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CALCULATION OF A BWR "PARTIAL ATWS" USING RAMONA-3B

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ABSTRACT

The RAMONA-3B code has been used to simulate a boiling water reactor (BWR) transient initiated by the closure of the main steam line isolation valves in which all the control rods in one-half the core fail to scram after reactor trip. The modeling of the nuclear steam supply system included three-dimensional neutron kinetics and parallel hydraulic channels (including a bypass channel). The transient is characterized by an initial pressure spike and then by oscillations in the pressure due to the opening and closing of relief valves. These oscillations in turn affect all thermohydraulic properties in the vessel. The simulation was continued for 7 minutes of reactor time at which point boron began to accumulate in the core. The calculation demonstrates the importance of using three-dimensional neutron kinetics in conjunction with the modeling of the nuclear steam supply system for this type of transient. RAMONA-3B is unique in its ability to do this type of calculation.

INTRODUCTION

Anticipated BWR transients in which there is no reactor trip (i.e., no scram) continue to be an unresolved licensing issue. Since the reactor does not shut down immediately, it is particularly important to adequately analyze the core neutronics and thermohydraulics during the transient. For those transients in which only part of the reactor is scrammed, as occurred in Browns Ferry Unit No. 3 (on June 28, 1980), a multidimensional treatment of the spatial effects occurring in the core is essential. RAMONA-3B [1] offers the unique capability to dynamically represent the spatial neutronics coupled explicitly with the thermohydraulics of the entire system. For this reason, a transient accompanied by partial scram failure was simulated to provide a test of the overall code capability to perform such analyses.

The accident analyzed with RAMONA-3B is initiated by the (inadvertent) closure of the main steam line isolation valves (MSIVs) from (approximately) rated core power and flow conditions. It is assumed that all control rods in one-half the core fail to scram after the reactor trip signal is received. Other safety systems are assumed operable; the recirculation pumps trip, pressure relief valves operate, and the high pressure coolant injection system, the reactor core isolation cooling system and the standby liquid control system all actuate when required.

The MSIV closure event is one of the most limiting for peak vessel pressure and suppression pool temperature considerations. Although RAMONA-3B does not calculate the pool temperature, it does provide the steam flow through the safety and relief valves which is required for the pool temperature calculation.

Analyses of this accident have been performed at General Electric [2] and at BNL [3]. Both of these analyses used auxiliary steady-state core calculations to help determine the core power and used assumptions slightly different than those used with RAMONA-3B. Nevertheless, the results reported herein are consistent with the previous work.

MODELING OF ACCIDENT CONDITIONS

The reactor modeled for this calculation was a BWR/4 at end-of-cycle conditions similar to those used previously [3]. The initial reactor state corresponded to operation at 104.5% of rated power (3440 MW) and 100% rated flow (12800 kg s^{-1}). All control rods were initially withdrawn.

A coarse mesh was used in both the steam line (8 nodes) and the core (12 axial nodes). This mesh was found to be acceptable based on comparisons with calculations using a finer mesh [1]. The core model used had 32 nodes in the x-y plane for the neutron kinetics calculation, but took advantage of half-core symmetry. Figure 1 shows the half-core configuration and the numbering for the neutron kinetics nodes and thermohydraulic channels. When the rods in half the core are inserted, it is the left side (i.e., nodes 1-3, 7-9, 13 and 14) that becomes rodded.

The cross sections used were originally derived for use with a two dimensional (R,Z) coupled neutron kinetics and core thermohydraulics code. The method used to obtain the data [4] did not include core burnup calculations. Beginning-of-life data were systematically varied until the core-average axial power distribution gave agreement with the expected end-of-life (Haling) power distribution. This procedure resulted in eleven cross-section sets which are distributed to 16 material (or exposure) zones based on eight axial and two radial divisions.

In transforming the actual core geometry into the configuration shown in Fig. 1, no attempt was made to homogenize fuel bundles with reflector water for the nodes at the core periphery. The good agreement in radial power distribution with a more detailed calculation [1] justifies this approach.

