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RAMONA-3B APPLICATION TO BROWNS FERRY ATWS*

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1.0 INTRODUCTION

The Anticipated Transient Without Scram (ATWS) is known to be a dominant accident sequence for possible core melt in a Boiling Water Reactor (BWR). A recent Probabilistic Risk Assessment (PRA) analysis[1] for the Browns Ferry nuclear power plant indicates that ATWS is the second most dominant transient for core melt in BWR/4 with Mark I containment. The most dominant sequence being the failure of long term decay heat removal function of the Residual Heat Removal (RHR) system.

Of all the various ATWS scenarios, the Main Steam Isolation Valve (MSIV) closure ATWS sequence was chosen for present analysis because of its relatively high frequency of occurrence and its challenge to the residual heat removal system and containment integrity. Therefore, this transient has been, and continues to be, analyzed by various organizations using various computer codes. However, most of the prior efforts have been carried out using point-kinetics codes.

Early deterministic analyses revealed a large variation in predicted power levels during an ATWS with the water level lowered to the top of the active fuel (TAF), as required by the Emergency Procedure Guidelines (EPG).[2] RELAP5/MOD1.6 results[3,4] predicted power levels of ~8%, which compared well with General Electric's statement[2] and Oak Ridge National Laboratory's prediction[5] of ~9%. On the other hand, the Electric Power Research Institute using spatial kinetics codes predicted[6,7] power levels of 15-18%. Therefore, with so many different predicted power levels, RAMONA-3B with 3D neutronics was used by the SASA program to provide best estimate ATWS calculations[8] with plant specific neutronic macroscopic cross sections from a TVA nuclear power plant using the P8X8R fuel.

The objective of this paper is to discuss four MSIV closure ATWS calculations using the RAMONA-3B code. The paper is a summary of a report being prepared[8] for the USNRC Severe Accident Sequence Analysis (SASA) program which should be referred to for details.

*Work performed under the auspices of the U.S. Nuclear Regulatory Commission.

2.0 RESULTS

The four MSIV closure ATWS sequences will be discussed in this section. The scenario and conclusions are presented with each transient.

2.1 Transient 1

The transient scenario overview is presented in Table 1. Essentially, this transient models a total failure to SCRAM followed by a recirculation pump trip at high pressure after the MSIV closure. The safety injection water (HPCI and RCIC) is throttled such that the downcomer water level drops to and then remains at the top of active fuel (TAF). Depressurization of the reactor vessel is initiated when the pressure suppression pool (PSP) water temperature reaches the value imposed by the heat capacity temperature limit (HCTL) curve for the PSP. This calculation has been performed mainly to evaluate the water level control of the Emergency Procedure Guidelines (EPGs). However, no boron injection or manual control rod insertion was performed.

The predicted reactor power (Figure 1) attained a maximum of 265% in about three seconds after the closure of the MSIV and returned to ~30% with the water level near the normal operating point. (For better resolution, the initial power spike on the plot is not shown.) As the water level was lowered to TAF (Figure 2) and maintained there, the power decreased to ~20% of rated power. This quasi-steady power level persisted until the HCTL was reached, requiring the operator to depressurize the system (Figure 3). This action increases the negative worth of the void reactivity feedback (Figure 4) in the core because of the increase in the core average void fraction because of flashing in the vessel and the increase in the specific volume of the vapor. The overall effect was the reduction of the relative power to ~15% after the depressurization. (The reactivity plots of Figure 4 are used for qualitative analysis only.)

The integral effect of the predicted power history can be seen in Figure 5 where the PSP water temperature reaches the assumed HPCI failure point of 190°F in ~23 minutes. This may lead to the overpressurization failure of the primary containment in about 20 additional minutes[5] because of the large amount of water injected into the vessel (causing core power increase) after the LPCI and condensate booster pumps (CBP) become active by low system pressure.

Transient 2

This transient has the same scenario as Transient 1 with the addition of the effect of manual rod insertion (MRI) superimposed on the calculation. Thus, in Table 2, the events are identical with the previous transient except that the MRI action begins after 150s.

The most difficult part of this transient was to determine a realistic control rod insertion strategy that an operator would choose along with a practical insertion speed. While it is true RAMONA-3B could easily be programmed to insert the high worth rods during the transient, the knowledge gained would be of minimal use since the code would insert rods based on data an operator does not have. There is no optimum insertion strategy because of the nature of the burnup process and the constantly changing control rod pattern during the life of a fuel cycle.

The information on the realistic insertion strategy was obtained from Mr. S.A. Hodge of Oak Ridge National Laboratory, who supplied Brookhaven National Laboratory with video tapes of ATWS simulation sessions conducted at Browns Ferry Simulator on August 20, 1983, under the auspices of USNRC/SASA program. The simulator session chosen to be used as the model for the RAMONA-3B calculation is shown in Figure 6. The resulting RAMONA-3B insertion pattern is shown in Figure 7 where it should be noted that the insertions after 1229s were assumed since the TVA simulator session ended by that time.

Additional information used in the transient was the fact that the rod worth has a maximum between notches 30 and 8 (i.e., 4.5 to 10 ft.). Thus any rod near the maximum worth range would be driven in by an operator before a rod that would be inserted from the withdrawn position. Also, the RAMONA-3B control rod insertion was ended at 10.4 ft. to simulate the operator's stoppage of the insertion process outside the maximum worth band of a control rod.

Another important piece of information taken from the video was the fact that the operator, during the ATWS, could not devote all of his time to the control rod insertion process. Therefore, in RAMONA-3B calculation, a speed of 2 in./s (out of a maximum of 3 in./s) was used to insert any given rod. As a final point, it should be noted that a 1/4 core model was used (the checked lines in Figures 6 and 7) in the RAMONA-3B calculation implying that the negative reactivity insertion history for the model is different from reality where only one rod is driven into the core at a time. However, after the four (or two for boundary) rods have been inserted, the correct rod worth is present.

The results of the MRI action can be seen in Figure 8 where the relative power is reduced to 14% at 389s with 6 rods inserted and the power drops to 11% by 1229s with 20 rods inserted. This should be compared against the previous calculation (i.e., Transient 1) where the average relative power over the simulation was ~18% (Figure 1). The depressurization began at 1229s when the HCTL (i.e., 160°F) for the system pressure was reached (which was the same PSP water temperature used before). The negative SCRAM (or control rod) reactivity shown in Figure 9 along with the increase in void reactivity (i.e., increase in core average void) caused by the depressurization is enough to put the reactor on a negative period resulting in a predicted relative power of ~5% at 1410s. However, power/void oscillations can occur since the positive reactivity from the Doppler (i.e., fuel temperature) and moderator temperature feedback is larger than the negative scram reactivity. The power would increase until a certain void fraction is reached (i.e., when the reactivities sum to zero and turn negative) causing the power to decrease. This will generate less voids resulting in an insertion of positive void reactivity - causing the cycle to repeat itself. Power spikes should be avoided because of the effects on the system and their effects on the operator's instrumentation. Of course, these power spikes will eventually be damped out as soon as the operator inserts enough rods to introduce sufficient negative SCRAM reactivity to nullify the positive reactivity. To determine how many rods are needed for the control rods to become the dominant controlling factor for these mitigative actions, another calculation was performed without depressurization.

In Figure 10 the MRI calculation without depressurization indicates that the relative power drops to about ~6% after 28 rods (at ~1700s) have been inserted. Figure 11 shows that by 1800s the SCRAM reactivity is clearly the dominant negative reactivity controlling the effort to bring the plant to a hot shutdown. The void reactivity (i.e., core average void) still determines the resultant power level, but its effect is greatly reduced. (An example of a power/void spike can be seen at ~1650s in Figure 10, and the corresponding void reactivity swing can be found in Figure 11 at the same time.) Consequently, if the operator depressurized the system after inserting ~32 rods, the core should attain the decay heat level with low amplitude power/void spikes until a sufficient number of rods have been inserted to completely remove the void effect (i.e., hot shutdown).

The heatup rate for the PSP for all three transients discussed so far can be seen in Figure 5. While Transient 1 went into HPCI failure at 23 minutes, both MRI predictions show a large delay in the time to HPCI failure. The MRI heatup rates would level off since both calculations were terminated at low power. The PSP water temperature could eventually be turned around if the residual heat removal coding system (i.e., the RHR with its ~3% of rated power cooling ability) could be turned on. Otherwise, HPCI failure is inevitable with all its repercussions.

