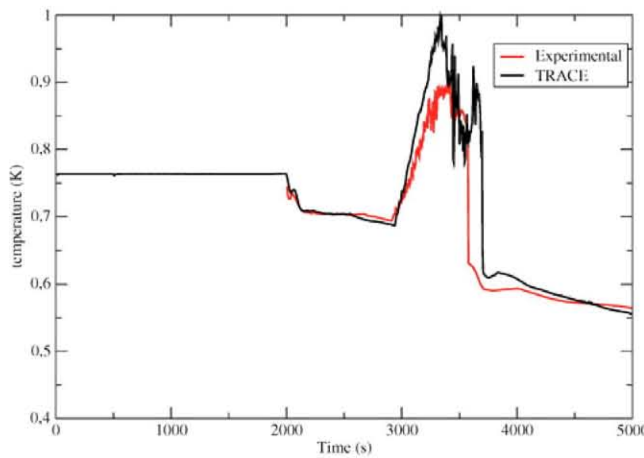

Fig. 12. Maximum CET temperature. Simulation of OECD/NEA ROSA Test 6.1.

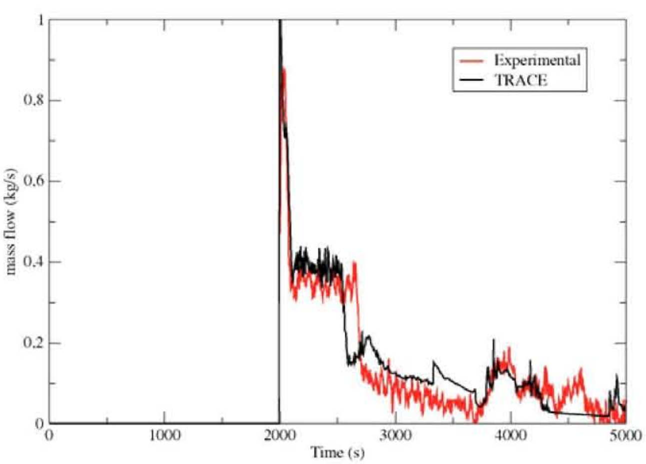

Fig. 10. Break mass flow. Simulation of OECD/NEA ROSA Test 6.1.

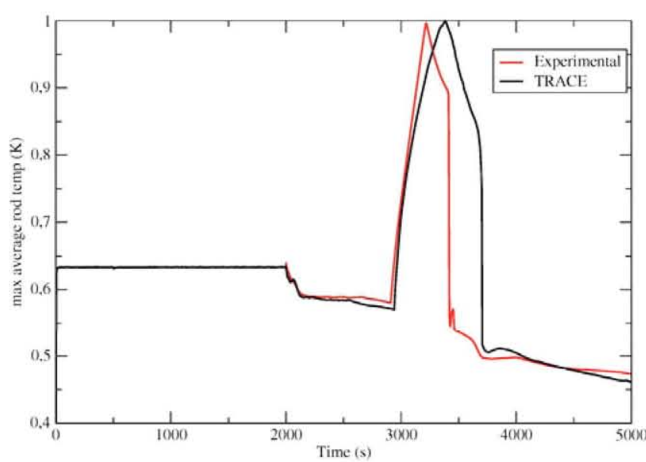

Fig. 13. Peak cladding temperature. Simulation of OECD/NEA ROSA Test 6.1.

TABLE II

Upper-Head SBLOCA Experiments Performed in LSTF

Test	Program	Conditions	Final State
SB-PV-02 (equivalent to 0.5% of cold-leg break)	ROSA-IV, May 1987	Effect of high-pressure-injection initiation was studied when temperature at hot leg reached T_{sat} + 10.0 K.	Heatup; quenched.
SB-PV-07 (1%, half-size of one CRDM nozzle ejection)	ROSA-V June 2005	Operator actions of high-pressure-injection recovery were initiated when temperature at CET reached 623 K.	Heatup; quenched
SB-PV-08 (0.1%)	ROSA-V October 2005	Steam generator depressurization (full opening of reactor vessel) was initiated when temperature at CET reached 623 K.	Heatup; quenched
SB-PV-09 (1.9%) OECD/NEA ROSA Test 6.1	ROSA-V November 2005	Steam generator depressurization (full opening of reactor vessel) was initiated when temperature at CET reached 623 K.	Heatup; power trip.

to unexpectedly high values. This issue was of concern for people involved in this test because this effect was previously detected in some former upper-head SBLOCA tests in LSTF (see Table II and Refs. 33 through 39), but the delay to detect core uncovering was never found to be so long [~230 s (Ref 37)]. This large delay made the accident management action ineffective because it was implemented too late in the transient.

As a result, at a May 2007 ROSA meeting, several technical reports and presentations raised concerns with the CETs and their role in the accident management of OECD/NEA ROSA Test 6.1 (Ref. 35). Consequently, the NEA Working Group on Analysis and Management of Accidents decided to study the problem more deeply. Many meetings took place between 2008 and 2009 (Refs. 40 through 43), but the conclusions had not been made public at the time this paper was written.

In order to analyze the delay problem, the relationship between clad and CET temperature evolution could be approximated by the linear expression $\Delta T_{CET} = C\Delta T_{CLAD} - T_0$. This analysis was performed by JAERI with several experiments; see Refs. 34, 37, and 38. The adjustment obtained for the upper-head tests and the simulation with TRACE are included in Table III.

The comparison of the slopes C shows that the simulation of OECD/NEA ROSA Test 6.1 with TRACE provides larger values of the CET temperatures than the experimental ones. The ratio between the experimental and the simulated increase of CET during core uncovering in OECD/NEA ROSA Test 6.1 is 0.65; see Fig. 12. This value was obtained from several simulations with different nodalizations, in which the most conservative value (the lowest value) was chosen. Therefore, it is necessary to take into account this relationship in the plant applications that are described later in this paper.

TABLE III

Relationship Between T_{CET} and T_{Clad} During the Upper-Head SBLOCA Tests Performed in LSTF

Test	C	T_0
SB-PV-02 (0.5%)	1.47	16.9
SB-PV-07 (1%)	1.98	28.1
SB-PV-08 (0.1%)	1.96	28.9
SB-PV-09 (1.9%) OECD/NEA ROSA Test 6.1	2.75	0
Simulation of OECD/NEA ROSA Test 6.1	1.65	2.0

IV. ALMARAZ-1 TRACE MODEL

Almaraz NPP has two PWR units; it is located in Cáceres (Spain) and is owned by a consortium of three Spanish utilities: Iberdrola (53%), Endesa (36%), and Gas Natural Fenosa (11%). The commercial operation started in April 1981 (Unit I) and in September 1983 (Unit II). Each unit is a Westinghouse three-loop PWR. The nominal power is 2739 MW(thermal) and 977 MW(electric), respectively. The original Westinghouse steam generators were replaced between 1996 and 1997, and since then, three Siemens KWU 61W/D3 steam generators have been used. The RCPs, designed by Westinghouse, are single stage and centrifugal. The auxiliary feedwater system (AFWS) consists of one turbine-driven pump and two motor-driven pumps.

The Almaraz-1 TRACE model has 255 thermal-hydraulic components (2 VESSEL, 73 PIPE, 43 TEE, 54 VALVE, 3 PUMP, 12 FILL, 33 BREAK, 32 HEAT STRUCTURE, and 3 POWER components), 740 SIGNAL

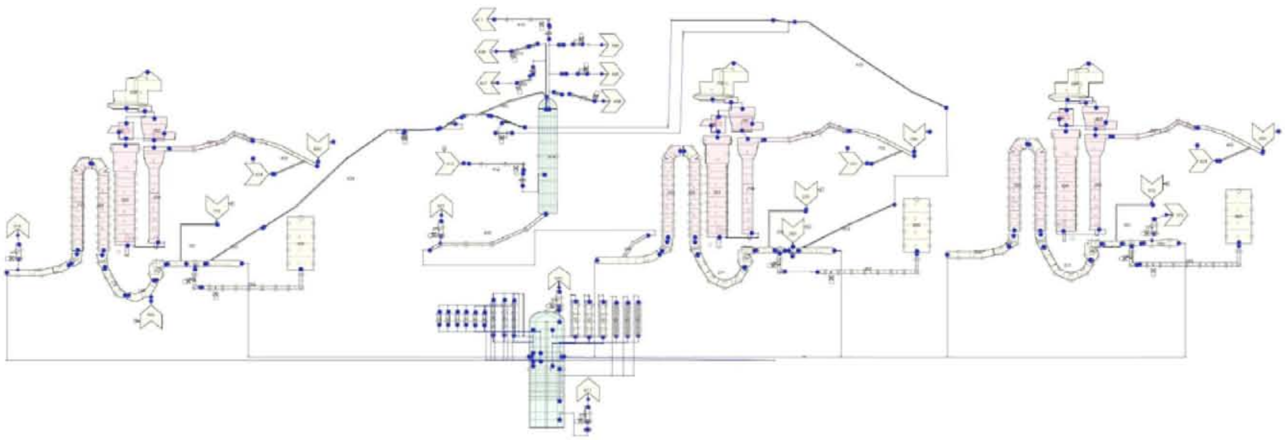

Fig. 14. Simplified scheme of the Almaraz-1 TRACE model.

VARIABLES, 1671 CONTROL BLOCKS, and 58 TRIPS; see Fig. 14.

Regarding the primary and secondary circuits, the following components have been modeled:

1. reactor vessel, modeled by a VESSEL component (Fig. 15), which includes the core region, guide tubes, support columns, core bypass, and the bypass to the vessel head via the downcomer and via guide tubes

2. The nuclear core power is modeled with axial cosine power shape distribution. The core power is distributed among nine HEAT STRUCTURE components, with one HEAT STRUCTURE per sector.

