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Overview of TRAC-BD1/MOD1 Assessment Studies

Briant L. Charboneau

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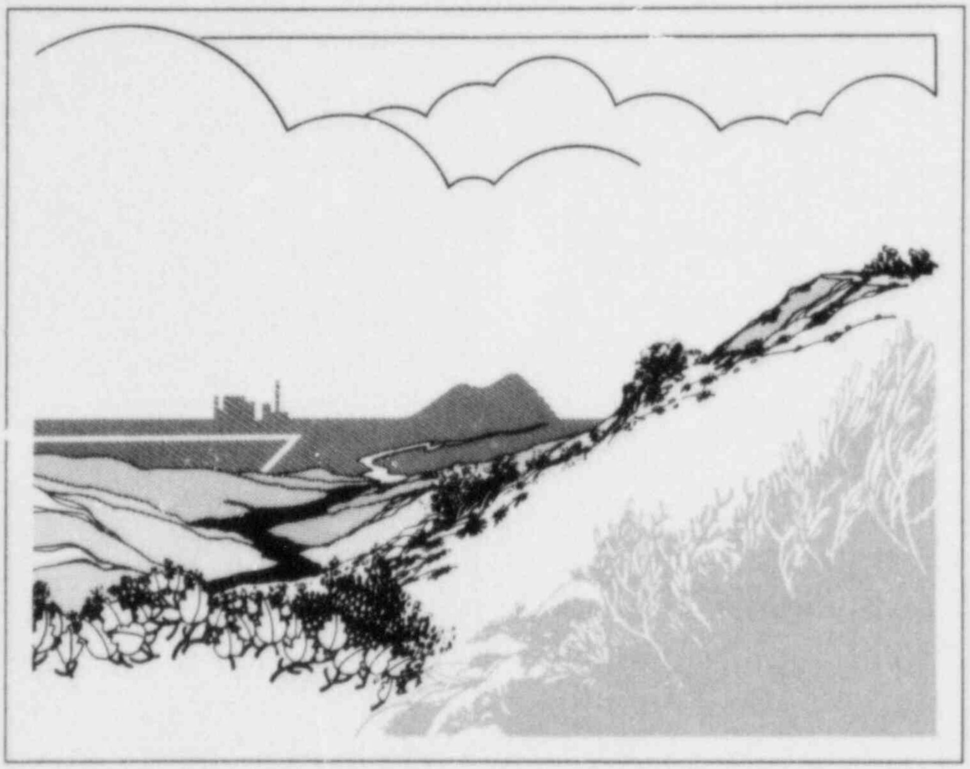

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OVERVIEW OF TRAC-BD1/MOD1 ASSESSMENT STUDIES

Briant L. Charboneau

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ABSTRACT

This report summarizes a series of computer simulations sponsored by the United States Nuclear Regulatory Commission (USNRC) performed at the Idaho National Engineering Laboratory (INEL) to continue the advancement of boiling water reactor (BWR) safety research. The simulations were performed to evaluate the analysis capabilities of the Transient Reactor Analysis Code BWR version (TRAC-BD1/MOD1) to calculate operational transients, including anticipated transients without scram (ATWS) and loss-of-coolant accidents (LOCAs). The assessment simulations were performed for a broad range of scenarios, to encompass as many different phenomena as possible. Comparisons are made between the measured and calculated data. Conclusions are made with respect to the calculated system pressure response, thermal response, and break flow response, as well as the capabilities to model containment and natural circulation conditions. Recommendations are made with respect to user guidelines.

FIN No. A6047—Code Assessment and Applications—Transients

SUMMARY

A series of TRAC-BD1/MOD1 computer code simulations sponsored by the United States Nuclear Regulatory Commission (USNRC) were performed at the Idaho National Engineering Laboratory (INEL) to continue the advancement of boiling water reactor (BWR) safety research. The simulations were assessed with data from separate effects test facilities, integral test facilities, another computer code, and a full-scale power plant. The separate effects simulations consist of two blowdown/containment tests and a large break reflood test. The integral simulations include three LOCAs of varying sizes, a power transient, three natural circulations tests, and four power plant start-up tests. Results, conclusion, and recommendations from this series of computer calculations are summarized. Comparisions are made between the measured and calculated data. Conclusions are made with respect to the calculated system pressure response, thermal response and break flow response, as well as the capabilities to model containment and natural circulation conditions. Recommendations are made with respect to user guidelines.

The development of the TRAC-BWR code is continuing with additional improvements. The latest version of the code (TRAC-BF1) is currently undergoing developmental assessment prior to independent assessment and its release.

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OVERVIEW OF TRAC-BD1/MOD1 ASSESSMENT STUDIES

1. INTRODUCTION

TRAC-BD1/MOD1 computer code is a best estimate code for the analysis of postulated accidents and transients in boiling water reactor (BWR) systems and related experimental facilities. A series of assessment studies were conducted on TRAC-BD1/MOD1 at the Idaho National Engineering Laboratory (INEL) for the United States Nuclear Regulatory Commission (USNRC). This assessment series used test data from many different facilities, including the Swedish FIX-II Facility, the Full Integral Simulation Test (FIST) Facility, the General Electric Steam Sector Test Facility (SSTF), the Swedish Marviken Facility, Browns Ferry Nuclear Plant Unit 3, and the Japanese Rig of Safety Assessment (ROSA-III) Facility. TRAC-BD1/MOD1 calculations were also compared with some CONTEMPT/LT-28 calculations of the MARK I containment and the Marviken facility. This report summarizes the results and conclusions from this series of assessments.

1.1 TRAC-BWR Evolution

Development of the BWR version of the Transient Reactor Analysis Code (TRAC) began at the INEL for the USNRC in October 1979. The initial goal of the TRAC-BWR development program was to provide a basic analysis capability for design basis loss-of-coolant accidents (DBLOCAs). Two versions of the TRAC-BWR code have been released and another version is currently being developed.

TRAC-BWR was created from an interim version of TRAC-PF1. TRAC-PF1 was developed by the Reactor Safety Code Development Group at Los Alamos National Laboratory for analysis of LOCAs in pressurized water reactors (PWRs). TRAC-PF1 contains a full two-fluid thermal-hydraulic model in both one- and three-dimensional component models. Several major modifications were required of TRAC-PF1 before BWR systems could be accurately modeled. Most of the modifications relate to the differences in the core geometry. The fuel in a PWR is contained in an open lattice, whereas the fuel in a BWR is contained within individual channel boxes. The difference in core structure required that many additional thermal-hydraulic phenomena be modeled. The most important of these were rod-to-rod and rod-to-channel wall radiation heat transfer, separate fluid streams caused by the channel walls (channel and bypass regions), and countercurrent flow limiting (CCFL) at the upper tie plate (UTP) and the side entry orifice (SEO). Other major areas requiring changes or additions were the critical heat flux correlation, jet pump modeling, and separator-dryer modeling.

TRAC-BWR is of modular construction to allow its adaptation to any BWR plant or BWR experimental system. The code consists of modular components that are based on the physical components of a system, such as a pipe or pump component. The code also contains some more complex components designed to make the construction of a model easier, such as a vessel component and a jet pump component. Most of the components can be subdivided into a number of cells, to allow for a more detailed thermal-hydraulic analysis.

The first released version of TRAC-BWR (TRAC-BD1/MOD0[1] Version 8), was sent to the National Energy Software Center[a] in February 1981. TRAC-BD1/MOD0 provided the capability for analyzing a LOCA accident scenario, including blowdown, core heat-up, reflood with quenching, and refill.

Two interim versions of TRAC-BD1/MOD0, Versions 11 and 12, were widely used by the USNRC and its contractors before a second version of the code, TRAC-BD1/MOD1,[2] was released in April 1984. Many modeling improvements were made in these two interim versions; improvements to interfacial drag, critical flow, wall heat transfer, and subcooled boiling. The purpose of the second version is to provide the capability for analyzing operational transients, including anticipated transients without scram (ATWS), as well as providing

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a. National Energy Software Center, Bldg. 208—Room C-230, 9700 South Cass Avenue, Argonne, Illinois 60439.

an improved analysis capability for both design basis and small break LOCAs. Development of the TRAC-BWR code is continuing, with expanded modeling capabilities and numerical efficiency improvements. The third version of the code, TRAC-BF1, is scheduled for release in late FY-1986.

1.2 TRAC-BD1/MOD1 Description

TRAC-BD1/MOD1 was developed to analyze operational transients including ATWS as well as provide an improved analysis capability for both large and small break LOCAs. Many new models were added, and numerous existing models were significantly revised to provide these expanded capabilities. The major models added or significantly revised to create the TRAC-BD1/MOD1 version include:

- Balance of plant models, such as turbine, feedwater heaters, and condenser
- Soluble boron transport model
- Two-phase, level tracking model
- Improved constitutive relations for heat, mass, and momentum transfer between the fluid phases and between the fluid phases and structure
- Control systems model
- Reactivity feedback model, for use in the point reactor kinetics model
- A simple, lumped parameter containment model
- Noncondensable gas transport model
- Generalized component heat and mass transfer models
- User convenience features, such as free formatted input and extensive input error checking.

1.3 Assessment Objectives

The assessment analyses summarized in this report were performed to evaluate the capabilities of TRAC-BD1/MOD1 to simulate BWR transients. The analyses were chosen to include a wide variety of scenarios for which the code could be evaluated. These included a small break LOCA, an intermediate break LOCA, a large break refill/reflood, a power transient, natural circulation experiments, containment transients, and several operational transients. The assessment calculations were evaluated on (a) how realistically governing phenomena were represented, (b) how well key parameters were determined, and (c) how well overall system behavior was simulated.

1.4 Report Outline

This report summarizes and discusses the assessment results and conclusions from the different assessment calculations already conducted on TRAC-BD1/MOD1. The goal is to create a well developed perspective of the code's overall performance, thermal-hydraulic accuracy and validity, capabilities, and areas in need of improvement. Complete descriptions of the TRAC-BD1/MOD1 assessment calculations are contained in Appendixes A through G. An attempt has been made to repeat only enough information about each calculation to make the discussion of the assessment results and conclusions clear. No attempt has been made to include information pertaining to the construction of the individual models (nodalization, and the establishment of initial and boundary conditions) since this information is discussed in detail within each assessment calculation report duplicated in the appendices. An uncertainty analysis using the ROSA-III assessment calculation was performed. This study is presented in Appendix H.

2. FACILITY AND TEST DESCRIPTIONS

This section describes the facilities and tests used in the TRAC-BD1/MOD1 assessment studies. Table 1 lists and briefly describes the tests. Appendixes A through G describe in detail the tests and information about the facility models. The references to the report contains complete information about each of the facilities.

2.1 Separate Effects Facilities and Test Descriptions

The three assessment calculations pertaining to the analysis of separate effects are (a) BLOWDOWN 18, conducted at the Marviken facility, (b) SEQUENCE 483, for a MARK I containment as calculated by CONTEMPT/LT-28,[3] and (c) Test EA3.1, conducted at the Steam Sector Test Facility (SSTF). These tests and facilities are described in the following paragraphs.

2.1.1 Marviken. The Marviken facility, converted from a heavy water boiling reactor (never fueled) to an experimental blowdown facility,[4] is located in Sweden. The facility consists of a full-scale pressure vessel within a pressure suppression type containment. The containment is a multiple room type, and, therefore, not entirely prototypical of U.S. commercial BWR containments. The drywell consists of many rooms interconnected by doorways, vent shafts, and other openings. The wetwell is a large room located under the rooms that compose the drywell.

BLOWDOWN 18 was conducted in the Marviken facility to simulate a large pipe rupture in the primary loop of a BWR. The Marviken vessel was initially filled with water and heated to ~520 K at 4600 kPa. The liquid in the vessel was then discharged as a two-phase mixture into one of the drywell containment rooms. Twenty-eight of the 57 vents connecting the drywell to the wetwell were open during BLOWDOWN 18. Neither drywell nor wetwell cooling was used during the experiment.

2.1.2 MARK I Containment. A MARK I containment is composed of two main components: drywell and wetwell.[5] The pressure vessel is completely contained within the drywell. The wetwell is a toroidal-shaped vessel that surrounds the lower elevation of the drywell. The wetwell is normally about half full of water and is connected to the drywell by eight large vent pipes.

SEQUENCE 483, simulated to occur in the MARK I containment, was initiated by closure of the main steam isolation valve (MSIV) with subsequent failure of the reactor to scram. The boron injection system was not activated. The power oscillated as the system pressure varied between the safety relief valve (SRV) set points. High-pressure core injection (HPCI) was activated, and at 250 s HPCI suction shifted to the torus, upon activation of the torus high water level signal. HPCI is assumed to have failed because of high torus temperature (366.5 K), and the torus continued to heat until steam breakthrough was assumed. The automatic depressurization system (ADS) and low-pressure core injection (LPCI) were activated with high drywell pressure and the vessel water level at the triple low point.

2.1.3 SSTF. The Steam Sector Test Facility (SSTF) was a BWR simulator, volume scaled 30/360 degrees, or 1/12, of a full-scale BWR/6.[6] The principal features of the facility used for the simulated test were a full-scale 30 degree sector mockup of a BWR/6-218, low pressure core spray (LPCS), LPCI, recirculation lines capable of simulating a DBLOCA, 58-rod bundles, lower plenum, and guide tubes/bypass. The unpowered rod bundles used steam injection to simulate core heat. The facility was designed to operate at low pressures (~1 MPa) and to simulate multiple channel behavior.

Test EA3.1, RUN 111, was a reference BWR/4 refill/reflood transient. The test was initiated by the following sequence: (a) closure of the vessel steam vent valve, (b) opening of the blowdown valve, (c) actuation of emergency core cooling system (ECCS) injection, and (d) control of the steam injection to simulate vapor generation from the core and vessel structures. The test was characterized by rapid reflood of the core channels and refill of the lower plenum.

2.2 Integral Test Facilities and Test Descriptions

Integral test facilities were used to simulate various sizes of LOCAs, a power transient, natural circulation experiments, and several different operational transients. Brief descriptions of the facilities and tests are described in the following subsections, arranged according to facility.

