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Research Article

BWR/5 Pressure-Suppression Pool Response during an SBO

Javier Ortiz-Villafuerte, Andrés Rodríguez-Hernández, Enrique Araiza-Martínez, Luis Fuentes-Márquez, and Jorge Viais-Juárez

Departamento de Sistemas Nucleares, Instituto Nacional de Investigaciones Nucleares, 52750 Ocoyoacac, MEX, Mexico

Correspondence should be addressed to Javier Ortiz-Villafuerte; javier.ortizvillafuerte@gmail.com

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RELAP/SCDAPSIM Mod 3.4 has been used to simulate a station blackout occurring at a BWR/5 power station. Further, a simplified model of a wet well and dry well has been added to the NSSS model to study the response of the primary containment during the evolution of this accident. The initial event leading to severe accident was considered to be a LOOP with simultaneous scram. The results show that RCIC alone can keep the core fully covered, but even in this case about 30% of the original liquid water inventory in the PSP is vaporized. During the SBO, without RCIC, this inventory is reduced about 5% more within six hours. Further, a significant pressure rise occurs in containment at about the time when a sharp increase of heat generation occurs in RPV due to cladding oxidation. Failure temperature of fuel clad is also reached at this point. As the accident progresses, conditions for containment venting can be reached in about nine hours, although there still exists considerable margin before reaching containment design pressure. Detailed information of accident progress in reactor vessel and containment is presented and discussed.

1. Introduction

Even before the Fukushima accident in 2011, many projects and programs existed for research on transients and severe accident analysis at nuclear power plants ([1, 2], e.g.). Following the Fukushima accident, several studies on severe accident at nuclear power plants have been released, using different computer codes for simulation [3, 4]. Naturally, particular attention has been given to accident scenarios in which boiling water reactors are the power source. One research article particularly important to this study is that by Allison et al. [4], since the nuclear steam supply system (NSSS) is practically the same nodalization model used in here, as well as the same code: RELAP/SCDAPSIM [5, 6], which is based on the RELAP5 [7] and SCDAP codes [8]. In this present analysis, version Mod 3.4 was used for simulation of a station blackout (SBO). In [4], the authors state that a combination of versions Mod 3.4 and Mod 3.5 was used for simulation of different scenarios during an SBO, and results about reactor core and pressure vessel (RPV) damage were presented. In that paper there is a very detailed schematic of the NSSS nodalization model of the Laguna Verde nuclear power plant (NPP) located in the Gulf of Mexico. However, it is not clear if all steam passing through the safety/relief valves (SRVs) goes to a *time-dependent volume* (TDV) or to an *annulus* (A), which would serve as the primary containment. In any case, no results about pressurization and mass inventory changes in the pressure-suppression pool (PSP) of the Mark II containment are presented in that paper.

2. Simplified Model for Primary Containment

In this study, a primary containment model, including both wet well and dry well volumes, is added to the NSSS nodalization model. This containment model was developed to represent a Mark II containment, as shown in Figure 1. The wet well is composed of six *single volumes*, but only one large volume is used for the dry well. Both wells discharge to environment (*time-dependent volumes*) through venting valves, although these valves were kept shut during the simulation. The venting valves were located at volumes 6 and 7, as shown in Figure 1. The heights of the six volumes of the wet well allow us to track the water level in the PSP at points of interest. While the nodalization can be coarse, only those parameters involved in setpoints of the guidelines for control of containment are followed here.

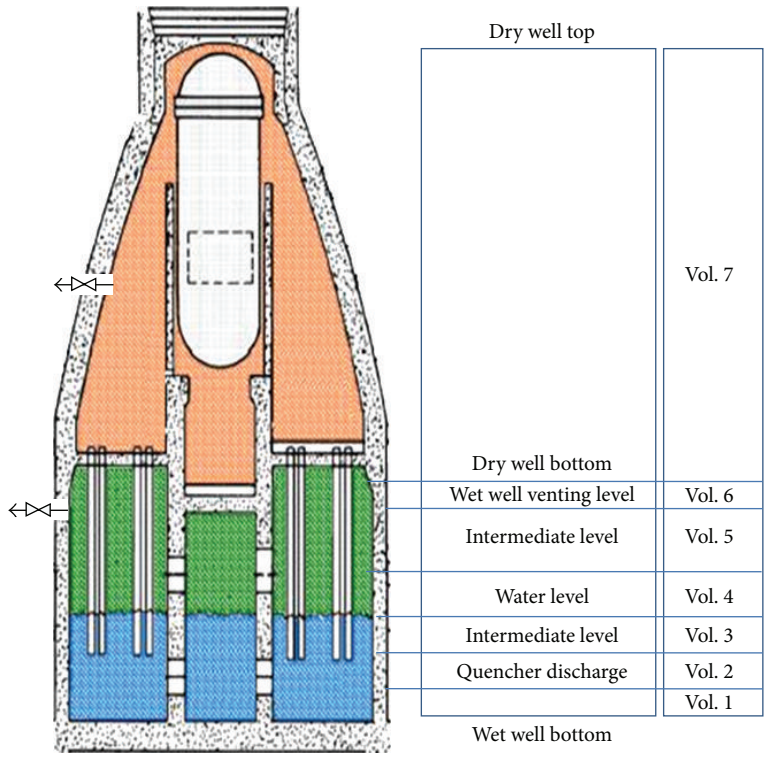

Figure 1: Simplified nodalization model of a BWR Mark II containment.