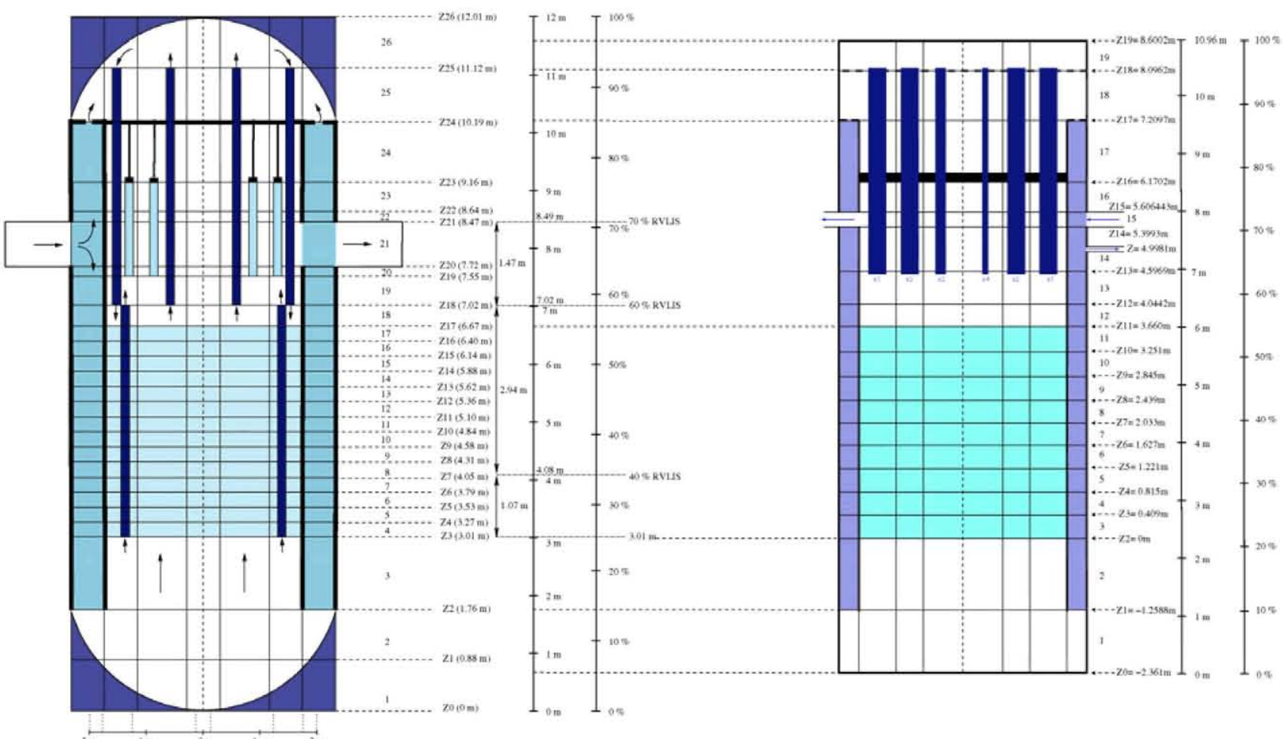

Fig. 15. Comparison of the vessel nodalizations of Almaraz-1 and ROSA/LSTF models.

3. primary circuit, including steam generators and pressurizer in loop 2 (containing heaters, relief/safety valves, and pressurizer spray system)
4. chemical and volume control system (CVCS)
5. ECCS: safety injection system and accumulators
6. steam lines up to the turbine stop valves, with the relief, safety, and isolating valves
7. steam dump with eight valves
8. feedwater system and AFWS. Feedwater pumps coast down, and auxiliary mass flows are included as boundary conditions.

The control, protection, and engineering safeguard systems and signals modeled are the following:

1. pressurizer level control, which includes the CVCS isolating discharge signal, the CVCS charge flow, and heaters
2. pressurizer pressure control, which includes proportional and backup heaters, spray lines, and pilot-operated relief valves (PORVs)
3. steam generator level control system
4. steam dump control
5. turbine control
6. protection and engineering safeguard system signals, which include the emergency shutdown system (SCRAM); safety injection; pressurizer safety valve logic; AFWS activation; relief, safety, and isolating valve logic of steam lines; normal feedwater system isolation; turbine trip; and pump trip.

This model has been validated with steady and transient conditions and verified with a large set of transients.[44–51]

In these kinds of transients, it is necessary to include the RVLIS to measure the water level in the reactor vessel as it is measured in the plant. In Almaraz-1 there are two calibrations: a dynamic calibration (with all RCPs running) and a static calibration (all RCPs tripped). A model for these RVLIS measures was implemented in the Almaraz-1 model taking into account the descriptions and model of several references: Refs. 52, 53, and 54. Figure 15 shows the relationship among several values of the RVLIS and heights in Almaraz-1 and as compared to the ROSA/LSTF vessel model.

V. UPPER-HEAD SBLOCA WITHOUT HPSI: REFERENCE CASE

In this first analysis the secondary-side cooling is not taken into account in order to check if it is necessary to avoid high cladding temperatures.

In this first group of simulations, several conditions were imposed in the model:

1. The break area is adjusted to the CRDM section of Almaraz-1 [6.985 cm (2.75 in.)].
2. No HPSI is available.
3. All accumulators are available (3/3).
4. One train of LPSI is available.
5. The main steam isolation valve is closed by high pressure inside containment.
6. There is no secondary-side depressurization.
7. An upper-head SBLOCA takes place at $t = 4650$ s.
8. There is no RCP trip.

The results show that the reference case needs secondary-side depressurization in order to avoid high cladding temperatures; see Figs. 16 and 17.

Therefore, we decided to perform a broad spectrum of sensitivity analyses with respect to several variables in order to find the most limiting cases: break area, discharge coefficient, break localization within the upper head, friction factors in the accumulators' exit, upper downcomer area, steady-state upper-head mass flows, number of accumulators available, and RCP trip delay.

The results of peak cladding temperature (PCT) sensitivity cases show the following: low sensitivity to break location, friction factors at the accumulators' exit, and steady-state upper-head mass flows; medium sensitivity to discharge coefficients and upper downcomer area; and high sensitivity with respect to break area size, RCP trip delay, and number of accumulators available.

Taking into account the results of the sensitivity analysis, we decided to carry out two analyses: first, an extensive sensitivity analysis with respect to the break area

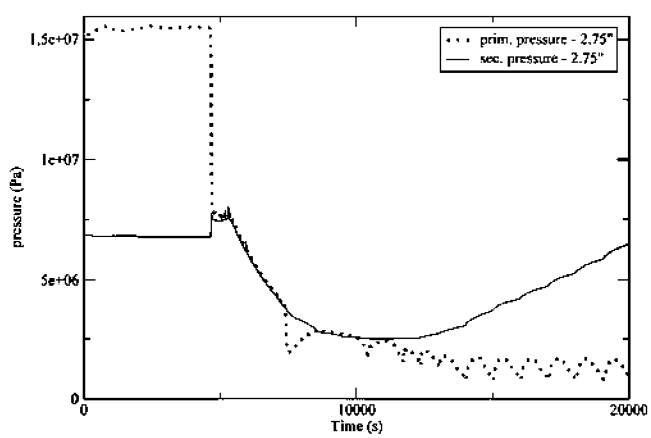

Fig. 16. Primary and secondary pressures. Reference case.

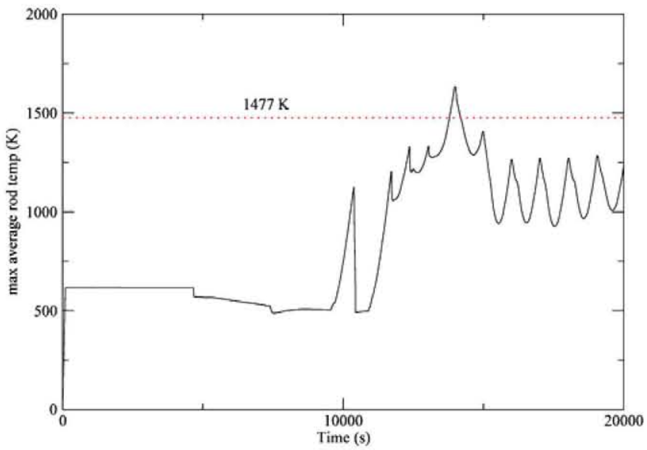

Fig. 17. Maximum cladding temperature. Reference case.

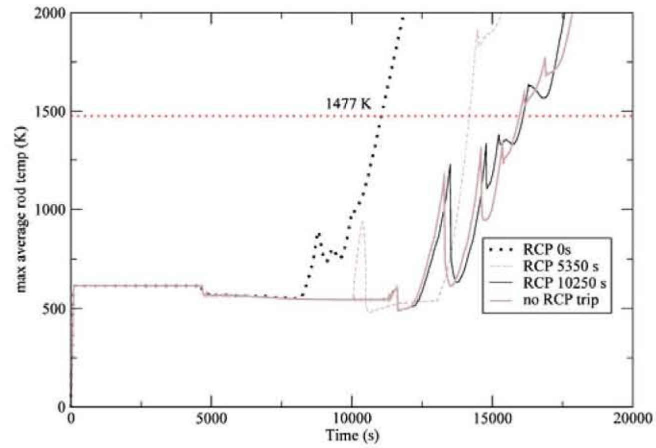

Fig. 19. Sensitivity to RCP trip delay with all accumulators available. Case with 5.1-cm (2-in.)-diam break. Cladding temperature.

size and RCP trip delay (with all accumulators available)—Secs. VI, VII, and VIII—and second, a sensitivity analysis with respect to the number of accumulators available—Secs. IX and X.