Table 1. Test descriptions

Facility	Designation	Description
Marviken (separate effects blowdown facility)	BLOWDOWN 18	Large break blowdown containment test, assessed with test data and CONTEMPT/LT-28
MARK I (CONTEMPT/LT-28 calculation for a BWR MARK I containment)	SEQUENCE 483	Containment calculation, with MSIV closure, scram failure, and steam breakthrough, with HPCI and LPCI
FIX-II (scaled Swedish BWR integral facility)	3025	Intermediate size split recirculation line break, with no ECCS
FIST (scaled BWR/6-218 integral facility)	6PMC2	Power transient, MSIV closure, failure to scram, with only RCIC
	6PNC1-2b	Natural circulation test at 1-MW core power
	6PNC1-4	Natural circulation test at 2-MW core power
	6PNC1-6	Natural circulation test at 3-MW core power
SSTF (scaled BWR/6, 30 degree sector integral)	EA3.1 RUN 111	Large break DBA reflood/ refill test, BWR/4 ECCS configuration, with LPCS and LPCI
Browns Ferry Nuclear Plant Unit 3 (1100 MW BWR/4)	PT	One recirculation pump trip test
	TPT	Two recirculation pumps trip test
	FTT	Feedwater turbine trip transient
	GLR	Generator load rejection test
ROSA-III (scaled BWR facility)	RUN 912	Small recirculation line break, with HPCI failure

2.2.1 FIX-II. The FIX-II facility is a volumetrically scaled (1/777) BWR simulator, designed to investigate simulated fuel rod heat transfer during LOCAs and other transients. FIX-II is scaled from the Swedish Oskarshamn-2 Reactor, which is a 570-MW BWR with external pumps.[7,8] The simulator is powered with an electrically heated, 6 x 6-rod bundle. The pressure vessel contains a lower plenum, core, upper plenum, simulated steam separators, and steam dome. External to the pressure vessel are the downcomer, guide tubes, bypass, and the two recirculation loops. The broken loop simulated one actual loop, and the intact loop simulated three actual loops.

The experiment simulated was Test 3025, International Standard Problem 15. The break flow changed from subcooled liquid to a high quality mixture after approximately 20 s. The guide tubes and lower plenum flashed at about 50 s, as the final core dryout started. The test was terminated at 76 s on low steam dome pressure.

2.2.2 FIST. The Full Integral Simulation Test (FIST) facility is a full-height BWR simulator, scaled 1/624 of a BWR/6-218.[9] It contains a single full-size electrically heated 8 x 8 bundle. The flow areas are scaled to yield velocities similar to the full-scale reactor. The system contains two complete recirculation loops, a heated feedwater system, a reactor core isolation cooling (RCIC) system, and safety relief valves. The only significant scaling compromises are the vessel metal mass and heat transfer areas. The facility size and operating conditions required the facility to contain several times the scaled amount of metal mass in some areas.

Four tests conducted at the FIST facility were used to assess TRAC-BD1/MOD1. These tests consisted of a power transient (6PMC2), and three natural circulation tests (6PNC1-2b, 6PNC1-4, and 6PNC1-6). The power transient was initiated with the closure of the MSIV. This was followed by the failure of the reactor to scram, and a high-pressure core spray (HPCS) system failure. The system pressure remained below the ADS high-pressure activation set point. RCIC was the only inventory make-up system available. The programmed bundle power was based on a BWR/6 plant calculation with a one-dimensional neutronics code and boron injection at 120 s. The power calculation, however, assumed that the HPCS system was active, resulting in a higher power condition due to the increased mass in the bundle. The transient is characterized by high reactor power and a low coolant inventory make-up rate caused by the mismatch in the programmed power and the assumed HPCS failure.

The three natural circulation tests were designed to simulate natural circulation in a BWR-6/218 under normal and off-normal conditions. The three test were similar, except they were initiated at different core powers: 1, 2, and 3 MW. Each test was started from steady state conditions and natural circulation flow rates corresponding to the core power. Independently controlled parameters such as power, pressure, feedwater flow, and feedwater temperature were held steady during the tests. The tests were initiated with a step decrease in the feedwater flow to cause a steam-feedwater flow mismatch. This resulted in a slowly falling liquid level and decreasing natural circulation core flow.

2.2.3 Browns Ferry Nuclear Plant. The Browns Ferry Nuclear Plant Unit 3 is an 1100-MW General Electric BWR/4, owned and operated by the Tennessee Valley Authority.[5,10] The plant has been operating since March 1977. Four of the startup tests conducted at this facility were used in this assessment study. The tests include a single recirculation pump trip, a two recirculation pump trip, a feedwater turbine trip, and a generator load rejection transient.

The three operational transients all started from steady state and were initiated by preset boundary conditions. The boundary conditions primarily controlling the transients were the pump speeds and feedwater flow rates. A zero electrical load initiated the generator load rejection transient, resulting in a power/load unbalance. The unbalance caused a trip signal to be generated; however, the signal interlock to the turbine stop valves was disconnected. As a result, the turbine began to overspeed; the turbine control valve began to close; and the bypass valve began to open in response to the overspeed condition. A turbine overspeed trip eventually tripped the turbine stop valves and scrammed the reactor. At about 6 s, the downcomer water level reached the low-level set point; the MSIVs began to close; and the recirculation pumps were tripped.

2.2.4 ROSA-III. The Rig of Safety Assessment-III (ROSA-III) is a volumetrically scaled (1/424) BWR system with an electrically heated core.[11,12] Four half-length, electrically heated bundles power the core. There are two recirculation loops, each loop containing two jet pumps and one recirculation pump. ROSA-III containing all the major BWR components and has them arranged accordingly, except for the jet pumps. The jet pumps had to be placed external to the pressure vessel, owing to scaling constraints with the downcomer flow area. The facility also contains prototypical BWR ECCS which include a HPCS, a LPCS, and a LPCI.

Run 912, a small break test, was simulated at ROSA-III using the TRAC-BD1/MOD1 code. The test was initiated with a 5% break in the recirculation pump suction piping. After MSIV closure, the SRVs were used to control the system pressure until the ADS was activated. Then LPCS and LPCI were initiated to quench the rods.

3. TRANSIENT CALCULATION RESULTS

The accurate simulation of global BWR responses requires the accurate calculation of the local phenomena occurring within the system. This section identifies and discusses the key phenomena controlling the scenario of events being calculated. Detailed discussions of the individual calculations are presented in the assessment topical reports appended to this report. The focus of this report is on the similarities and differences of the assessment findings, so that a well-developed perspective of the code's capabilities may be gained. The following subsections correspond to the parameters that have been judged key indicators of the code's predictive capabilities, which are system pressure, core thermal behavior, break flow, containment temperature, and natural circulation flow.

3.1 Simulated System Pressure Response

The correct system pressure is critical for simulation of most phenomena occurring in a BWR transient. The system pressure depends highly on many of the phenomena present in the experiment. Therefore, correct simulation of the system pressure indicates that most of the complex interdependent phenomena are also correctly simulated. The primary factors controlling and being controlled by the system pressure are as follows:

- Break flow
- Safety system actuations
  - ECCS
  - ADS
  - SRV
- Flashing/condensation
- Heat transfer
  - Ambient heat loss
  - Stored energy
  - Bundle to coolant heat transfer.

LOCAs have many event timings that are directly linked or influenced by the system pressure. Table 2 presents a comparison between some calculated and measured parameters. The power transient, Test 6PMC2 (FIST), is included in the table even though it is not considered a typical LOCA. The diversity of the simulated transients prohibits a meaningful composite uncertainty analysis; nevertheless, Table 2 indicates that the pressure responses were generally well predicted for the simulated LOCAs. Table 3 presents a summary of key events for operational transients. ECCS timings are not included in Table 3 because the two transients they were simulated in used time trips for their actuations. Table 4 summarizes the simulated pressure responses for all the studies. Several areas of discrepancy exist between data and calculations, including modeling uncertainties, break flow, condensation, pool boiling, and ECCS flow. Modeling uncertainty simply refers to uncertainties in the measured data and in the facility configuration. The interactions between system pressure and the above parameters are discussed in the following paragraphs.

Break flow was an important parameter controlling system pressure in all the transients that experienced a break. In Test 3025 (FIX-II), the break flow was the only significant parameter identified as adversely affecting the blind calculation. The break mass flow was underpredicted by as much as 25%. Figure 1 illustrates that the simulated pressure response was delayed relative to the data because of a lower calculated break flow rate.

It is suspected that the system pressurization in RUN 912 (ROSA-III) was also sightly overpredicted because of the underpredicted break flow. As indicated in Figure 2, the SRV mass flow was actuated earlier because of the faster pressurization. The ADS initiation time was not directly affected by the higher pressure, since it was tripped by a downcomer liquid level signal. The discrepancies associated with the LPCS and LPCI initiation times are directly attributable to the system pressure calculation, since they were initiated with a pressure trip.

Excessive steam condensation is believed to have caused an accelerated depressurization during the EA3.1 Test (SSTF) simulation. During the first 6 s of the subcooled LPCI liquid injection into the jet

Table 2. Comparison of calculated and measured key event timings for LOCAs

Event	3025 (FIX-II) Measurement	3025 (FIX-II) Calculation (Blind)	3025 (FIX-II) Calculation (Sensitivity)	RUN 912 (ROSA-III) Measurement	RUN 912 (ROSA-III) Calculation	6PMC2 (FIST) Measurement	6PMC2 (FIST) Calculation
Maximum steam dome pressure	7.5 (7.35 MPa)	8.4 (7.4 MPa)	9.5 (7.28 MPa)	84.0 (8.40 MPa)	62.0 (8.40 MPa)	6.0 (8.43 MPa)	7.0 (8.52 MPa)
First SRV opening	1.0	1.0	1.0	84.0	62.0	3.4	3.4
Second SRV opening	12.1	12.1	12.1	NA	NA	107.0	108.0
LPCS initiation	NA	NA	NA	318.0	335.0	NA	NA
LPCI initiation	NA	NA	NA	406.0	392.0	NA	NA
RCIC initiation	NA	NA	NA	NA	NA	85.0	97.3
Lower plenum flashing	22.0	23.4	19.7	159.0	166.0	—[a]	600.0
Whole core uncovery	64.0	81.0	56.0	275.0	300.0	NA	NA
Peak cladding temperature	1.5 (614 K)	—[b] —[b]	1.5 (660 K)	410.0 (839 K)	394.0 (748 K)	355.0 (895)[c]	325.0 (895)[c]
Bundle quench	NA	NA	NA	444.0	436.0	990.0	660.0

a. Indeterminable from data.

b. No peak occurred during this part of the simulation.

c. Administrative temperature limit.

pumps, the simulation overpredicted the depressurization of the system. The pressure dropped in each cell as the liquid entered, causing additional steam to be drawn into the jet pump. This event is believed to have caused an accelerated depressurization in the EA3.1 (SSTF) calculation. The relatively course nodalization is believed to have contributed to the problem but not to have been the primary cause. The condensation calculation used cell averaged conditions for void and temperature, which can create severe distortions in large cells. As the fluid conditions in the jet pumps changed to subcooled liquid, the excessive depressurization rate was terminated. Further discussion on this subject is presented in Appendix E.

The pressure response in the containment model was assessed in two separate calculations: BLOWDOWN 18 (Marviken) and SEQUENCE 483 (MARK I). The pressure response was well-simulated throughout the system in BLOWDOWN 18. This included the drywell pressures, the wetwell pressure, and the vessel pressure, see Figures 3, 4, and 5. The drywell and wetwell pressure responses were also well-simulated in SEQUENCE 483, until the suppression pool reached its saturation temperature. TRAC failed to produce an accurate simulation beyond this point because it does not contain a containment pool boiling model. The drywell and wetwell pressure responses are shown in Figures 6 and 7, respectively. TRAC issues a warning message when the saturation point is reached, and it also has an option to stop the calculation at this point. Refer to Appendix A for a complete discussion of this subject.

The Marviken model was also used in a sensitivity study to investigate TRAC's numeric stability when the pressure difference between the vessel and

Table 3. Summary of key system parameters for operational transients[a]

Test	Maximum Pressure (MPa)		Minimum Pressure (MPa)		Time of Pressure Maxima (s)		Time of Pressure Minima (s)	
	Measurement	Calculation	Measurement	Calculation	Measurement	Calculation	Measurement	Calculation
Blowdown 18[b] (Marviken, blowdown/containment)	0.33	0.29	0.10	0.10	135.0	88.0	0.0	0.0
SEQUENCE 483[c] (MARK I, blowdown/containment)	0.12	0.12	0.11	0.10	1130.0	1130.0	0.0	0.0
EA3.1 (SSTF, large break blowdown)	1.03	1.06	0.36	0.23	0.0	0.0	50.0	50.0
PT (Browns Ferry, one-pump trip)	7.03	7.03	6.85	6.90	0.0	0.0	18.5	18.5
TPT (Browns Ferry, two-pump trip)	7.03	7.03	6.73	6.81	0.0	0.0	20.0	20.0
FTT (Browns Ferry, feedwater turbine trip)	7.03	7.03	6.95	6.99	0.0	0.0	40.0	40.0
6LR (Browns Ferry, generator load rejection	7.63	7.52	6.95	7.00	4.4	4.9	1.6	1.7

a. The natural circulation tests were not included in the table because the system pressure was held constant.

b. Pressure comparisons were made with the break outlet (room 122) pressures.

c. Pressure comparisons were made with the wetwell pressures.

Table 4. Summary of simulated pressure responses

Test	Results	Comments
3025 (FIX-III medium break)	Pressure response was well-calculated relative to the data.	System pressure was improved during the subcooled and transition periods by adjusting the break discharge coefficient.
6PMC2 (FIST, power transient)	Overall pressure response was in good agreement with the data.	There were some deviations in the calculated pressurization rates. It is suspected that not modeling the vessel flanges caused the calculated pressure response to be accelerated. This resulted in an additional SRV cycle prior to the core power termination.
PT (Browns Ferry, one-pump trip)	Pressure trends were in excellent agreement with data.	Maximum discrepancy in steam dome pressure was 1.3%, though the total pressure change was within the data uncertainty (3%) of the initial pressure.
TPT (Browns Ferry, two-pump trip)	Calculated and measured steam dome pressures showed excellent agreement.	Maximum discrepancy in steam dome pressure was 0.7%; however, the total pressure change was within the data uncertainty (3%) of the initial pressure.
FTT (Browns Ferry, feedwater turbine trip)	Simulated pressure trends showed a good comparison with the data.	A minor discrepancy occurred as the pumps were run back. This discrepancy was attributed to modeling uncertainties in the pressure controller.
GLR (Browns Ferry, generator load rejection)	System pressure response was very good.	The slight differences were thought to be a result of modeling uncertainties in the pressure controller and MSIV steam leakage.
RUN 912 (ROSA-III, small break)	Pressure response was generally well-calculated.	The differences in depressurization rates seems to have been caused by lower calculated break flow.
BLOWDOWN 18 (Marviken, blowdown containment)	Pressure response in the primary loop, drywell, and wetwell was in good agreement with the data and CONTEMPT.	Modeling uncertainties were believed to be responsible for a slight disagreement during the first 15 s.
SEQUENCE 483 (MARK I, blowdown containment)	Calculated trends in the wetwell and drywell were in excellent agreement until the suppression pool reached the saturation temperature.	TRAC does not contain a containment pool boiling model.

Table 4. (continued)

Test	Results	Comments
EA3.1 (SSTF, large break blowdown)	Pressure responses were adequately simulated. An initial overcalculation in the depressurization rate resulted in an undercalculated pressure.	Excessive steam condensation occurring as the jet pumps filled is believed to cause this overcalculated depressurization.
6NPC1-2b, 4, 6 (FIST, natural circulation)	Pressure was held constant by the pressure control system. Agreement with the data was very good.	The main steamline flow was in complete agreement.

