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ASSESSMENT OF TRAC-BD1 AND RAMONA-3B CODES FOR BWR ATWS APPLICATION*

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1. ABSTRACT

Analysis of a typical BWR/4 Anticipated Transient Without Scram (ATWS) has been performed at BNL with both TRAC-BD1[1] and RAMONA-3B[2] codes within the scope of the NRC code assessment program. The objective of the program is to evaluate performance of various codes such as TRAC, RELAP5[12], and RAMONA-3B in predicting plant operational/accident transients or separate effect tests. Work is currently underway on modeling an ATWS-type experiment conducted in the FIST facility[3]. Results obtained in these calculations will be complementary (in thermal-hydraulics area) to those produced in the present typical BWR/4 ATWS calculations.

Of all various ATWS events, the Main Steam Isolation Valve (MSIV) closure ATWS sequence is the most severe one because of its relatively high frequency of occurrence and its challenge to the heat removal and containment integrity systems. Therefore, this transient has been, and is still being, analyzed by different organizations using various computer codes[4,5].

The transient was initiated by an inadvertent closure of all MSIVs with subsequent failure to scram the reactor. However, all other plant safety features, namely, the safety and relief valves, recirculation pumps trip, high pressure coolant injection and the standby liquid (boron) control systems were assumed to function as designed. No operator actions were assumed except for activation of the boron injection system. The calculations have been run until the reactor reached the hot shutdown mode of operation.

It was found that both TRAC-BD1 and RAMONA-3B produced similar results for the global parameters such as reactor power, system pressure, and the suppression pool water bulk temperature. Both calculations showed that the reactor can be brought to hot shutdown in approximately 20 to 25 minutes with the borated water mass flow rate of 2.78 kg/s (43 gpm) with 23800 ppm of boron. The suppression pool temperature (assuming no pool cooling) at this time could be in the range of 77 - 96°C (170 - 205°F).

An additional TRAC-BD1 calculation performed with RAMONA-3B power indicates that the thermal-hydraulic models in RAMONA-3B, although simpler than those in TRAC-BD1, can adequately represent the system behavior during an ATWS-type transient. Moreover, for reactor power calculation, RAMONA-3B with three dimensional space-time neutron kinetics is preferable to TRAC-BD1 with point kinetics since, as it was found in the RAMONA-3B calculations, the

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spatial core power distribution varies strongly during a BWR ATWS. The computer running time for RAMONA-3B was also significantly less than that for TRAC-BD1. Therefore, it is recommended that RAMONA-3B be used for best-estimate BWR ATWS analysis.

Further assessment of both codes is needed to resolve certain differences found in the predictions. Basically, they are related to: a) void distribution calculations in the vessel and, b) condensation on ECC water jets after the feedwater spargers are uncovered.

2. INTRODUCTION

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[6] for the Browns Ferry, Unit 1, nuclear power plant indicates that ATWS is the second most dominant transient for core melt in a BWR/4 with Mark 1 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 events, the Main Steam Isolation Valve (MSIV) closure ATWS sequence is the most severe one because of its relatively high frequency of occurrence and its challenge to the heat removal and containment integrity systems. Therefore, this transient has been, and is still being, analyzed by various organizations using various computer codes[4,5].

The main objective of this paper is to provide a comparative analysis of a BWR/4 MSIV closure ATWS calculation using two advanced, best-estimate codes namely, RAMONA-3B/MODO/Cycle 6[2] and TRAC-BD1/Version 12[1]. Although both are BWR codes, they were conceived from two different stand-points: RAMONA-3B was developed primarily for analyzing operational transients with an emphasis on three-dimensional reactor kinetics and multi-channel core thermal hydraulics, whereas, TRAC-BD1 was developed primarily for analyzing loss-of-coolant accidents with an emphasis on three-dimensional thermal hydraulics with point reactor kinetics.

In a BWR, vapor void fraction varies significantly in space, particularly in the vertical or axial direction. Because of strong void-reactivity feedback, the space-time neutron kinetics, as employed in the RAMONA-3B code is expected to be very important in the ATWS analysis. However, the thermal-hydraulic models of RAMONA-3B must also be adequate, at least for the operational transients such as ATWS. Thus, the second objective of this paper is to verify the thermal-hydraulic models of RAMONA-3B with those of TRAC-BD1 for the same reactor power history.

Before details of the particular calculation are presented, it is worthwhile to point out some specific (common and different) features of the RAMONA-3B and TRAC-BD1 codes. For the sake of brevity, this is accomplished with Table 1.

3. TRANSIENT SCENARIO

The transient was assumed to be initiated by an inadvertent closure of all Main Steam Isolation Valves (MSIVs) in a typical BWR/4 with rated power of 3293 MWt and operating pressure of approximately 7 MPa. As a result, the pressure in the reactor vessel increased rapidly causing void collapse in the core and increase in reactor power. The relief and safety valves were assumed to operate as designed, and the recirculation pumps were tripped at the high pressure set point (8.03 MPa).

The feedwater was assumed to be lost at 35 seconds into the transient. The basis of this assumption is explained later. Since RAMONA-3B did not yet include a feedwater control system, the feedwater flow rate was calculated by using the TRAC-BD1 code with its control system, and was imposed on the RAMONA-3B calculation as a boundary condition. As the downcomer water level dropped, the High Pressure Coolant Injection (HPCI) and the Reactor Core Isolation Cooling (RCIC) systems were activated and cold water at a rate of 337 kg/s and 37.8°C was continued to be introduced through the feedwater sparger. At 120 seconds into the transient, the operator was assumed to start one of the two Standby Liquid Control System (SLCS) pumps. After a delay of 45 seconds, the highly concentrated borated water at a rate of 2.78 kg/s (43 gpm) with 23800 ppm of boron started to enter the reactor vessel to achieve the final hot shutdown condition. No other operator action was assumed during this transient.

It should be noted that the selected transient is just one of the many possible sequences during an MSIV closure ATWS event. The intent here is to examine RAMONA-3B and TRAC-BD1 capabilities for analyzing BWR ATWS-type transients, not to recommend corrective actions during all possible ATWS sequences. Thus, the selected transient is sufficient to serve the main purpose of this study.