VI. SENSITIVITY ANALYSIS WITH RESPECT TO THE BREAK AREA SIZE AND RCP TRIP DELAY

An extensive analysis with respect to the break area size and RCP trip delay (with all accumulators available) was performed, as shown in Figs. 18, 19, and 20. The results shown in Table IV and Fig. 19 point out that all cases lead to core damage (PCT > 1477 K). The cladding temperature growing rate since core uncovering ($\approx$1 K/s) is similar to the experimental values from the PKL, LOFT, ROSA, PSB, and NEPTUN tests (from 0.2 to 2 K/s); see Refs. 21, 38, 39, and 55 through 58.

These analyses confirm the necessity of secondary-side depressurization at 55 K/h cooling of the primary circuit in order to avoid core damage. Therefore, a new analysis including secondary-side depressurization was performed. This analysis is described in Sec. VII.

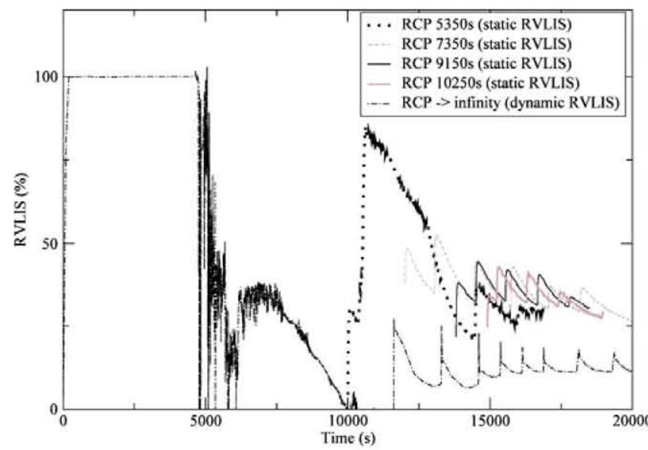

Fig. 20. Sensitivity to RCP trip delay with all accumulators available. Case with 5.1-cm (2-in.)-diam break. RVLIS.

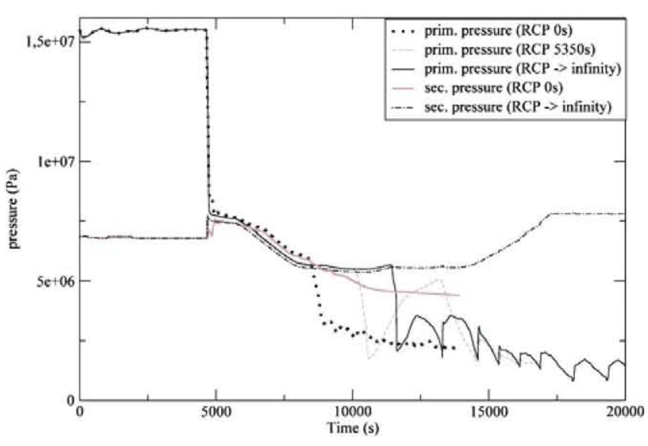

Fig. 18. Sensitivity to RCP trip delay with all accumulators available. Case with 5.1-cm (2-in.)-diam break. Primary and secondary pressures.

VII. UPPER-HEAD SBLOCA WITHOUT HPSI: CASES WITH RCS COOLING RATE OF 55 K/h

As was mentioned earlier, the operator follows EOP E-0, EOP E-1, and EOP ES-1.2 in this kind of sequence. In EOP E-1, step 11, the operator checks the primary pressure, and if it is not below 15 bars, there is

TABLE IV

Time to Damage in Upper-Head SBLOCA Without Depressurization

Break Diameter	RCP Trip Delay from SCRAM (s)					
	0	5350	7350	9150	10 250	No Trip
2.5 cm (1 in.)	29 120	27 343	27 366	27 422	27 438	42 171
5.1 cm (2 in.)	13 938	14 179	15 536	16 630	16 094	15 966
6.985 cm (2.75 in.)	14 192	13 296	13 874	13 641	16 222	13 790

a transition to EOP ES-1.2. In EOP ES-1.2, the operator will cool and depressurize the primary system, opening the relief valves in the secondary side, making sure to control that the primary cooling is close to 55 K/h. The time needed for the operators to arrive at EOP ES-1.2 is ~600 s from reactor scram.[59]

The simulations were performed including a manual cooling control system in the Almaraz-1 model. Several options of this manual control have been checked similar to the models described in Ref. 60. After adjusting the parameters of all the manual control models, a proportional control was selected.

The results of these simulations are shown in Figs. 21 through 26. The condition of damage or success for all these transients is included in Fig. 27. The region of Fig. 27 in which there are damage conditions is defined as the damage domain of the sequence; this kind of diagram is used as part of the Integrated Safety Assessment methodology developed by the Modelization and Simulation Area of the Spanish Nuclear Safety Council [Consejo de Seguridad Nuclear (CSN)]; see Refs. 61, 62, and 63 for more details of this methodology. Figures 27 and 28 show that the damage domain corresponds only to the cases with early RCP trip, $t < 10$ min, and break size close to maximum break size [6.985 cm (2.75 in.)] could lead to core damage. These results show that the present EOPs are adequate for this kind of sequence because if the operator follows these EOPs, i.e., does not trip RCP

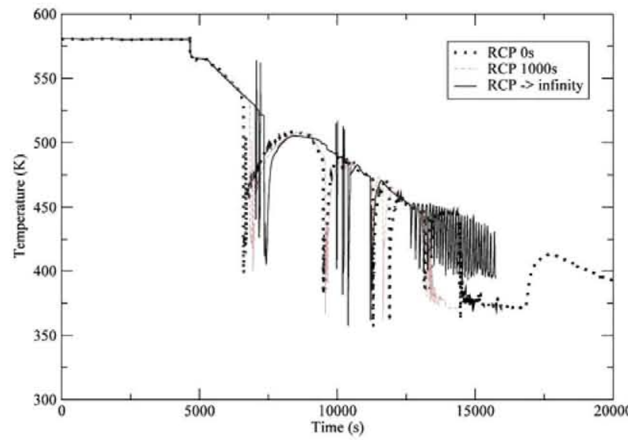

Fig. 22. Sensitivity to RCP trip delay with all accumulators available and 55 K/h primary cooling. Case with 6.985-cm (2.75-in.)-diam break. Average primary-side temperature.

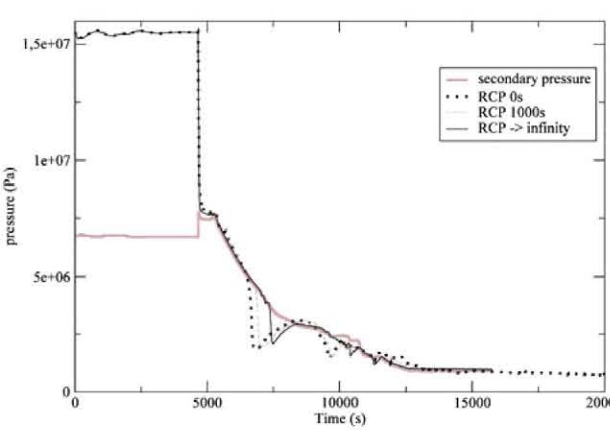

Fig. 21. Sensitivity to RCP trip delay with all accumulators available and 55 K/h primary cooling. Case with 6.985-cm (2.75-in.)-diam break. Primary and secondary pressures.

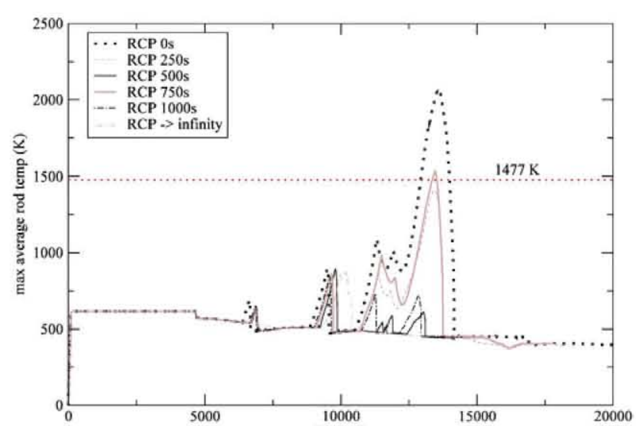

Fig. 23. Sensitivity to RCP trip delay with all accumulators available and 55 K/h primary cooling. Case with 6.985-cm (2.75-in.)-diam break. Peak cladding temperature.

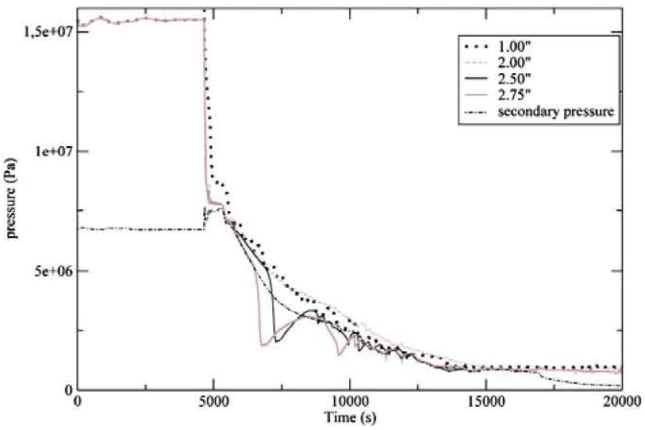

Fig. 24. Sensitivity to break size with all accumulators available and 55 K/h primary cooling. Cases with RCP trip simultaneously with SBLOCA. Primary and secondary pressures.