Figure 2 shows the part of the nodalization model related to steam discharge to the PSP. In this figure, the volume A920 in [4] has been replaced by the proposed containment model. The *single volumes* representing the wet and dry wells are linked by *single junctions*. In this simplified model, the discharge into PSO from the safety relief valves is at the point shown in Figure 2.

3. Steady State and Transients Test

The plant model has been tested for consistency of the steady state. Although only 300 s of steady state are used in the accident simulation, 3000 s were used to qualify the steady state, following the directions given in some International Atomic Energy Agency's (IAEA) documents [1]. Initial conditions for hydrodynamic components and heat structures, as well as relevant control variables, used at the start of simulation correspond to a time when reactor power, dome pressure, RPV water level, feedwater temperature, and so forth were the closest to their operating nominal values. The automatic actuation of safety systems was adjusted and verified with the simulation of a turbine trip with both opening of bypass valves and failure to open and the additional simulation of the closure of main steam isolation valves [9, 10].

4. Accident Scenario

The simulation is started considering that a scram has occurred, due to an earthquake for example, and simultaneously a loss of off-site power (LOOP) happens. The LOOP leads to a load rejection, without bypass in this study, and trip of both recirculation pumps. While it is clear that the initial load rejection transient itself (in the LOOP event) is more severe that the scenario chosen here, because of delay of scram, it does not change significantly this accident evolution because of the automatic actuation of SRVs.

Because of loss of external energy source, the reactor is isolated and there is a consequent loss of feedwater. Water level thus drops until reaching the setpoints for actuation of emergency core cooling systems (ECCS). Both the high pressure cooling spray (HPCS) and the reactor core isolation cooling (RCIC) systems are allowed to deliver cooling water for one hour, and then the HPCS is disabled due to an assumed loss of emergency diesel generators. Then, SBO is considered to happen. The RCIC continues to work for another three hours. Assumptions and more details of the simulation are given in each section in which the whole SBO scenario has been divided for analysis. The simulation results are presented for the different time periods occurring during the accident scenario, as the LOOP, start of ECCS, and the SBO.

5. LOOP Transient

The first 5 to 10 seconds of the LOOP are the most important of this transient, since both power and pressure reach their maximum values, as shown in Figure 3, and the automatic actuation of the SRVs starts. During this time period, the maximum outer cladding temperature rise in any axial node of any of the four fuel channels is 11 K but it only takes

