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ANALYSIS OF HIGH PRESSURE BOIL-OFF SITUATION DURING MSIV CLOSURE ATWS IN A TYPICAL BWR/4*

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Anticipated Transient Without Scram (ATWS) is recognized as one of the Boiling Water Reactor (BWR) accident sequences potentially leading to core damage[1]. Of all the various ATWS initiating events, the Main Steam Isolation Valve (MSIV) closure ATWS has a relatively high frequency of occurrence and poses a challenge to the Residual Heat Removal (RHR) and containment integrity systems. Although under investigation for quite a long period of time[2-3], different aspects of this type of transient are still being analyzed[4-6]. The final outcome of these studies should be a well defined set of recommendations for the plant operator to mitigate an ATWS accident.

The objective of this paper is to provide a best-estimate analysis of the MSIV Closure ATWS in the Browns Ferry Unit 1 BWR with Mark 1 containment. The calculations have been performed using the RAMONA-3B code[7] which has a three-dimensional neutron kinetics model coupled with one-dimensional (multi-channel core representation), four-equation, nonhomogeneous, nonequilibrium thermal hydraulics. The code also allows for one-dimensional neutronic core representation. The 1-D capability of the code has been employed in this calculation since a thorough sensitivity study showed that for a full ATWS, a one-dimensional (axial) neutron kinetics adequately describes the core behavior. (Note that the core steady-state symmetry in this case was preserved throughout the transient so that radial effects could be neglected.) The calculation described in the paper was started from a steady-state fuel condition corresponding to the end of Cycle 5 of the Browns Ferry reactor.

From a number of different scenarios being considered by the USNRC Severe Accident Sequence Analysis (SASA) program for BWR ATWS analysis, the following transient scenario will be discussed in this paper: all MSIVs close in 5 seconds, the control rods fail to insert on SCRAM signal, the feedwater is lost by 8 seconds, and the recirculation pumps are tripped on the high vessel pressure signal. The water level in the downcomer drops and finally reaches the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems set point elevation, i.e., Level 2. These emergency systems are designed to maintain the reactor vessel water inventory in the event of loss of coolant accidents by injecting cold water from the condenser storage tank into the downcomer region of the vessel. However, according to the present scenario, only the RCIC system which provides approximately 39 kg/s of water flow rate is available. Another minor source of the vessel inventory make-up water, namely, the Control Rod Drive (CRD) cooling water with the flow rate of approximately 6 kg/s, is also available. However, the primary source of safety injection, i.e., HPCI with 320 kg/s of water flow rate, is assumed to be unavailable. The reactor is left in the automatic mode of operation with the vessel pressure oscillating between the safety/relief valve pressure set points; no operator actions including boron injection were assumed to be taken.

The results of the RAMONA-3B calculation for this transient are shown in Figures 1 through 9. Figure 1 shows that the downcomer water level was dropping very rapidly reaching the safety injection systems set point level, i.e., Level 2, at approximately 15 seconds. The TAF elevation was crossed about 58 seconds later. The water level continued to drop since the combined RCIC and CRD water injection was not sufficient to make up the vessel water inventory being depleted by steam losses through the steam line safety and relief valves. The corresponding water injection and steam line flow rate histories are shown in Figure 2. The rate of downcomer water level drop is proportional to the difference between these curves.