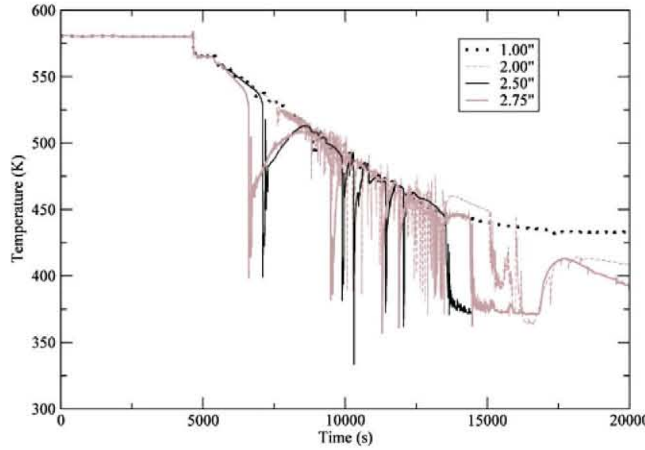

Fig. 26. Sensitivity to break size with all accumulators available and 55 K/h primary cooling. Cases with RCP trip simultaneously with SBLOCA. Average primary-side temperature.

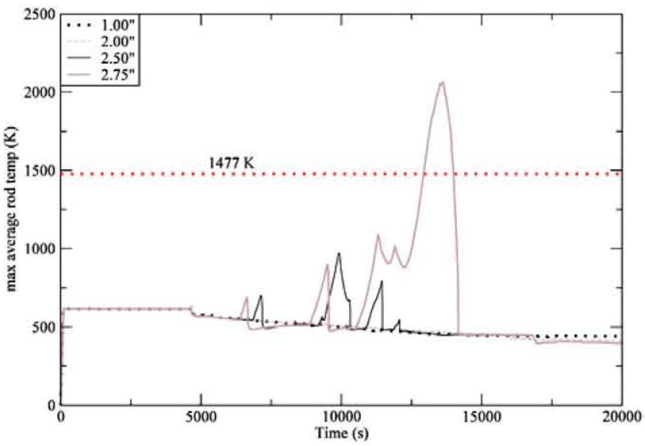

Fig. 25. Sensitivity to break size with all accumulators available and 55 K/h primary cooling. Cases with RCP trip simultaneously with SBLOCA. Peak cladding temperature.

and perform the 55 K/h primary-side cooling, there is no fuel damage.

However, if there is an operator error related to inadequate manual operation, it is still possible that RCP trip will arrive at damage conditions. Therefore, a new analysis has been performed for the transients that lead to damage conditions, and it is described in Sec. VIII.

VIII. UPPER-HEAD SBLOCA WITHOUT HPSI: CASES WITH FULL OPENING OF RELIEF VALVES

In the damage transients discussed in Sec. VII, all the transients included the manual actions corresponding to EOP ES-1.2, like a primary-side cooling rate of 55 K/h. Therefore, it is interesting to analyze the transition to EOP FR C.1. The transition to EOP FR C.1 requires the following conditions:

1. $T_{CET} > 921$ K or

2. $T_{CET} > 649.13$ K, $VL < 40\%$, and all RCPs tripped.

The time of transition to EOP FR C.1 corresponding to the transients of the damage domain of Sec. VII is shown in Table V and has been obtained from Figs. 29 and 30 (not all simulated cases are shown in these figures).

Some of the results of the previous damage transients with full opening of all secondary-side relief valves (3/3) at the time of EOP FR C.1 transition (plus an assumed operator delay of 60 s) are shown in Figs. 31 and 32. In all the cases, fully opening the relief valves avoids the damage limit, as shown in Fig. 31.

However, the simulated CET temperature being higher than the real one during core uncovering, as is described in Sec. III, must be taken into account. Therefore, we decided to obtain the available time to fully open all of the secondary-side relief valves for all the cases arriving at core damage with 55 K/h secondary-side cooling and to compare this time with the corrected transition time to EOP FR C.1. The corrected transition time is obtained taking into account that the ratio between the experimental and the simulated CET temperature for OECD/NEA ROSA Test 6.1 is 0.65; see Sec. III for more details. Therefore, if the values of the simulated CET temperatures are corrected, a new transition time to EOP FR C.1 is obtained; see Table VI and Fig. 33. The results show that there is a large margin of time between the corrected transition time to EOP FR C.1 and the available time to begin EOP FR C.1.

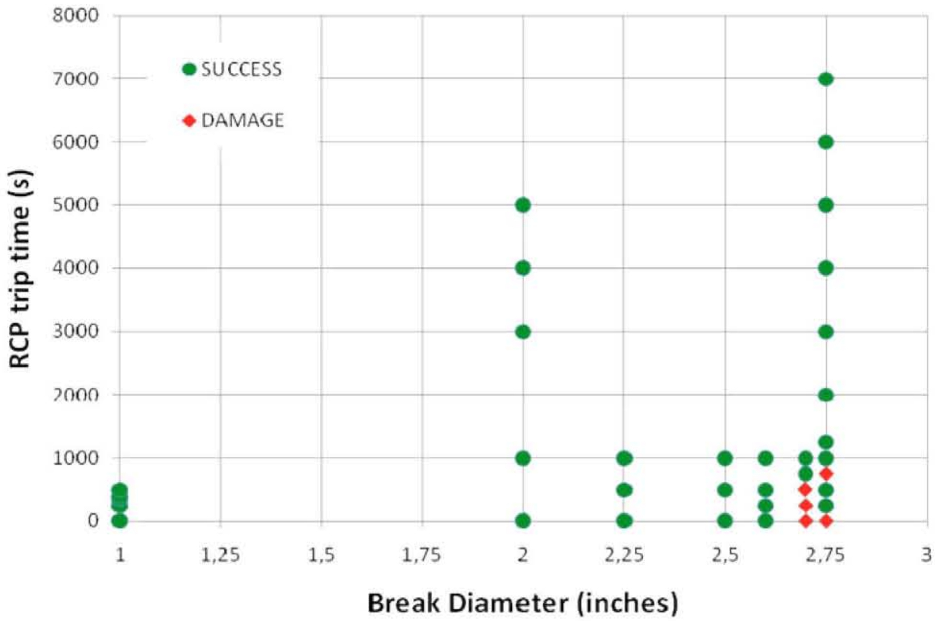

Fig. 27. Sensitivity to RCP trip delay and break size with all accumulators available and 55 K/h primary cooling. Damage domain of the sequence.

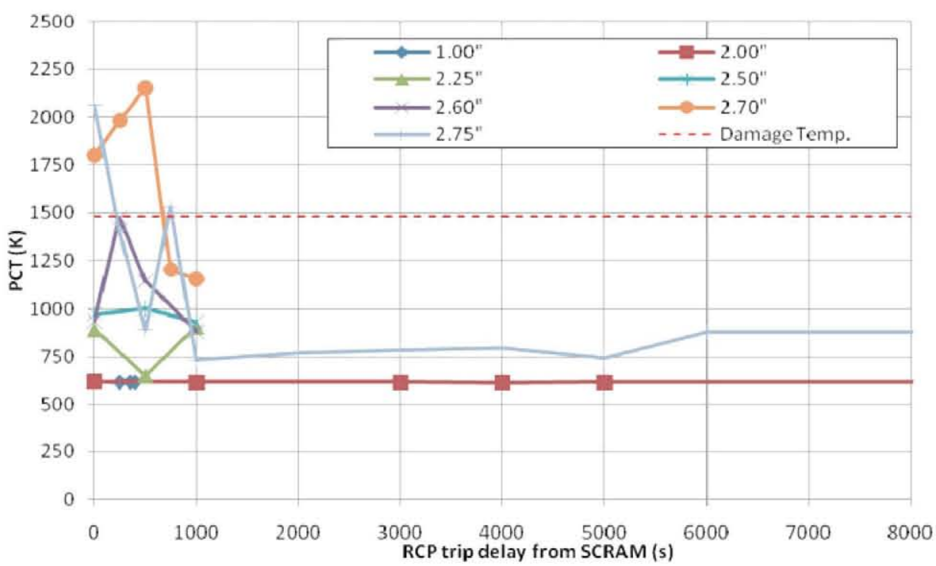

Fig. 28. Sensitivity to RCP trip delay and break size with all accumulators available and 55 K/h primary cooling. Peak cladding temperature.

Therefore, these results also show that the present EOPs are adequate for this kind of transient, also in the case of an operator mistake related to an inadequate manual RCP trip.

These results have been obtained with the availability of all accumulators (3/3). In order to complete the analysis, a new sensitivity study with respect to the number of accumulators available was performed, as discussed in Sec. V.