As a final comment for Transient 2 (which applies to all ATWS best estimate calculations), strong spatial effects in neutronics were obvious throughout the calculation because of the strong void feedback. A good example of this effect can be seen in Figure 14 were the axial power shape at 1229s completely inverts as compared to the power shape at other times. This demonstrates that point kinetics cannot be used for ATWS calculations. These conclusions have recently been stated in a letter[10] discussing knowledge learned in using the TRAC-BF1 code at INEL with 1D neutronics. The statement in the letter reads as follows:

> "During the mitigation of an ATWS in a BWR, the operator can implement a reactor power reduction technique termed level control. Level control is implemented subsequent to the tripping of the recirculation pumps, and introduces into the reactor an independent degree of freedom which is normally controlled by the automatic level control system. By dropping and maintaining the liquid level in the downcomer at a level equivalent to the top of the active fuel, the overall power of the reactor can be reduced. Such a dramatic change in the liquid level in the reactor has equivalent effects on the void, flow, and power patterns within the core. It has been observed that during the mitigation of hypothetical ATWS, large variations in these patterns can take place. The large

changes in, for example, core void profile, which level control necessitates results in large variations on the neutron flux and reactor power profiles within the core. These variations couple back to the thermal-hydraulics in a manner which cannot be separated. Thus, it becomes very difficult, if not impossible, to apply a point kinetics model to the simulation of such a transient."

Transient 3

The scenario for this transient can be found in Table 3. Essentially, the events are similar to the previous scenarios except for the fact that the HPCI system fails to operate. However, before discussing the RAMONA-3B predicted results, a brief description of the 1D neutronic core model used for this calculation should be given.

The 1D set of macroscopic cross sections used for this transient was generated from the 3D set of cross sections used for other transients. By using the RAMONA-3B method[8] to collapse the 3D cross sections down to an equivalent set of 1D cross sections, an accurate and fast running plant model was created to run RAMONA-3B in the 1D mode. An example of the success of this effort is shown in Figure 12 where the power histories of the two calculations almost coincide for 800s. This in itself is extremely convincing that the 1D cross sections are accurate. In addition, comparison of the total reactivities, Figure 13, proves that both sets of cross sections produce similar core reactivities responses for the same stimuli. This not only validates the RAMONA-3B 3D to 1D collapsing method, but also verifies the 1D coding in RAMONA-3B. The 1D version ran at a CPU-TO-REAL Time ratio of about 2.

The downcomer water level history is presented in Figure 15. After 70s the water level drops below TAF since the RCIC and control rod drive hydraulic system (CRDHS) flow is not sufficient to maintain the vessel inventory. The effect of lowering the downcomer water level was a reduction in the hydrostatic driving head causing low flow through the core, as seen in Figure 16. The overall result was that a relative power level of 4% of rated power was predicted by RAMONA-3B at 150s, as shown in Figure 17. No CHF was detected during the simulation.

Transient 4

The scenario for Transient 4 is identical to that found in Table 1 except that the recirculation pumps do not trip off, i.e., they continue to run during the transient. Although it is recognized that this event is highly unlikely to occur, the simulation was performed to determine the reactor power, system pressure, and water level drop during such an event to study the importance of this trip.

The results are shown in Figure 18 through Figure 20. In Figure 19, the power is shown to stabilize at about ~80% of rated, making the steam flow below the rated maximum of the SRVs, which is 85% of full power steam flow rate. The system pressure, Figure 19, peaks at 1340 psia and levels off to ~1310 psia, dropping below the fracture pressure for the vessel. The most significant graph is shown in Figure 20 where the downcomer water level is shown dropping quickly because the mass of steam leaving through the SRVs is larger than the ECC water entering the reactor vessel. Essentially, this calculation dictates that the operator must verify that the recirculation pumps have tripped after an ATWS has been identified to be in progress, and this must be done quickly.

3.0 CONCLUSIONS

RAMONA-3B has been used to calculate four MSIV closure ATWS scenarios. Conclusions resulting from the study are summarized in this section.

3.1 Level control (Contingency #7 of EPGs) and pressure control reduce the reactor power from 30 to ~18% during the course of the transient. HPCI failure occurs in ~23 minutes. No MRI or SLC was modeled during the transient.

3.2 Level and pressure control along with MRI will delay the time to HPCI failure. Although the reactor power can be reduced to 6% with ~20 rods inserted after the system has been depressurized, power spikes are expected because the void reactivity is comparable to the SCRAM (or control rod) reactivity and both are below the positive reactivity supplied from the Doppler and Moderator feedback. By waiting until ~32 rods have been inserted before depressurization, the negative SCRAM reactivity would be the dominant feedback effect and large enough to lower power close to the decay heat level. However, the PSP water temperature is very high and HPCI failure will eventually occur unless the RHR cooling is turned on. No SLC was modeled during this transient.

3.3 The High Pressure Boil Off calculation (i.e., Transient 3) predicted that the power would drop to ~4% at 150s with the water level ~4ft below TAF. No CHF was detected during the first 150s. RCIC and CRDHS flow is enough to sustain ~2.7% of rated power without loss of liquid inventory in the vessel. Thus, if core power can be sustained below 2.7%, no fuel damage would be expected.

3.4 If the recirculation pumps do not trip during an MSIV closure ATWS, the core power stabilizes around 80% of rated. Thus, the steam flow rate is below the SRV maximum capacity of 85% of full power steam flow rate. The peak pressure of 1340 psia was calculated. While the calculation showed that the reactor vessel was safe from an overpressurization failure, the water level was dropping rapidly. This calculation demonstrates that the operator should verify the recirculation pump trip as his first action during an ATWS event.

3.5 The RAMONA-3B calculations showed very strong neutronic spatial effects during the transients because of the void feedback in a BWR. Thus, point kinetics may not be appropriate for these transients.

4.0 RECOMMENDATIONS FOR FUTURE WORK

The boron injection (or SLC) issue was not addressed during these calculations. To evaluate the effectiveness of the SLC system, the problem of boron stratification in the lower plenum must be studied. While applying level control to an ATWS will reduce the power, it will also lower the flow rate into the core creating a reduction in the boron mixing efficiency. The effectiveness of the SLC system in mitigating an ATWS is a technical issue which remains to be solved.

5.0 REFERENCES

1. S.E. Mays et al., Interim Reliability Evaluation Program: Analysis of Browns Ferry, Unit 1, Nuclear Plant, NUREG/CR-2802, EGG-2199, July, 1982.

2. BWR Emergency Procedure Guidelines, Revision 2 (Draft) Appendix B, July, 1982.

3. W.C. Jouse, "INEL BWR Severe Accident ATWS Study," Eleventh Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, October 24-28, 1983.

4. W.C. Jouse, "Sequence Matrix for the Analysis of an ATW in a BWR/4; Phenomena, Systems and Operation of Browns Ferry Nuclear Plant Unit 1," Draft, May 1984.

5. R.M. Harrington and S.A. Hodge, ATWS at Browns Ferry Unit One, NUREG/CR-3470, ORNL/TM-8902, July, 1984.

6. B. Chexal et al., Reducing BWR Power by Water Level Control During an ATWS: A Quasi-Static Analysis, NSAC-69, May, 1984.

7. C.E. Peterson et al., Reducing BWR Power by Water Level Control During an ATWS: A Transient Analysis, NSAC-70, August, 1984.

8. P. Saha et al., "RAMONA-3B Calculations for Browns Ferry ATWS Study," to be published November, 1985, Brookhaven National Laboratory.

9. G.C. Slovik et al., "RAMONA-3B Applications to Browns Ferry ATWS, Twelfth Water Reactor Safety Research Information Meeting, Gaithersburg, Maryland, October 22-26, 1983, NUREG/CP-0058 (Vol. 3).

10. Letter, Mr. T.R. Charlton (Manager, Reactor Simulation and Analysis program) to Mr. F.L. Simo (Director, Reactor Research and Technology Division), dated September 10, 1985. Subject: Transmittal Documenting the Reduction of Neutronic Cross Section Data for TRAC-BF1-TRC-120-85. (See Attachment by Mr. Wayne C. Jouse dated September, 1985, bottom of page 2.)

Table 1. Sequence of Events for Transient 1.

- MSIV closure in 5 sec.
- Failure to SCRAM.
- Feedwater flow ceases in 8 sec.
- Recirculation pump trip at high pressure.
- Downcomer water level hits lo-lo level and HPCI and RCIC ramp up to 5600 gpm in 25 sec.