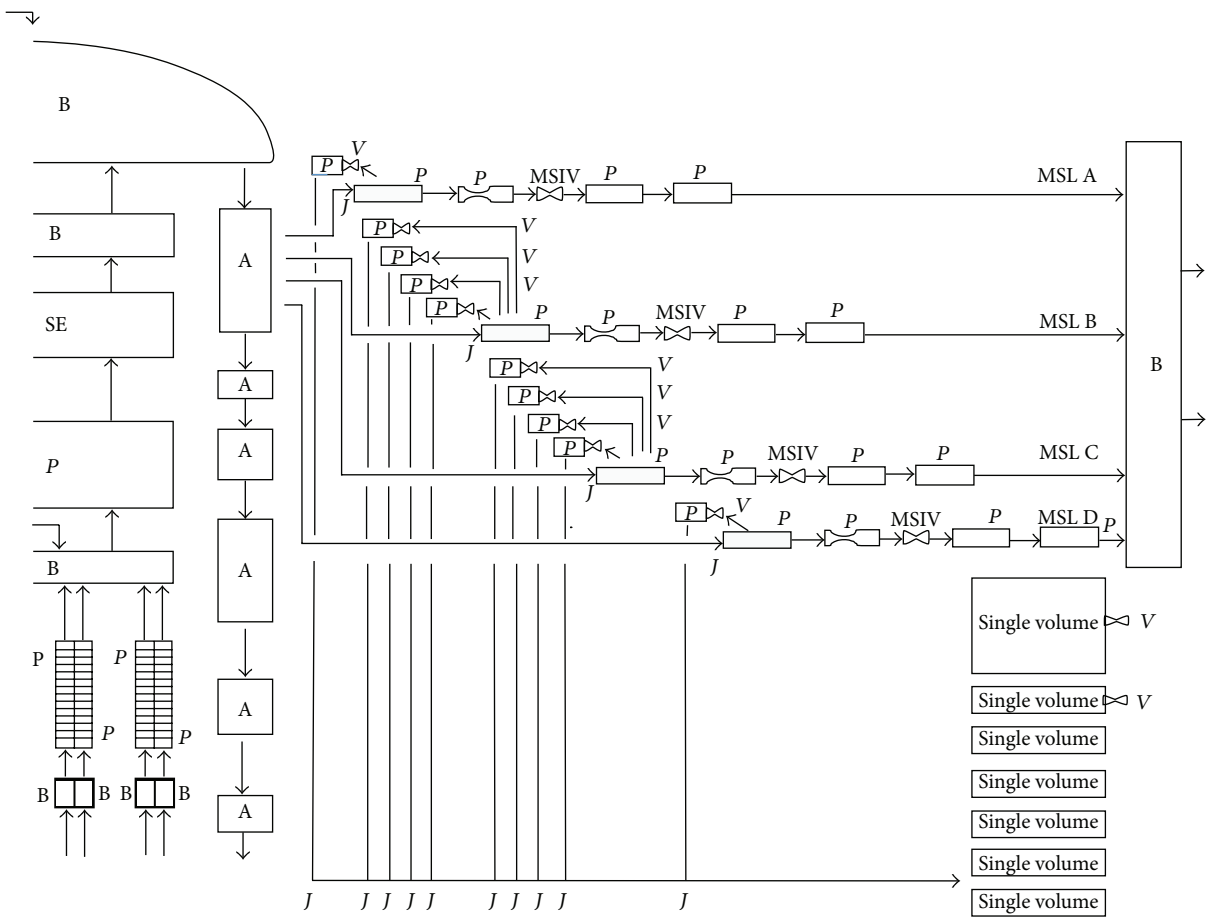

Figure 2: Schematic of connections from RPV to PSP, passing through the SRVs.

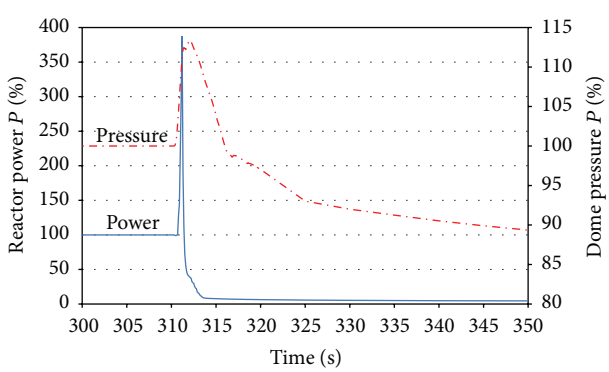

Figure 3: Reactor power and dome pressure evolution during the first seconds of the LOOP.

about 4 seconds to drop to the initial value, and continues decreasing. Power and pressure rise and drop shown in this figure agree both qualitatively and quantitatively with other transient analyses of a load rejection [9, 10].

In Figure 3, as well as all following plots, the reference values are the rated operational values. Figure 4 shows the pressure change in each of the containment model volumes. Although pressure rise in the lowest volumes (in the water pool) is quite significant, pressure in the whole wet well does not reach the setpoint, which in conjunction with PSP water level determines actuation of the venting valve. In the dry well, pressure rise does not reach 10%. In our model, the SRVs vertical (downwards) pipelines end up on *trip valves*, at the junction between volumes 1 and 2 (see Figures 1 and 2); that is, quenchers were not included in the model. A detailed modeling of quenchers would be a nice feature if preliminary hydrodynamic load computations were required.

Primary containment control guidelines, from the emergency operating procedures (EOPs), require monitoring the water level in wet well, as well as pressure. The steam injected in the water pool initially causes a rise of the (two phase) level, but as liquid increases its temperature more steam will be generated and level will eventually drop below its nominal value. As it is shown in Figure 5, in the first seconds of the LOOP, water level rise will be noted, but it falls back quickly to practically its original level. Both Figures 4 and 5 show that the PSP can handle the strength of the initial steam discharge quite rapidly.

6. ECCS Actuation

With the reactor already scrammed and no feedwater injection, vessel water level drops until reaching the setpoint to

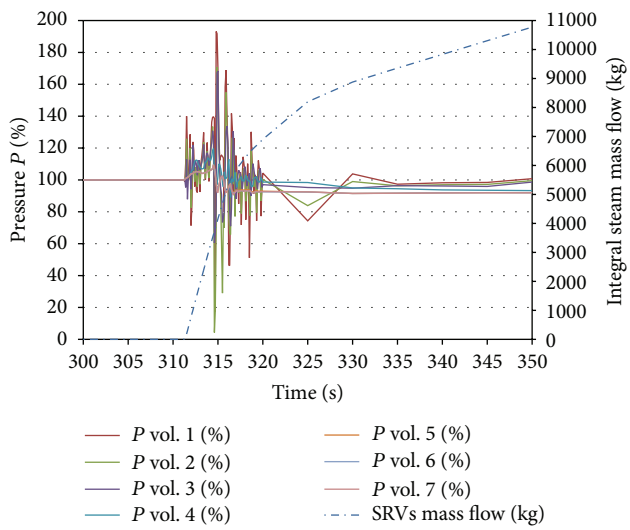

Figure 4: Pressure evolution in the volumes that compose the containment and integral steam mass flow towards the PSP, during the first 40 seconds of the LOOP.