4. INPUT MODEL DESCRIPTION

The modeling of a typical BWR/4 reactor system was performed by modifying a plant data set used at Idaho National Engineering Laboratory for the analysis of the Browns Ferry generator load rejection operational transient. The thermal-hydraulic part of the data was carefully checked and compared with the information given in Peach Bottom 2 and 3 FSAR[7] and the EPRI report NP-563[8]. Effort was made to ascertain that the reactor vessel geometric data were correct, and the thermal-hydraulic steady-state conditions for both RAMONA-3B and TRAC-BD1 adequately described a typical BWR/4 nominal full power operating condition, as shown in Table 2.

Since the RAMONA-3B code was developed specifically for the analysis of Boiling Water Reactors, the system under investigation was already "preassembled" in the code using major components typical for BWRs. However, the geometric and operational data had to be specified to represent the particular BWR being modeled.

The RAMONA-3B representation of the BWR/4 system is shown in Figure 1 which includes a Reactor Pressure Vessel with a Steam Dome Region, Downcomer, Lower Plenum, Core, Riser and Steam Separator, Steam Line with the Main Steam Isolation Valves, and four banks of Safety and Relief Valves. The reactor vessel has one combined recirculation loop with jet and recirculation pumps.

In the RAMONA-3B calculation the fuel assemblies in the core were represented by assuming a half-core mirror symmetry with 16 neutronic channels; all control rod groups were assumed to be completely withdrawn from the core region and remained so throughout the calculation. Six parallel hydraulic or heated channels and one bypass channel each with 12 axial nodes were used.

The TRAC-BD1 model of the reactor system utilized 20 components each consisting of a number of cells as shown in Figure 2. Only one CHAN component was used to represent the core region which was divided into 12 axial nodes. Although the input requirements for the TRAC-BD1 and the RAMONA-3B codes were considerably different, care was taken to ensure that all the geometric input data and the various trip signals used in the two codes were consistent with each other. No balance-of-plant modeling was necessary for the MSIV closure ATWS analysis presented in this paper.

Modeling of the core region is more elaborate in RAMONA-3B (with 192 neutronic and 84 hydraulic cells for half-core) because the code performs a three-dimensional time-dependent neutron kinetics calculation. Two group cross sections generated for the Peach Bottom 2 End-of-Cycle 2 conditions have been used in the present RAMONA-3B calculation. To be consistent, the reactivity feedback coefficients [9-10] developed from the BNL-GWIGL calculations of the Peach Bottom 2 turbine trip tests were used in the TRAC-BD1 calculation.

Table 3 compares the RAMONA-3B and TRAC-BD1 input models for the same BWR/4 plant. Information on recirculation pump trip, HPCI, RCIC, and boron injection is also included. The opening and closing set points and the rated flow rates for the S/R valves are given in Table 4.

5. COMPARISON BETWEEN RAMONA-3B AND TRAC-BD1 RESULTS

The RAMONA-3B and TRAC-BD1 predictions for the initial part of the transient following the MSIV closure but before boron injection, will be discussed first. Selected results for the first 150 seconds are depicted in Figures 3 through 9 (comparison of sequence of events for the entire transient is shown in Table 5). The results are discussed below with emphasis on system parameters considered to be the most important from the plant safety viewpoint.

5.1 Short Term (0-150 sec) Results

As expected, immediately after the MSIV closure initiation, the reactor vessel pressure experienced a rapid increase (Figure 3) which in turn, caused void collapse in the core (Figure 4). This introduced a positive reactivity insertion, and a rapid increase of the power (Figure 5) in the first 4 seconds of the transient. Differences in peak power predictions (230% in RAMONA-3B vs. 520% in TRAC-BD1) as well as the reactor power up to approximately 30 seconds can be attributed to the differences in the void fraction predictions, difference in the void reactivity feedback parameter and the three-dimensional neutronics in RAMONA-3B vs. point kinetics in TRAC-BD1. Since TRAC-BD1 predicted a higher reactor power, opening of three banks of relief valves could not arrest the pressure rise as it occurred in the RAMONA-3B calculation; so even the fourth bank, i.e., the safety valves, had to open in the TRAC-BD1 calculation (Figure 6). Note that the higher peak pressure (Figure 3) predicted by TRAC-BD1 was consistent with the higher reactor power (Figure 5) calculated by the code during the first 30 seconds. However, due to the strong pressure-void-reactivity coupling, it is difficult to determine the exact reason for this prediction. An additional TRAC-BD1 calculation has, therefore, been performed with RAMONA-3B reactor power to separate the thermal-hydraulic and neutronic effects. This is discussed later in the paper.

At approximately 30 seconds, the S/RVs actions together with the recirculation pump trip brought the power, pressure, and other system parameters to a quasi-steady-state condition with reasonable agreement between the two calculations (Figures 3, 5, 6 and 7). However, as shown in Figure 4, the core average void fraction including the bypass was an exception. Some possible reasons for this discrepancy are discussed later. No critical heat flux (CHF) condition was experienced in either calculation.

As seen in Figure 5, the reactor power started to increase slightly after 100 seconds. This was due to positive reactivity insertion when the cold HPCI and RCIC water reached the core. There was no boron in the HPCI and RCIC systems. The cold water injection was activated by a low water level signal at slightly different times in these two calculations (Figure 8) in accordance with the water level predictions as shown in Figure 9. In the TRAC BD1 calculation, the collapsed water level dropped at a faster rate because of higher reactor power.

One of the major reasons for the differences observed in the detailed results produced by the RAMONA-3B and TRAC-BD1 codes was due to the differences in the neutronics and power calculation area. As indicated in Tables 1 and 3, TRAC-BD1 calculation was performed with point kinetics assuming the same axial power distribution as the RAMONA-3B steady-state distribution. The power distribution had to be kept invariant in the TRAC-BD1 calculation throughout the transient, whereas RAMONA-3B used a three-dimensional time dependent neutron kinetics. The effect of this difference can be seen in Figure 10 where the RAMONA-3B axial core power distributions at different times are presented. The corresponding axial void fraction profiles are shown in Figure 11. It is seen that a slight variation in axial void profile can indeed produce a large change in the axial power distribution which a point kinetics code like TRAC-BD1 cannot predict.