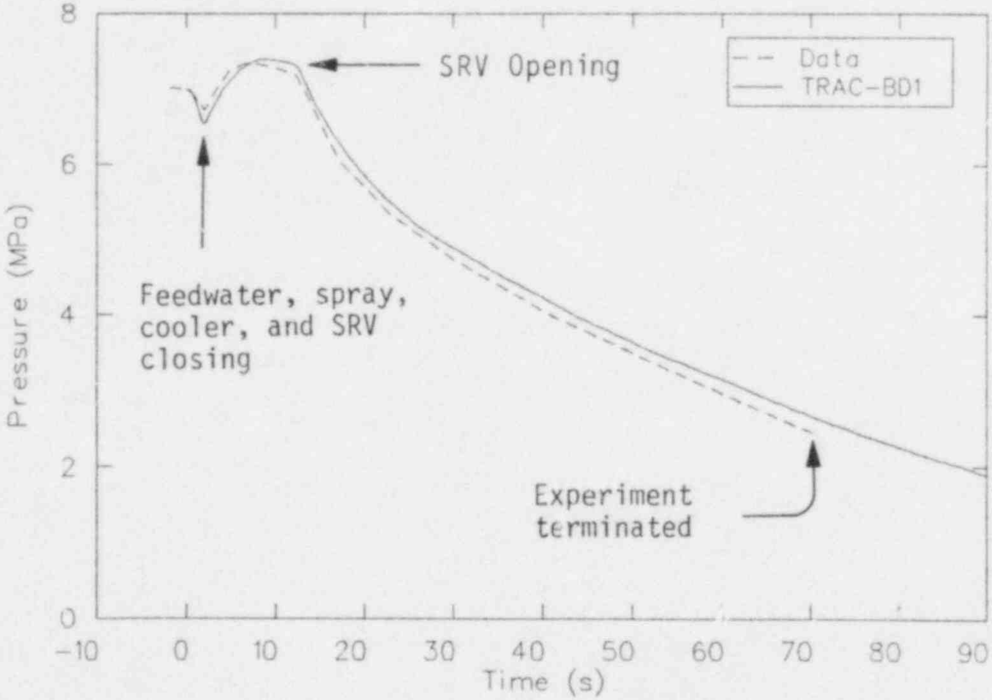

Figure 1. Comparison of measured and calculated steam dome pressure for Test 3025 (FIX-II).

the containment is small. The Marviken vessel pressure was arbitrarily reset so that the vessel/containment pressure difference was small. A 150-s calculation was then performed, and no instabilities were observed with the explicit numeric techniques used in the containment models. The pressures and mass flow rates appeared very stable.

Accurate SRV modeling was an important parameter controlling the pressure response in the 6PMC2 (FIST) calculation. It was determined that special modeling of the FIST SRVs was required to accurately simulate the measured mass flow rates. The code-data comparisons showed that slight deviations in the SRV flow caused deviations in the pressure responses. A discussion of this SRV model technique is presented in Section 4.6.

As indicated in the pressure response summary (Table 4), the simulation performed using the Browns Ferry Plant model showed excellent agreement with the data. The two-pump trip (TPT) test matched the data slightly better than the single-pump trip (PT) test.

In summary, the calculated pressure response was well-predicted, and most of the discrepancies

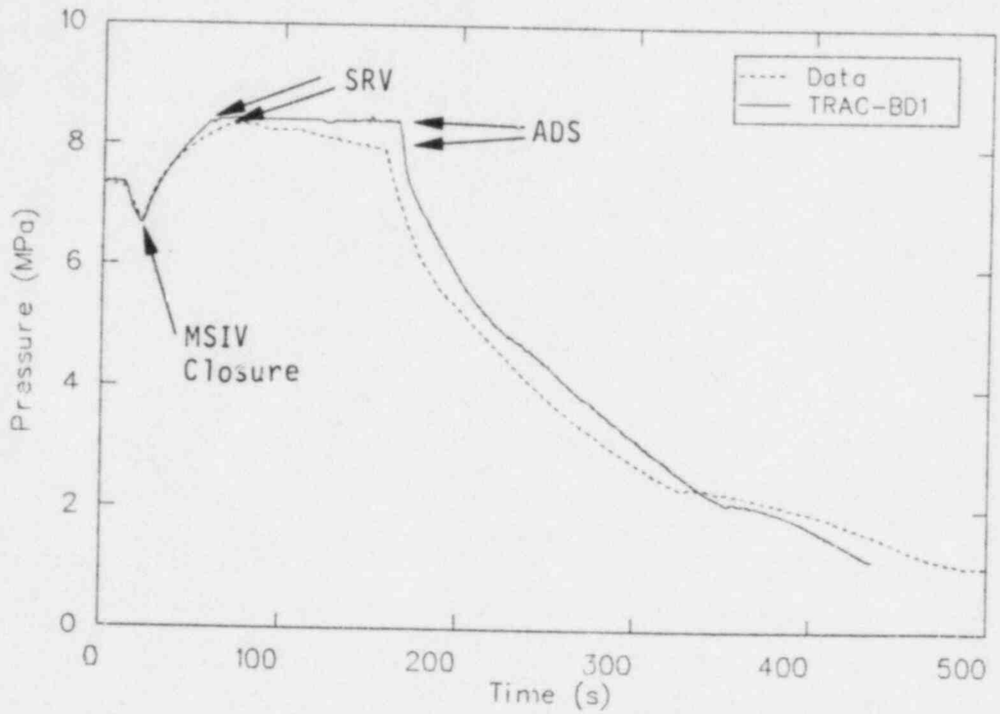

Figure 2. Comparison of measured and calculated steam dome pressure for RUN 912 (ROSA-III).

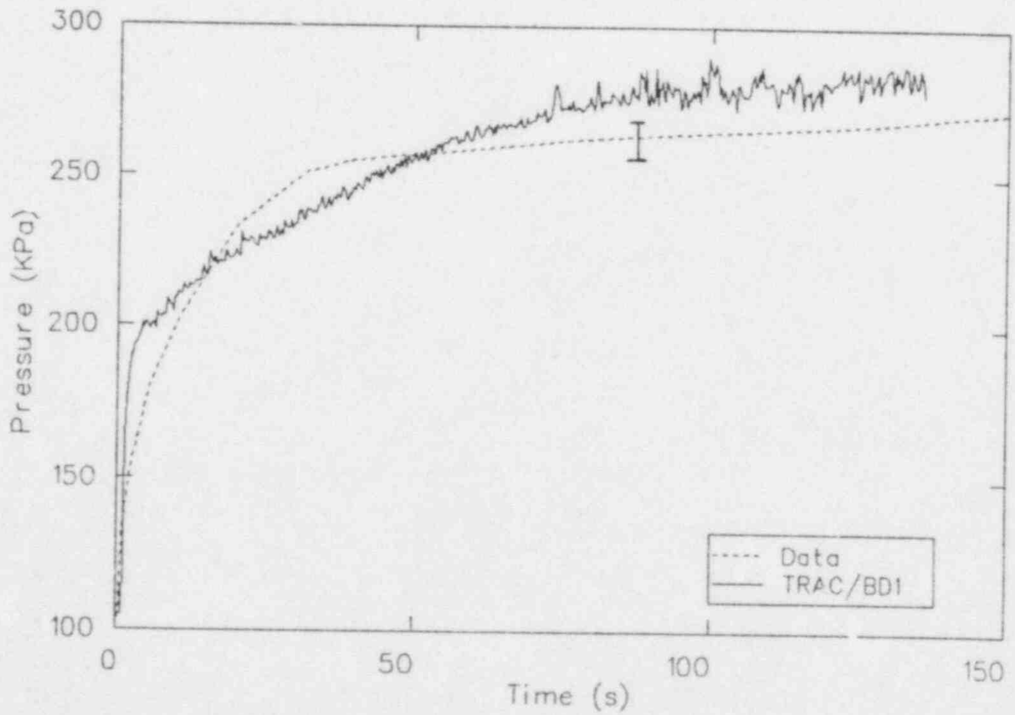

Figure 3. Comparison of drywell pressure, Room 104, for BLOWDOWN 18 (Marviken).

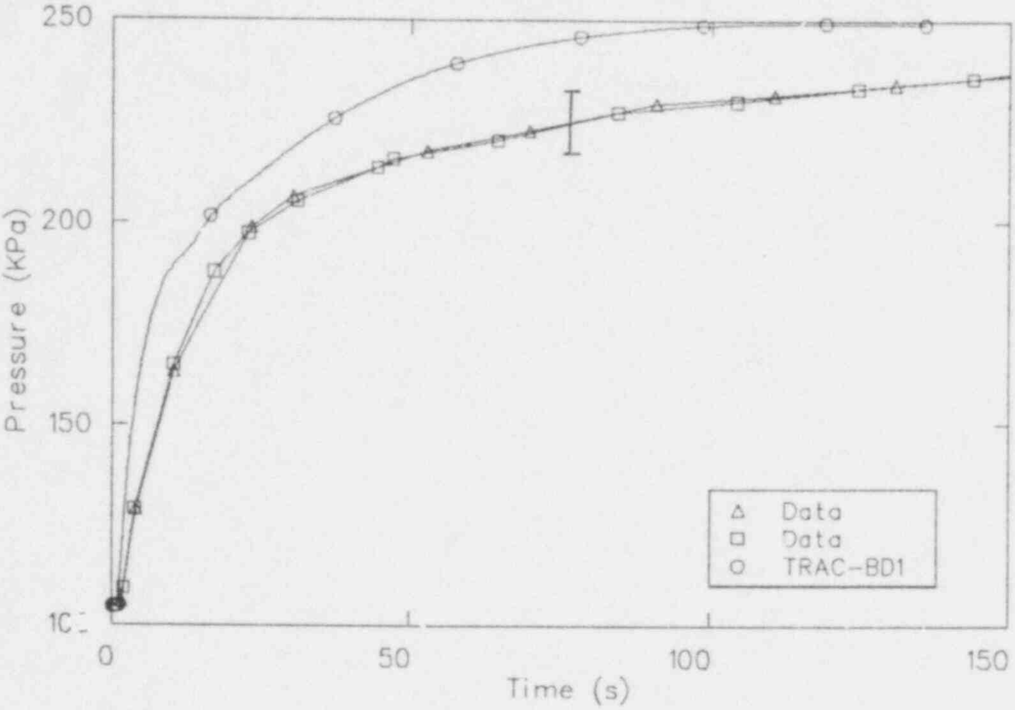

Figure 4. Comparison of wetwell pressure, Room 105, for BLOWDOWN 18 (Marviken).

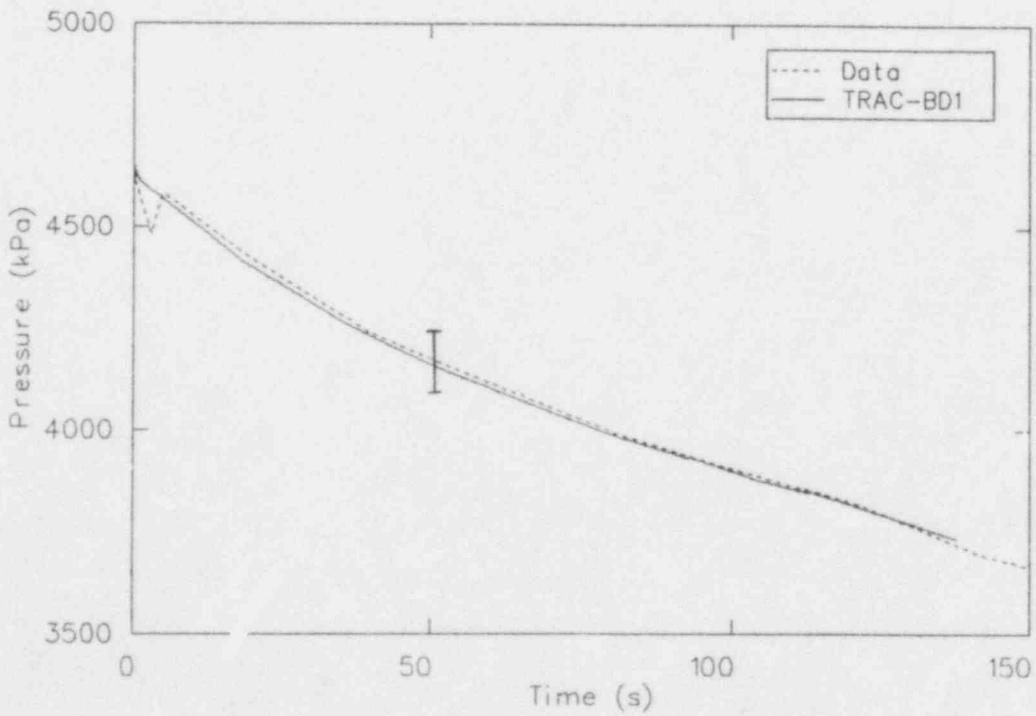

Figure 5. Comparison of reactor vessel pressure for BLOWDOWN 18 (Marviken).

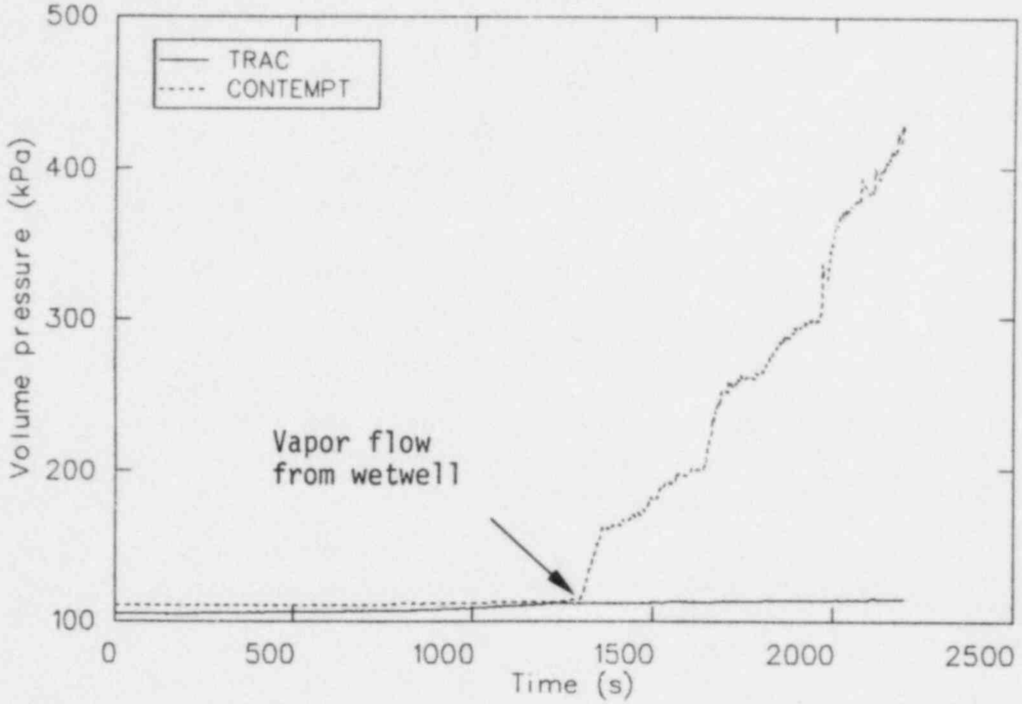

Figure 6. Comparison of drywell pressure for SEQUENCE 483 (Mark I).