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                            ___________________________________
                            |1    |2    |3    |4    |5    |6    |
                            |     |     |     |     |     |     |
                            |    1|    2|    2|    3|    3|    4|
                            |7    |8    |9    |10   |11   |12   |
                            |     |     |     |     |     |     |
                            |    1|    2|    2|    3|    3|    4|
Neutron Kinetics Nodes            |13   |14   |15   |16   |
                                  |     |     |     |     |
Thermohydraulic Channels          |    5|    5|    6|    6|
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Figure 1 Core Configuration for Partial Scram Calculation

Reactor trip would normally occur due to the signal for MSIV closure, however, that signal is ignored and the assumption is that the trip signal will be due to high power. The control rods in half the core move at a speed of 0.91 m s^{-1}. The low level signal for both the high pressure coolant injection (HPCI) and the reactor core isolation cooling (RCIC) systems is -1.77 m relative to the initial water level outside the steam separator skirts. The HPCI and RCIC systems are assumed to take 3 s to reach their full flow rate after a delay time of 27 s.

The 13 safety and relief valves are grouped into four banks. The opening and closing flow rates are assumed to be exponential with a time constant of 0.1 s. The feedwater control system is represented by an input boundary condition. The flow rate was obtained from GE [2] where it was calculated for a similar accident situation. Figure 2 is a plot of the flow rate from the feedwater sparger. The early flow (< 200 s) is due to the feedwater system (water at 196°C) and the latter flow (> 300 s) is due to the HPCI and RCIC systems (water at 48.9°C). The standby liquid control system was initiated on a time signal to represent operator action at 255 s coupled with a delay time of 45 s. The rate of boron addition to the vessel at the location of the jet pump instrumentation lines corresponds to an injectant flow rate of 2.7 kg s^{-1} (43 gpm) with a boron concentration of 23,000 ppm.

RESULTS

The transient calculated with RAMONA-3B was initiated by an MSIV closure and then calculated for 400 s. During this period, control rods were inserted, the recirculation pumps tripped, pressure relief valves opened and closed, the amount of feedwater changed and the HPCI/RCIC and SLCS systems were actuated. The effect of these actions and the interaction of the different feedback mechanisms makes the local and global system behavior very complex. The major events and trends during the transient as calculated by RAMONA-3B are given in Table I. The following discussion is separated into five parts in order to explain certain features of the calculation.

Initial Overpressurization Transient

The MSIV closure causes an increase in pressure in the vessel (cf. Fig. 3). This pressure pulse is less severe than that caused by a turbine stop valve closure due to the relatively long time (4 s) it takes for the MSIV to close. The increase in pressure collapses steam voids which has the effect of increasing core power (cf. Figs. 4 and 5). This tends to be self-limiting because an increase in power increases the void fraction. However, the insertion of control rods due to an overpower (120% of rated power) signal, the decrease in flow due to the RPT and the decrease in pressure due to the opening of relief valves combine to terminate the early phase of the transient.

Figure 5 shows the fission rate during this early phase for (neutron kinetics) channel 5 (cf. Fig. 1) in which all control rods are inserted and channel 2 in which no control rods move. Reactor trip occurs at 2.5 s and by 10 s only the decay heat level is significant in channel 5. Note that the ability to monitor behavior such as in Fig. 5 requires a code with spatial neutron kinetics. The changes in fission rate that are most pronounced on Fig. 5 (and not seen on Fig. 4 because of the different scale) occur after 2.8 s with a frequency of 3 Hz. These are due to the insertion of control rods and the coarseness of the axial mesh. This can be eliminated with a finer noding or with an appropriate control density function for a node. This effect should not change the behavior of the fission rate averaged over several periods ($\sim$1 s). Figures 6 and 7 which show the radially averaged axial power at different times during the transient also demonstrate the importance of spatial neutron kinetics. The figures show that the spatial and temporal behavior is non-separable.

Effect of Recirculation Pump Trip (RPT)

The RPT occurs due to the pressure exceeding 8.03 MPa at 5 s. The resulting flow rates for (thermohydraulic) channels 2 and 3 (cf. Fig. 1) are shown in Fig. 8. (The drive loop flow rate and the pump speed after a RPT with only a feedwater trip and no control rod insertion have been compared with GE results for up to 10 s [5] and found to be within 3%. Channel 3 contains control rods after the reactor trip and channel 2 is unrodded. Because the power is higher in channel 2 than in channel 3, the flow