IX. UPPER-HEAD SBLOCA WITHOUT HPSI: CASES WITH FULL OPENING OF RELIEF VALVES: SENSITIVITY TO THE NUMBER OF ACCUMULATORS AVAILABLE

In this sensitivity analysis the worst previous case with a cooling rate of 55 K/h was selected. This case corresponds to the sequence with a break size of 6.985 cm (2.75 in.) and RCP trip at the same time as the SBLOCA event, because the time between conditions for EOP FR

TABLE V

Transition Conditions to EOP FR C.1

Break Size	Time of RCP Trip from SCRAM (s)	Time (s) When $T_{\mathrm{CET}} > 921.93$ K	Time (s) When $T_{\mathrm{CET}} > 649.13$ K	Time (s) When RVLIS $< 40\%$	Transition Time to EOP FR C.1 (s)	Damage Without EOP FR C.1 Accident Management (s)
6.90 cm (2.70 in.)	0	11 260	6684	6478	6684	13 007
6.90 cm (2.70 in.)	250	11 018	6626	6436	6626	12 880
6.90 cm (2.70 in.)	500	10 984	6608	6406	6608	12 764
6.985 cm (2.75 in.)	0	11 223	9128	6421	9198	12 928
6.985 cm (2.75 in.)	750	11 475	9423	6728	9423	13 467

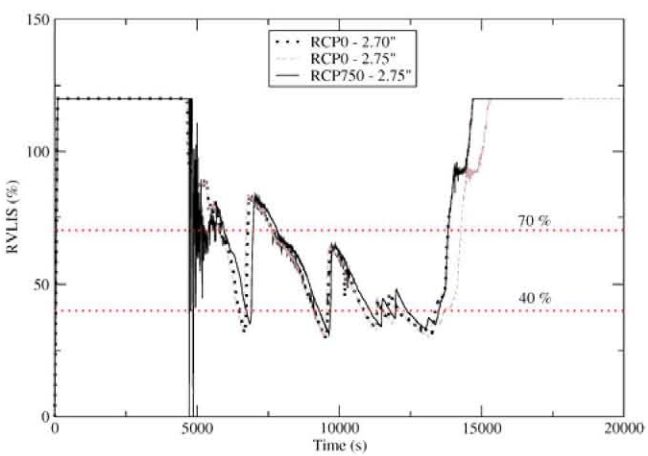

Fig. 29. Vessel level. RVLIS. Damage cases with three accumulators and 55 K/h primary cooling.

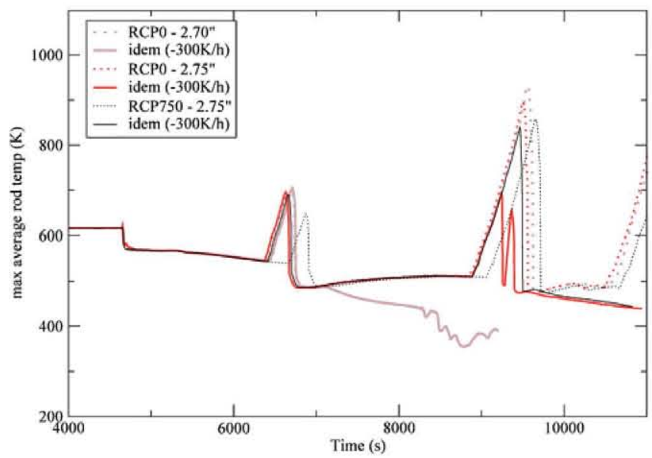

Fig. 31. Comparison of the cases of the damage domain with secondary-side cooling at 55 K/h and fully opened relief valves. Peak cladding temperature.

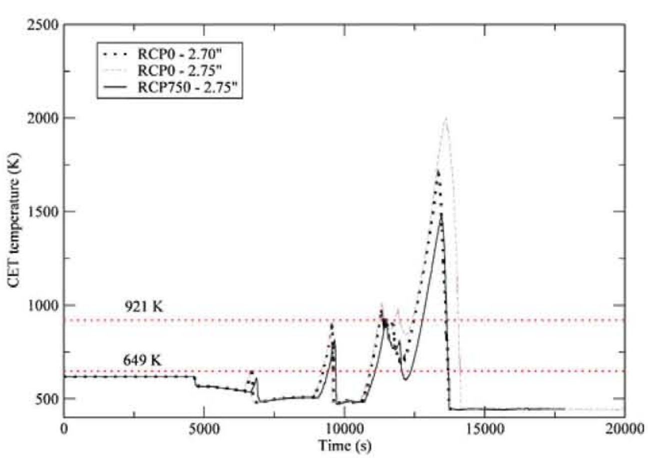

Fig. 30. Maximum CET temperature. Damage cases with three accumulators and 55 K/h primary cooling.

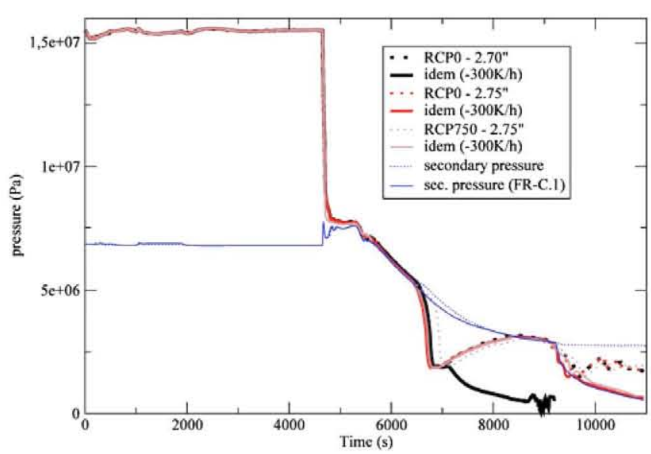

Fig. 32. Comparison of the cases of the damage domain with secondary-side cooling at 55 K/h and fully opened relief valves. Primary and secondary pressures.

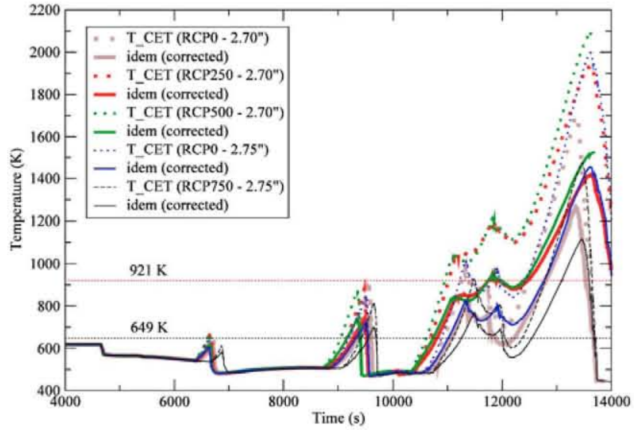

Fig. 33. Corrected values of CET temperatures for the cases with core damage.

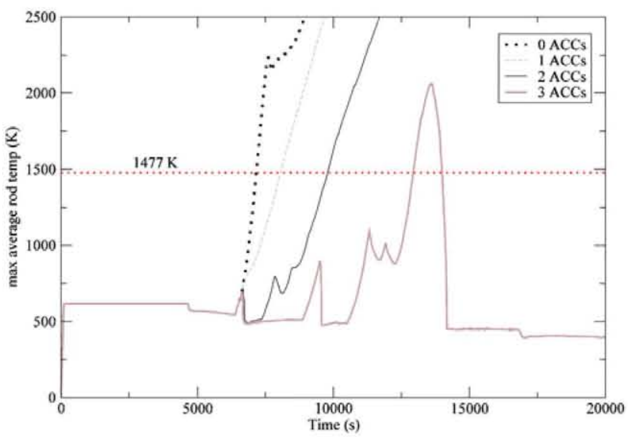

Fig. 34. Sensitivity to the number of accumulators available and 55 K/h primary cooling. Cases with RCP trip simultaneously with SBLOCA. Peak cladding temperature.

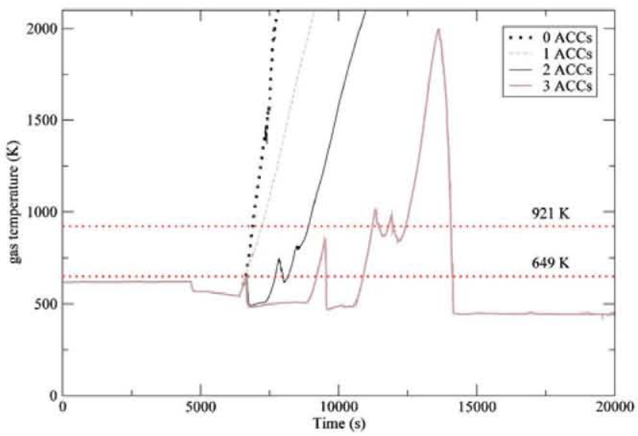

Fig. 35. Sensitivity to the number of accumulators available and 55 K/h primary cooling. Cases with RCP trip simultaneously with SBLOCA. CET temperature.

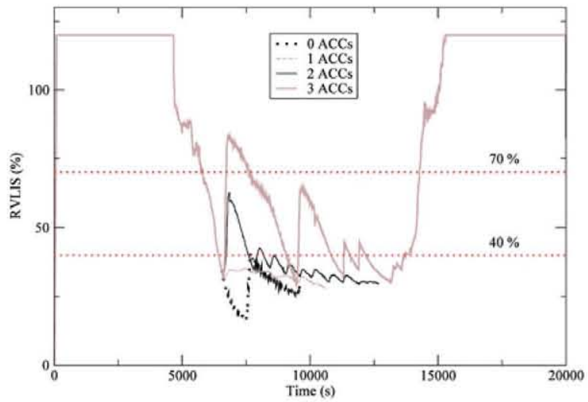

Fig. 36. Sensitivity to the number of accumulators available and 55 K/h primary cooling. Cases with RCP trip simultaneously with SBLOCA. Vessel level (static RVLIS).