At 150 sec. operator takes control.

- Lowers HPCI and RCIC flow to drop the downcomer water level to top-of-active (TAF), and maintains there.
- Operator follows depressurization line according to PSP heat capacity temperature limit curve.
- HPCI shifts suction from CST to PSP at high PSP water level (high level of 15.2 ft).

Table 2. Sequence of Events for Transient 2.

- MSIV closure in 5 sec.
- Failure to SCRAM.
- Feedwater flow ceases in 8 sec.
- Recirculation pump trip at high pressure.
- Downcomer water level hits lo-lo level and HPCI and RCIC ramp up to 5600 gpm in 25 sec.

At 150 sec. operator takes control.

- Operator starts manually inserting control rods one by one.
- Lowers HPCI and RCIC flow to drop the downcomer water level to top-of-active fuel (TAF), and maintains there.
- Operator follows depressurization line according to PSP heat capacity temperature limit curve.
- HPCI shifts suction from CST to PSP at high PSP water level (high level of 15.2 ft).

Table 3. Sequence of Events for Transient 3.

- MSIV closure in 5 sec.
- Failure to SCRAM.
- Feedwater flow ceases in 8 sec.
- Recirculation pumps fail to trip at high pressure.
- HPCI and RCIC injections start when downcomer water level hits lo-lo level.

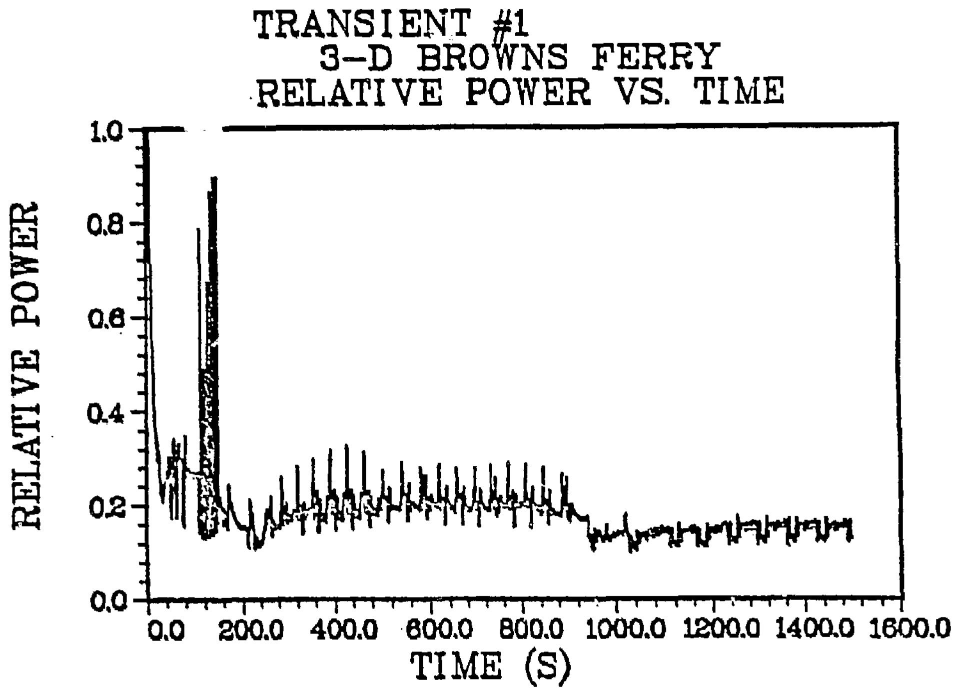


Figure 1. Reactor power prediction for Transient 1.

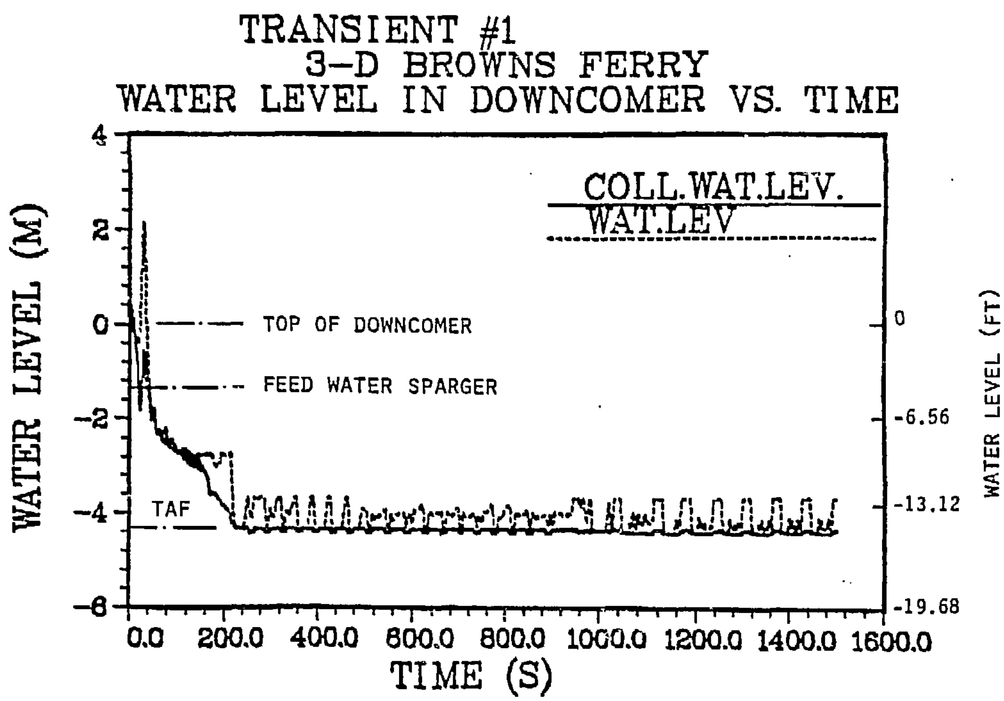


Figure 2. Downcomer water level prediction for Transient 1.

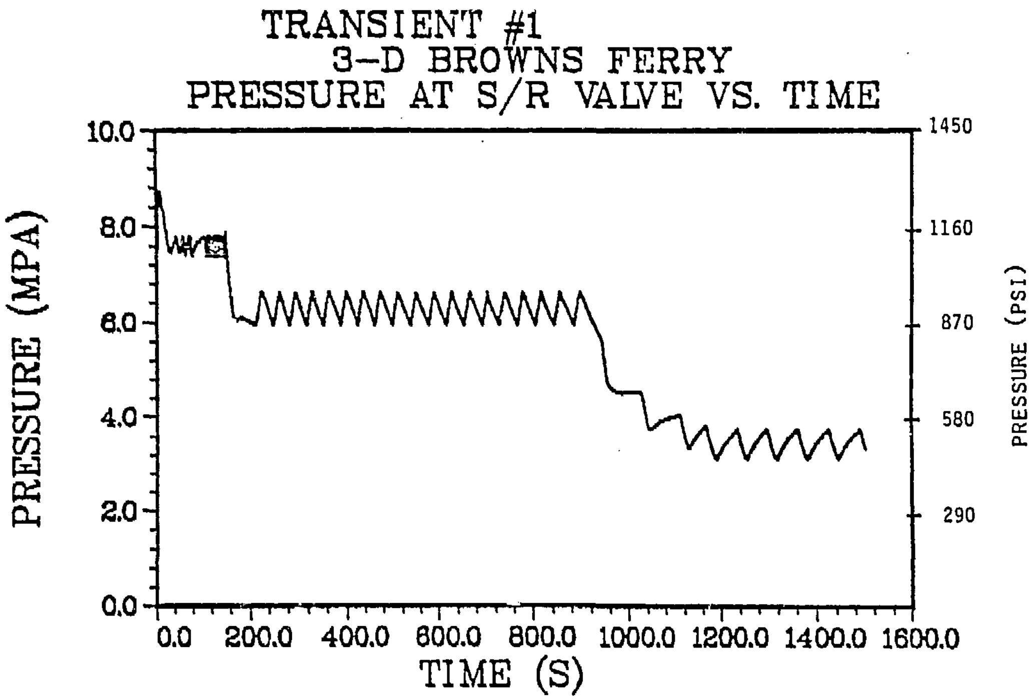


Figure 3. System pressure for Transient 1.