TABLE I

TRANSIENT CHRONOLOGY

Time (s)	Event/Comment
0	MSIV begins to close initiating the transient. It closes completely in 4 seconds. System pressure rises, voids collapse, and reactor power rises.
2	Feedwater trip. Feedwater reduced to zero in 18 s (see Fig. 2).
2.2	Power reaches 120% of rated power. After 0.3 s delay half of the control rods are inserted in 4 s.
3.8-4.0	System pressure reaches the setpoints of relief valve banks 1-3. All valves stay open until approximately 15 seconds.
4.1	Overall reactor power peaks at 2.9 times the steady-state power. Unscrammed side of reactor peaks at 3.5 times the steady-state power.
4.5	System pressure reaches 8.03 MPa. After 0.53 s delay recirculation pump trips.
5-6	Power in scrammed half of reactor at decay heat levels and remains so for duration of transient.
6-7	Average fuel temperature for hottest node peaks at 1300°C.
8	System pressure peaks at approximately 8.3 MPa.
15	Liquid velocity in downcomer drops below bubble rise velocity setting up situation for countercurrent flow.
20	Feedwater shut off completely.
30-400	Relief valve bank 1 opening and closing drives the system pressure in an oscillatory manner. Other system variables including reactivity follow.
35	Downcomer water level at temporary low. Feedwater turned on to simulate control system response.
35-120	Water level rises above steam separator skirt.
60	Recirculation pump coastdown completed.
155	Feedwater flow rate reduced to zero.
140-350	Water level decreases.
180-220	Liquid in vessel approaches saturation temperature following removal of subcooled feedwater. Average void fraction in core increases from 0.23 to 0.30. In response to system pressure oscillations, liquid in vessel goes from being superheated to subcooled and vapor generation rate changes accordingly.
300	Boron injection is initiated.
320	Boron enters core.
350	HPCI flow initiated.
360	Water level stabilized. HPCI flow approximates the steamline flow.
375	Subcooled water from HPCI injection reaches core entrance.
400	End of demonstration run.