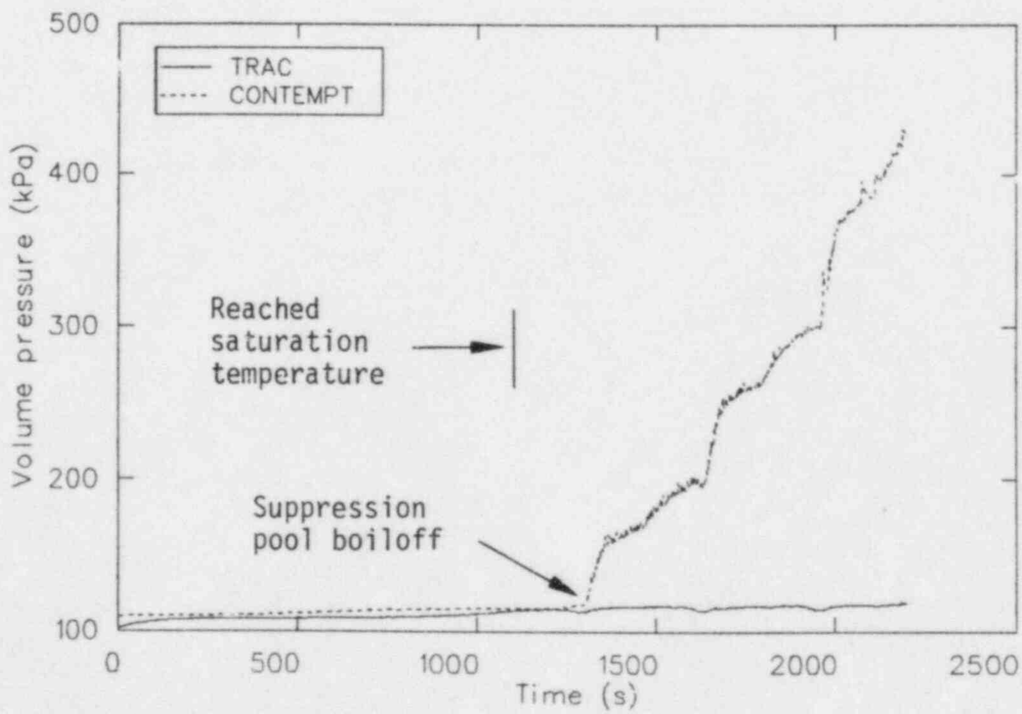

Figure 7. Comparison of wetwell pressure for SEQUENCE 483 (Mark I).

that did occur were caused by modeling sensitivities. The four major influences of the calculated pressures were break flow, condensation, containment pool boiling, and SRV flow.

The following conclusions were made relative to TRAC's capability to simulate the system pressure response:

- System pressure was well-predicted by TRAC during the LOCA simulations, the natural circulation transients, the power transient, the containment assessments, and the operational transients. Most discrepancies between measured data and calculated results are attributed to modeling sensitivities rather than code deficiencies.

- The system pressure response is very sensitive to SRV flow, break mass flow, core thermal response, and flashing/condensation. The calculated response of each parameter had a direct and significant influence on the system pressure during the calculations.

- Excessive vapor condensation in the jet pump is suspected of causing accelerated depressurization rates. It is recommended that the TRAC interfacial heat transfer model be reviewed with regard to this problem. Finer nodalization is one modeling technique that may reduce the effect of this problem.

- The containment pressures were well-simulated until the suppression pool reached saturation conditions. After this point, an accurate pressure response is impossible, since TRAC does not include a containment pool boiling model.

3.2 Simulated Core Thermal Response

In this section, the core thermal response and the major variables controlling it are discussed. The primary variables controlling the core thermal response are as follows:

- System pressure

- Event timings

- Vessel liquid mass distribution
  - Upper tie plate CCFL
  - Side entry orifice CCFL.

Generally, the simulated core thermal responses were in good agreement with the data.

In Test 3025 (FIX-II), the core thermal response agreed well with the data, provided the calculated break mass flow was correct. A sensitivity study regarding the break flow model is discussed in Section 3.3. Figure 8 shows typical rod temperature comparison. The initial offset corresponds to the temperature variation through the thickness of the cladding. The calculated surface temperature is being compared with the measured internal cladding temperature. As the power decayed, the temperature profile through the cladding thickness was significantly reduced (less than 4 K after 10 s).

During this same calculation, there was excellent agreement with the first departure from nucleate boiling (DNB); however, the calculation overpredicted the severity by approximately 45 K. Some of the thermocouples indicated a second DNB at about 22 s, but the code did not. This heat-up averaged only about 10 K and did not occur across any one level of the core. Therefore, it was attributed to localized behavior. The third DNB was a top down dryout. As shown in Figure 8, the time it occurred and the calculated rate of heat-up were in very good agreement with the data. Figure 9 illustrates the rate at which the heat-up propagated through the core. The propagation rate and dryout times were well-calculated.

Several of the CHAN modeling improvements made in TRAC-BD1/MOD1 were apparent in the core thermal response calculated for RUN 912 (ROSA-III). The fine-mesh model in the CHAN eliminated a pressure spike that occurred during a simulation of this test using a previous version of TRAC.[13] The pressure spike occurred at about 400 s, as indicated in Figure 10. The pressure spike was caused by quenching of the large nodes. The fine-mesh model provides a larger number of smaller nodes, which results in a more continuous quench behavior. There were also significant improvements in the calculated heater rod temperatures and core inventory results. The improved performance of the code is attributed to the modifications of the interfacial drag model and heat transfer relationships, and the inclusion of the fine mesh model.

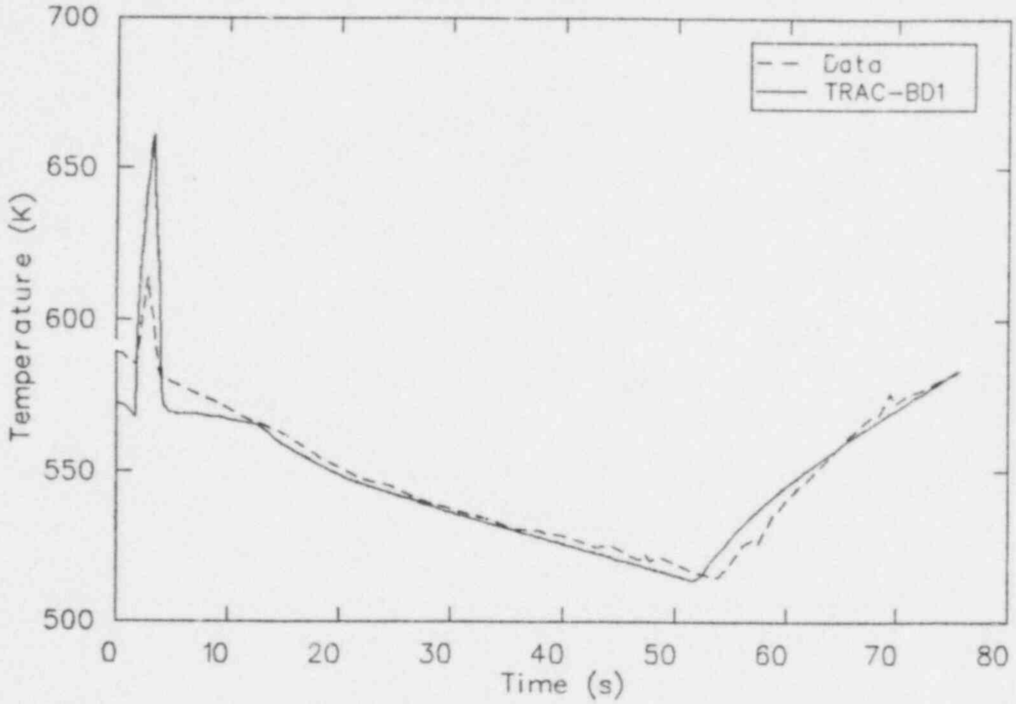

Figure 8. Comparison of experimental and sensitivity calculation rod cladding temperature for Test 3025 (FIX-II).

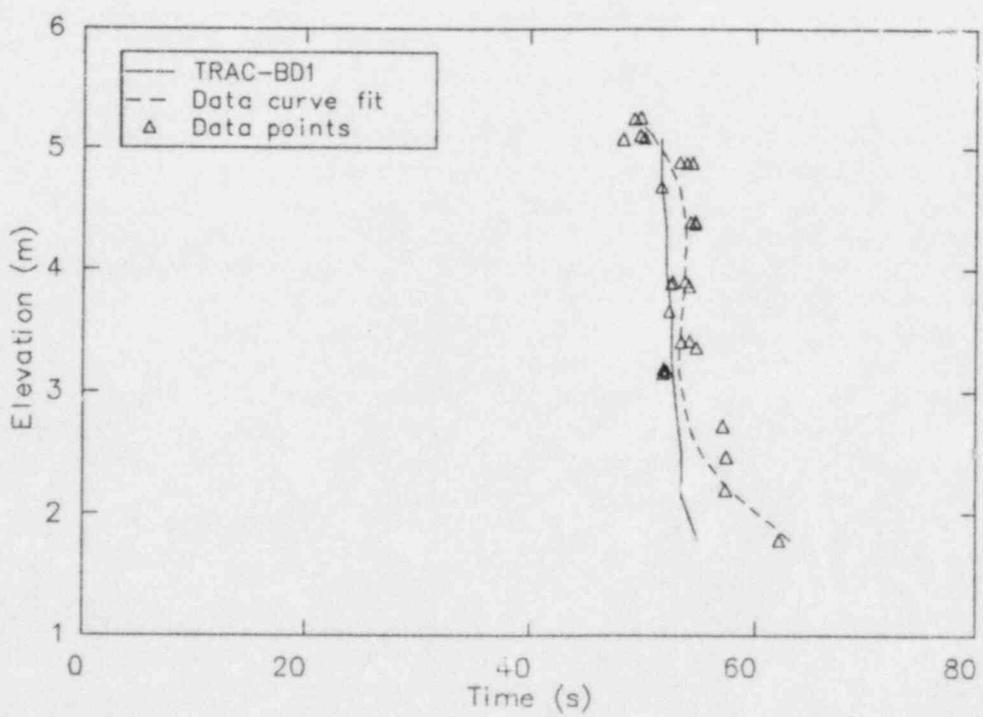

Figure 9. Comparison of experimental and calculation dryout front propagation for Test 3025 (FIX-II).

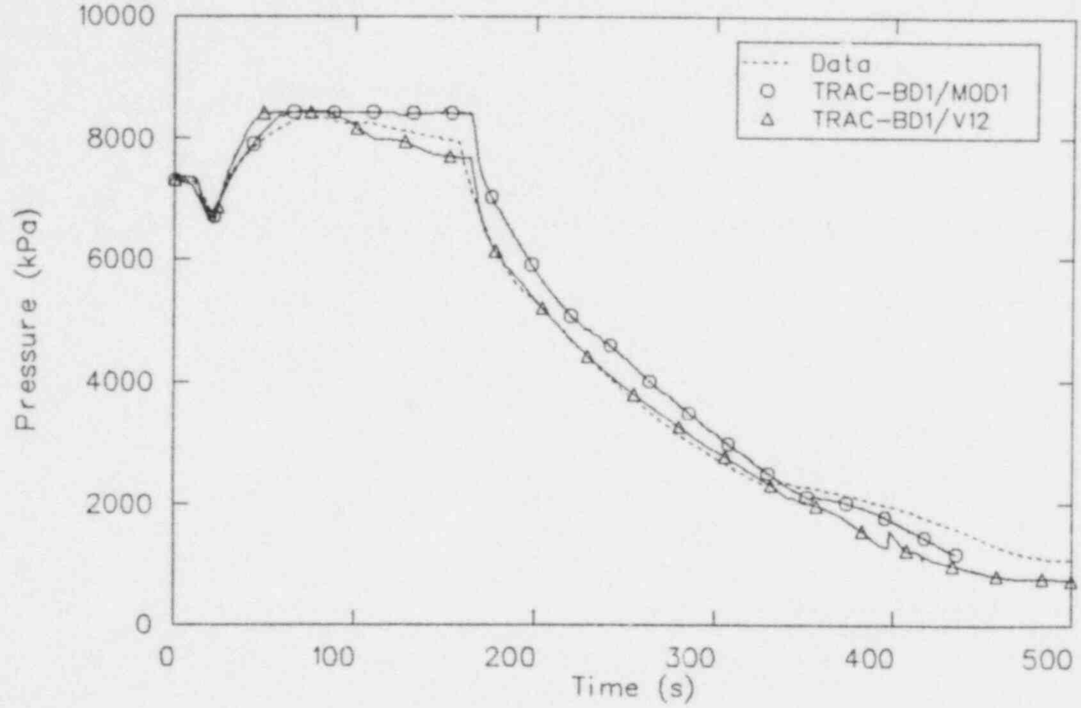

Figure 10. Steam dome pressure comparison between TRAC-BD1/Version 12, TRAC-BD1/MOD1 and the data for RUN 912 (ROSA-III).

Test 6PMC2 (FIST), calculated heater rod temperature response, matched the data very closely; however, there are two areas of disagreement, as shown in Figure 11. First, TRAC calculated the heat-up to occur slightly earlier, owing to an additional SRV cycling. Second, the heater rods quenched early because of an overcalculation of the liquid entrainment in the core. The data indicate the quench front was moving with the core liquid level. The calculation produced a separate quench front ahead of the core liquid level. This separate quench front was caused by the excessive entrainment above the calculated liquid level. The final quench was caused by the downcomer, lower plenum, and lower channel cells flashing, triggering a sudden swell in the channel. The data showed the lower downcomer reached saturation but did not flash. The reason that flashing occurred during the simulation is unclear.

There were no rod heat-ups measured or calculated during the natural circulation tests; however, a heat-up did occur in a sensitivity study of the 3-MW test. This was caused by excessive numerical diffusion triggered by the omission of the level tracking option. This subject is discussed further in Section 4.1.

Multi-channel behavior is a very important phenomenon controlling the core thermal response during the refill/reflood stage of a BWR LOCA. The behavior can only be experimentally produced in facilities containing many fuel bundles, such as the SSTF facility. Multiple-bundle geometry allows different flow regimes to occur and thus control the localized heat transfer differently. The flow regimes in a bundle usually depend on the radial location of the bundle and the power level of the bundle. Multi-channel phenomena has been determined to effect timing of the lower plenum and lower bundle reflood. These differences do not invalidate overall experimental results from few- or single-channel BWR subscale facilities. Rather, the differences show that the multi-channel phenomena occurring in the reflood/refill phase of a LOCA are an experimental gap in one- or few-bundle test facilities.