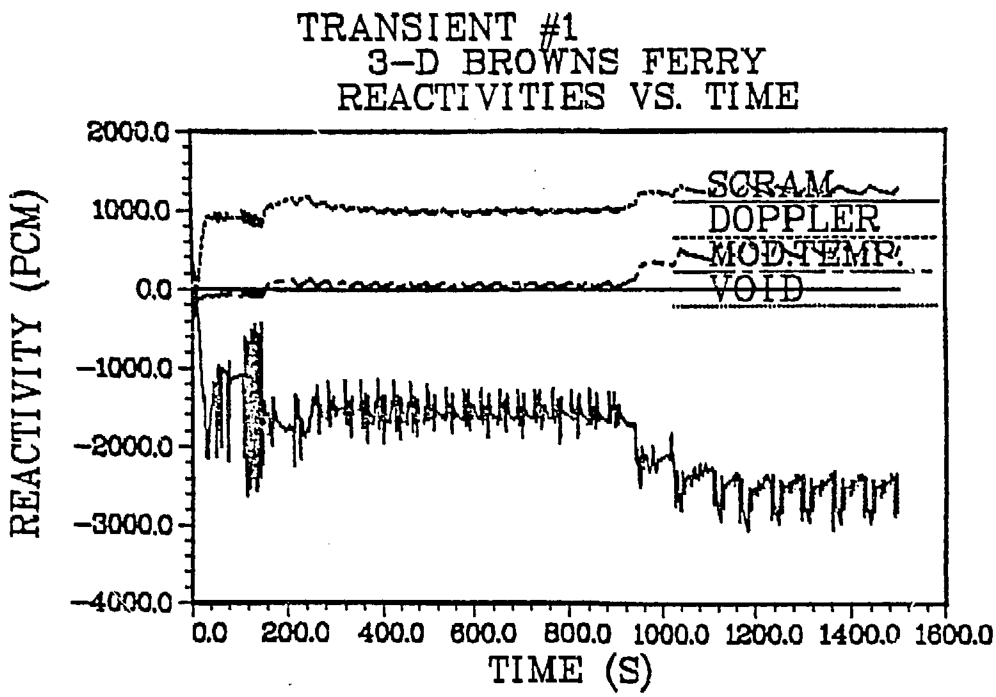


Figure 4. Reactivity predictions for Transient 1.

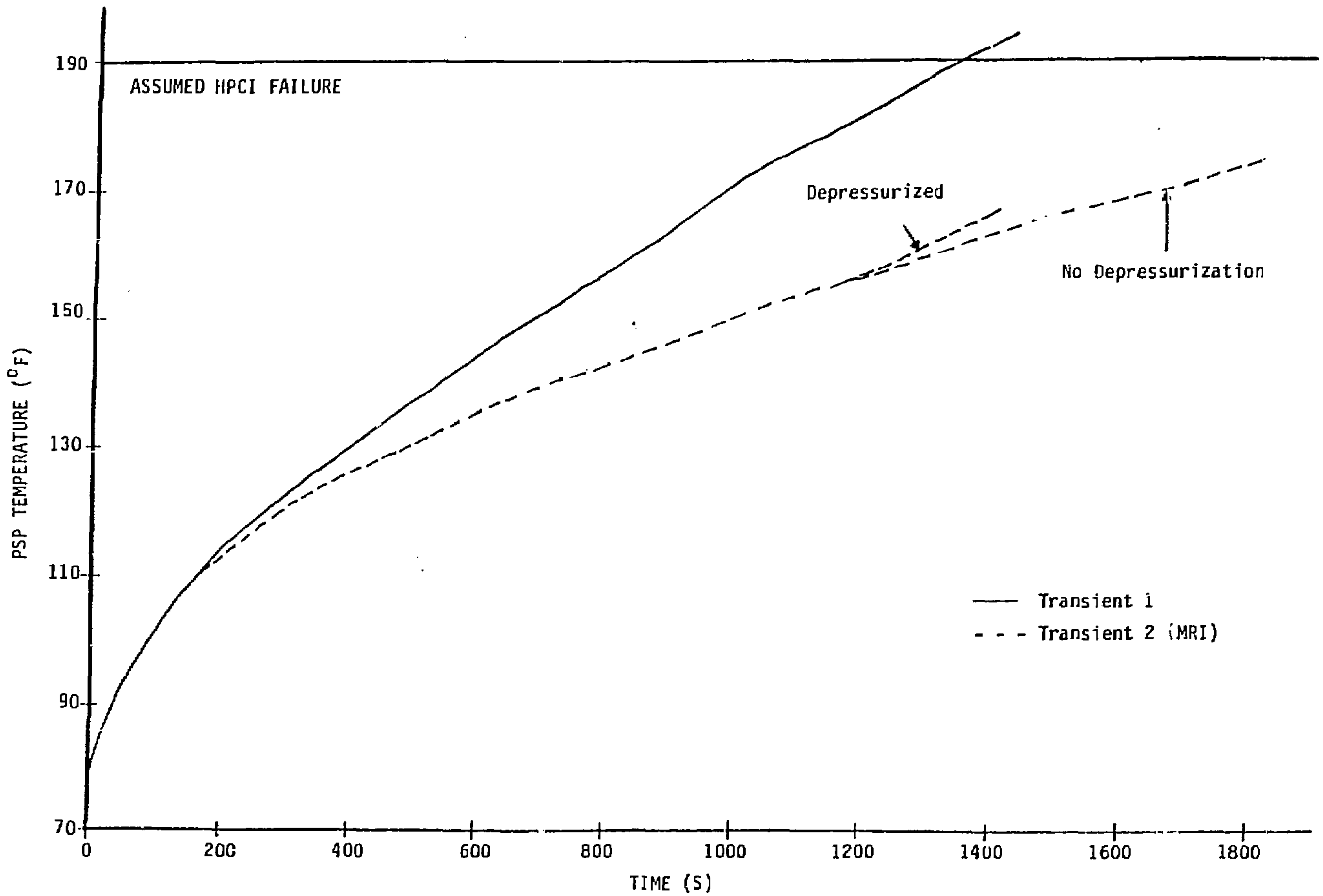
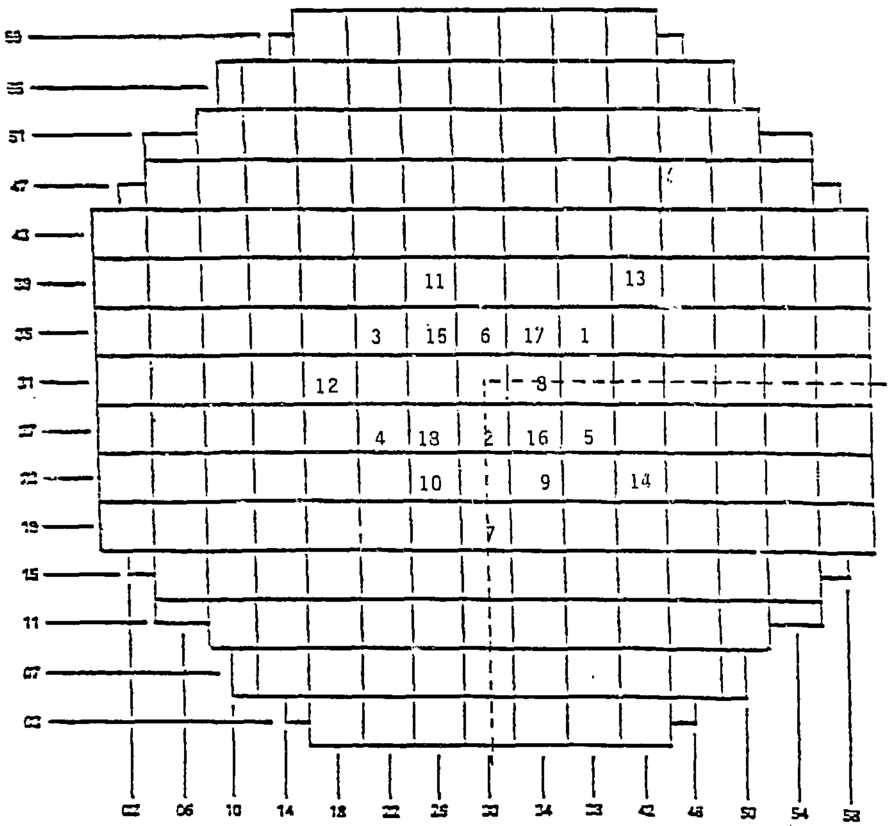