Another area of concern is the differences in the void predictions as shown in Figure 4. A large difference in the core average void fraction (including bypass) can be seen between the RAMONA-3B and TRAC-BD1 calculations (Figure 4). The difference was present even at the steady-state condition when the reactor power, core flow and temperature conditions were either identical or very close as shown in Table 2. RAMONA-3B uses a slip correlation to calculate the void fraction for a given flow quality. TRAC-BD1, however, solves two phasic momentum equations to calculate the individual phase velocities. Therefore, the correlations which affect the void prediction are completely different in these two codes. Thus some differences in the void fraction prediction should be expected. There were also some differences due to the single channel vs. multi-channel treatment of the reactor core thermal hydraulics in the TRAC-BD1 and RAMONA-3B codes, respectively. However, it should be noted that it is the change in void fraction, rather than the absolute value of void fraction, which is more important in the reactor power calculation. This explains why the total reactor powers as calculated by RAMONA-3B and TRAC-BD1 were in reasonable agreement although the core average void fractions were quite different.

5.2 Long Term (0-1500 sec) Results

Both calculations were continued until a hot reactor shutdown condition (~2% of steady state power) was achieved as a result of boron injection. Highly concentrated borated water at a rate of 2.78 kg/s (43 gpm) with 23800 ppm of boron was injected starting at 165 seconds. As the boron concentration in the core started to increase, the power dropped temporarily resulting in a drop in the void fraction which, in turn, increased the power again. These competing effects of negative boron reactivity and positive reactivity insertion due to void collapse kept the reactor critical for a long time. Meanwhile, the downcomer water level reached the high level shut-off point due due to continuous injection of HPCI and RCIC water. After this water injection was terminated, the boron concentration in the core started to increase at a higher rate, and it eventually overcame the competing void-reactivity effect. The qualitative behaviors of RAMONA-3B and TRAC-BD1 results were quite similar,and as an example, selected results from the TRAC-BD1 calculations are shown in Figures 12 through 16 for core-average boron concentration, downcomer water level, reactor power, and core-average void fraction. TRAC-BD1 predicted the hot shutdown condition at approximately 1100 seconds, whereas RAMONA-3B predicted the same condition at approximately 1400 seconds. This difference is believed to be mainly due to the differences in the boron reactivity feedback coefficients in the two calculations. No attempt was made to adjust these parameters to achieve a better agreement between the RAMONA-3B and TRAC-BD1 calculations.

As seen in Figure 15 (RAMONA-3B), the reactor remained in hot shutdown condition for approximately 40 seconds. This state was interrupted soon after initiation of the HPCI and RCIC cold water injection on the low downcomer water level had occurred: due to positive reactivity insertion the reactor became critical again.

In the course of this study a question about effects of condensation on the ECCS cold water jets was raised: the condensation starts after the feed-water sparger nozzles are uncovered and the water is injected into predominantly steam environment. This issue may become important because the reactor power depends strongly on the water subcooling at the core inlet. Neither of the two codes had a model for the condensation on cold water jets so that developmental work on such a model for RAMONA-3B was initiated.

5.3 Suppression Pool Water Temperature

Since steam released through the S/R valves is dumped into the suppression pool, the pool water temperature starts to increase. To maintain the containment integrity, it is important to keep the suppression pool water temperature at a sufficiently low value. The pool water temperature is, therefore, an important variable from the plant safety viewpoint.

A stand-alone computer program was written to solve the mass and energy conservation equations for the suppression pool water. Steam flow rates and enthalpies calculated by RAMONA-3B and TRAC-BD1 were used as input to this program. The calculated suppression pool temperatures (assuming no pool cooling by the RHR system) are shown in Figure 17. Since RAMONA-3B predicted a longer time for achieving the reactor shutdown condition, the final suppression pool water temperature calculated by RAMONA-3B was ~20°F compared to ~170°F for TRAC-BD1.

5.3.1 Non-Perfect Boron Mixing

It is known that during a low flow or natural circulation cooling mode, all the boron injected into a BWR lower plenum may not be carried into the core. This is because of higher specific gravity of the injected borated water, and the presence of hundreds of control rod guide tubes in the lower plenum. However, in the present calculations no such boron stratification effect was considered. Thus the boron concentration shown in Figure 12 is probably higher-than-actual, which has probably resulted in a shorter-than-actual hot shutdown time.

An attempt has been made to take into account the effect of possible boron stratification based on RAMONA-3B calculation results. Based on the boron mixing efficiency vs. recirculation flow as presented in Reference 11, a value of 0.75 can be assumed for the boron mixing efficiency in the present estimate. Thus, the actual boron concentration in the core would be about 25% lower than the values shown in Figure 12. This would delay the drop in reactor power from ~1300 seconds (as shown in Figure 15) to ~1450 seconds. Even with the assumption of no HPCI and RCIC injection at 1400 seconds (so that the reactor does not regain criticality), the additional reactor power would increase the suppression pool water temperature by another 12°F (6.7°C). So without the RHR cooling, the pool water bulk temperature would be~217°F (102.8°C).

5.3.2 Suppression Pool Cooling

During a BWR ATWS, the operator could be expected to activate the RHR system to reduce the suppression pool heat-up rate. However, the RHR system is designed to remove only about 3% of the rated power. Therefore, even if the pool cooling is activated at the early stage of the transient, the maximum reduction of pool water temperature would be approximately 15°F (8.3°C).

A realistic boron mixing model coupled with maximum pool cooling by the RHR system can, therefore, result in a pool water bulk temperature of~202°F (94.4°C) at the time of reactor hot shutdown. This temperature may still be high from the plant safety viewpoint. Thus, the effects of other mitigative features such as manual rod insertion, use of two SLCS pumps with total capacity of 86 gpm, lowering the downcomer water level to the top of active fuel (TAF), etc., should be investigated. The RAMONA-3B code is already being used for this purpose under the Severe Accident Sequence Analysis (SASA) program.