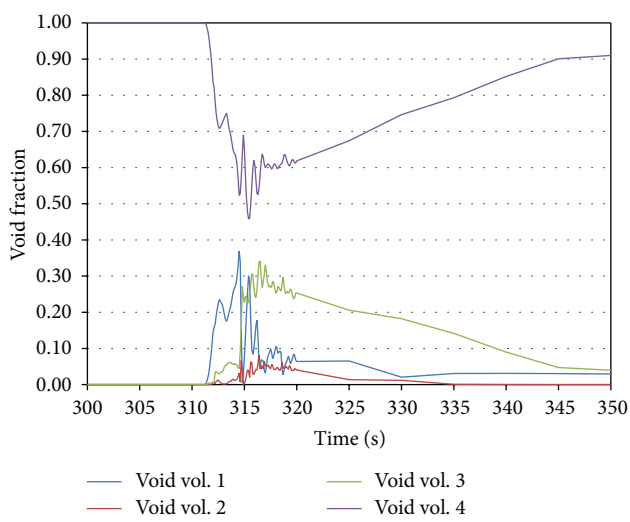

Figure 5: Void fraction in PSP during the first seconds of the LOOP.

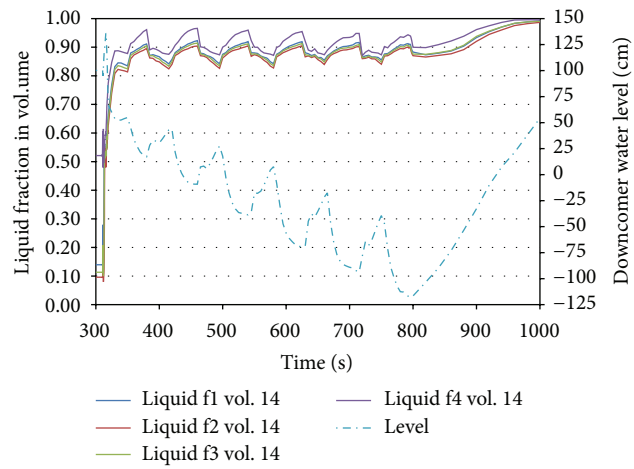

Figure 6: Water level drop in downcomer during the LOOP and recovery when starting the HPCS and RCIC, and liquid fraction in node at top of fuel.

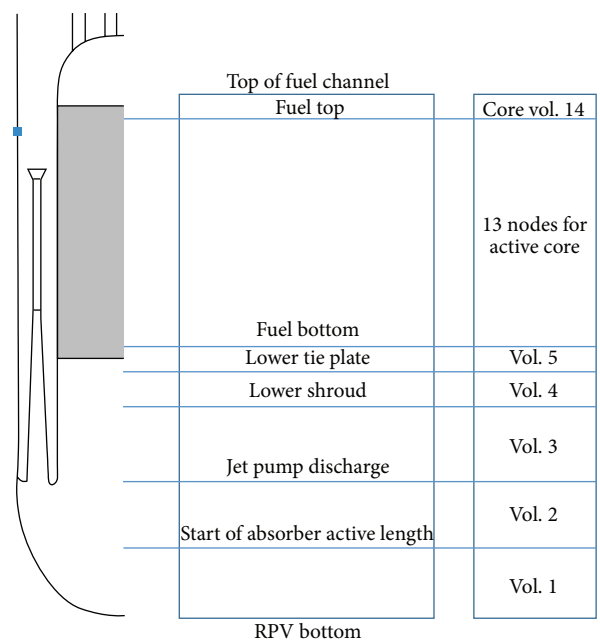

Figure 7: Schematic of the nodalization model of core and sections of RPV below core.

start the ECCS. Since reactor pressure is being controlled by the SRVs actuation, there is no depressurization, and thus only the HPCS and the RCIC systems can start. Figure 6 shows water level in downcomer and liquid fraction in core node 14, which starts at the top of the active core and ends at the top of the channel box, as it is shown in Figure 7.

In Figure 6, f1 means fuel channel 1 and so on. Note that at rated power, f1, f2, and f3 have a void fraction close to 90% in core node 14, but 30 seconds after the initial transient, void fraction drops to less than 20% in all fuel channels. HPCS and RCIC start when downcomer water level reaches −90.2 cm; see Figure 6, but there is a delay in actually delivering cooling water to top of core. Once these cooling systems are fully discharging, the whole channel boxes are practically immersed in liquid.

In the containment, pressure spikes occur only at the bottom volume, as the steam is being discharged, but the general trend of pressure is to drop, until the start of the ECCS, when pressure stabilizes at about 85% of the initial pressure of each volume, as shown in Figure 8. Note that void fraction in volume 3 has reached almost 30%. Thus, the initial water level in PSP has already decreased since no liquid exists in volume 4. During this time period, clad temperatures cycle in a range from 10 to 12 K, with maximum values of about 563 K, following the trend of dome pressure.