Test EA3.1 (SSTF) was simulated to assess the capability of TRAC to calculate the multiple-channel flow effects occurring during a refill/reflood stage of a BWR LOCA. The only major limitation in the facility regarding this phenomenon was that the bundles were not powered, thus, eliminating the possibility of rod temperature comparisons.

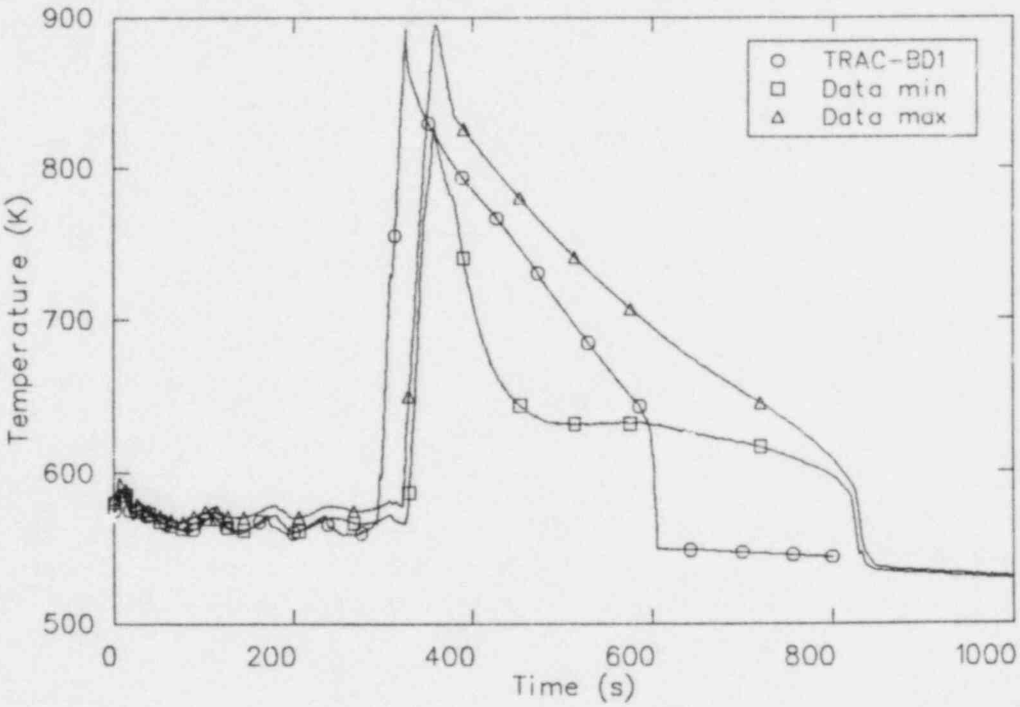

Figure 11. Comparison of calculated and measured heater rod temperatures for Test 6PMC2 (FIST).

The next few paragraphs discuss the differences between the measured and the calculated multichannel effects during the EA3.1 Test. In the data, LPCS formed a nearly uniform pool above the upper tie plate (UTP) allowing liquid drainage to the bypass and channel regions. The side entry orifice (SEO) and lower tie plate (LTP) leak paths permitted flow to the lower plenum, though the bypass and channel regions showed a net increase in mass. The LPCS liquid drainage lead to the formation of three principle regimes:

- Co-current upflow in the central, high-powered bundles
- CCFL conditions at the SEO and UTP located in the middle, average-powered bundles
- Liquid down flow located in the peripheral, low-powered bundles.

In the base case simulations, a substantial fraction of the LPCS fluid was convected into the steam dome region rather than draining into the bypass and channel regions. An insufficient amount of LPCS liquid drained into the channels to cause any of the channel flows to change to a liquid down flow regime. The lack of drainage to the bypass and channel regions resulted in these areas having a net mass loss instead of a net gain, as the data indicate. The TRAC/BWR Development Group has determined that the failure to calculate LPCS drainage was a result of TRAC's nonconserving form of the momentum equation and its staggered mesh finite difference approximation. This subject is discussed further in Section 4.10.

A sensitivity study was performed to find a method of minimizing the severity of this momentum problem. The method was to increase the bypass/upper plenum interface junction flow areas and adjust the junction form loss factors to produce equivalent hydraulic resistances. The sensitivity calculation was terminated at 50 s because of flow oscillations at the upper plenum/channel peripherally interface. The calculation indicated that the LPCS liquid pooled above the UTP as in the data, except it was skewed, with the bulk of the fluid being in the peripheral regions. The peripheral LPCS drainage was sufficient to cause CCFL breakdown in the peripheral regions; however, it was also much greater than the data indicated. The overcalculated peripheral flow seemed to retard the drainage in the middle and central regions. The upper plenum and bypass mass inventories were in

adequate agreement with the data until ~28 s, when computational problems started. Overcalculation of the drainage from the bypass region was then observed, a result of CCFL modeling compromises (discussed in Section 4.3). The channel inventories were underpredicted because of insufficient LPCS drainage in the middle and central regions. In summary, upper plenum drainage was not correctly calculated during the base case, but upper plenum drainage trends were improved during the sensitivity calculation.

The following conclusions were made relative to TRACs capability to simulate the core thermal response:

- TRAC demonstrated the capability to accurately simulate the core thermal response, provided the mass inventory in the bundle is correct.

- Improvements in TRAC since the TRAC-BD1/Version 12 assessments, provided for a better simulation of the data. Heater rod response and core fluid behavior were in good agreement with the data.

- The fine-mesh model in the CHAN component worked well during the reflood portion of the transients. The fine mesh model provided a good simulation of the rod quench behavior; and averted a pressure spike found in earlier versions of the code when large coarse nodes quenched.

- Upper plenum drainage trends were difficult to calculate with TRAC because of its nonconserving form of the momentum equation.

3.3 Simulated Break Flow Response

The most important parameters controlling the simulation of the break flow are as follows:

- Break plane upstream conditions
  - Void fraction
  - Subcooling
- Choking plane location
- Event timings of ECCS, ADS, SRV, etc.

The effects of these parameters on break flow are described in the following paragraphs.

Break flow was modeled in three of the assessment simulations: a large break containment test, an intermediate split break LOCA, and a small break LOCA. Several break flow sensitivity calculations were also conducted while performing these assessments.

BLOWDOWN 18 was a large blowdown/containment simulation conducted at the Marviken facility. The Marviken vessel and break piping were modeled as two pipes. A break component connected the primary piping and the containment. This is the first time the containment was directly coupled to the primary system. Figure 12 compares the calculated and measured break mass flow rates. The calculated values showed excellent agreement with the measured data. Measured break flow data were determined from differential pressure measurements.

The intermediate split break LOCA was conducted in the FIX-II facility. No ECCSs were activated during the test. The first attempt to simulate the experiment was prior to release of the data, and the initial simulated break flow response appeared reasonable. The data comparison showed the calculated break mass flow was underpredicted by as much as 25%. The simulation was then reconducted after increasing the break discharge coefficient from 0.7 to 0.9. The calculated break flow was then considered to be in good agreement with the data. Figure 13 illustrates the calculated and measured break flows. There are two major areas of uncertainty present in the reported break flow comparison. First, the calculated system depressurization rate showed a trend very similar to the measured pressure; however, the measured subcooled break flow did not seem to correspond with the depressurization rate. Second, the calculated break flow indicates an abrupt reduction in the break flow rate, because the recirculation lines flashed and the measured data did not. These discrepancies are believed to be within the uncertainty of the data.

A discharge coefficient is used to reduce the break flow area to adjust the break mass flow rate. The assessment calculations performed to date have indicated the discharge coefficient needs to be between 0.7 and 1.0; however, no obvious methodology has arisen to determined what value within this range to use. As indicated in Test 3025 (FIX-II), this range of coefficients may easily alter the break flow by more than 25%.

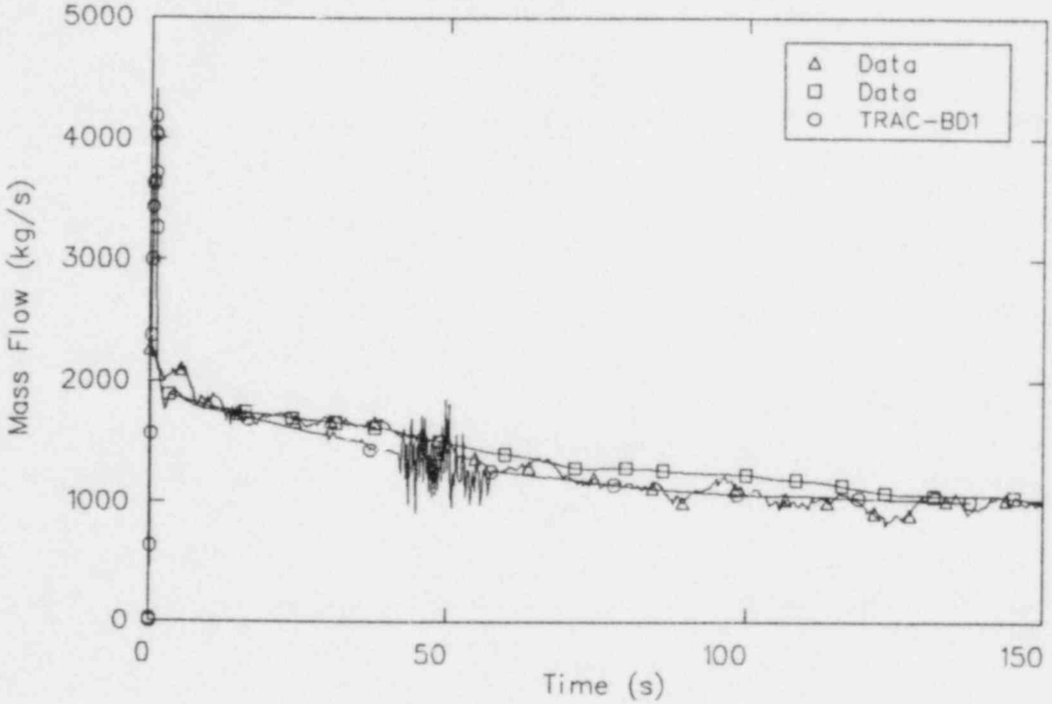

Figure 12. Comparison of break mass flow rate from the reactor vessel for BLOWDOWN 18 (Marviken).

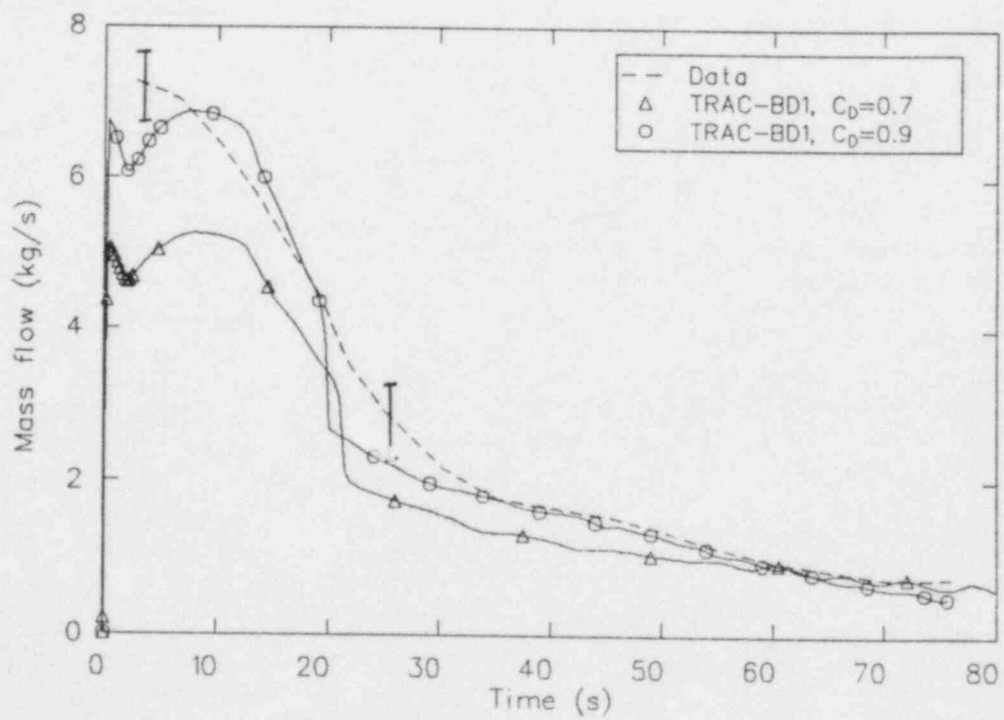

Figure 13. Comparison of experimental and calculation break flow for Test 3025 (FIX-II).

An attempt was made to estimate a general discharge coefficient based on previous assessment calculations (see Reference 14). It was determined to be 0.75; however, this has proven to be inadequate. In Test 3025, the initial discharge coefficient, 0.7, was determined by comparing the geometry of the discharge nozzle with similar nozzles having predetermined discharge coefficients. This method also proved to be inadequate. It is recommended that further investigation be conducted on guidelines to determine appropriate break discharge coefficients.

The last calculation in which a break was modeled was RUN 912, a small break LOCA conducted at ROSA-III. The break flow data were reported to have large uncertainties; a direct comparison of the break flows was therefore of limited value. The rest of the measured data indicated the break flow was probably underpredicted.

The amount of subcooling in the recirculation loops strongly influenced break mass flow. The steady state lower plenum subcooling was approximately 6 K in the calculation and 11 K in the measured data. As shown in Figure 14, the break mass flow rate increased 15% during a sensitivity calculation when the break flow subcooling was increased from 6 to 11 K. Although several attempts were made to adjust the amount of subcooling, the model always steadied out to about 6 K subcooling.

In a later assessment study using the ROSA-III model, it was determined the downcomer level control system was causing this subcooling problem.[15] The control system monitored the downcomer level and the steam-feedwater mismatch signal. It used a proportional-integral controller to adjust the feedwater flow to achieve the desired downcomer level. To solve the subcooling problem, the gain on the steam-feedwater mismatch signal and the limits on the feedwater flow set point were adjusted. The steady state feedwater flow increased, which increased the amount of subcooling.

The following conclusions were made relative to TRAC's capability to simulate the break flow response:

- Calculated break flows adequately simulate the data.

- Uncertainties remain in the use of the TRAC critical flow model. Adjustment of the break discharge coefficient is required to correctly simulate the break mass flow. Break discharge coefficients producing good simulation range from 0.7 to 1.0. Development of more specific user guidelines is considered to be a worthy effort.

- The subcooled critical flow model is very sensitive to the amount of fluid subcooling. A sensitivity study showed that a 5-K change in break flow subcooling can cause a 15% decrease in break mass flow.

3.4 Simulated Containment Temperature Response

This section examines the vapor and liquid temperature response of the containment model. It is important to note that the temperature is calculated with the assumption that each volume or compartment is well-mixed. This assumption is consistent with the rest of the code and generally true; however, some containment designs, such as the multiple-room design in the Marviken facility, can lead to temperature stratifications within the compartments. Marviken data indicate some compartments with a 20-K temperature variation; however, the effect on most simulations is likely to be minimal. The following paragraphs summarize the simulated responses in the containment assessments reported in Appendix A.