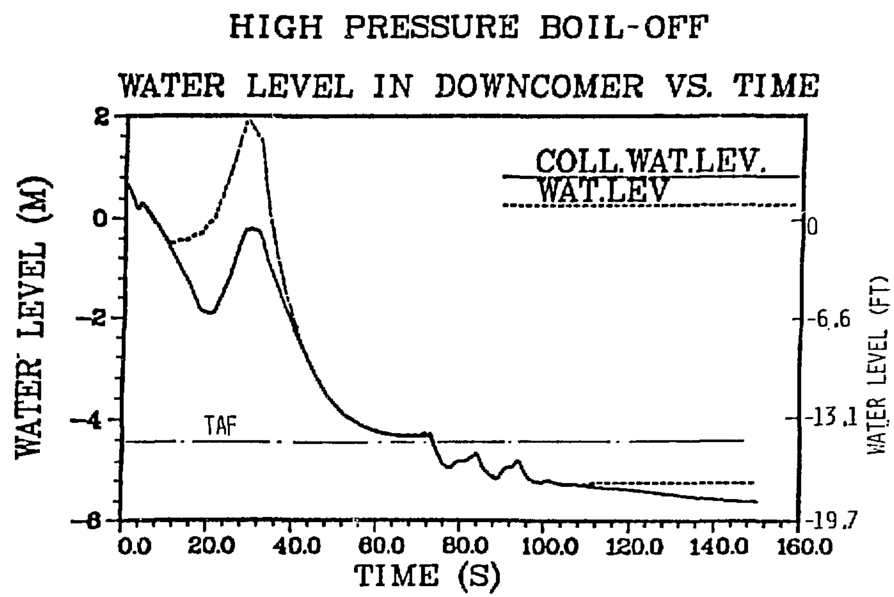

Figure 1: Water Level Prediction for the High Pressure Boil Off Calculation

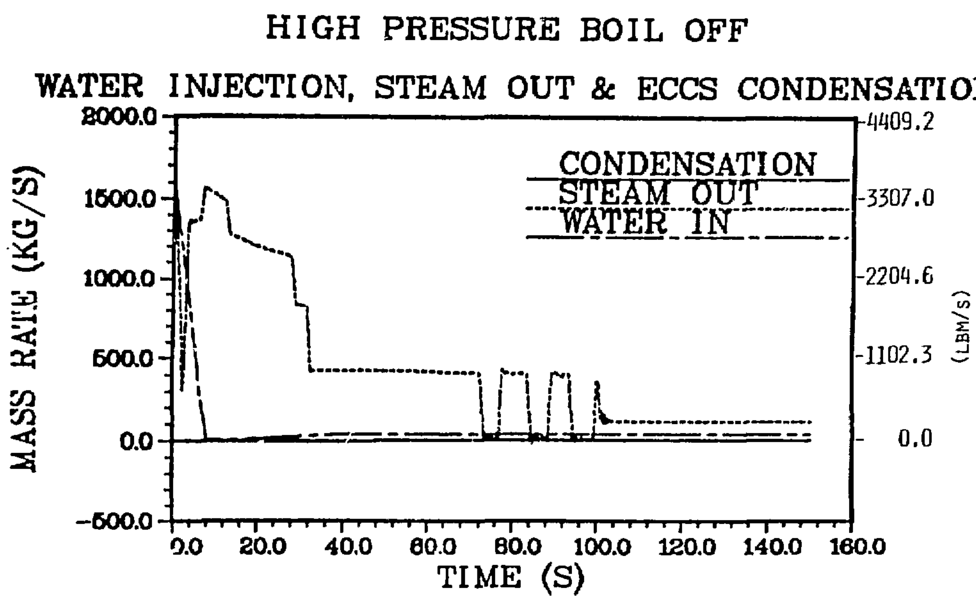

Figure 2: Water Injection and its Associated Amount of Condensation Along with the Existing Steam Flowrate

After an initial spike in the vessel pressure and reactor power due to the MSIV closure (Figures 3 and 4), the core flow rate was continuously decreasing (Figure 5) following the water level drop in the downcomer. (After the recirculation pumps trip, the core flow rate becomes a function of the downcomer static head). As a consequence, the core voids started to increase, causing a strong negative reactivity insertion. (The average core in-bundle void fraction histories are shown in Figure 6.). Again, the histories of the in-bundle void fraction, core void reactivity (Figure 7), and power (Figure 4) demonstrate the tight coupling between the thermal hydraulics and reactor neutronics typical for the BWR core.

The core inlet subcooling is an important parameter contributing both to the axial core power distribution and the total reactor power. Unlike the transients when the HPCI injection flow is available, the safety injection flow rate for this calculation is very small, which results in a low core inlet subcooling (Figure 8). The low injection flow rate is also accompanied by low condensation rate after the downcomer level drops below the feedwater spargers, as seen in Figure 9. Incidentally, this figure also shows that the feedwater sparger level was crossed at approximately 39 seconds.