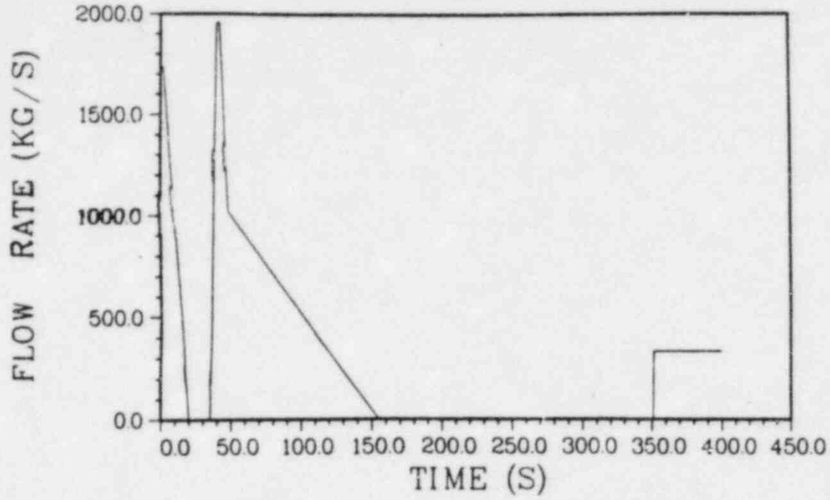

Figure 2 Flow Rate at Feedwater Sparger

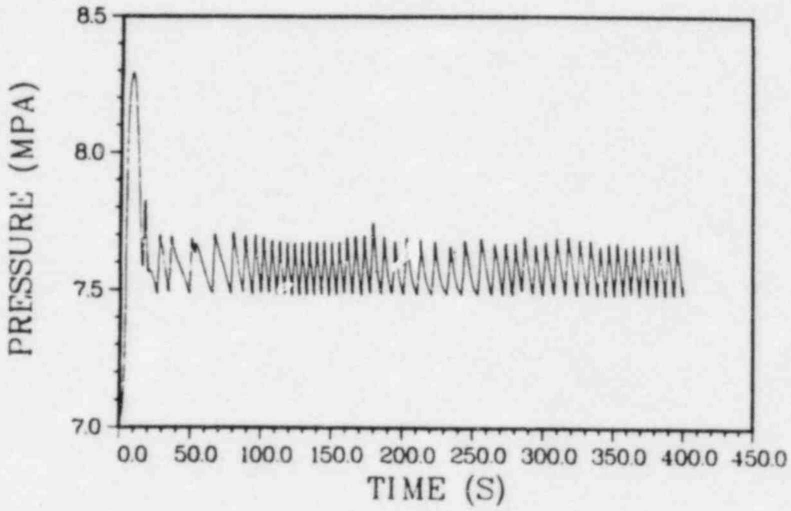

Figure 3 System Pressure

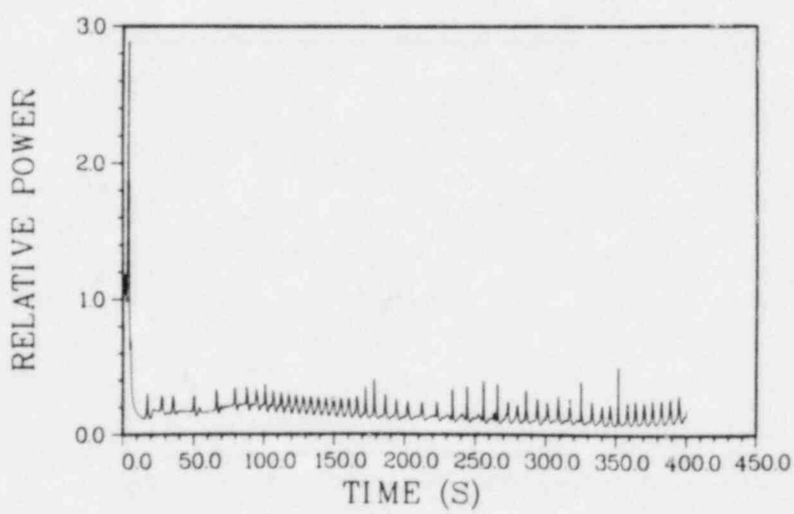

Figure 4 Relative Core Power

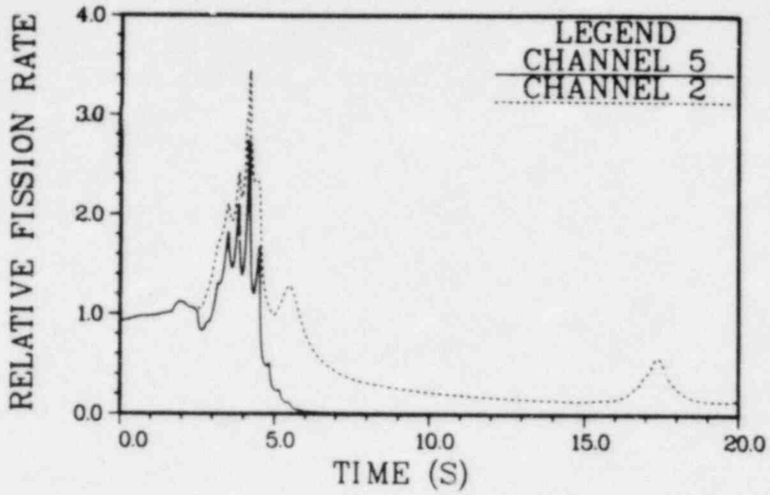

Figure 5 Fission Rate in Neutron Kinetics Channels

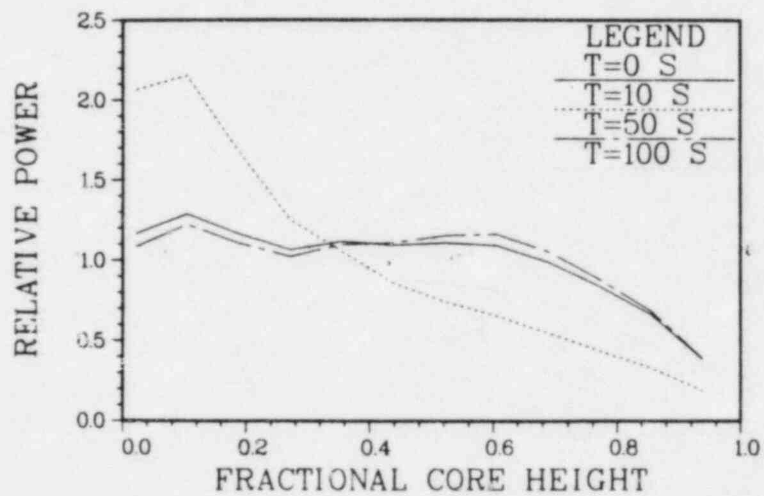

Figure 6 Axial Power Distribution at Different Times

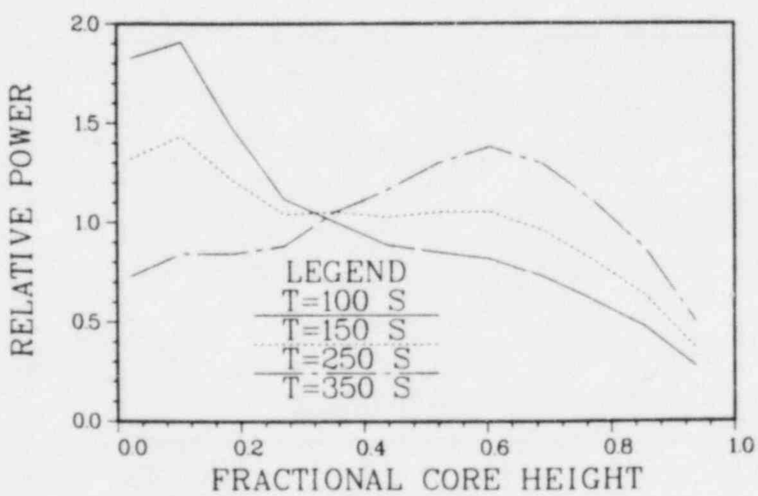

Figure 7 Axial Power Distribution at Different Times

rate is also higher. Figure 8 shows that the natural circulation flow rate in this transient is 25-30% of the steady-state flow rate.

By reducing the flow rate to natural circulation, the void fraction is maintained reasonably high. Because of void feedback the reactor power is kept lower than might have been attained otherwise. (Note that the pressure is approximately 7.6 MPa during the transient which is higher than the initial value of 7.0 MPa, and this by itself would decrease the void fraction.) This is an important strategy for dealing with this type of accident. Figure 4 shows that the time-average relative power is $\sim$ 20% up to $\sim$150 s and $\sim$15% during the latter part of the transient.

Effect of Relief Valve Cycling

The initial overpressurization (cf. Fig. 3) is sufficient to open all three banks of relief valves. However, after a 10 s period of venting and a reduction in core power, the pressure is only sufficient to actuate one bank of valves. Since the MSIVs have closed, the vapor generated in the core exhausts through this bank. The opening of valves reduces pressure, and hence, the valves open and close in a fairly regular cycle. The flow rate through all the valves is given in Fig. 9. The period of the valve cycling ranges from 5-15 s. Hand calculations at two different time intervals ($\sim$70 s and 130 s) confirmed the periods calculated by the code. The variation of this time period is expected since the rate of pressure drop when the valve is open is primarily governed by the difference between the steam flow rate out of the vessel and the total vapor generation rate in the vessel, i.e., a small difference between two large numbers.