The vapor temperatures in the wetwell and drywell were well-calculated until pool boiling was indicated. As discussed in the Section 3.1, TRAC does not calculate pool boiling in the containment model.

The suppression pool temperature in the Marviken calculation was increasingly undercalculated. By the end of the transient, the data were underpredicted by about 20 K, as shown in Figure 15. This is attributed to lack of communication between the drywell and the wetwell. The spill option was not used during the initial calculation. If the amount of mass the data indicate was added to the suppression pool (at the saturation temperature), this could account for the temperature difference.

The suppression pool temperature in the MARK I calculation compared well with the CONTEMPT calculation. The calculated and measured suppression pool temperatures are compared in Figure 16.

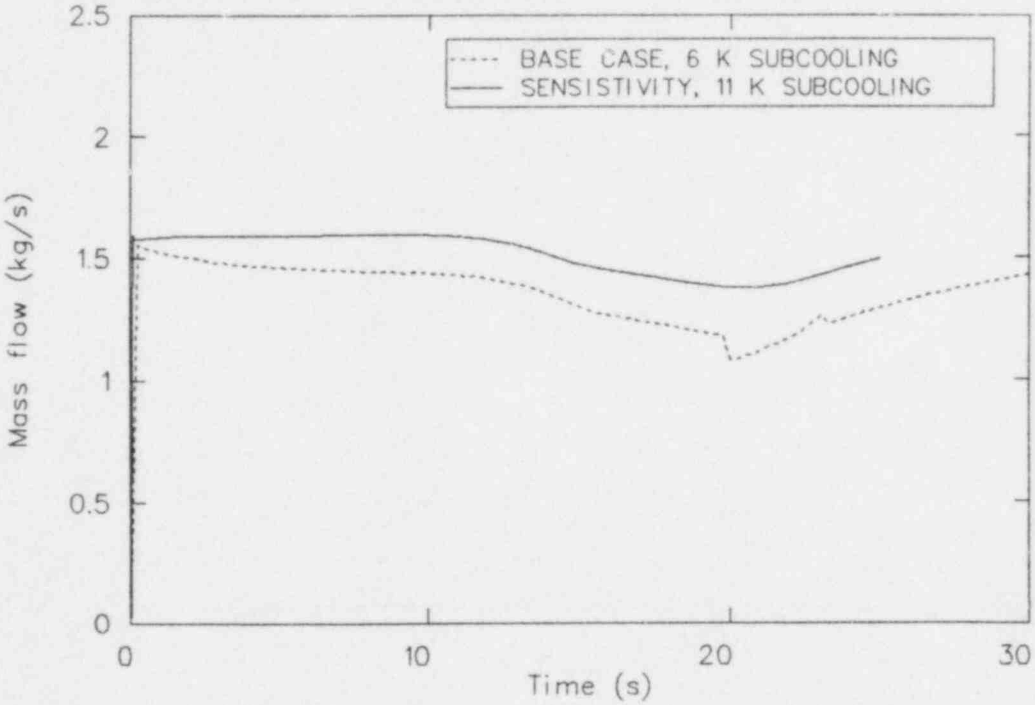

Figure 14. Comparison of base case and sensitivity calculations of the break flow for RUN 912 (ROSA-III).

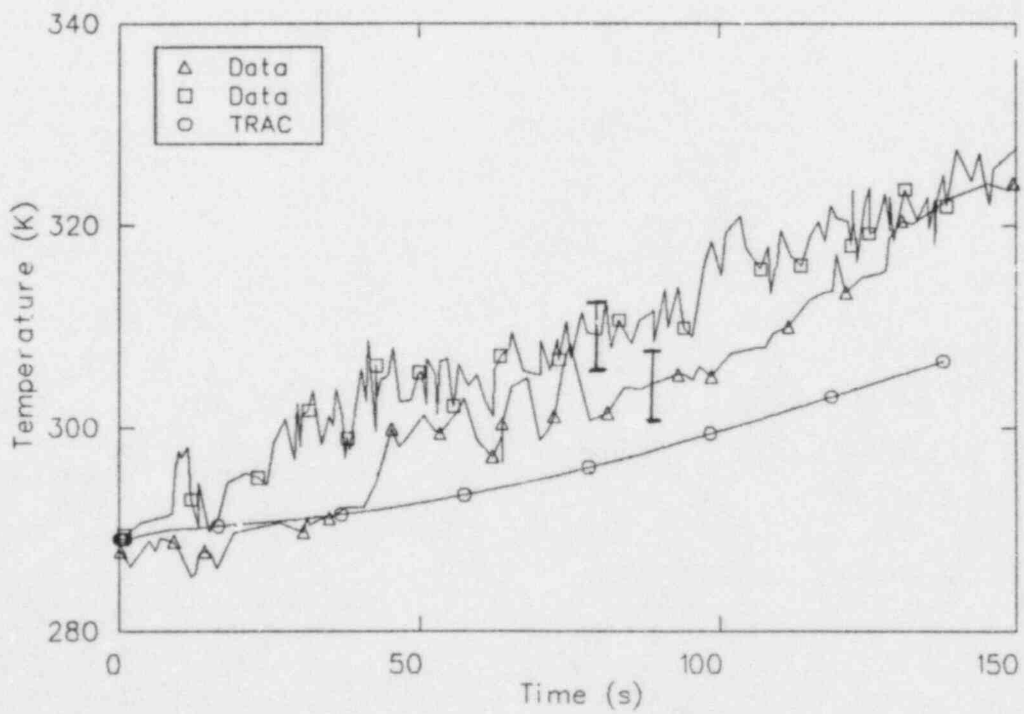

Figure 15. Comparison of wetwell liquid temperature for BLOWDOWN 18 (Marviken).

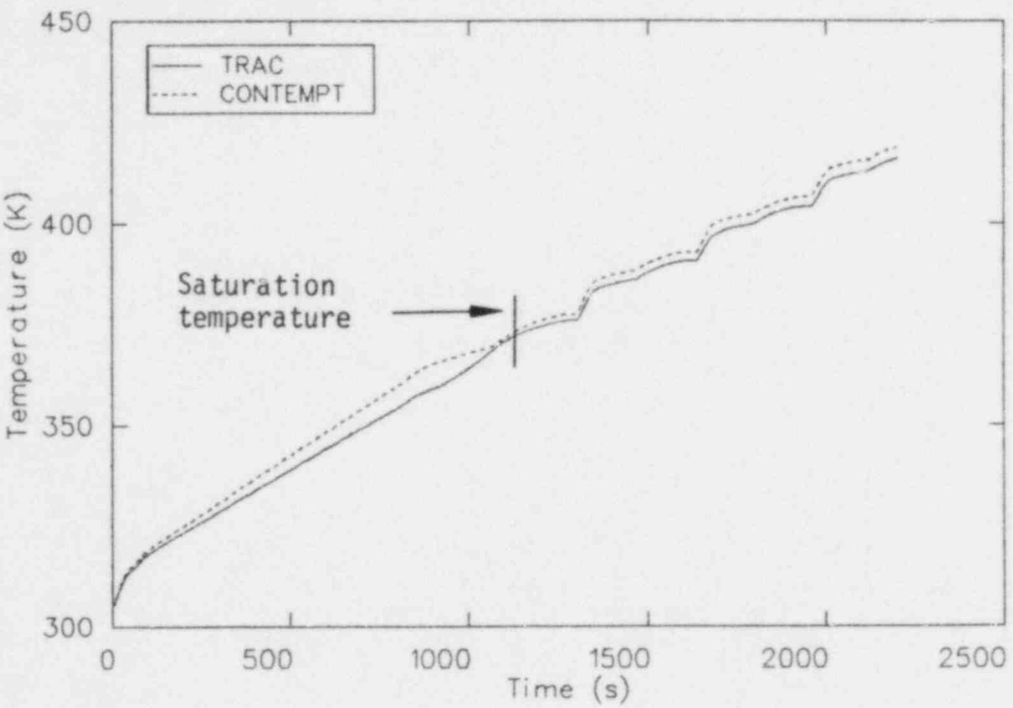

Figure 16. Comparison of suppression pool liquid temperature for SEQUENCE 483 (Mark I).

The following conclusions were made relative to TRAC's capability to simulate the containment response:

- TRAC demonstrated the capability to calculate BWR containment behavior during both a large break and an ATWS transient. Results from transients in which the pool temperature nears saturation should be used with caution.

- The multi-room capability demonstrated by the Marviken calculation worked very well. The multi-room capability may also have application to the secondary containment of BWRs.

3.5 Simulated Natural Circulation Flow Response

This section summarizes the natural circulation flow response calculated during the three assessment simulations reported in Appendix D. The feedwater was near saturation conditions in all three tests. The calculated response is discussed and compared to the data in three regions: downcomer, bypass, and lower plenum.

The natural circulation flow versus the downcomer level matched the data relatively well. There were two major discrepancies between the data and the simulations. First, the simulations showed step changes in the flow rates after a downcomer cell boundary was crossed by the liquid level, see Figure 17. This problem was caused by the liquid level tracking model. Second, large flow oscillations occurred as the liquid level crossed a flow area reduction. This is illustrated in Figure 18. Both of these problems are discussed in Section 4.1 and Appendix D, where it is concluded that further refinement of the level tracking model is required to better simulate slowly falling liquid levels.

The calculated flow in the secondary natural circulation path between the core and the bypass was also compared with the data. This flow was from the bundle to the bypass flow. A typical code-data comparison of the natural circulation bypass core flow is shown in Figure 19. The slightly higher calculated flow to the core from the bypass is attributed to a decreasing inlet core flow rate. The calculated bypass flow generally agreed with the data except for two points: the level crossing caused step changes in the flow and large flow oscillations. The step changes in the flow rate were caused by the level tracking model. The flow oscillations were

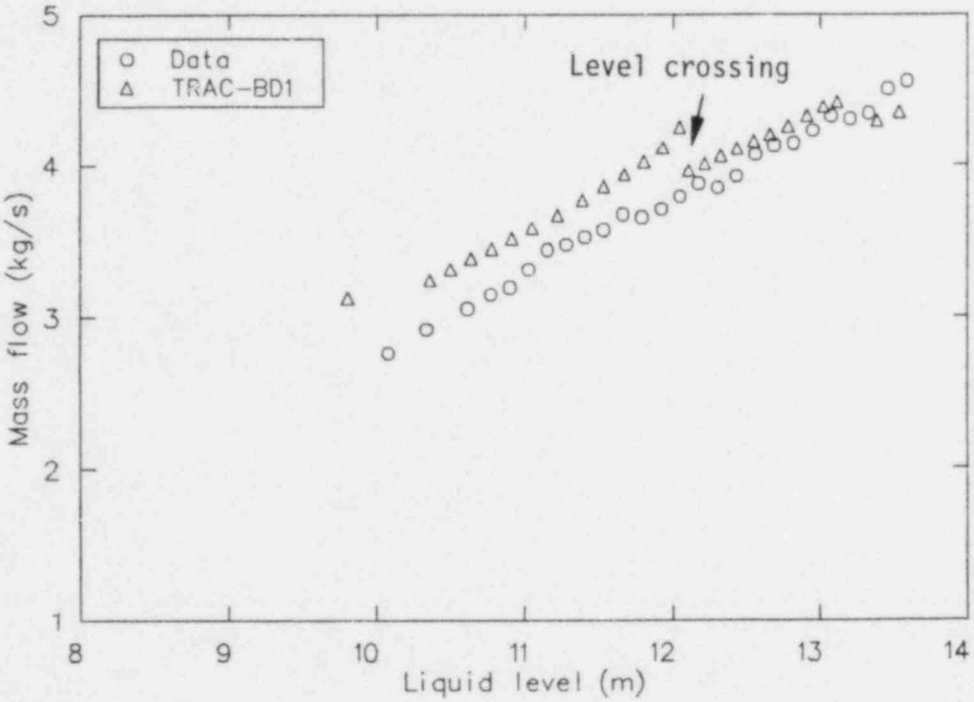

Figure 17. Comparison of the measured and calculated natural circulation flow versus downcomer level for Test 6PNC1-4 (FIST).

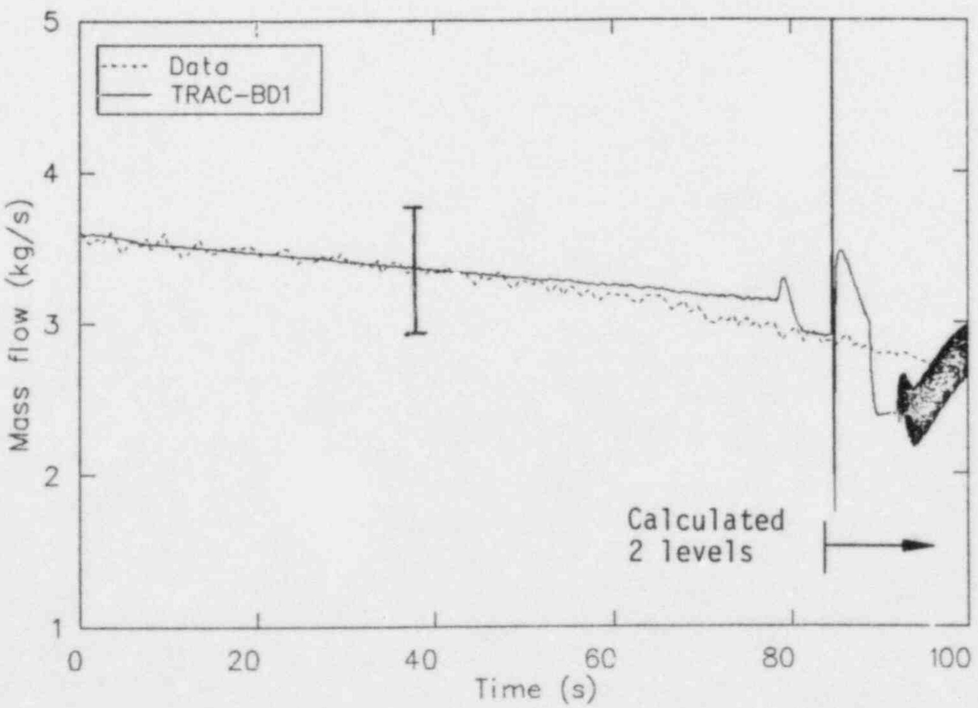

Figure 18. Comparison of the measured and calculated natural circulation flow versus downcomer level for Test 6PNC1-6 (FIST).