The calculation shows that during the first 150 seconds, the Doppler (fuel temperature), void (core flow rate, vapor generation rate), and moderator temperature (core inlet temperature and reactor vessel pressure) reactivity effects (Figure 7) brought the total reactor power down to approximately 4.1% of the rated power (Figure 4). It is worthwhile to mention that throughout the transient the Doppler and void reactivity effects were the principal contributors to the total core reactivity.

An important result is that by 150 seconds only about 1.2% in the pre-

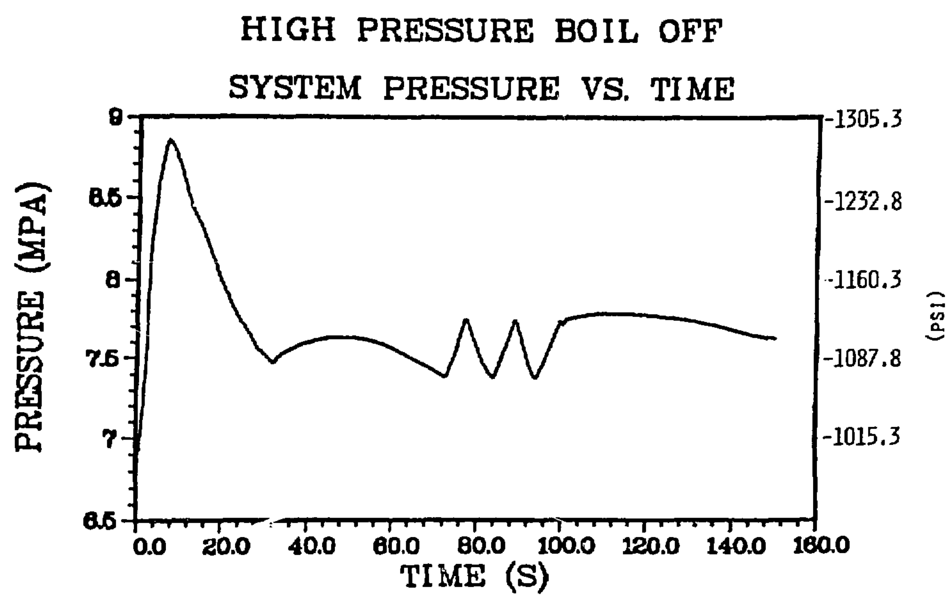

Figure 3: System Pressure for the High Pressure Boil Off Calculation

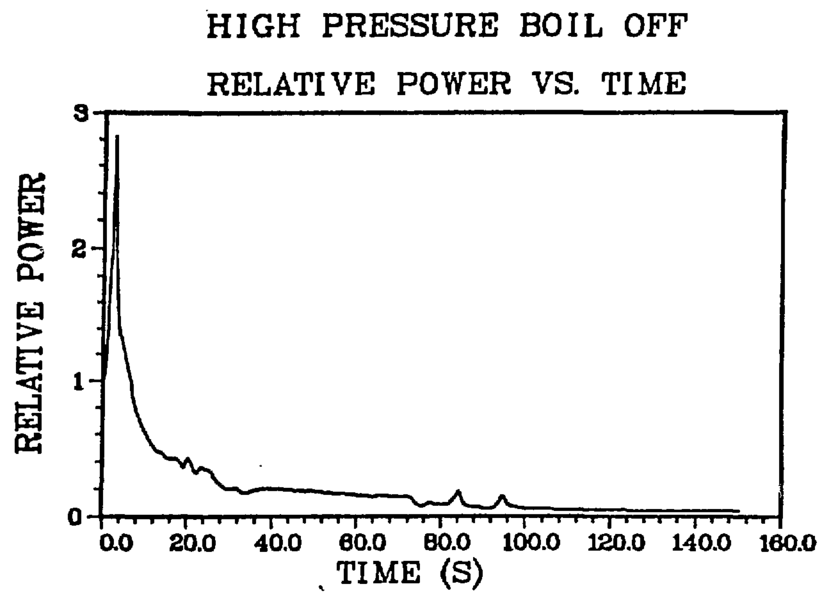

Figure 4: Relative Power for the High Pressure Boil Off Calculation

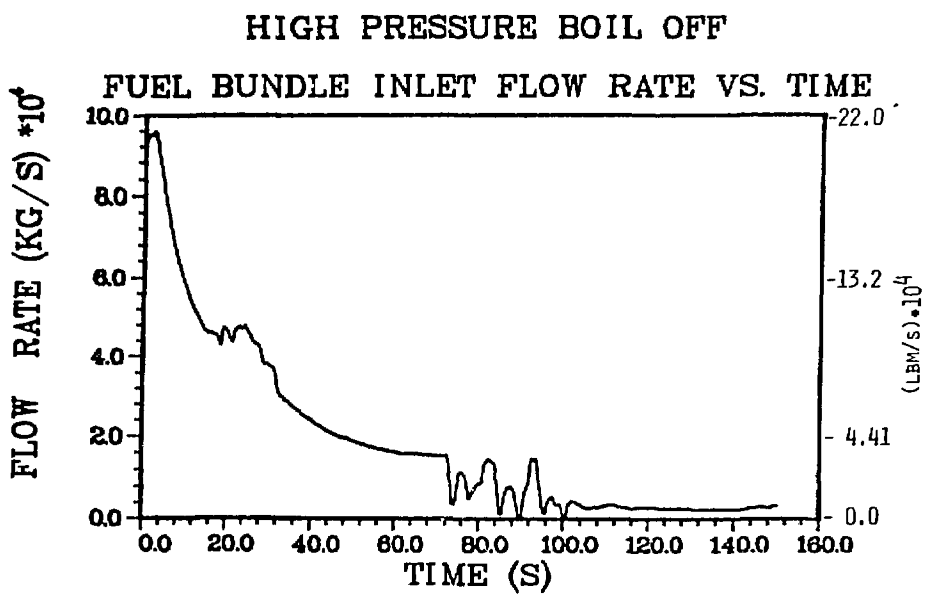

Figure 5: Core Inlet Flow Rate to the Fuel Assemblies for the High Pressure Boil Off Calculations