The cycling of the relief valves affects the system pressure (cf. Fig. 3) and this in turn cause the oscillation in power during the transient (cf. Fig. 4). These oscillations can also be seen in the flow rates shown in Fig. 8 and in the core-average void fraction.

The flow rates at the core exit for a channel in which the power is relatively high and for a channel with only decay heat are not in phase. During the latter half of the transient they are close to 180° out-of-phase with the flow rate increasing in one channel when it is decreasing in the other and vice-versa.

Water Level in the Vessel

Figure 10 shows the water level in the downcomer region. The water inventory in the vessel and the level in the downcomer are closely related to the feedwater flow, as can be seen by comparing Fig. 2 with Fig. 10. The water level calculation in RAMONA-3B takes into account that steam voids are present in the downcomer; the measurement in a BWR uses an instrumentation line in which the void condition may be different. Hence, care must be exercised in interpreting the calculated water level.

Core Inlet Subcooling

The core inlet subcooling is a strong function of the feedwater flow rate. The inlet subcooling (along with the flow rate and power) helps determine the void fraction in the core. When the inlet subcooling is close to zero at times greater than 200 s, the void fraction significantly increases. This reduction in inlet subcooling corresponds to a reduction in subcooling throughout the vessel. When this occurs, the pressure oscillations cause the vessel liquid state to oscillate between subcooled and superheated and flashing and condensation can take place.

The Effect of Boron

Boron enters the vessel at the location of the jet pump at 300 s. It starts to enter the core approximately 20 s later and increases to a concentration of 20 ppm by the time the calculation is terminated at 400 s. This is insufficient to shut