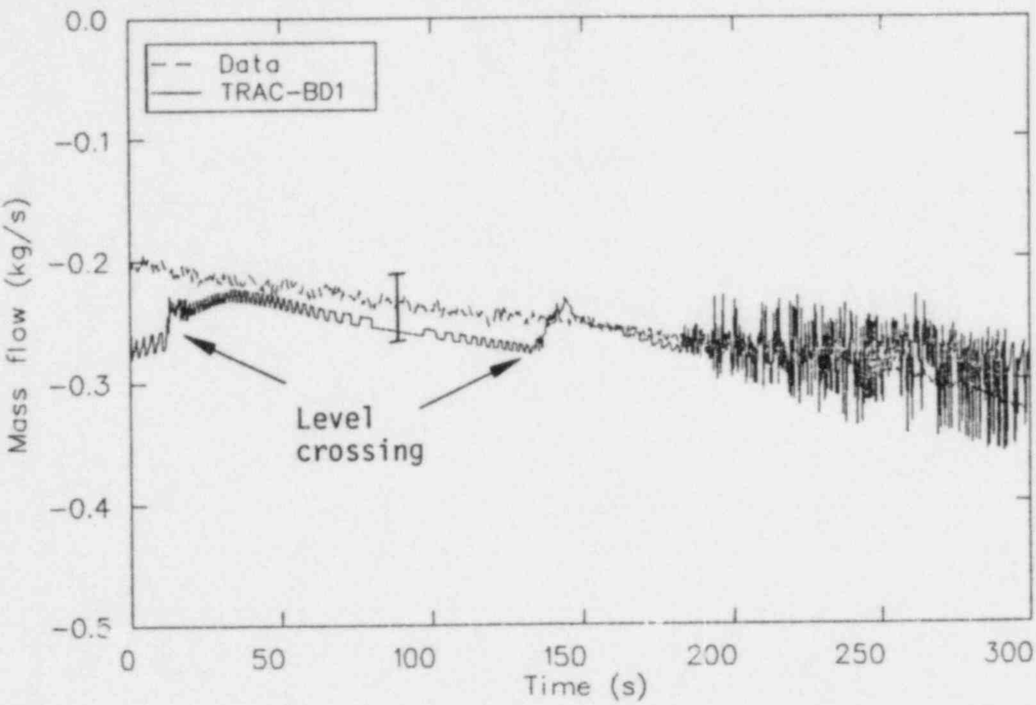

Figure 19. Comparison of the measured and calculated core bypass flow for Test 6PNC1-4 (FIST).

caused by the explicit leak flow junction used to model the bypass. It is believed an implicit solution for the leak path would stabilize the flow and eliminate the problem.

There were two paths for circulation to the bundle from the one-dimensional jet pumps in the FIST facility. The first was through the lower plenum and the second was through a cross-flow piece connecting the jet pump discharge and the bundle. In all three tests, TRAC calculated the cross flow to always flow from the downcomer to the core regions, but the data indicate a flow reversal in the cross piece occurred during two of the tests. TRAC failed to calculate these flow reversals because TRAC does not contain a source momentum term for a one-dimensional connection to the vessel. The effect was minimal in the natural circulation tests, but could be significant in a BWR/4 calculation with boron injection.

The following conclusions were made relative to TRAC's capability to simulate the natural circulation flow response:

- The key parameters were well-calculated by TRAC during the natural circulation assessments. The calculated core flow, downcomer level, bypass flow, and vessel temperature distribution were in adequate agreement with the data.

- A secondary natural circulation path between the core and bypass was calculated correctly; liquid drained from the upper plenum into the bypass and back into the core.

- Incorrect pressure distributions caused by the level tracking model created step changes in the natural circulation flow rates. Nevertheless, these fluctuations did not severely affect the transients. The influence of this problem was less with a rapidly changing level than with a slowly changing level.

- The leak path calculation should be made implicit. In the FIST model, the core bypass flow was modeled using an explicit leak path junction. The explicit nature of the leak flow calculation caused flow oscillations at the bypass junction. At low flow rates, these oscillations might adversely affect the core flow.

- The code needs to be assessed for natural circulation with subcooled conditions and varying downcomer levels. The available experimental data base does not include tests with these conditions. It is recommended that future experimental test programs consider including tests with these conditions.

- The inclusion of a source momentum term for a one-dimensional component connection to the vessel should be investigated. The absence of a momentum term calculation at the jet pump discharge connection to the vessel inhibited the code from correctly calculating the lower plenum flows.

4. SENSITIVITY CALCULATION RESULTS AND USER GUIDELINES

This section discusses the additional information determined during the sensitivity calculations not directly related to the parameters discussed in Section 3. It also discusses user guidelines which have been identified during the various assessment calculations.

4.1 Level Tracking Model

This section reviews the assessment of the level tracking model. The advantages of using the model are presented and then the limitations of using the model.

The application of the level tracking model in the vessel permitted a coarser nodalization to be used, which substantially reduced computer cost during the ROSA-III simulation. In addition, the level tracking model still permitted accurate modeling of the vessel source locations and prevented numerical diffusion. It also significantly improved the calculated trends in the vessel lower plenum and annulus regions compared with assessment calculations performed using earlier versions of the code which did not contain this model.

The use of the level tracking model in the downcomer is recommended in natural circulation simulations. Using the natural circulation data, three sensitivity calculations were performed to assess the performance of the level tracking model. The omission of the level tracking model caused; excessive downward diffusion of vapor in the downcomer, the downcomer to drain in a nonphysical manner, and nonphysical rod heat-ups during the calculation.

Using the level tracking model significantly improved the calculated results of the assessment simulation; however, there were three limitations encountered with its use: flow oscillations, conflicting levels, and pressure disturbances.

Flow oscillations occurred when a slowly falling level, about 0.02 m/s in the natural circulation tests, approached a cell boundary. The default buffer size, TRAC parameter EPSALPL, determines when the level is transferred. The default value of EPSALPL is inadequate for these situations, and making EPSALPL a user-definable variable should be investigated. The inadequacy can be minimized for slowly changing levels by reducing the level tracking void parameter EPSALPL.

Flow instabilities were erroneously calculated as the liquid level crossed a flow area reduction during the 6PNC1-6 natural circulation test (FIST). The liquid level was calculated to occur above and below the flow reduction in the downcomer as the level approached the area reduction. This caused large level and pressure oscillation, which adversely affected the transient. The transient could only be successfully completed by removing the flow area restriction in the downcomer. The necessary conditions for two different levels to occur in adjacent cells should be investigated. See Section 4.10 for a description of how the momentum solution in TRAC affects this behavior.

The balance between the static head and the dynamic flux momentum terms in the cell with the liquid level was also incorrectly calculated. The pressure distribution was accurate for static conditions, however step changes occurred in the downcomer pressure after each level crossing. This problem is believed to be caused by the momentum equation used with the level tracking model. The effect of this problem can be minimized by using a finer nodalization. It was also observed that the behavior just described was significantly minimized for transients having fast level changes. See Appendix D for a detailed discussion.

4.2 Extract Feature

The extract feature in TRAC is provided to enable the user to withdraw the thermal hydraulic component information for a specified restart dump time. This allows the user to change parameters for a sensitivity study without having to rerun the entire transient. An attempt was made to use this feature during the 6PMC2 test simulation, but it produced a dramatic difference in thermal response of the sensitivity calculation. Careful examination of the calculations revealed a problem with the sensitivity calculation caused by the lack of significant digits transferred during the extract operation. The extract feature was only transferring four significant digits, which was insufficient to accurately represent the void and pressure conditions in the core. The code has since been updated

to increase the number of significant digits transfered. The number of significant digits now varies between 6 and 10, depending on the specific programming in each area of the code. This update has not been fully assessed, but it is believed to have eliminated the problem.

4.3 CCFL Option

The countercurrent flow limiting (CCFL) option was found to be very limited and was a source of code-data differences in several of the simulations. In particular, the SSTF model showed premature drainage of the bypass liquid into the guide tubes. This behavior was partially induced by the limited CCFL coefficients available for modeling CCFL. As done in the SSTF calculation, the only method available to modify the two sets of CCFL default coefficients is to alter the wetted perimeter input. This was an inconvenient method of adjusting the CCFL coefficients. The CCFL phenomenon directly depends on the specific geometry characteristics and needs to be more flexible. Since this assessment series was performed, the code has been updated to allow the input of 10 user defined CCFL coefficient sets.

4.4 One-Ring Models

This section describes problems encountered while using the one-radial-ring two-theta-section model for the FIST facility. In the FIST facility, the downcomer and a section of the lower plenum are external to the vessel. This is illustrated in Figure 20, where the FIST vessel is compared with a BWR/6 vessel. The FIST model is shown in Figure 21. It was necessary to model the FIST facility in this manner to allow independent ambient heat loss for each cell, one-dimensional flow throughout the vessel, and proper modeling of the vessel walls for both the downcomer and the core regions. A problem associated with the separator model is described, then a problem with heat transfer in the theta direction.

The TRAC vessel steam separator requires at least a two-ring-vessel model. This allows the liquid to return to the downcomer in the outer ring. Therefore, one ring geometry models, such as FIST, need to use either the perfect pipe separator or the mechanistic pipe separator. In the FIST model, these separator models triggered severe oscillations in the return liquid flow rate. These flow rate oscillations were caused by the explicit nature of the leak path used for the return liquid flow calculation. The flow rate oscillations were sufficient to cause the entire system to be unstable. Consequently, an update was created by the TRAC-BWR Development Group that allowed the vessel separator to return the liquid through the theta face. The update solved this modeling difficulty in the FIST facility. The pipe separator models have been replaced with tee separator models in the TRAC-BF1. The tee separator models are based on a tee component rather than a pipe component to allow for a more implicit leak path. The tee separator models have not been fully assessed at the present time.

No thermal communication was modeled between the downcomer (cell 1) and the standpipe, separator, and dryer regions (cell 2) of the FIST model, because TRAC does not allow heat transfer in the theta direction. The impact on the calculation was judged to be significant. In the 6PMC2 Test, the standpipe was cooled by the RCIC fluid entering the top of the downcomer. It is believed the downcomer would have supplied enough cooling to significantly reduce discrepancies in the following areas: steam dome superheat, system depressurization rate, core exit flow, and amount of entrained vessel fluid. See Appendices C and D for more information about these problems.

The one-ring and two-theta-section model used in this assessment is considered un[illegible] for BWR models; however, it was considered necessary to correctly model the ambient heat loss in this facility. Because of its limited application, the TRAC-BWR Development Group is reviewing whether to include the VESSEL separator model update in the code. The update is currently available to users upon request. The priority of developing a theta heat transfer model is also currently being reviewed.

4.5 Flow Restriction Modeling

It was determined in the course of modeling flow restrictions that use of the minimum flow areas and the loss coefficients based on these areas resulted in underpredicted mass flow rates. This is because of the backward differencing formulation in TRAC, which fails to calculate the full pressure recovery downstream of a flow restriction. It was determined that the reference upstream flow areas and their corresponding loss coefficients yielded mass

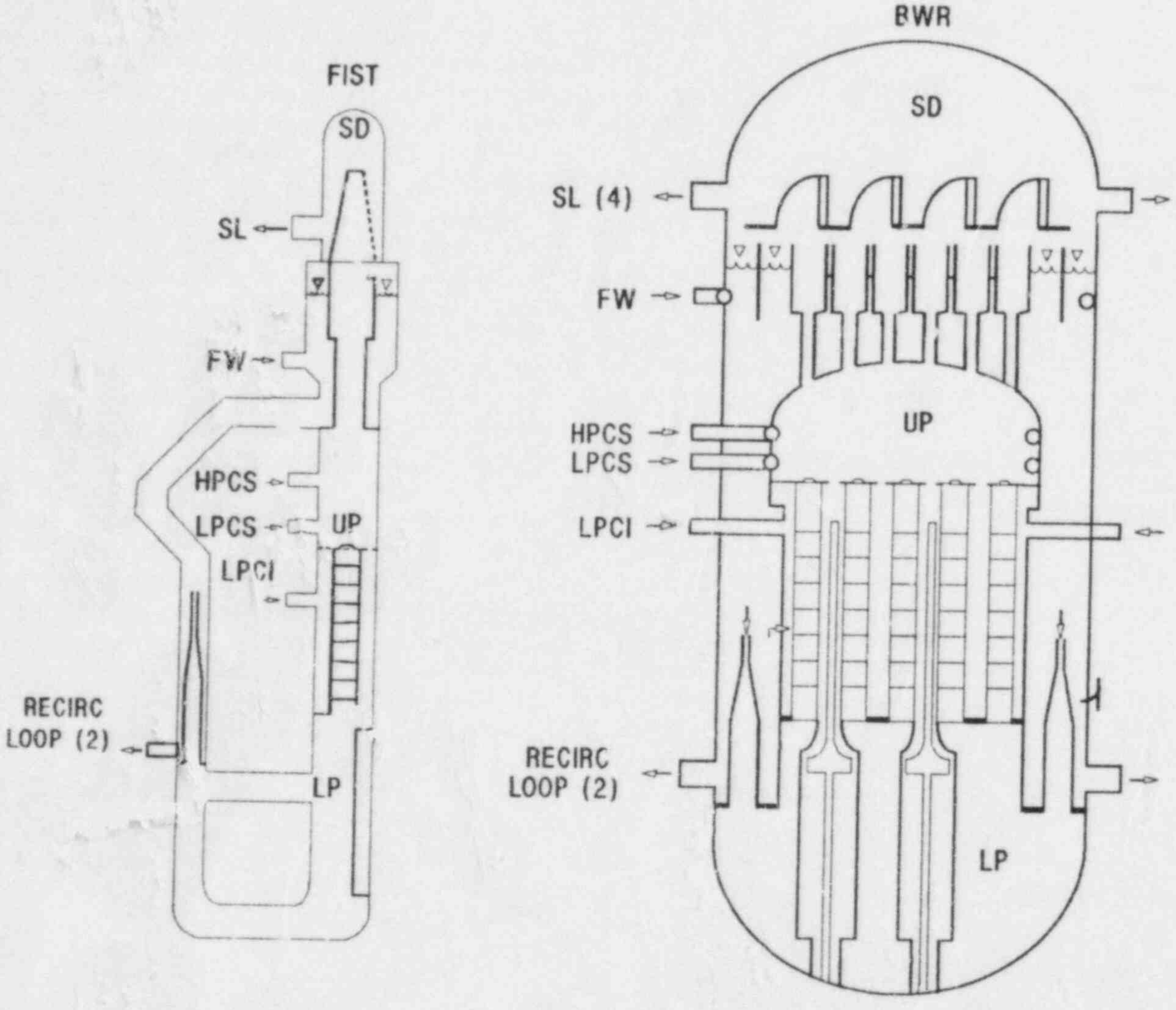

Figure 20. Comparison of FIST vessel with a BWR/6 vessel.

flow rates comparable to the data. Caution should be used when modeling flow restrictions without pressure loss data.

A central differencing scheme is currently under development, which will eliminate the irreversible pressure loss calculated in the backward differencing scheme. The central difference scheme will be incorporated in TRAC-BF1 when it is completed.