7. Containment Behavior during ECCS Injection Period

At the Laguna Verde NPP, four hours of continuous cooling to RPV at high pressure are guaranteed by the final safety analysis report (FSAR) [9]. In this study, both HPCS and RCIC run together for one hour, and then the HPCS is disabled due to an assumed failure of emergency diesel generators. At this point the SBO scenario begins. The RCIC system is however allowed to continue injecting coolant for another three hours, to comply with the FSAR. Figure 9 is the continuation of Figure 8. It is quite important to note that between 30 and 32% of the initial PSP liquid inventory has already vaporized, even before the RCIC system stopped. This implies that the pressure setpoint for containment (wet well in this case) venting should be lowered too, according to the guidelines, but as Figure 9 shows, pressure has also been decreasing while core cooling is provided. Note, however, that about 1000 seconds before the SBO scenario begins, the containment pressure started rising again as liquid in volume 3 reaches its saturation temperature. In the core, all fuel channels are still covered and clad temperatures cycle in the same trend as dome pressure, in a range of 5 and 6 K, even after the SBO has begun.

In the containment, it was shown before (Figure 4) that immediately after the discharge of the 10 SRVs, significant pressure spikes occur in the water pool, but pressure rise in the "air volumes" is just about 10%. Therefore, in a real situation, the reactor operator would not take any actions for emergency depressurization of the RPV, containment venting, or containment spray. After the loss of the RCIC system, an increase in wet well pressure is noted. However, throughout the whole period simulated, the pressure is below those limit values established in the EOPs, so no actions are required again. The actions established in the EOPs, as those mentioned above, are foreseen as means to preserve the pressure suppression function and to protect the primary containment integrity.

8. SBO Severe Accident

The results for this time period are divided into two ranges. The first period goes up to 32000 seconds, and the second period lasts 3000 seconds. This is done so to clearly differentiate the times before and after core melting. Figure 10 shows liquid fraction in four axial nodes of the hottest fuel channel, which corresponds to f2 in this case, and in the channel nose (vol. 5 in Figure 7). As shown in Figure 10, it took one hour the first time to vaporize all the liquid water in the region between the top of active fuel (TAF) and the top of channel box (node 14), but after only four more minutes, this node will not be covered again. The top 46 cm of the fuel will uncover only 6 minutes later. For f2, it took four hours to completely and permanently uncover the active fuel. Note that the channel nose is still immersed in water. Figure 10 also shows the evolution of the outer clad temperature of the hottest axial node (node 9) in f2. Once the node 9 is uncovered, temperature rises quite significantly.

In the containment, a continuous pressure increase and water level decrease will demand eventually the reactor operator to make a decision, according to the EOP guidelines, to depressurize the rector or to continue discharging steam to the PSP, even if the suppression capability could be lost. This decision, of course, would be made if the operator has the necessary information and possibility. As shown in Figure 11, in this simulation, it is good to note that while the water level reaches 65% of the initial level for about 75 minutes, the pressure in the PSP is still low enough to comply with EOP guidelines, and thus the PSP can continue receiving steam, and, therefore, there is no need to start venting the wet well yet.

The last 50 minutes of simulation are presented next. Figure 12 shows that the water level in the RPV falls until just below the lower shroud support, at the end of this simulation. This figure also shows those axial nodes in fuel

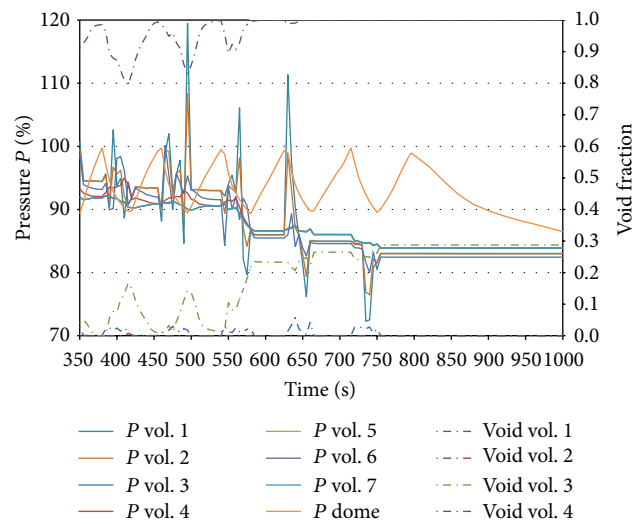

Figure 8: Pressure and void fraction evolution in containment before and after start of ECCS.

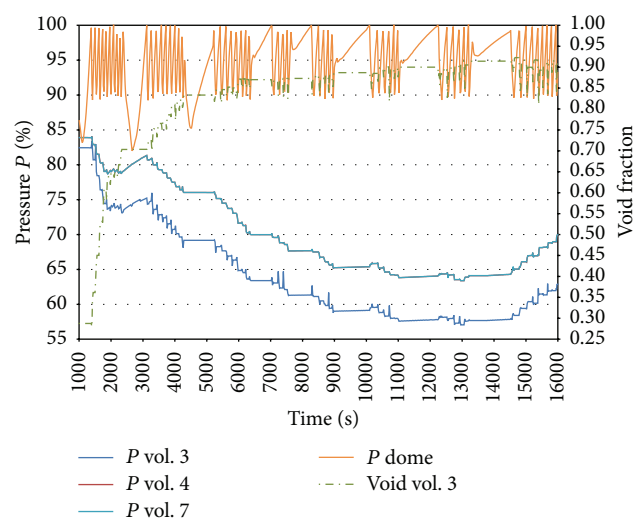

Figure 9: Pressure and void fraction evolution in containment during ECCS injection period.