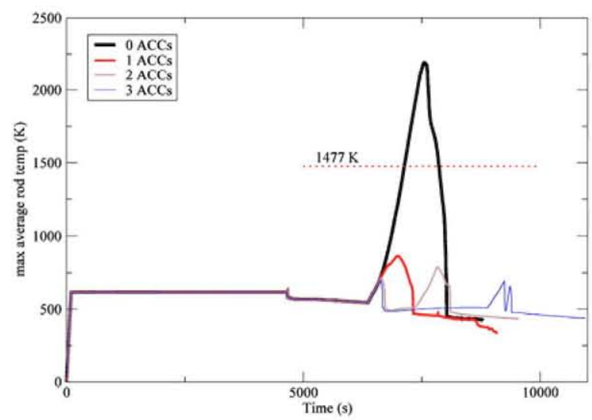

Fig. 37. Sensitivity to the number of accumulators available and full opening of reactor vessel. Cases with RCP trip simultaneously with SBLOCA. Peak cladding temperature.

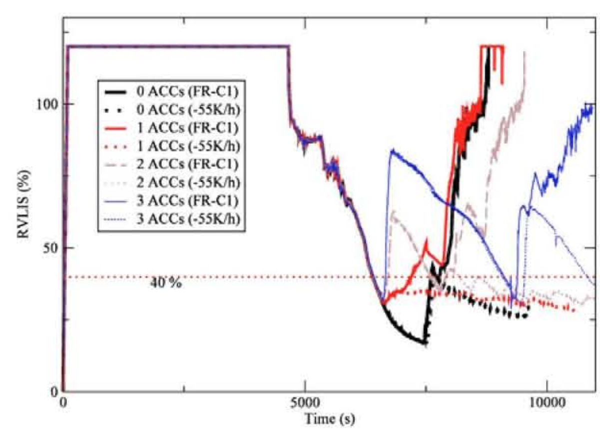

Fig. 38. Sensitivity to the number of accumulators available and full opening of reactor vessel. Cases with RCP trip simultaneously with SBLOCA. RVLIS.

TABLE VI

Transition Conditions for EOP FR C.1 with Corrected CET Temperatures

Break Size	Time of RCP Trip from SCRAM (s)	Transition Time to EOP FR C.1 (s)	Corrected Transition Time to EOP FR C.1 (s)	Available Time to Begin EOP FR C.1 (s)	Damage Without EOP FR C.1 (s)
6.90 cm (2.70 in.)	0	6684	9363	12 877	13 007
6.90 cm (2.70 in.)	250	6626	9287	12 626	12 880
6.90 cm (2.70 in.)	500	6608	9166	12 108	12 764
6.985 cm (2.75 in.)	0	9198	9330	12 448	12 928
6.985 cm (2.75 in.)	750	9423	9545	12 173	12 879

C.1 transition and core damage is the smallest of all cases; see Table V. The results shown in Fig. 34 show that all cases with three, two, one, and no accumulators lead to core damage. Therefore, the RVLIS and CET temperature shown in Figs. 35 and 36 are analyzed in order to obtain the transition times to EOP FR C.1; see Table VII. The results with full opening of all relief valves with these transition times show that only the case with no accumulators leads to core damage conditions, as shown in Figs. 37 and 38 and Table VII. In this case the CET temperatures have not been corrected, like in Sec. VIII, because those results have shown a large time margin of actuation in all cases except in the case of no accumulators.

X. UPPER-HEAD SBLOCA WITHOUT HPSI: CASES WITH FULL OPENING OF RELIEF VALVES: MODIFIED STATUS TREE F.0.2

In the previous sensitivity analysis with respect to the number of accumulators available, it was found that

TABLE VII

Times at Which T_{CET} and VL Reach the Transition Values in Current Status Tree F.0.2

Accident Management Condition	Time (s) When RVLIS $<$ 40%	Time (s) When $T_{CET} > 649.13$ K	Time (s) When $T_{CET} > 921.93$ K	Transition Time to EOP FR C.1 (s)	Damage Without EOP FR C.1 (s)	Damage with EOP FR C.1 Transition (s)
Three accumulators	6421	9198	11 223	9198	12 928	—
Two accumulators	6421	6624	8 911	6624	9 784	—
One accumulator	6421	6631	7 217	6631	8 051	—
No accumulators	6421	6628	6 880	6628	7 157	7172

TABLE VIII

Proposed Modifications to Status Tree F.0.2

	Modification	Conditions to EOP FR C.1 Transition
Current status tree F.0.2	None	$T_{CET} > 921$ K or ($T_{CET} > 649.13$ K and $VL < 40\%$)
Mod 1	Change level condition from 40% to 70%.	$T_{CET} > 921$ K or ($T_{CET} > 649.13$ K and $VL < 70\%$)
Mod 2	Eliminate T_{CET} condition in EOP FR C.1 transition.	$VL < 40\%$
Mod 3	Eliminate T_{CET} condition in EOP FR C.1 transition, and change level condition from 40% to 70% (level below hot legs; see Fig. 15).	$VL < 70\%$

only the case with no accumulator available could lead to core damage (PCT > 1477 K); see Fig. 37. However, in all cases, core uncovering was observed. Therefore, we analyze several modifications of Status Tree F.0.2 transition conditions in order to avoid core damage and core uncovering in all cases.

Taking into account that RCPs are always tripped when the operator checks the condition $T_{CET} > 649.13$ K, as shown in Fig. 35, there are only a few possible FRG transition conditions to EOP FR C.1 in the current Status Tree F.0.2; see Table VIII. The modifications proposed by the Universidad Politécnica de Madrid are similar to those proposed by several members of the OECD/NEA ROSA project; see Refs. 35 and 37.

The transition time to EOP FR C.1 can be obtained from the data shown in Table IX; see Table X. These transition times point out that it is necessary to simulate only the transients corresponding to Modification 2 (Mod 2) and Modification 3 (Mod 3).

The results of Mod 2 and Mod 3 point out that Mod 2 gives better results than the current Status Tree F.0.2 (see Table XI), but Mod 2 and Mod 3 do not avoid core damage for the case of no accumulators nor do they avoid core uncovering in the other cases; see Figs. 39 and 40. On the other hand better results are obtained with Mod 3 because core damage is avoided for the case of no accumulators; core uncovering is also avoided in the other cases; see Figs. 41 and 42. Therefore, the only modification of interest could be Mod 3, but one should take into account that this management action (three PORVs fully opened) is very aggressive and the cooling rate of the primary side could be very large. Additionally and according to some references, the operator behaviour must be conservative, and decisions must be based on more

TABLE IX

Times at Which T_{CET} and VL Reach the Transition Values in the Current Status Tree F.0.2 and Proposed Modified Versions

Accumulator(s) Available	Time (s) When RVLIS < 70%	Time (s) When RVLIS < 40%	Time (s) When $T_{CET} > 649.13$ K	Time (s) When $T_{CET} > 921.93$ K
Three accumulators	5721	6421	9198	11 223
Two accumulators	5721	6421	6624	8 911
One accumulator	5721	6421	6631	7 217
No Accumulators	5721	6421	6628	6 880

TABLE X

Time of Transition for Each FRG (EOPs FR C.1 and FR C.2)

Modifications in Status Tree F.0.2	Transition to EOP FR C.1 (s) Three Accumulators	Transition to EOP FR C.1 (s) Two Accumulators	Transition to EOP FR C.1 (s) One Accumulator	Transition to EOP FR C.1 (s) No Accumulators
Current Status Tree F.0.2	9198	6624	6631	6628
Mod 1: Change RVLIS setpoint to 70%.	9198	6624	6631	6628
Mod 2: Eliminate $T_{CET} < 649$ K condition.	6421	6421	6421	6421
Mod 3: Change RVLIS setpoint to 70%, and eliminate $T_{CET} < 649$ K condition.	5721	5721	5721	5721

TABLE XI

Peak Cladding Temperature for Each Case

	PCT at −55 K/h (K)	PCT with FR-C.1 (K)	PCT with Mod 2 (K)	PCT with Mod 3 (K)
Three accumulators	2063	696	620	620
Two accumulators	>3000	790	634	620
One accumulator	>3000	864	710	620
No accumulators	>3000	2192	2005	683

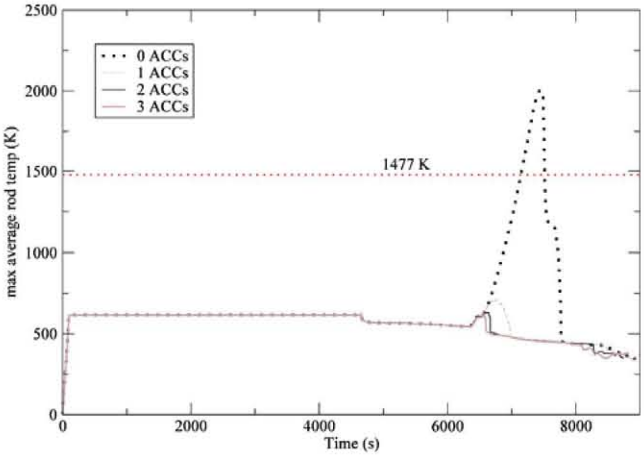

Fig. 39. Modification 2 of Status Tree F.0.2. Maximum cladding temperatures.

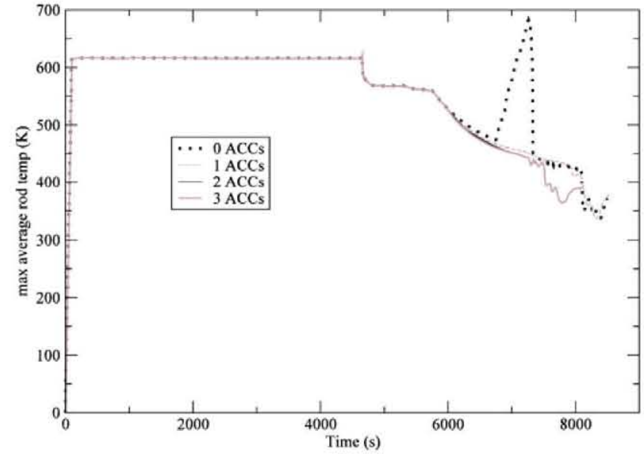

Fig. 41. Modification 3 of Status Tree F.0.2. Maximum cladding temperatures.