5.4 TRAC-BD1 Calculation with RAMONA-3B Power

Close coupling between the neutronics and thermal-hydraulics in a BWR makes interpretation of the code predictions a very complex task. An attempt to break up the above coupling was made by performing an additional TRAC-BD1 calculation for the first 150 seconds with the RAMONA-3B core power history as a boundary condition. The RAMONA-3B power was imposed because of the code's detailed treatment of the neutronics part of the calculation. Spatial power variation as a function of time, however, could not be imposed on the TRAC-BD1 code due to code limitation. Therefore, only the total core power was imposed. Results of this calculation answer the questions concerning the differences in the thermal-hydraulic modeling only, and their impact on the code predictions (as it was mentioned earlier, more information on purely thermal-hydraulic performance of both codes at MSIV closure ATWS-type conditions will be available when the BNL code assessment program with FIST results is completed).

A few selected results of the additional TRAC-BD1 calculation are compared with the RAMONA-3B results in Figure 18 through 21. As it is seen now, the system pressures, S/RV flow rates, and the average fuel temperatures (Figures 18 through 20) are in much better agreement. Differences in the void fraction prediction (Figure 21) still remain due to different thermal-hydraulic models used in these codes. Further assessment with experimental data is required to resolve this issue.

6. SUMMARY AND CONCLUSIONS

Based on comparisons between the TRAC-BD1 "power imposed" calculation and the RAMONA-3B results, it can be said that the thermal-hydraulic models of both RAMONA-3B and TRAC-BD1 provide adequate representation of an ATWS event in a BWR. However, for the reactor power calculation, RAMONA-3B with space-time neutron kinetics is a superior and preferable tool to the TRAC-BD1 with point kinetics for ATWS type events where the spatial core power distribution varies with time. Also, the computer running time for RAMONA-3B (with 115 hydraulic cells and 192 neutronic cells has been found to be about four times lower than TRAC-BD1 (with 63 hydraulic cells and point kinetics). Therefore, it is recommended that RAMONA-3B be further used for best-estimate analysis of BWR ATWS-type events.

The following conclusions can be drawn based on the above results:

a) In the event of a BWR/4 MSIV closure ATWS, the reactor can be shutdown with recirculation pumps trip, safety/relief valves actions, and boron injection in approximately twenty-five minutes. With all safety systems (except reactor scram) working as designed, no core uncovery or damage is predicted.

b) Either the RAMONA-3B or TRAC-BD1 code can be used to predict the global variables such as reactor power, pressure, suppression pool temperature during a BWR ATWS event. However, for a best-estimate analysis, the use of the RAMONA-3B code with space-time neutron kinetics is preferable since the spatial core power distribution significantly varies during an ATWS event. Moreover, it is difficult to determine the point kinetics feedback coefficients (as required for a TRAC-BD1 calculation) a priori. Computer running time is also significantly lower for RAMONA-3B than for TRAC-BD1.

c) The thermal-hydraulic models of RAMONA-3B, although simpler than those of TRAC-BD1, are adequate for analyzing abnormal BWR plant transients such as ATWS. Further assessment of both codes is needed to establish the correctness of core void distribution predictions.

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Table 1. GENERAL FEATURES OF RAMONA-3B AND TRAC-BD1 CODES

Item	RAMONA-3B	TRAC-BD1
A. Neutronics	Two-group, three-dimensional neutron kinetics.	Point kinetics.
B. Thermal hydraulics		
1. Balance equations	One-dimensional, four-equation modeling.	Three-dimensional, six equation modeling.
2. Core hydraulics	Set of parallel coolant channels and one bypass channel.	Same as RAMONA-3B.
3. Phasic velocities	Slip model ($v_g = Sv_l + V_o$).	V_g and V_l are calculated separately.
4. Subcooled boiling	Present	Present
5. Fuel conduction	Present	Present
6. Safety systems	Present	Present
7. Plant controls	BWR-specific	General

Table 2. Steady State Conditions of RAMONA-3B

	RAMONA-3B	TRAC-BD1
Thermal Power, MWt	3293	3293
Pressure, Pa	6.99×10^6	6.99×10^6 (steam done)
Mass Flow Rate of Coolant at:		
Core Entrance, kg/sec	12780	12880
Steam Line/Feedwater, kg/sec	1639	1679
Feedwater Temperature, °C	191.2	191.4
Core Inlet Fluid Enthalpy, kcal/kg	288.39	288.38
Core Bypass Flow/Total Core Inlet Flow, %	11.32	11.20
Recirculation Drive Flow Rate, kg/sec	4356	4306

Table 3. Comparison of the RAMONA-3B and TRAC-BD1 Models

	TRAC-BD1	RAMONA-3B
Core Region (Hydraulics)	1 heated + 1 bypass channel, 14 (12 active) axial nodes.	6 heated + 1 bypass channel, 12 axial nodes
Core Region (Neutronics)	Imposed axial power shape; reactivity feedback coefficients were developed from BNL-GWIGL and RELAP-3B calculation of Peach Bottom 2 turbine trip tests[9-10].	16 channels 192 nodes
Downcomer	1D, 3 axial nodes.	1D, 11 axial nodes.
Automatic Control System	Used to control downcomer water level, turbine inlet pressure & recirculation loop flow.	Not used. Feedwater flow rate taken from TRAC-BD1 calculations.
Recirculation Pump Trip Set Point	8.0324 MPa with 0.53 sec. delay.	Same as TRAC-BD1
HPCI + RCIC Initiation Set Point	Time at which the downcomer level reaches Level 2 with 30 sec. delay.	Same as TRAC-BD1
Boron Injection Set Point	t=120 sec with 45 sec. delay.	Same as TRAC-BD1
Boron Injection Rate and Concentration	2.79 ℓ/s, 23800 ppm.	Same as TRAC-BD1
Initial Pressure Suppression Pool Water Volume	3859 m^3.	Same as TRAC-BD1

Table 4. Safety and Relief Valves Set Points

Bank #	1 (4 Relief)	2 (4 Relief)	3 (3 Relief)	4 (2 Safety)
High Pressure (opening), 10^5 Pa	75.50	76.19	76.88	85.84
Delay Time (opening), sec	0.4	0.4	0.4	0.4
Low Pressure (closing), 10^5 Pa	73.23	73.90	74.57	83.22
Delay Time (closing), sec	0.0	0.0	0.0	0.0
Rated Flow/ Valve (kg/sec)	98.86	98.86	98.86	117.60