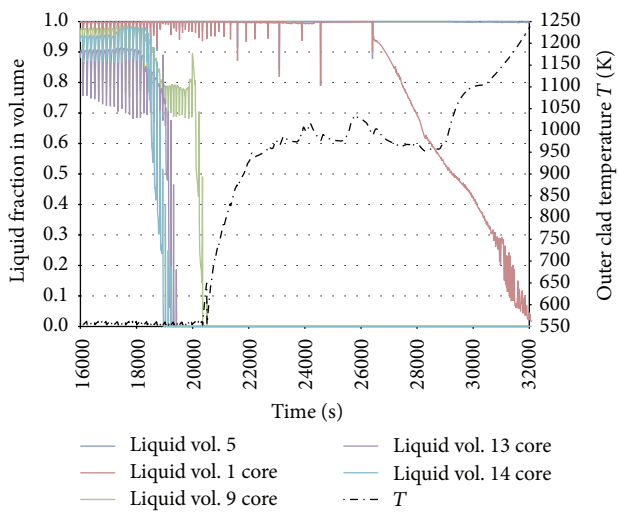

Figure 10: Void fraction in four axial nodes of fuel channel 2 and in its nose during the first hours of SBO. Clad temperature corresponds to the hottest axial node in this channel.

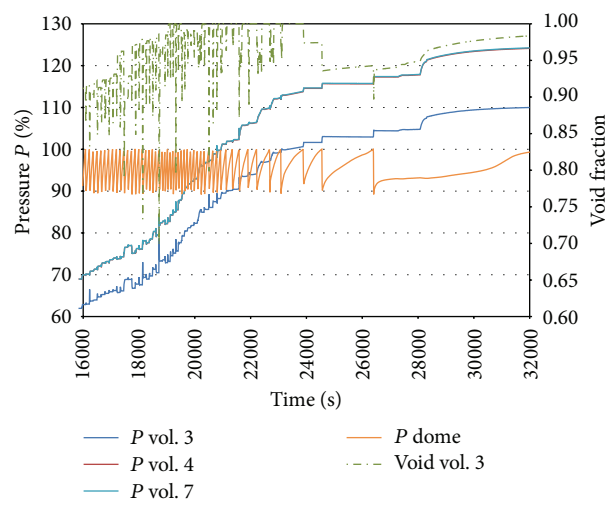

Figure 11: Pressure and void fraction in containment during the first hours of the SBO.

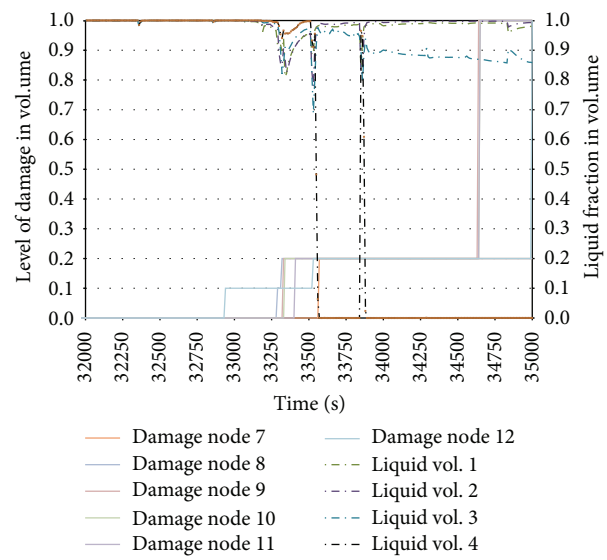

Figure 12: Axial nodes of f2 forming a molten pool in core and fraction of liquid in volumes below the core.

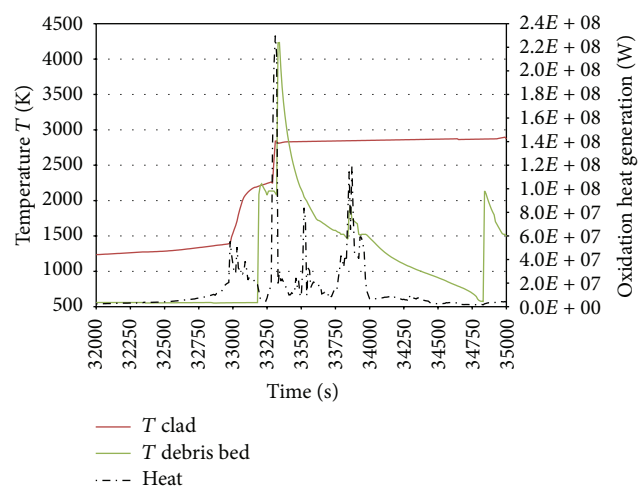

Figure 13: Temperature evolution on cladding and debris bed at RPV wall at the nodes where peak values were reached, and total heat generated due to oxidation.

channel 2 that have reached the status of *molten pool*. All other nodes reached at least the status of *fragmented rubble*. This combination of damage and water levels is important since the lower tie plate and supporting structures may be in a quite harsh environment, although the temperature of the lowest node in the fuel channel is a primary parameter to determine the integrity of those structures. In this case, the temperature of node 1 was only 775 K, even when already damaged.