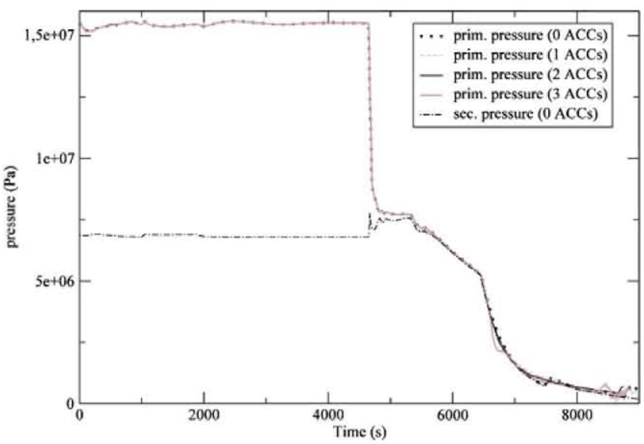

Fig. 40. Modification 2 of Status Tree F.0.2. Primary and secondary pressures.

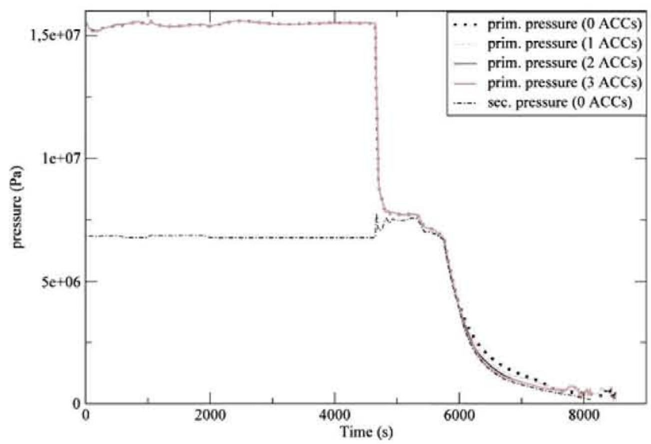

Fig. 42. Modification 3 of Status Tree F.0.2. Primary and secondary pressures.

than just vessel level measurement; see Refs. 64 and 65 for more details on this issue.

XI. CONCLUSIONS

The main conclusions of this research are the following:

1. With the actual procedures (Westinghouse design), only the case with HPSI failed and no accumulators available could lead to core damage. This conclusion has been obtained taking into account the correction of the simulated CET temperatures. This correction has been performed considering the most conservative relationship among the experimental data and several simulated cases. This kind of conservative correction could avoid the necessity of more detailed codes.

2. Changing the RVLIS condition (from 40% to 70%) and eliminating the CET condition in the EOP FR C.1 transition lead to better results than those with present transitions to EOP FR C.1. However, it should be taken into account that this management action (three PORVs fully opened) is very aggressive and the cooling rate of the primary side could be very large.

At present, it does not seem necessary to change Status Tree F.0.2 for the Westinghouse PWR design, because such change is useful only for SBLOCA accidents with HPSI failed and no accumulators available. Also, it must be taken into account that the full opening of three PORVs is a very aggressive management action and does not seem to be necessary in other accidents with higher probability, like SBLOCA with HPSI failed and one or more accumulators available.

ACKNOWLEDGMENTS

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REFERENCES

1. "Davis-Besse Reactor Pressure Vessel Head Degradation. Overview, Lessons Learned, and NRC Actions Based on Lessons Learned," NUREG/BR-0353, Rev. 1, U.S. Nuclear Regulatory Commission (Aug. 2008).

2. "Root Cause Analysis Report—Significant Degradation of the Reactor Pressure Vessel Head," CR 2002-0891 (Apr. 15, 2002) and CR 2002-0891, Rev.1 (Aug. 27, 2002), First Energy Nuclear Operating Company, Davis Besse Nuclear Power Station.

3. U.S. Nuclear Regulatory Commission Operating Experience Web Site: http://www.nrc.gov/reactors/operating/ops-experience/pressure-boundary-integrity/upper-head-issues/references-upper-head-issues.html (current as of July 14, 2010).

4. "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," NRC Generic Letter 97-01, U.S. Nuclear Regulatory Commission.

5. "Introduction to Accident Analysis," TECNATOM Operating Practices Course, PF3T-LA-M12 (1999) (in Spanish).

6. "Inadequate Core Cooling Studies of Scenarios with Feedwater Available," WCAP-9754, Westinghouse Electric Corporation (1980) (nonproprietary version).

7. "Analysis of Delayed Coolant Pump Trip During Small Loss of Coolant Accidents for Westinghouse Nuclear Steam Supply Systems," WCAP-9585, Westinghouse Electric Corporation (1979) (nonproprietary version).

8. B. SHERON, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors, NUREG-0623, U.S. Nuclear Regulatory Commission (1979).

9. "Systems Description of Trillo NPP," TECNATOM (1988) (in Spanish).

10. "EPR Design Description," Framatome ANP (2005).

11. "AP1000 Design Control Document," Rev. 17, Westinghouse Electric Company (2008).

12. "A Probabilistic Safety Assessment of the Standard French 900 MWe Pressurized Water Reactor," Commissariat à l'Energie Atomique (1990).

13. "AP1000 Probabilistic Risk Assessment Report," Rev. 1, Westinghouse Electric Company (2003).

14. Y. HAYASHI, G. SAIU, and R. F. WRIGHT, "Development of Emergency Response Guidelines (ERGs) for AP1000," *Proc. ICAPP'06*, Reno, Nevada, June 4–8, 2006, American Nuclear Society (2006) (CD-ROM).

15. D. G. EISENHUT, "Clarification of TMI Action Plan Requirements," NUREG 0737, U.S. Nuclear Regulatory Commission (1980).

16. "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Regulatory Guide 1.97, Rev. 3, U.S. Nuclear Regulatory Commission (1983).

17. "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Regulatory Guide 1.97, Rev. 4, U.S. Nuclear Regulatory Commission (2006).

18. IEEE Std. 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

19. R. J. LUTZ, "Post Accident Monitoring Instrumentation Re-Definition for Westinghouse NSSS Plants," WCAP 15981-NP, Westinghouse Electric Company (2004).

20. D. G. EISENHUT, "Inadequate Core Cooling Instrumentation System," Generic Letter 82-28, U.S. Nuclear Regulatory Commission (1982).

21. J. P. ADAMS and G. E. McCREERY, "Detection of Inadequate Core Cooling with Core Exit Thermocouples: LOFT PWR Experience," NUREG/CR 3386, EG&G Idaho (1983).

22. J. P. ADAMS and G. E. McCREERY, "Limitations of Detecting Inadequate Core Cooling with Core Exit Thermocouples," *Trans. Am. Nucl. Soc.*, **46**, 474 (1984).

23. "Modern Instrumentation and Control for Nuclear Power Plants: A Guidebook," Technical Reports Series 387, International Atomic Energy Agency (1999).

24. "Westinghouse Evaluation of RVLIS Performance at the Semiscale Test Facility," Westinghouse Electric Corporation (Dec. 1981).

25. S. KAERCHER, "New Reactor Water Level Instrumentation for PWR," presented at 32nd Annual Mtg., Tarragona, Spain, 2006, Spanish Nuclear Society (2006).

26. R. L. ANDERSON, J. L. ANDERSON, and G. N. MILLER, "Inadequate Core Cooling Instrumentation Using Heated Junction Thermocouples for Reactor Vessel Level Measurement," NUREG/CR 2627, Oak Ridge National Laboratory (1982).

27. ROSA-V GROUP, "ROSA-V Large Scale Test Facility (LSTF) System Description for the Third and Fourth Simulated Fuel Assemblies," JAERI-Tech 2003-037, Japan Atomic Energy Research Institute (Mar. 2003).

28. C. QUERAL, J. BARRERA, G. JIMENEZ, P. NIESUTTA, L. VALLE, and A. EXPÓSITO, "Simulation of OECD/ROSA

Tests 6.1 and 6.2," presented at 6th Mtg. Programme Review Group and Management Board of the OECD-NEA ROSA Project, Tokai-mura, Japan, November 7–8, 2007.

29. G. JIMENEZ, C. QUERAL, J. BARRERA, A. EXPOSITO, P. NIESUTTA, and L. VALLE, "Development of ROSA Model for TRACE 5.0," presented at 6th Mtg. Programme Review Group and Management Board of the OECD-NEA ROSA Project, Tokai-mura, Japan, November 7–8, 2007.

30. "TRACE V5.0 Theory Manual—Field Equations, Solution Methods and Physical Models," U.S. Nuclear Regulatory Commission (2007).

31. V. ABELLA, S. GALLARDO, and G. VERDÚ, "Comparison of Different Versions of TRACE5 Code in the Simulation of LSTF (ROSA V)," presented at Nuclear and Renewable Energy Conf. (INREC), Amman, Jordan, 2010.

32. J. FREIXA and A. MANERA, "Analysis of an RPV Upper Head SBLOCA at the ROSA Facility Using TRACE," *Nucl. Eng. Des.*, **240**, *7*, 1779 (July 2010).

33. M. SUZUKI, "Break Location Effects on PWR Small Break LOCA Phenomena—Inadequate Core Cooling in Lower Plenum Break Test at LSTF," JAERI-M 88-271, Japan Atomic Energy Research Institute (Jan. 1989).