Table 5. Sequence of Events

EVENT	RAMONA-3B	TRAC-BD1
MSIVs closure starts, sec	0.0	0.0
S/RVs start to open, sec	2.62	3.57
Recirculation pumps trip on high pressure, (8.03 Mpa), sec	3.7	3.95
MSIVs are completely closed, sec	4.0	4.0
Maximum fuel temperature is reached, sec	4.5 (773°C)	5.7 (835°C)
Maximum system pressure reached, sec	8.5 (8.56 MPa)	11.0 (9.30 MPa)
HPCI + RCIC flow starts on low water level signal, sec	86.6	75.8
Boron Injection begins (43 gpm, 23800 ppm), sec	165.0	165.0
HPCI + RCIC turned off on high water level signal, sec	980.0	1275.0
Hot shutdown achieved, sec	1400.0	1100.0
HPCI and RCIC reactivated on low downcomer water level, sec	1400.0	

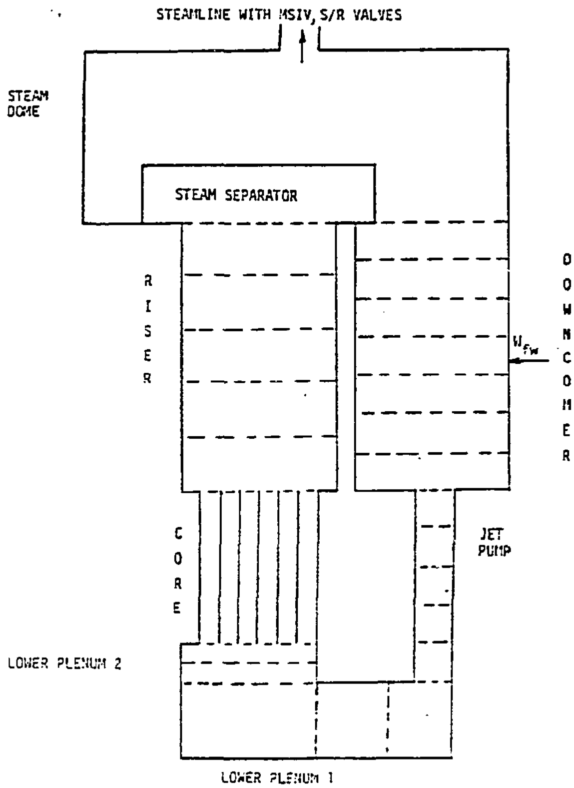

Figure 1 RAMONA-3B Representation of a Boiling Water Reactor Vessel

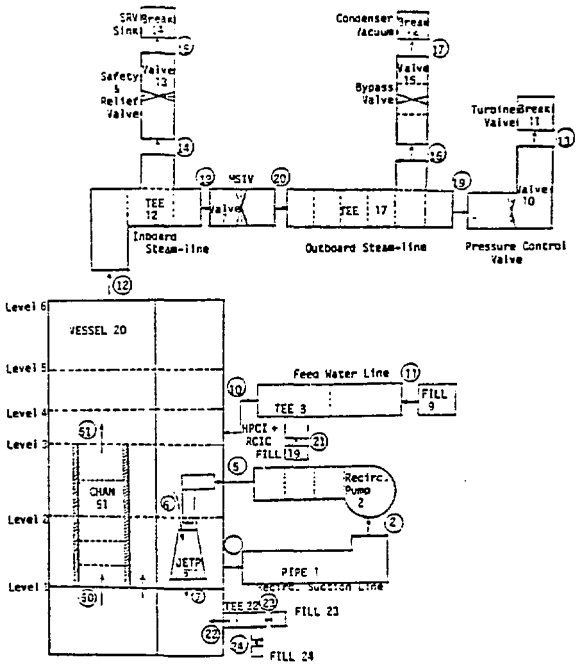

Figure 2 BWR/4 Noding Diagram for TRAC-BD1

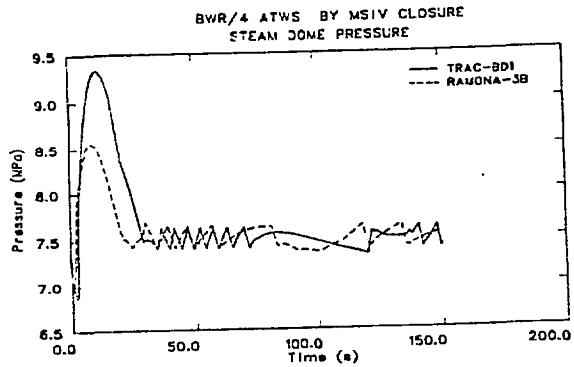

Figure 3. Comparison of Steam Dome Pressures as Calculated by RAMONA-3B and TRAC-BD1

Figure 3 Comparison of Steam Dome Pressures as Calculated by RAMONA-3B and TRAC-BD1

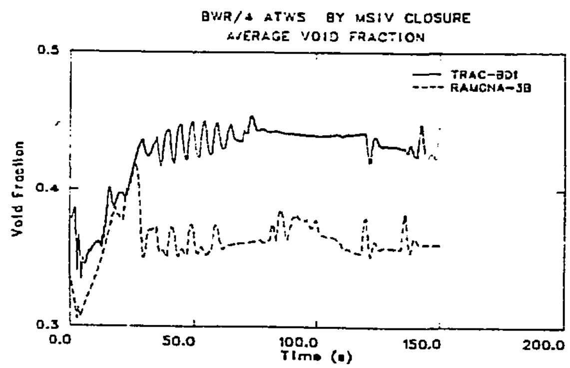

Figure 4. Comparison of Core Average Void Fractions (Including Bypass) as Calculated by RAMONA-3B and TRAC-BD1

Figure 4 Comparison of Core Average Void Fractions (Including Bypass) as Calculated by RAMONA-3B and TRAC-BD1