Figure 13 shows the evolution of the cladding temperature of the hottest axial node (node 9 of f2) and the evolution of the temperature of the debris bed at the node (node 86) where the debris bed peak temperature is reached, as calculated by COUPLE on the mesh along the RPV wall. This figure also shows the heat generated because of oxidation reactions in core. Regarding the temperature peaks, both temperatures increase significantly as RPV pressure increases. Then, as steam is discharged, oxidation heat rises sharply, driving further temperature rise.

Figure 14 shows the debris bed temperature on the COUPLE mesh at the time of peak temperature. These peak temperature values agree with those results reported in [4]. Although the peak value is above 4000 K, the RPV wall temperature was always less than 2200 K, as it is shown in Figure 15. In this figure, the maximum temperatures at any node of both debris bed and structural material (the RPV wall) are shown.

In the containment, there is an important pressure rise following the events just mentioned, as it can be seen in Figure 16. However, it is until the second largest oxidation heat generation peak occurs that the combination of PSP

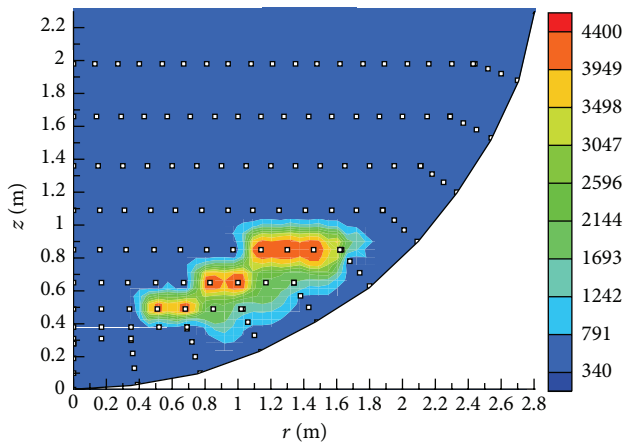

Figure 14: Debris bed temperature distribution on RPV wall at the time of reaching the maximum value.

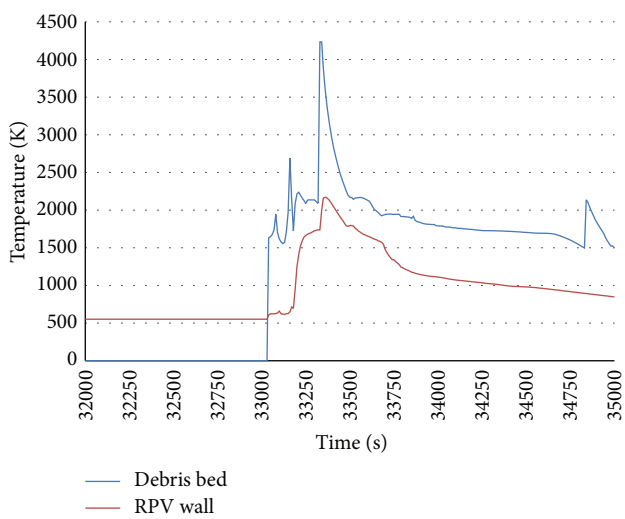

Figure 15: Maximum temperatures at any node on the COUPLE mesh of debris bed and RPV wall.

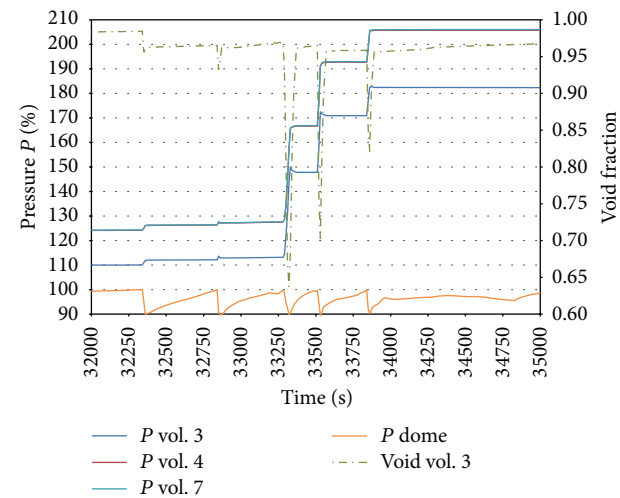

Figure 16: Pressure and void fraction evolution in primary containment.