34. M. SUZUKI et al., "CET Performance at ROSA/LSTF Tests—Twelve Tests with Core Heat-Up," JAEA-Research 2009-011, Japan Atomic Energy Agency (July 2009).

35. "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, Japan Atomic Energy Agency (Feb. 2008) (proprietary report to be released Apr. 2012).

36. M. SUZUKI, T. TAKEDA, and H. NAKAMURA, "ROSA-V/LSTF Vessel Top Head LOCA Tests SB-PV-07 and SB-PV-08 with Break Sizes of 1.0 and 0.1% and Operator Recovery Actions for Core Cooling," JAEA-Research 2009-057, Japan Atomic Energy Agency (2009).

37. 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," *J. Power Energy Systems*, **3**, *1*, 146 (2009).

38. M.SUZUKI et al., "Performance of Core Exit Thermocouple for PWR Accident Management Action in Vessel Top Break LOCA Simulation Experiment at OECD/NEA ROSA Project," *Proc. ICONE16*, Orlando, Florida, May 11–15, 2008, ASME (2008) (CD-ROM).

39. M. SUZUKI et al., "Final Data Report of ROSA/LSTF Test 6-1 (1.9% Pressure Vessel Upper-Head Small Break LOCA Experiment SB-PV-09 in JAEA)," Proprietary Report, Thermohydraulic Safety Research Group, Nuclear Safety Research Center, Japan Atomic Energy Agency (2006)

40. WORKING GROUP ON THE ANALYSIS AND MANAGEMENT OF ACCIDENTS, NEA/SEN/SIN/AMA(2009)7, Nuclear Energy Agency (Sep. 2009).

41. Organisation for Economic Co-operation and Development Nuclear Energy Agency Gamma Group Web Site: http://www.nea.fr/nsd/csni/gama-pow.html (current as of July 14, 2010).

42. R. PRIOR, "Criteria for the Transition to Severe Accident Management," presented at Organisation for Economic Co-operation and Development Workshop Implimentation of Severe Accident Management Measures (ISAMM-2009), Schloss Böttstein, Switzerland, October 25–28, 2009; http://sacre.web.psi.ch/ISAMM2009/isamm09-prog.html (current as of July 14, 2010).

43. C. QUERAL, A. EXPÓSITO, L. VALLE, G. JIMENEZ, E. VILLALBA, and S. BENEYTO, "Plant Applications of ROSA 6.1 Test. Accident Management Actions in an Upper Head SBLOCA," presented at 8th Mtg. Programme Review Group of the Organisation for Economic Co-operation and Development/Nuclear Energy Agency Rig of Safety Assessment Project, Paris, France, November 2008.

44. C. QUERAL, J. MULAS, I. COLLAZO, A. CONCEJAL, and N. BURBANO, "Problems Found in the Conversion of Almaraz NPP Model from RELAP5 into TRAC-M" (2002) (unpublished).

45. C. QUERAL, J. MULAS, I. COLLAZO, A. CONCEJAL, N. BURBANO, I. GALLEGO, and A. LÓPEZ, "Conversion of the Thermal Hydraulics Components of Almaraz NPP Model from RELAP5 into TRAC-M," presented at Int. Conf. Nuclear Energy for New Europe 2002, Kranjska Gora, Slovenia, September 9–12, 2002.

46. J. MULAS, C. QUERAL, I. COLLAZO, A. CONCEJAL, N. BURBANO, A. LÓPEZ, and I. TARREGA, "Conversion of Control Systems, Protection and Engineering Safeguard System—Signals of Almaraz NPP Model from RELAP5 into TRAC-M," presented at Int. Conf. Nuclear Energy for New Europe 2002, Kranjska Gora, Slovenia, September 9–12, 2002.

47. A. LÓPEZ, C. QUERAL, and I. GALLEGO, "Conversion of Almaraz NPP Model from RELAP5 into TRAC-M," *Trans. Am. Nucl. Soc.*, **89**, 408 (2003).

48. I. GONZÁLEZ, C. QUERAL, A. EXPÓSITO, and M. NOVALES, "TRACE Model of Almaraz Nuclear Power Plant," presented at Int. Conf. Nuclear Energy for New Europe, Bled, Slovenia, September 5–8, 2005.

49. C. QUERAL, I. GONZÁLEZ, A. EXPÓSITO, I. GALLEGO, and A. CONCEJAL, "Conversion of the Steam Generator Model of Almaraz NPP from RELAP5 into TRAC-M, TRAC-P and TRACE," presented at Topl. Mtg. Mathematics and Computation (M&C 2005), Avignon, France, September 12–15, 2005.

50. C. QUERAL, A. EXPÓSITO, G. JIMÉNEZ, L. VALLE, and J. C. MARTÍNEZ-MURILLO, "Verication and Validation of Almaraz NPP TRACE Model," *Proc. ICAPP '08*, Anaheim, California, June 8–12, 2008, American Nuclear Society (2008) (CD-ROM).

51. C. QUERAL, A. EXPÓSITO, G. JIMÉNEZ, L. VALLE, and J. C. MARTÍNEZ-MURILLO, "Assessment of TRACE 4.160 and 5.0 Against RCP Trip Transient in Almaraz I Nuclear Power Plant," NUREG/IA-0233, U.S. Nuclear Regulatory Commission (July 2010).

52. G. HELGESON, W. WEEMS, and R. JETT, "Reactor Vessel Level Indication System," in "System Training Guide A-2d," Diablo Canyon Power Plant (1998).

53. C. WANG, "Study on Nuclear Severe Accident Codes and Their Applications," Chung Yuang Christian University (July 2007).

54. T.-C. WANG, S.-J. WANG, and J.-T. TENG, "Simulation of a PWR Reactor Vessel Level Indicating System During Station Blackout with MELCOR 1.8.5," *Nucl. Technol.*, **156**, 133 (2006).

55. O. SANDERVAG, "The Role of Core Exit Thermocouples in Accident Management," *Proc. Organisation for Economic Co-operation and Development Rig of Safety Assessment Large Scale Test Facility Mtg.*, Paris, France, May 31–June 1, 2007, Organisation for Economic Co-operation and Development (2007).

56. S. N. AKSAN, F. STIERLI, and G. T. ANALYTIS, "Boil-Off Experiments with the EIR-NEPTUN Facility: Analysis and Code Assessment Overview Report," NUREG/IA 0040, Paul Scherrer Institute (Mar. 1992).

57. S. COOPER, "Analysis of LOFT Test L5-1 Using RELAP5/MOD2," NUREG/IA-0118, U.S. Nuclear Regulatory Commission (May 1993).

58. P. D. BAYLESS, "RELAP/MOD3.2 Assessment Using an 11% Upper Plenum Break Experiment in the PSB Facility," INEEL/EXT-0-00058, Idaho National Engineering and Environmental Laboratory (Jan. 2003).

59. J. PARK, W. JUNG, J. KIM, and J. HA, "Analysis of Human Performance Observed Under Simulated Emergencies of Nuclear Power Plants," KAERI/TR-2895/2005, Korea Atomic Energy Research Institute.

60. S.-J. HAN, H.-G. LIM, and J.-E. YANG, "An Estimation of an Operator's Action Time by Using the MARS Code in a Small Break LOCA Without a HPSI for a PWR," *Nucl. Eng. Des.*, **237**, 749 (2007).

61. J. M. IZQUIERDO, J. HORTAL, M. SANCHEZ-PEREA, E. MELENDEZ, R. HERRERO, J. GIL, L. GAMO, I. FERNANDEZ, J. ESPERÓN, P. GONZALEZ, C. QUERAL, A. EXPÓSITO, and G. RODRÍGUEZ, "SCAIS (Simulation Code System for Integrated Safety Assessment): Current Status and Applications," *Proc. European Safety and Reliability Association Annual Conf. (ESREL 08)*, Valencia, Spain, September 22–25, 2008.

62. J. GIL, I. FERNÁNDEZ, S. MURCIA, J. GOMEZ, H. MARRÃO, C. QUERAL, A. EXPÓSITO, G. RODRÍGUEZ, L. IBAÑEZ, J. HORTAL, J. M. IZQUIERDO, M. SÁNCHEZ, and E. MELÉNDEZ, "A Code for Simulation of Human Failure Events in Nuclear Power Plants: SIMPROC," *Nucl. Eng. Des.*, **241**, *4*, 1097 (Apr. 2011).

63. C. QUERAL, L. IBÁÑEZ, J. GONZÁLEZ-CADELO, I. CAÑAMÓN, G. RODRÍGUEZ-MARTÍN, J. ELORZA, A. HIDALGO, A. LÓPEZ, C. CONDE, J. M. IZQUIERDO, J. HORTAL, M. SÁNCHEZ-PEREA, E. MELÉNDEZ, J. GIL, I. FERNÁNDEZ, J. J. GÓMEZ, and S. MURCIA, "Application of the Integrated Safety Assessment Methodology to MBLOCA Sequences," presented at 8th Int. Topl. Mtg. Nuclear Thermal-Hydraulics, Operation and Safety (NUTHOS-8), Shanghai, China, October 10–14, 2010.

64. J. L. ANDERSON, R. L. ANDERSON, E. W. HAGEN, T. C. MORELOCK, and T. L. HUANG, "Post-Implementation Review of Inadequate Core Cooling Instrumentation," *IEEE Trans. Nucl. Sci.*, **36**, 1248 (1989).

65. "Dynamic Range Uncertainties in the Reactor Vessel Level Instrumentation," Information Notice 97-25, U.S. Nuclear Regulatory Commission (1997).