Figure 5. Comparison of PSP Water Temperature for Transient 1 and the Manual Rod Insertion Calculations

ROD INSERTION PATTERN AT SIMULATOR

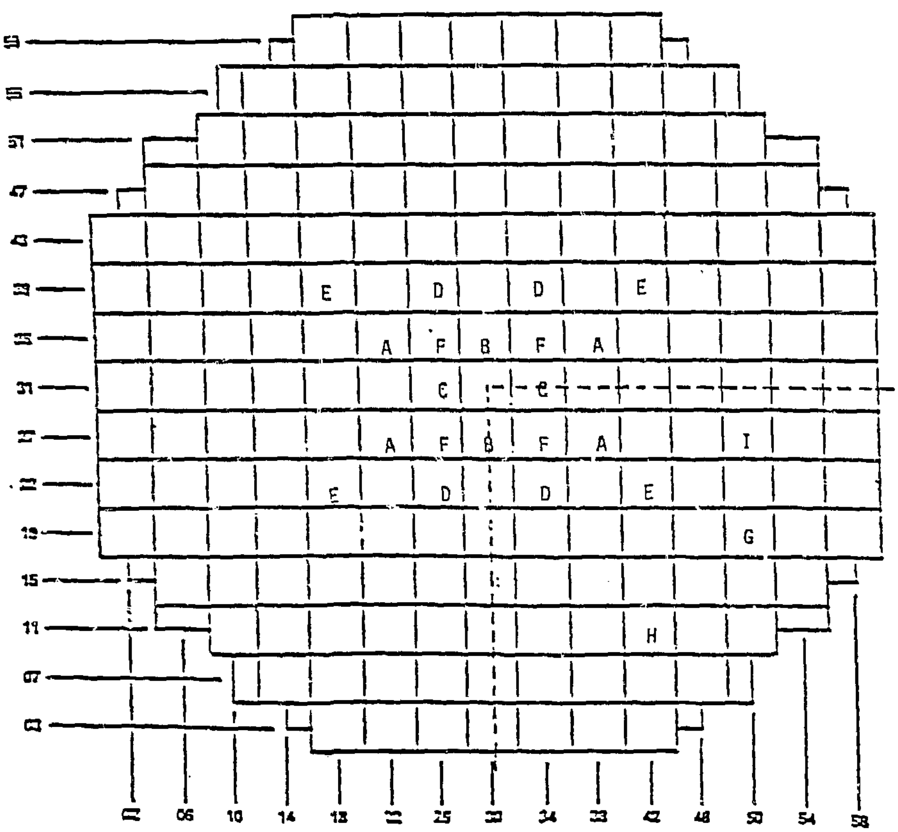
(TRANSIENT #7, AUGUST 20, 1983



* NUMBER INDICATES ORDER OF INSERTION

Figure 6. Control rod insertion pattern performed at TVA simulator during Session # 7.

RAMONA-3B ROD INSERTION PATTERN



ROD SEQUENCE	TIME	MOVEMENT
A	150 to 269 s	5.3 to 10.4 ft
B	269 to 389 s	0 to 10.4 ft
C	389 to 509 s	"
D	509 to 749 s	"
E	749 to 989 s	"
F	989 to 1229 s	"
G	1229 to 1469 s	"
H	1469 to 1709 s	"
I	1709 to 1949 s	"

Figure 7. Control rod insertion pattern used by RAMONA-3B.

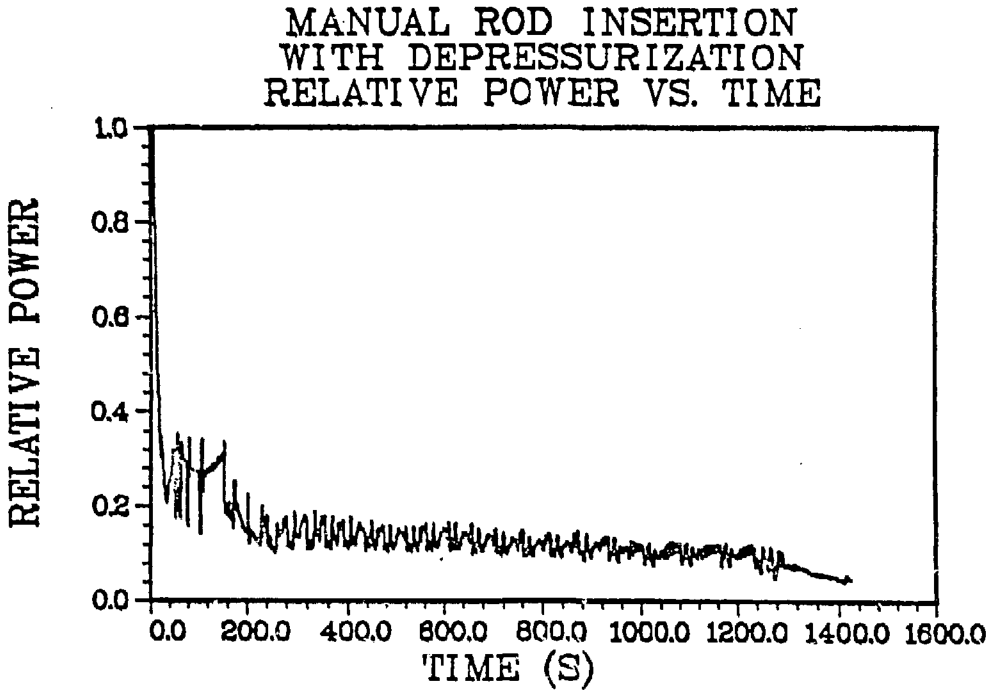


Figure 8. Reactor power history for Transient 2.

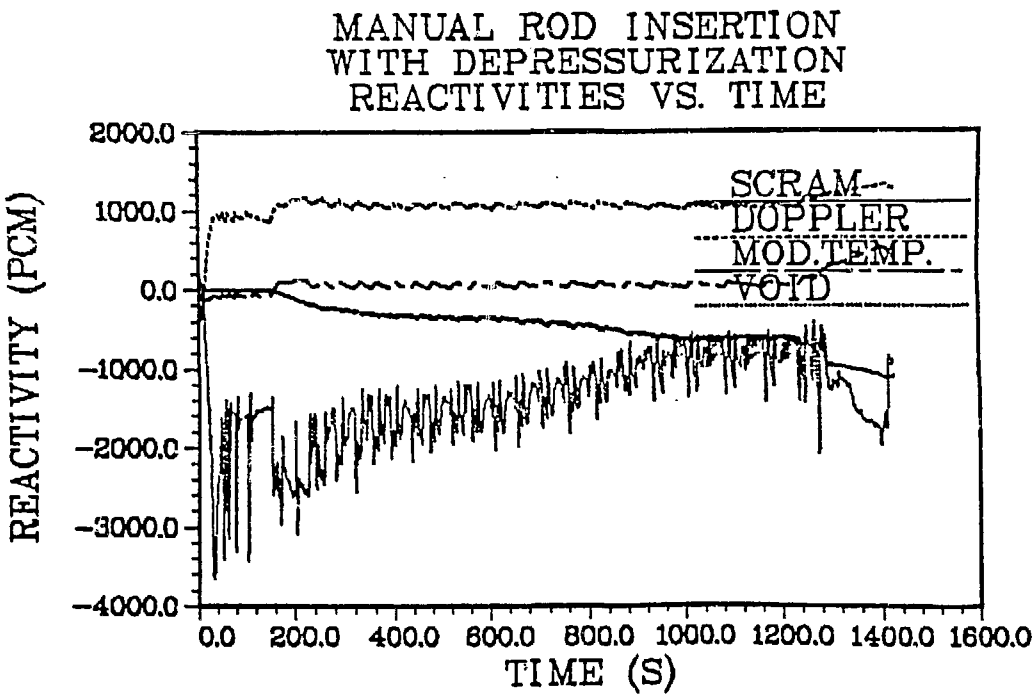


Figure 9. Reactivities for Transient 2.