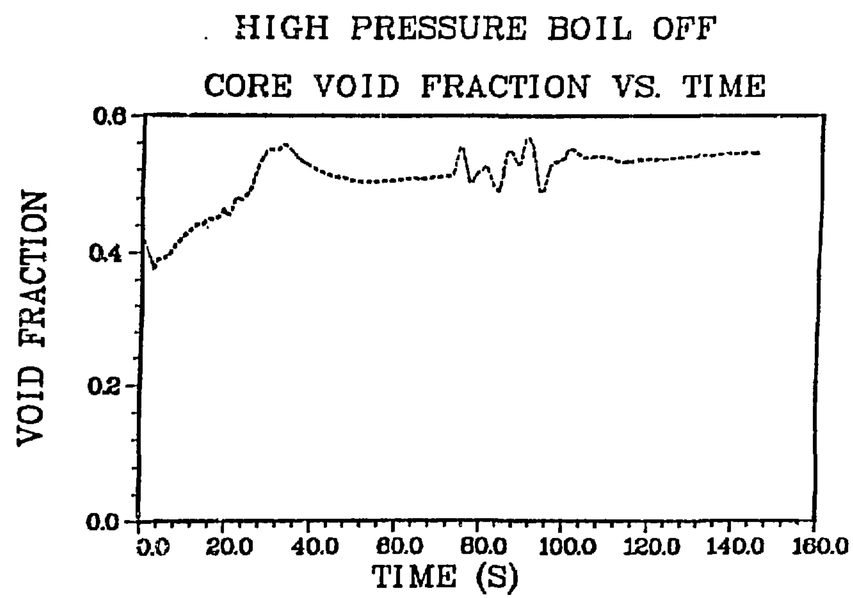

Figure 6: Core Average Void Fraction for the High Pressure Boil Off Calculation

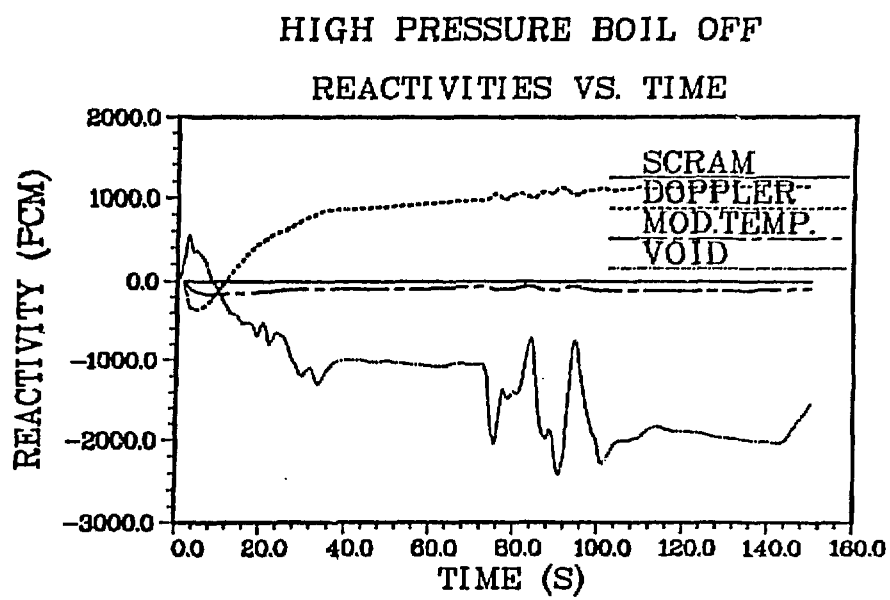

Figure 7: Components of the Total Reactivity for the High Pressure Boil Off Calculation