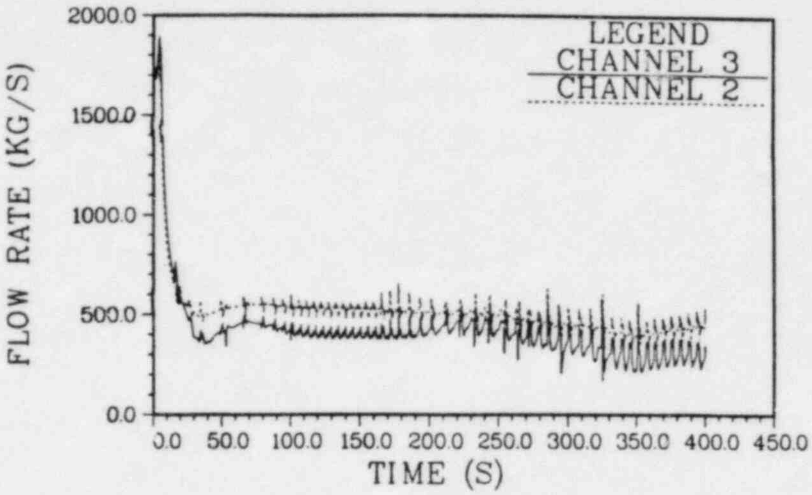

Figure 8 Inlet Flow Rate in Different Thermohydraulic Channels

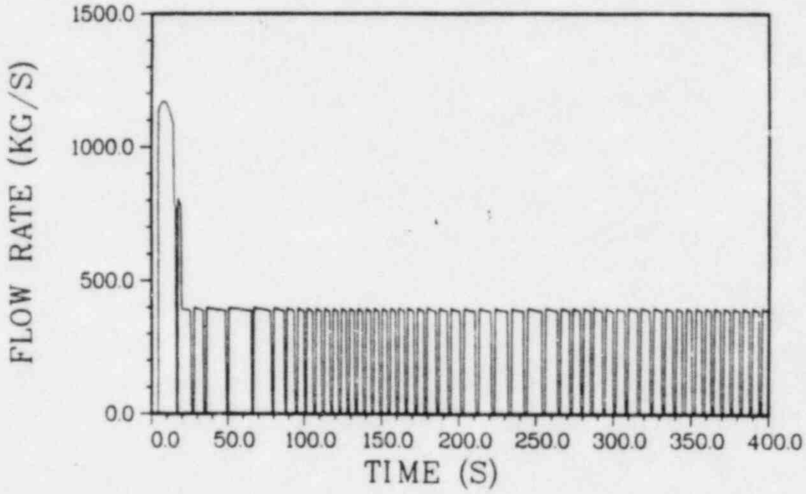

Figure 9 Steam Flow Through Relief Valves

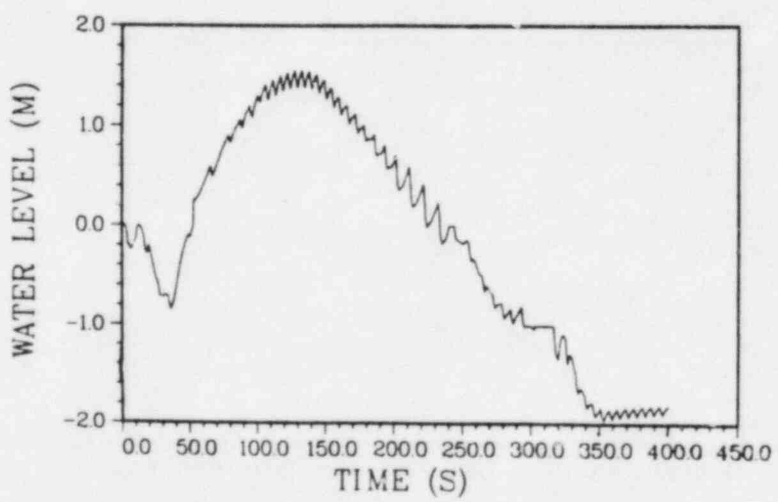

Figure 10 Mixture Water Level in Downcomer

down the reactor. It is expected to take $\sim$400 ppm of boron before the fission rate is reduced to a negligible level and this will take approximately 2000 s.

CONCLUSIONS

RAMONA-3B has been successfully applied to a transient initiated by MSIV closure in which only the control rods in half of the core are inserted after reactor trip. This accident is an excellent example of a situation in which it is important to model the steam supply system with a core component that includes three-dimensional neutron kinetics. RAMONA-3B is unique in its ability to satisfy this requirement. This calculation has involved not only the basic vessel and steam line thermohydraulic models and the neutron kinetics model, but also the activation of different parts of the control and protection system model.

The code calculated over a range of conditions encountered in 400 s of reactor time with a (reasonable) computing time on BNL's CDC-7600 equipment that averaged $\sim$15 times reactor time. The results are in qualitative agreement with those reported elsewhere [2], i.e., there is an initial overpressurization phase followed by a long period of operation with natural circulation characterized by oscillations with a frequency of $\sim$0.1 Hz. This oscillatory behavior is reflected in power, pressure, flow and other system variables. Quantitative comparisons with other calculations [2,3] are not possible because of differences in the assumptions used in each calculation.

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