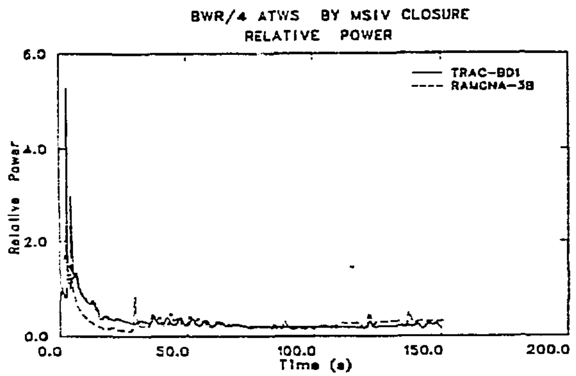

Figure 5 Comparison of Relative Core Powers as Calculated by RAMONA-3B and TRAC-BD1

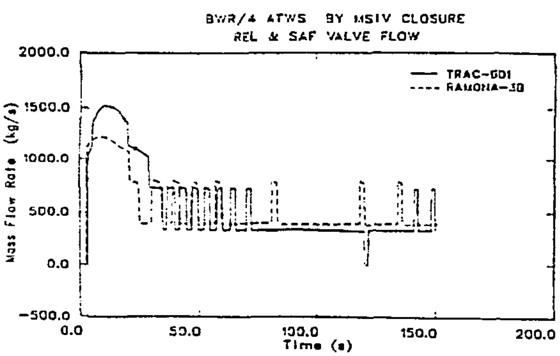

Figure 6 Comparison of Steam Discharge Rates Through Relief and Safety Valves as Calculated by RAMONA-3B and TRAC-BD1

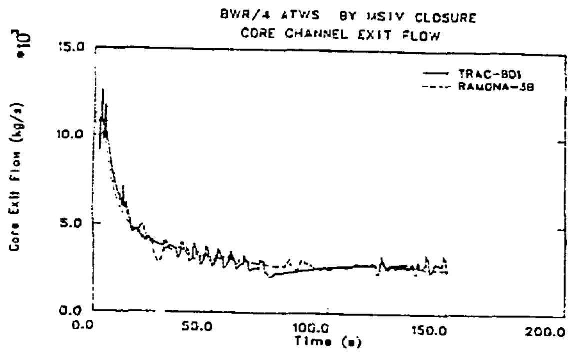

Figure 7 Comparison of Core Exit Flow Rates as Calculated by RAMONA-3B and TRAC-BD1

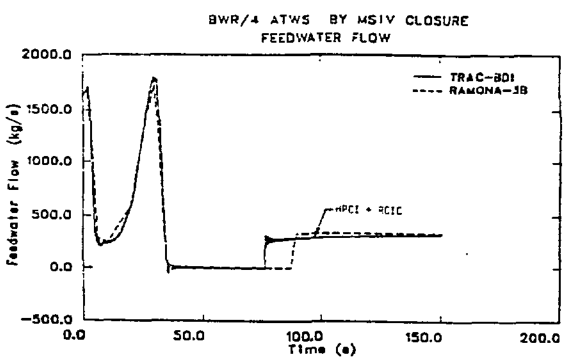

Figure 8 Comparison of Feedwater Sparger Flow Rates as Calculated by RAMONA-3B and TRAC-BD1

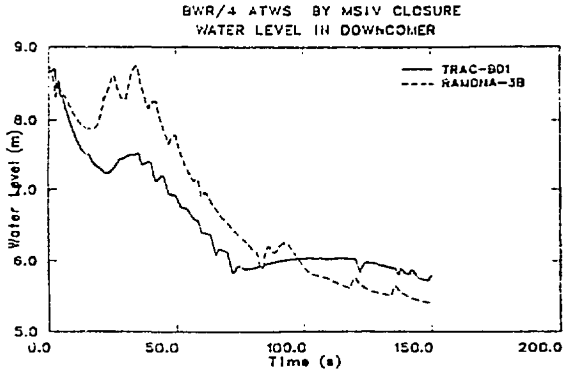

Figure 9 Comparison of Collapsed Downcomer Water Levels as Calculated by RAMONA-3B and TRAC-BD1

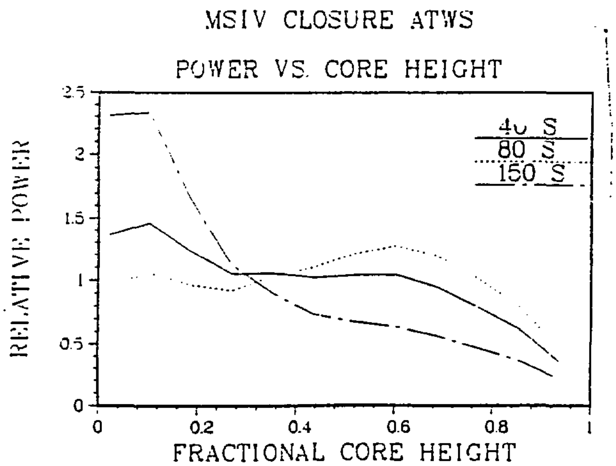

Figure 10 Axial Core Power Distributions as Calculated by RAMONA-3B

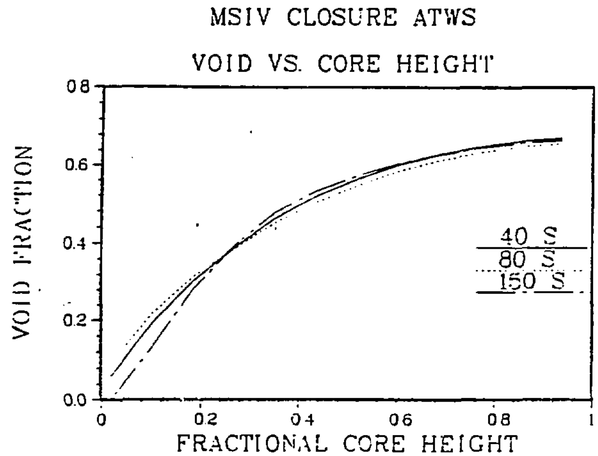

Figure 11 Axial Core Void Fractions as Calculated by RAMONA-3B

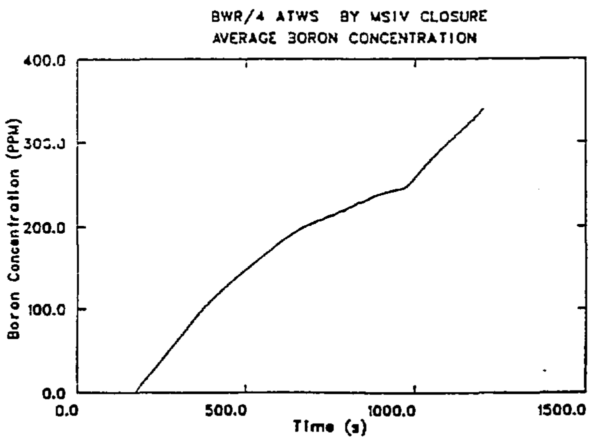

Figure 12 Core Average Boron Concentration (Including Bypass) as Calculated by TRAC-BD1