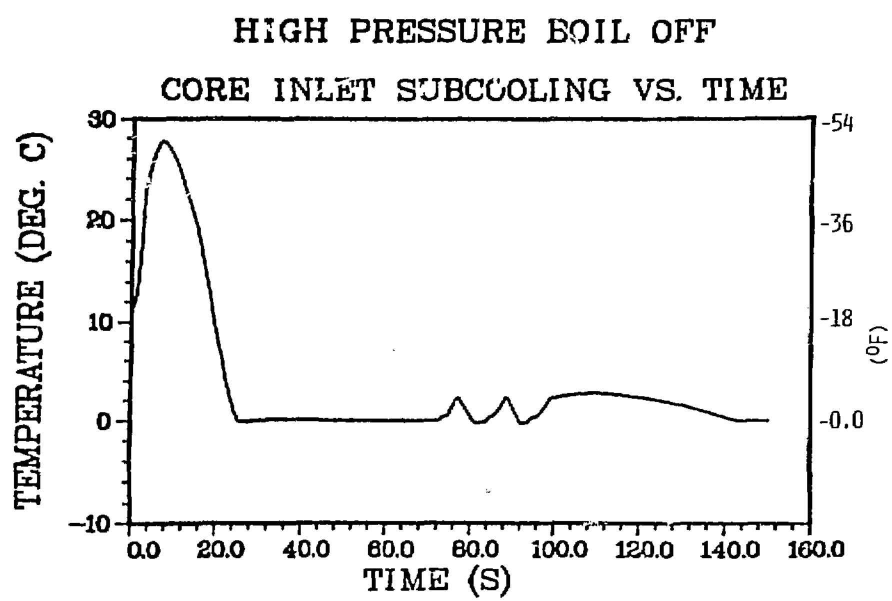

Figure 8: Core Inlet Subcooling for the High Pressure Boil Off Calculation

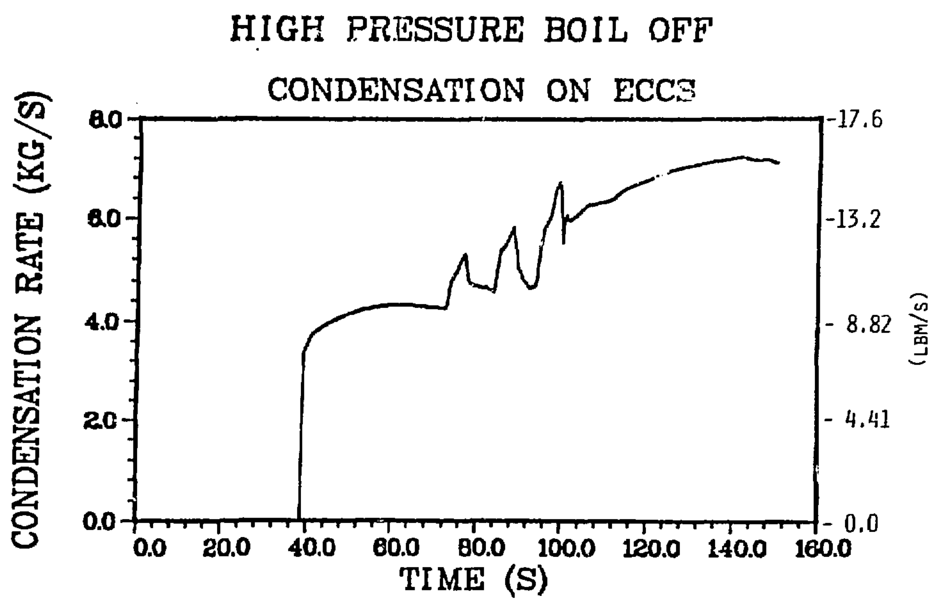

Figure 9: Condensation on Injected ECC Water for the High Pressure Boil Off Calculation

dicted 4.1% total power was produced by fission in the core, while the rest was the decay power. The results indicate that the reactor power was almost brought down to the decay heat level due to the large amount of negative reactivity caused by the core voiding after the water level in the downcomer dropped 1.2 meters (4 ft.) below TAF. The core, however, could regain power in response to any positive reactivity insertion, such as a drop in void fraction.

Gradually, the reactor should arrive at a quasi-steady state condition; the power will oscillate around the value corresponding to the power needed to evaporate the 45 kg/s of combined RCIC and CRD injected water which represents 2.7% of rated power. The quasi-steady state would be established in the following manner: any decrease in power leads to a lower vapor generation and, consequently, to a decrease in the core voids. At the same time, build-up of the downcomer water level and natural circulation driving head begins, and the core flow rate increases. As a result, power starts to increase and, eventually, the core vapor generation exceeds the RCIC and CRD water flow rates so that the downcomer water level, the driving head, and the core flow rate start to decrease. Consequently, the void fraction in the core increases, dropping the total reactor power back to the water injection controlled power level (~2.7%). It is clear, however, that regardless of the water injection flow rate, the power cannot drop below the decay heat level.

Another issue associated with the present transient scenario, besides the total reactor power, was the thermal hydraulic behavior of the core at low flow conditions. The question was whether the fuel and cladding temperatures would reach a point where the fuel rod damage could occur. In the RAMONA-3B calculation, no CHF condition in the core was predicted up to 150 seconds when the calculation was terminated because of intermittent full-core flow reversal. At this point a stand-alone sensitivity study based on the results obtained in the RAMONA-3B calculation was initiated in order to address the issue of the fuel rod temperatures.