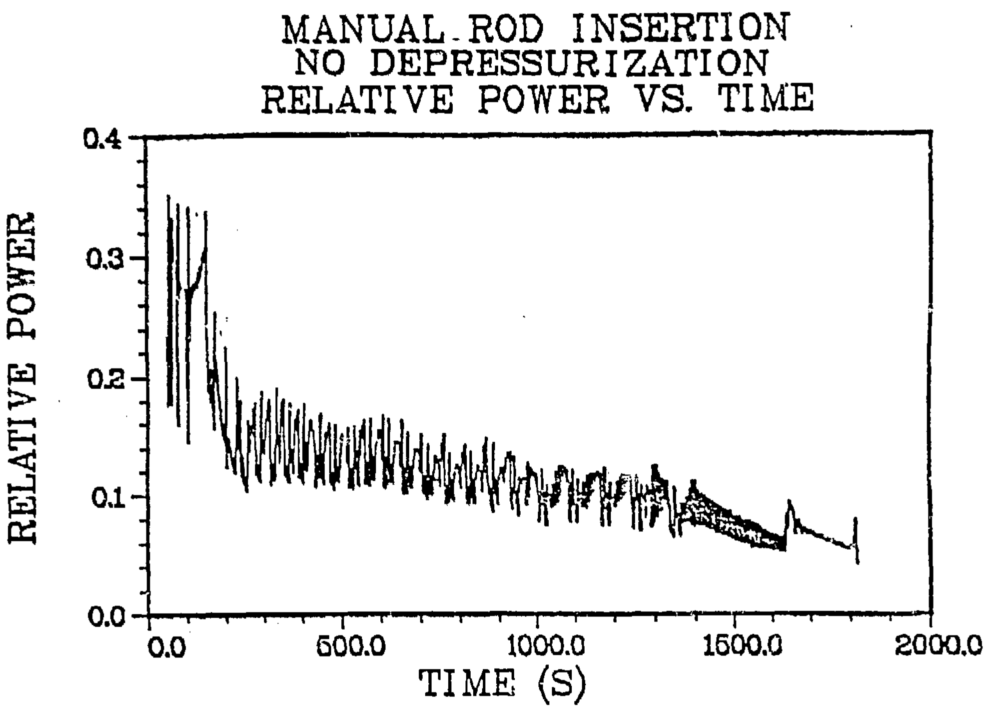


Figure 10. Relative power prediction for Transient 2 with no depressurization.

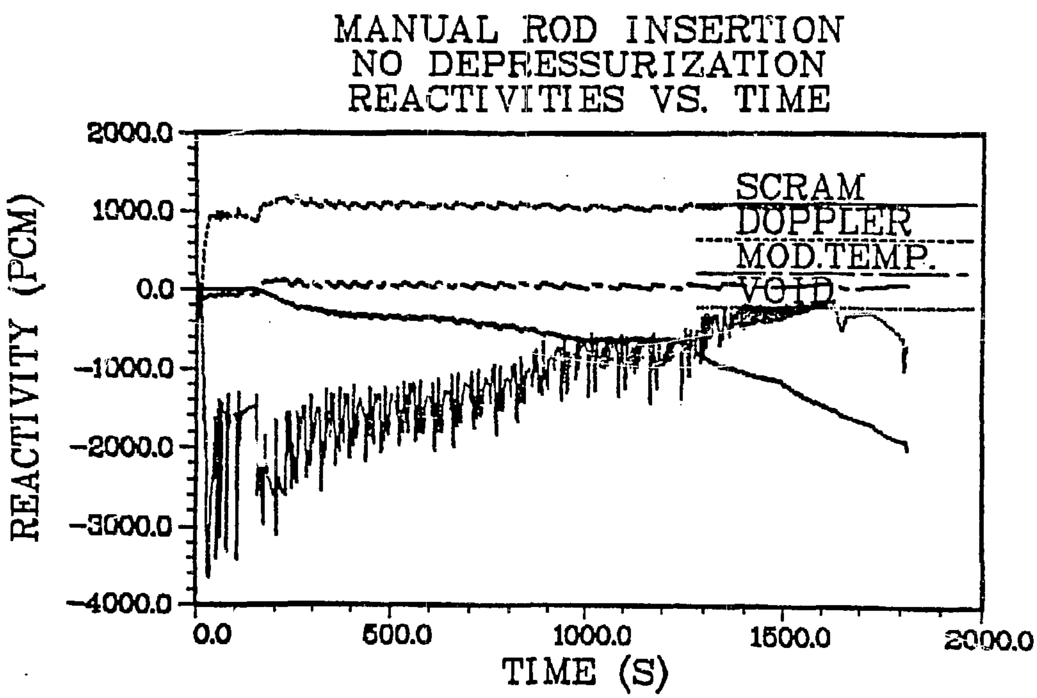


Figure 11. Reactivities for Transient 2 with no depressurization.

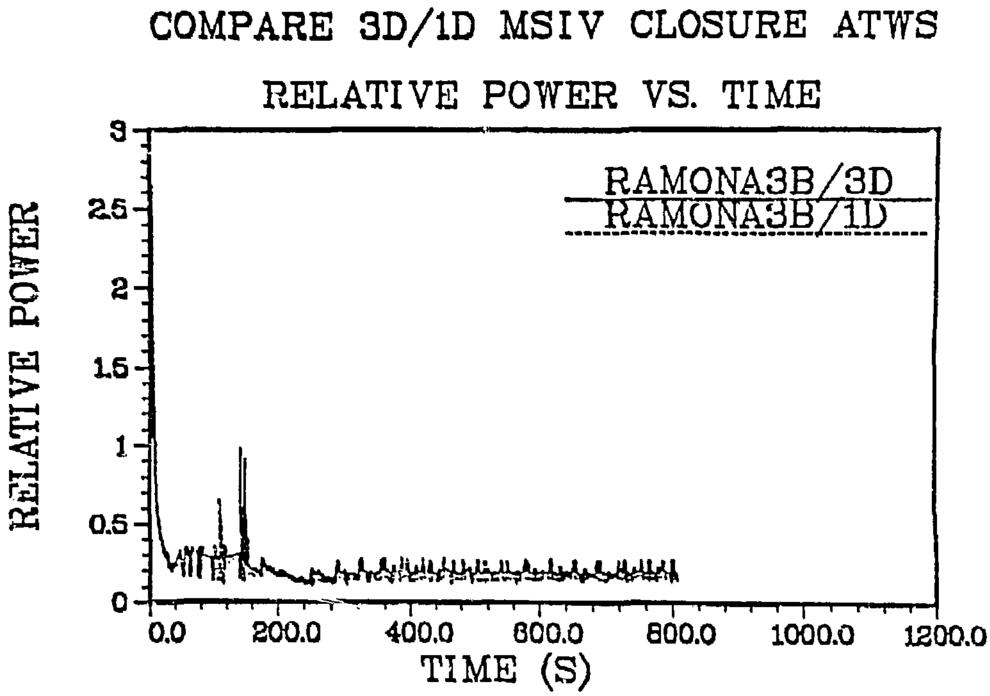


Figure 12. Comparison of reactor power for RAMONA-3B/3D and RAMONA-3B/1D.

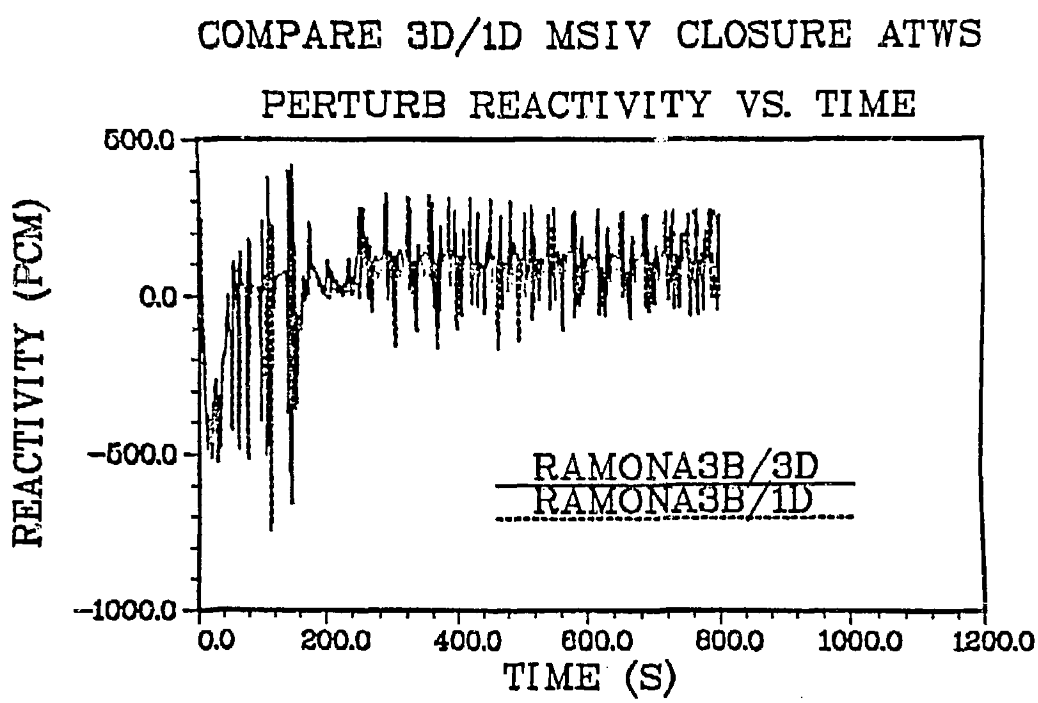


Figure 13. Comparison of total reactivities predicted for RAMONA-3B/3D and RAMONA-3B/1D.