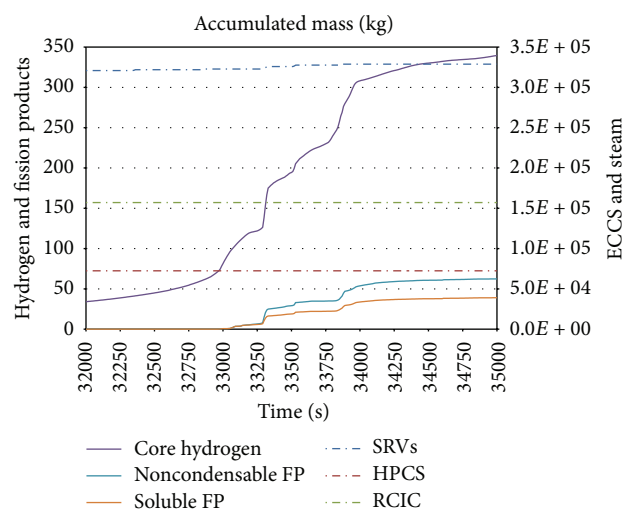

Figure 17: Total accumulated mass of hydrogen generated in core, fission products released, emergency cooling injected in vessel, and steam discharged to the PSP.

water level and pressure reaches the setpoint to start venting, according to EOP guidelines. In this simulation, venting valves were kept closed to find out the maximum pressure that could happen and how close it would be to design pressure. The results show that there is still much margin, in that aspect.

9. Limitations of This Simulation

The containment model used here for analysis is quite limited, since three-dimensional effects can be important when steam is being discharged, and thus pressure and temperature distribution in the volumes are not described accurately by this nodalization model. Further, actually the actuation of only one SRV is able to keep control of RPV pressure, but then this SRV is discharging in one particular zone of the PSP, so average pressure and temperature values for the whole *single volumes* used in the model have associated uncertainties that need further study. This is particularly important when considering the actual location of measuring instruments.

Another issue is that all those systems used to keep containment in its normal operational conditions are not included in the model. However, in this particular accident being analyzed, the lack of any electrical power does not allow the use of equipment or systems to help keeping pressure, temperature, and water level in safe ranges. One other important limitation in this study is that fission products and hydrogen generated inside the RPV were not tracked, so there is no knowledge of how much of these elements reached the containment. On this issue, it should be considered that reactor operator needs to know the potential source term to decide (if possible) either to vent or not, but recalling that hydrogen will continue to accumulate. Figure 17

shows the total accumulated mass of fission products, both noncondensable and soluble, and hydrogen generated during the event. Also, the total mass of coolant injected by the ECCS and the steam delivered to the PSP are shown in this figure.

Finally, there is still the issue of how long cooling liquid can be injected to RPV though the HPCS and RCIC from reservoir tanks. While the RCIC was shown to be capable of keeping the core fully covered and that its coolant injection can be satisfied for the four hours, the use of the HPCS for four hours might reduce significantly the cooling injection time.

10. Conclusions

A severe accident scenario for a BWR has been simulated, with an emphasis on the behavior of the pressure-suppression pool during the event. A simplified model of a wet well and dry well was added to an NSSS model of a BWR to determine if conditions for containment venting are reached. The simulation was performed with RELAP/SCDAPSIM Mod 3.4. The total duration of the scenario was 35000 seconds (~10 hours). The starting point was a LOOP with simultaneous scram. The LOOP is simulated as a load rejection without bypass and the two recirculation pumps being tripped. Thus, a significant amount of steam is discharged to the PSP in the first seconds of the transient. One hour of HPCS and RCIC actuation together was considered, and then only the RCIC worked for three hours more. Then station blackout conditions were assumed. In this study, the safety relief valves actuate throughout the event. The results show that the RCIC alone can maintain the core immersed in water, but once it stops, core melting and high temperatures on the RPV wall will occur, in agreement, both qualitatively and quantitatively, with other similar simulations reported in the scientific literature.

Regarding the evolution of pressure and water level in the PSP, even when RCIC is still injecting coolant to RPV about 30% of the original water inventory in the PSP is vaporized. During the SBO, the inventory reduces about 5% more. Further, a significant pressure rise occurs in containment at about the time when a sharp increase of heat generation occurs in RPV due to cladding oxidation. Failure temperature of fuel clad is also reached at this point. As the accident progresses, conditions for containment venting can be reached after about nine hours, under SBO scenario. However, there still exists plenty of margin before the containment pressure design value can be reached. But, since no three-dimensional capabilities exist in the computer code used in simulation and steam discharge is most of the time delivered in one particular zone of the PSP, average values of key parameters do have uncertainties that need some studying.

Furthermore, since both hydrogen and noncondensable fission products have been already released in the reactor vessel, the reactor operator may reach the point, if possible of course, of deciding the time to vent or how long to allow the accumulation of hydrogen in containment. In our current models, fission product and hydrogen tracking was not considered, so it cannot be predicted any of the potential hydrogen hazards or a preliminary source term.

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