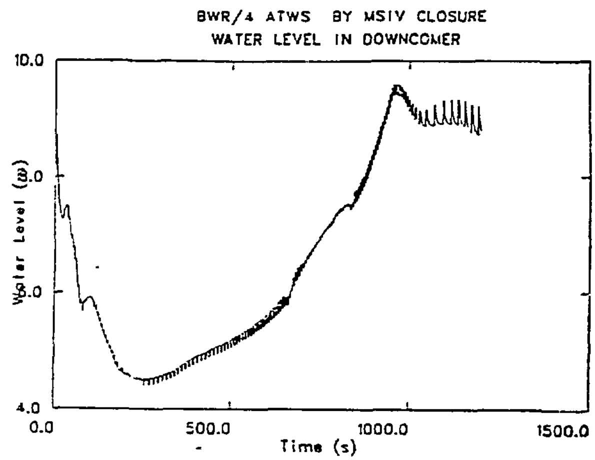

Figure 13 Collapsed Water Level in Downcomer as Calculated by TRAC-BD1

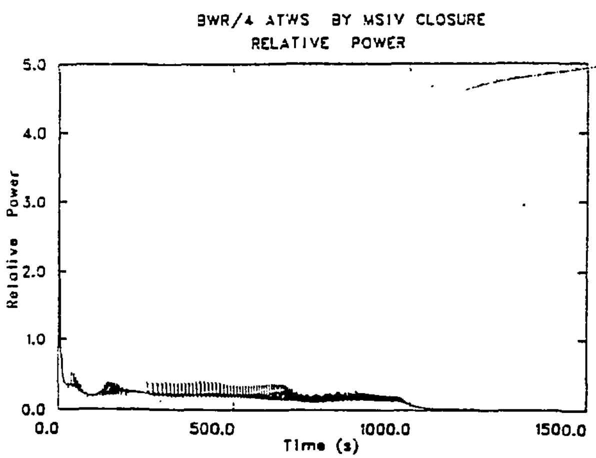

Figure 14 Long Term Reactor Power (Relative to Steady State) as Calculated by TRAC-BD1

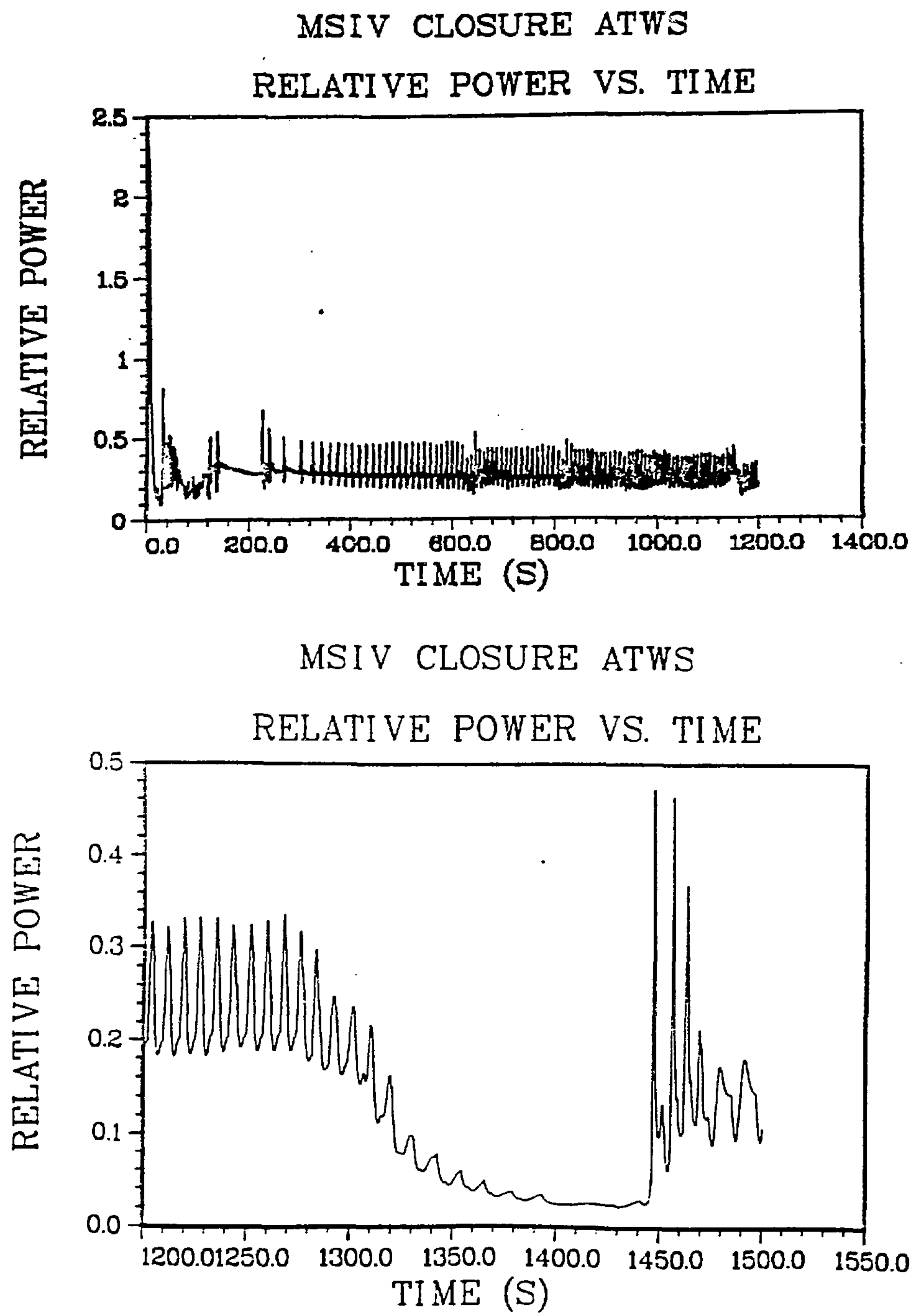

Figure 15 Long Term Reactor Power (Relative to Steady State) as Calculated by RAMONA-3B