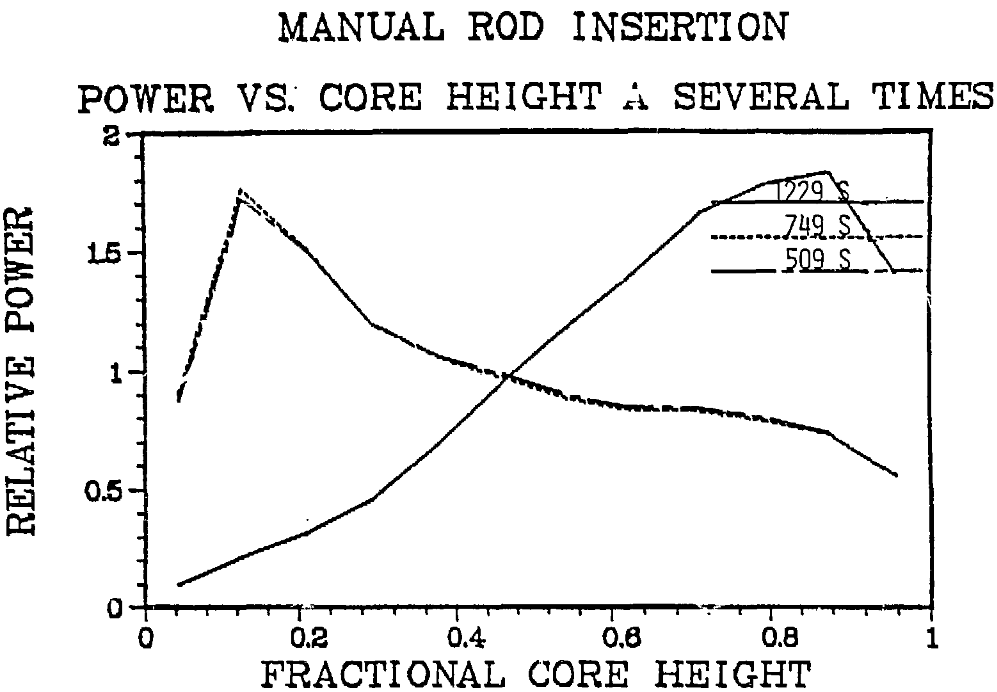


Figure 14. Axial power distributions for several times during Transient 2 (with depressurization).

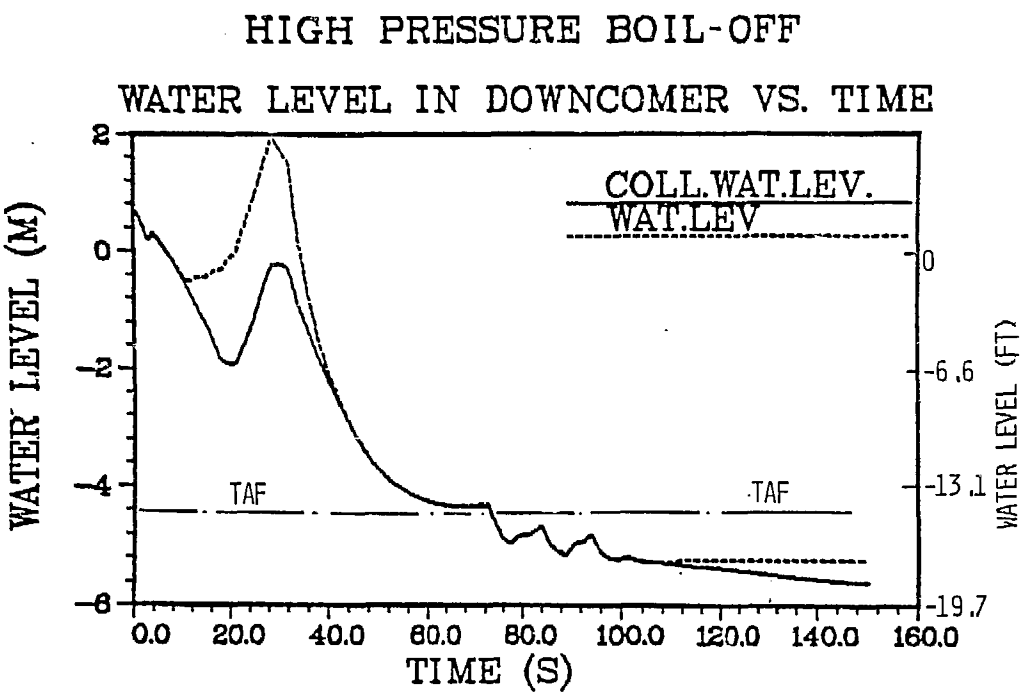


Figure 15. Downcomer water level predictions for Transient 3.

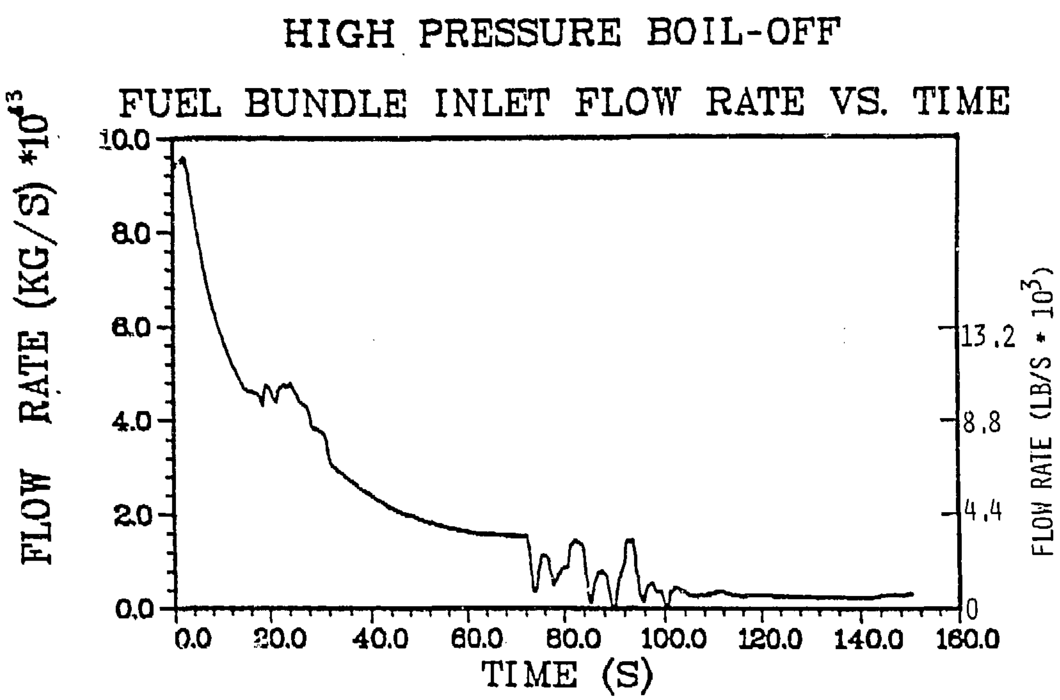


Figure 16. Core inlet flow rate for Transient 3.