The following conservative assumptions have been made for the simplified core thermal hydraulics model:

- there is a level separation in the core;
- the core mixture level resides at the elevation where the equilibrium flow quality equals to one;
- below the water level there exists a film boiling flow heat transfer regime;
- only pure superheated steam cooling takes place in the core above the water level;
- radiation heat transfer is negligible.

The stand-alone steady state calculation was performed based on the above assumptions and using the decay heat power distribution predicted by the RAMONA-3B code with a core flow rate of 45 kg/s from the RCIC and CRD. The post-CHF heat transfer coefficient for the superheated steam cooling above the water level was calculated according to the correlation suggested by Anklam (1981).

The results obtained for the core power of 2.7% of rated reactor power showed that the maximum temperature at the fuel rod center was 996°C (1825°F), which is approximately 10% above the corresponding temperature at the reactor nominal power condition. However, the cladding surface temperature was predicted to be around 980°C (1796°F). Note that this is the lower end of the temperature range where the exothermic steam-zirconium reaction should be taken into account while predicting the cladding temperature and the hydrogen generation[9]. Therefore, the temperature of 980°C (1796°F) is high enough to cause concern, although it may not immediately lead to fuel rod failure.

In summary, the following conclusions are drawn from the results presented in this section:

1. Total reactor power drops to 4.1% of rated power by 150 seconds (1.2% fission and 2.9% decay power) with the downcomer water level at about 1.2 meters (4 ft.) below TAF.

2. No CHF condition is expected up to 150 seconds into the transient time. Stand-alone long term estimate does not show fuel and cladding temperatures high enough for immediate fuel rod failure. However, the cladding temperature is high enough to cause concern about the fuel rod integrity because the steam-zirconium reaction could begin.

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