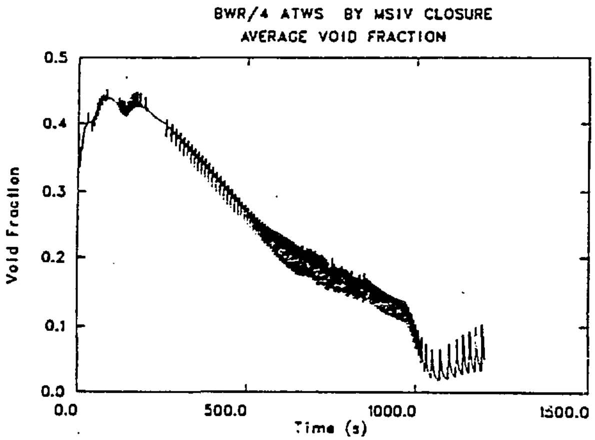

Figure 16 Core Average Void Fraction (Including Bypass) as Calculated by TRAC-BD1

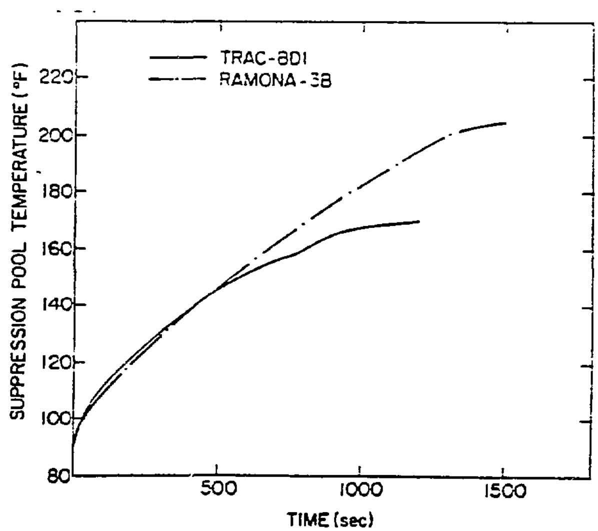

Figure 17 Comparison of Suppression Pool Water Temperature as Calculated by RAMONA-3B and TRAC-BD1

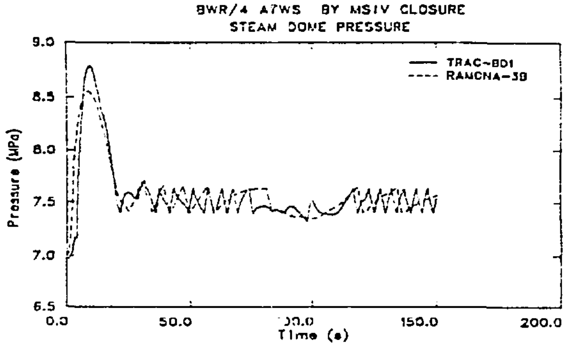

Figure 18 Comparison of Steam Dome Pressure as Calculated by RAMONA-3B and TRAC-BD1 with RAMONA-3B Reactor Power

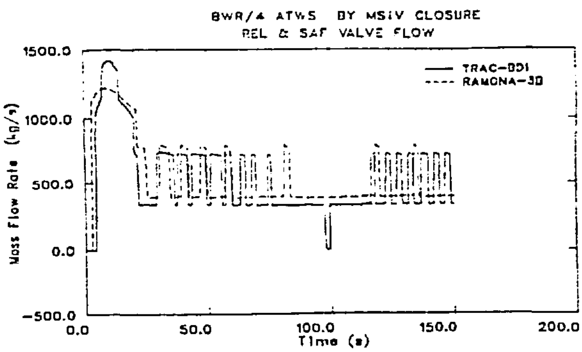

Figure 19 Comparison of Steam Discharge Rates Through Relief and Safety Valves as Calculated by RAMONA-3B and TRAC-BD1 with RAMONA-3B Reactor Power

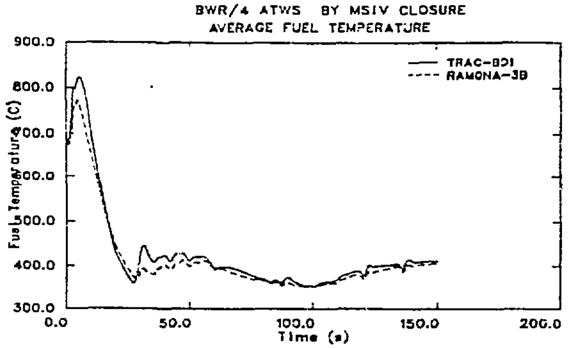

Figure 20 Comparison of Average Fuel Temperatures as Calculated by RAMONA-3B and TRAC-BD1 with RAMONA-3B Reactor Power

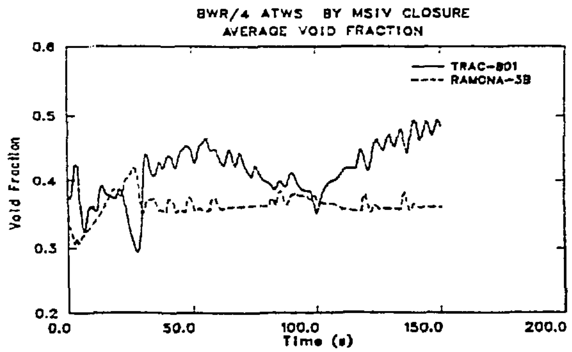

Figure 21 Comparison of Core Average Void Fractions (Including Bypass) as Calculated by RAMONA-3B and TRAC-BD1 with RAMONA-3B Reactor Power