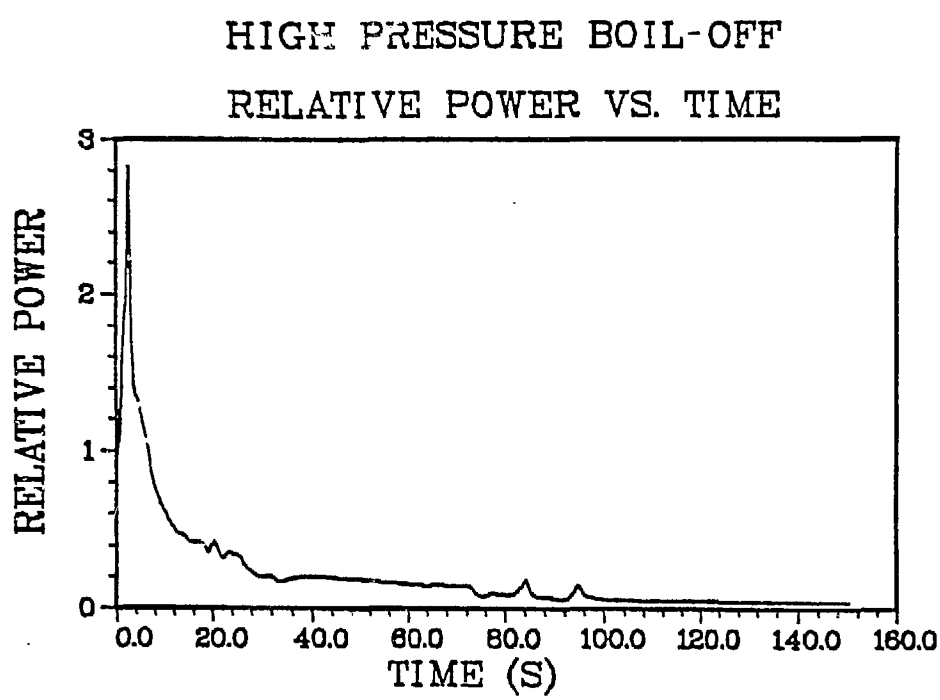


Figure 17. Reactor power history for Transient 3.

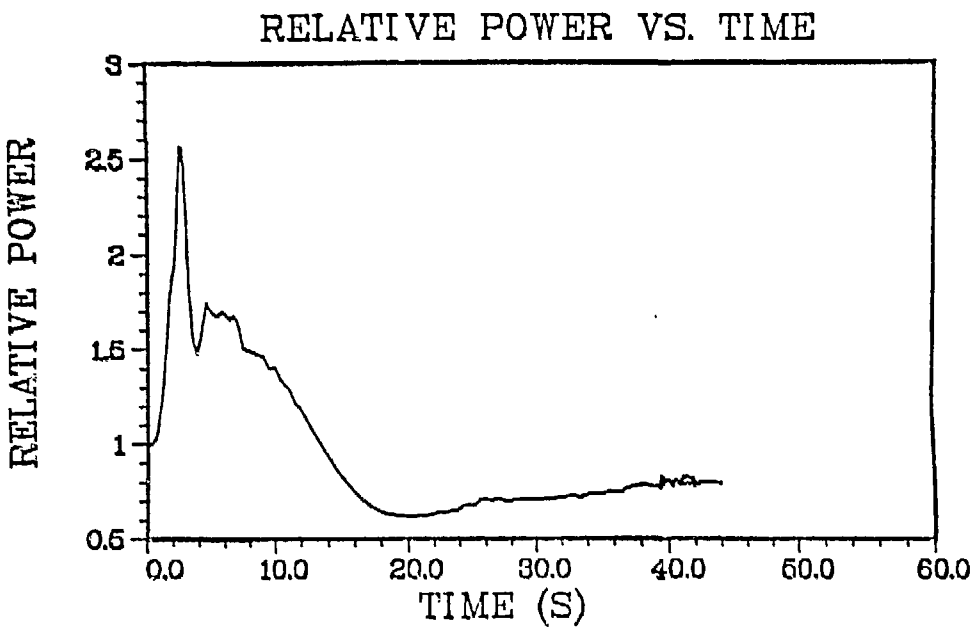


Figure 18. Relative power prediction for Transient 4.

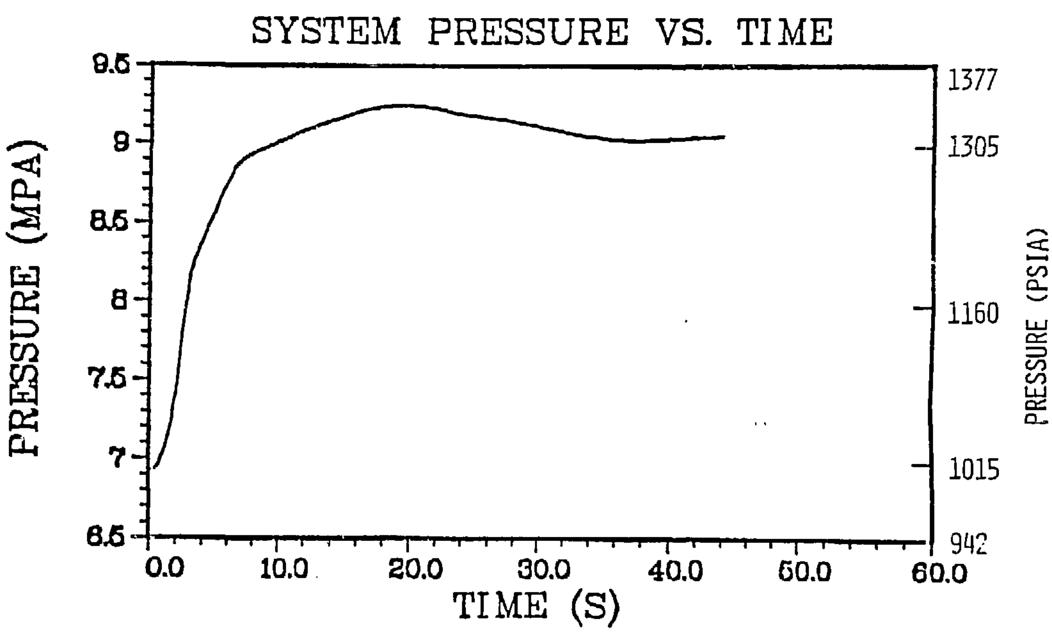


Figure 19. System pressure for Transient 4.

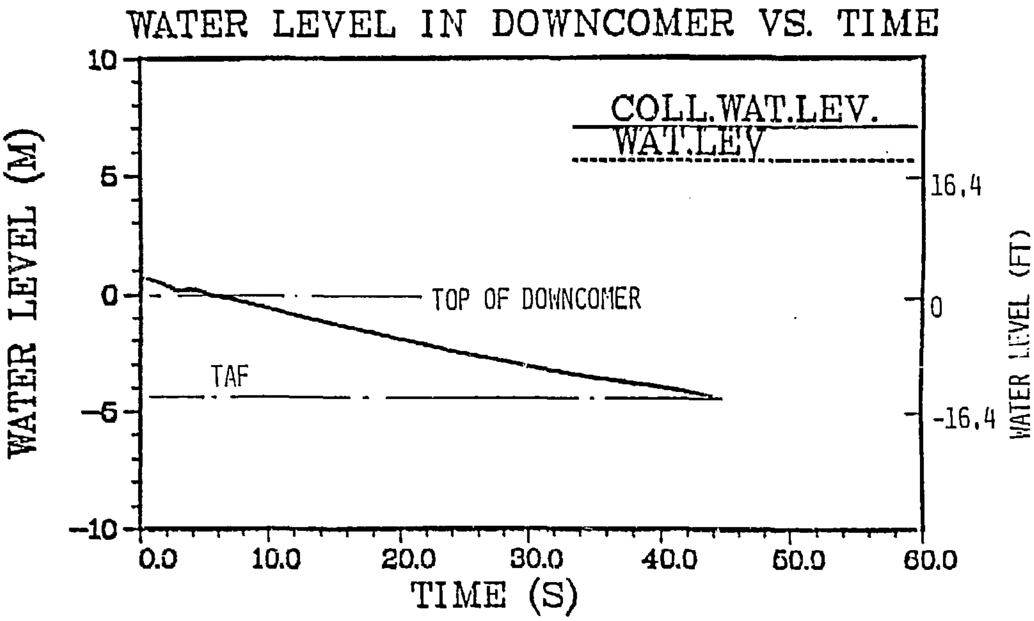


Figure 20. Downcomer water level for Transient 4.