4.6 Multiple Bank SRV Option

TRAC allows multiple banks of SRVs to be modeled with a single component called a multiple bank SRV (MBSRV). This option enables the model size to be reduced. The MBSRV option correctly models banks of valves when the valves are the same size. However, the MBSRV should be used with caution when modeling banks of valves containing different sizes of valves. Using the actual valve flow areas to model different sizes of valve banks in a MBSRV may result in the calculation of erroneous mass flow rates.

This discrepancy is caused by the method in which TRAC scales the flow area to represent different combinations of valves opened. A single hydraulic diameter and valve flow area are input to the code for the MBSRV. To model the various SRV operational conditions, the user inputs a table with the percentage of the specified flow area for each SRV valve combination (i.e., one valve open, two

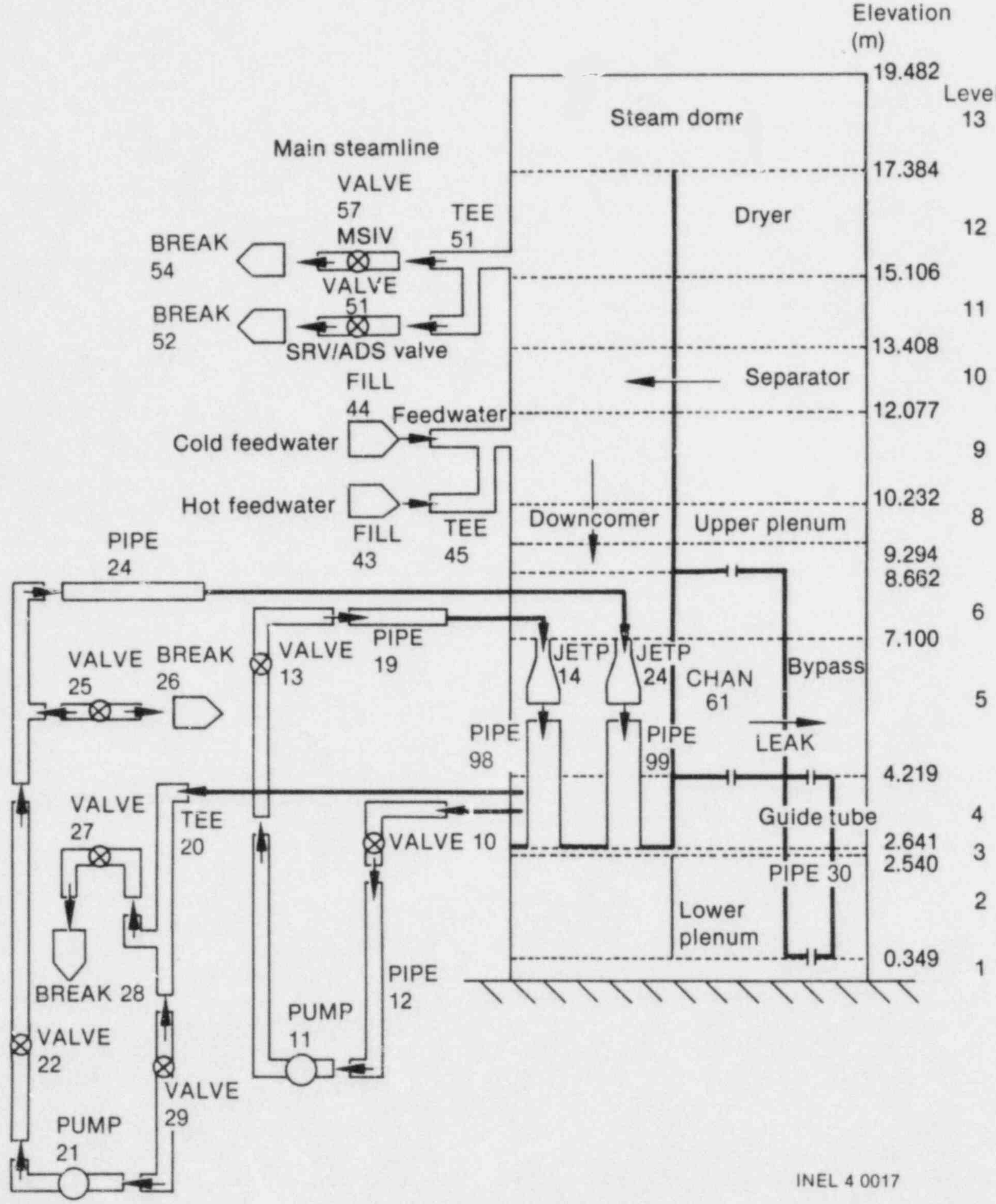

Figure 21. FIST system nodalization.

valves open, etc.). This model assumes the hydraulic diameter varies linearly with the valve area, which is only true if all the valves are the same size.

Modeling banks of different sizes of valves with a single MBSRV component requires measured mass flow data be available. A sensitivity study is required to determined the valve flow areas necessary to achieve the desired valve response. It is recommended that the MBSRV component be modified to include the hydraulic diameter with the flow area input. An immediate solution is to use a separate MBSRV or valve component for each size group; however, this will require many additional components in the model. The MBSRV option models multiple valves correctly when all of the valves are the same size.

4.7 Water Packing Option

Two sensitivity studies were conducted to evaluate the utility of the water packing model. First, the study using the EA3.1 (SSTF) data is discussed; second, the study using the RUN 912 (ROSA-III) data is presented.

The SSTF sensitivity study consisted of running the base calculations of the EA3.1 Test with the water packing option on. The first attempt to use the water packing option during steady state was unsuccessful because large flow oscillations occurred before the system stabilized. The second attempt was during a rerun of the base calculation. The calculation failed about 5 s after the break occurred owing to unrecoverable water packing problems in the jet pump ice. With the option off, TRAC was able to recover from these temporary computational problems. The third attempt was during another rerun of the base calculation, which failed about 22 s after the break was opened. The calculation stopped because of water packing in the jet pump, as in the base case calculation.

The conclusions from this study are that the water packing option imposed additional constraints on the code and generally should not be used. It was useful as a restart tool to pinpoint when the computational problems start (without the option, water packing can only be inferred from large pressure and flow spikes).

The water packing sensitivity study using the ROSA-III data was performed to evaluate the affects on numerical water hammer. The water packing option was designed to adjust the junction velocities surrounding cells where water packing is detected. Water packing is assumed when a water-solid cell surrounded by partially voided cells causes large pressure increases or decreases. It is considered a water hammer if the surrounding cells are also water-solid. TRAC's primary test to identify water packing is to check to see if the surrounding cells are water-solid. TRAC was unable to distinguish between water hammer caused by numerical problems and that caused by thermal-hydraulic phenomena.

After consulting with the TRAC-BWR Development Group and performing the ROSA-III calculation with the water packing option on, it was concluded that the numerical water hammer oscillations are either handled incorrectly or are ignored by the water packing model. The water packing model needs to be improved to prevent nonphysical water hammer problems between one-dimensional components and the VESSEL component.

4.8 Separator Modeling

The separator cell liquid return junction in the FIST model was connected to a liquid-filled cell. During the steady state calculations a very low void fraction was calculated in the separator cell. Because of the low flow conditions during the natural circulation tests, the pressure balance between the liquid-filled downcomer cell and the adjacent separator cell was maintained by filling the separator cell with liquid. An excessive amount of liquid in the separator would cause inventory problems during the transient. To alleviate this problem, the separating cell was moved up so that the bottom of the cell was at the lowest pickoff ring elevation (see Figure 22). With this configuration, the separator would fill only when the downcomer liquid level was higher than the lowest pickoff ring. However, since the liquid return was connected to a vapor-filled cell, excessive downward numerical diffusion of vapor existed. The influence of the downward numerical diffusion was minimized by using the level tracking model in the downcomer region. This modeling technique has been accepted practice for our modeling and is recommended when using the vessel separator.

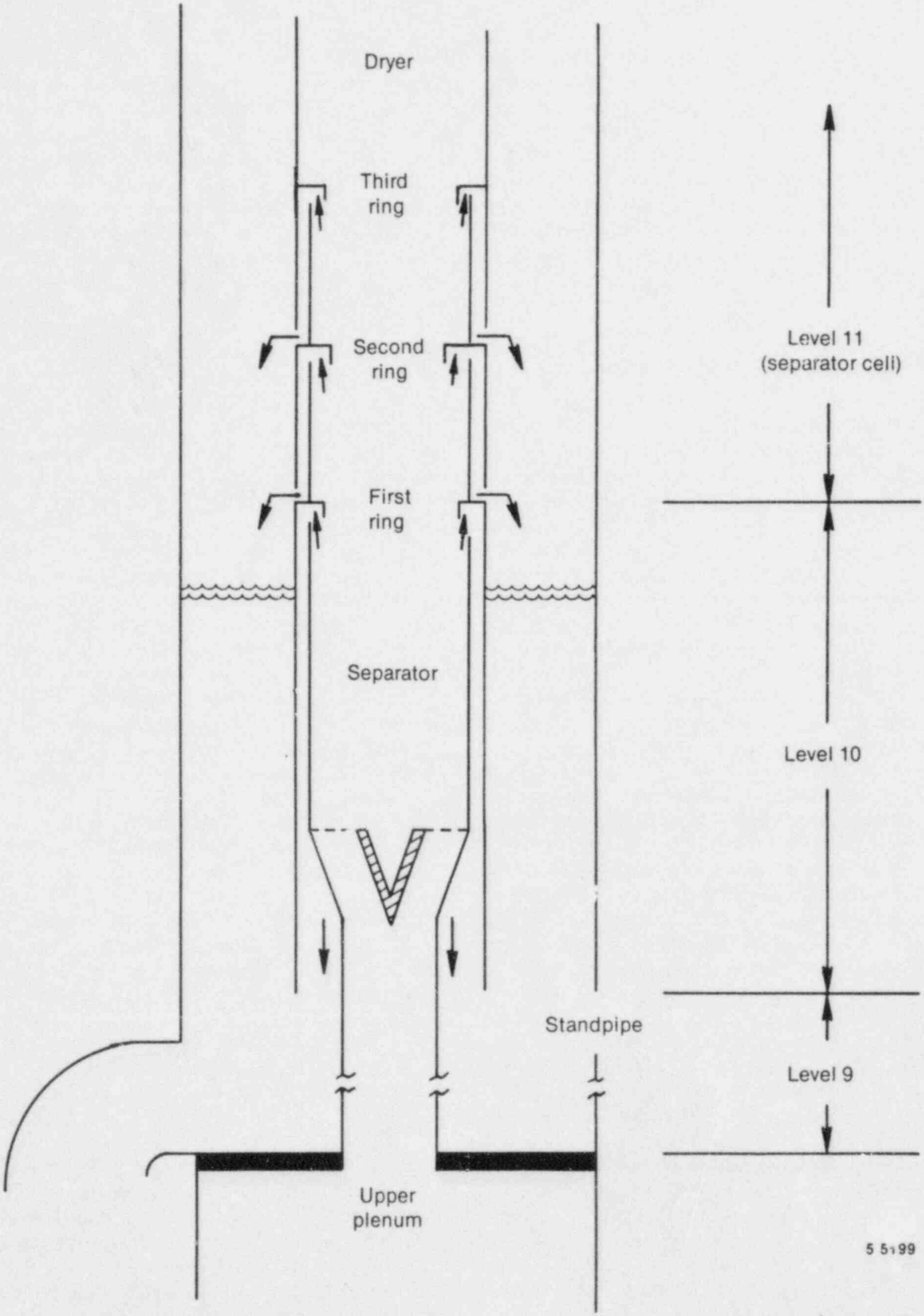

Figure 22. Separator model diagram.

4.9 Spill Option

The code does not allow liquid from the break or condensation within the drywell to flow from one compartment to the next to the wetwell without specifically turning on the spill option. If the spill is off, TRAC will stop the calculation when liquid completely fills a containment volume. When modeling a containment compartment, the user must be careful that the compartment has sufficient volume for the effluent or the condensation, or that spill option is activated to allow mass to be transferred to the next compartment.

4.10 Momentum Solution

This section summarizes the response of the TRAC-BWR Developmental Group to difficulties reported in obtaining correct velocity and pressure profiles in cells near void fraction discontinuities.[16] Investigation into the origin of these difficulties indicates that the assumptions used in TRAC's treatment of momentum transport and conservation are not reasonable at such points of discontinuity. TRAC's form of the momentum equation is nonconservative in the sense that the computed momentum transport out of one cell is not identically equal to the computed momentum transport into the adjacent cell, except in the limit as the spatial node size approaches zero.

The inaccuracies in momentum conservation have been acceptable up to now in light of the economy of numerical solution to this form versus other forms. However, the assumptions used to derive this form together with the accuracy of its approximated solution, become questionable in regions where strong gradients or discontinuities in any of the field variables exist. For example, the effective elimination of numerical diffusion of void by use of the level tracking model in TRAC can give rise to sharp void gradients, which approximate a true phase boundary (level). In addition, the use of course nodalizations can also generate large gradients in the field variables. Thus, by using either of these modeling techniques a user unknowingly generates precisely the conditions that can result in spurious results because of the nonconservative finite difference form of the momentum equation.

The upper plenum drainage calculation during the EA3.1 test (SSTF) simulation is reviewed as an example of the problems that can result from use of TRAC's nonconservative momentum equation. In this calculation, the steady state flow conditions resembled those shown in Figure 23. High quality steam rose through the vessel bypass region at $\sim$3 m/s, and was accelerated at the upper tie plate flow restriction to 25 m/s. The liquid droplets were carried with the steam and also reached a high velocity ($\sim$20 m/s) at the flow restriction. Subcooled liquid was added to the upper plenum region in Cell j, just above the tie plate. Though this liquid entered the cell with zero axial momentum the finite difference formulation of the nonconservative liquid momentum, equation at junction j + 1/2 treats all the liquid in Cell j as though it entered the cell with velocity $v_{\ell,j-1/2}$. Thus, the subcooled ECCS liquid was artificially given a momentum sufficient to carry it out of Cell j and into Cell j + 1, contrary to what was observed physically. The problem was caused by improper association of the liquid mass in Cell j with the velocity of the liquid entering Cell j.

In conclusion, the integral forms of the conservation equations for mass, energy, and momentum are those most appropriate for numerical solution using finite difference schemes. The use of pointwise nonconservative partial differential equations and their finite difference approximations can result in spurious computer results when regions of sharp discontinuities or steep gradients of field variables occur between nodes. Two situations where this conditions can occur are (a) where use of the level tracking model results in a steep void gradient between cells, and (b) where coarse nodalization results in large differences in mass and energy inventories in adjacent large cells.

The use of donor cells weighting in the spatial gradient term of the momentum equation results in inconsistent application of the forces represented in the equations; i.e., the fluid mass to which the spatial portion of the inertia term applies is not the same as the fluid mass to which the gravity term applies. This gives rise to spurious calculated velocity profiles in regions of steep void gradients.

It is the consensus of the TRAC-BWR Code Development Group that the problems encountered thus far with the current momentum formulation in TRAC are inherent in the nonconserving form of the momentum equation and its staggered mesh finite difference approximation. In order to achieve more physically reasonable results in all flow situations, the momentum equation should be

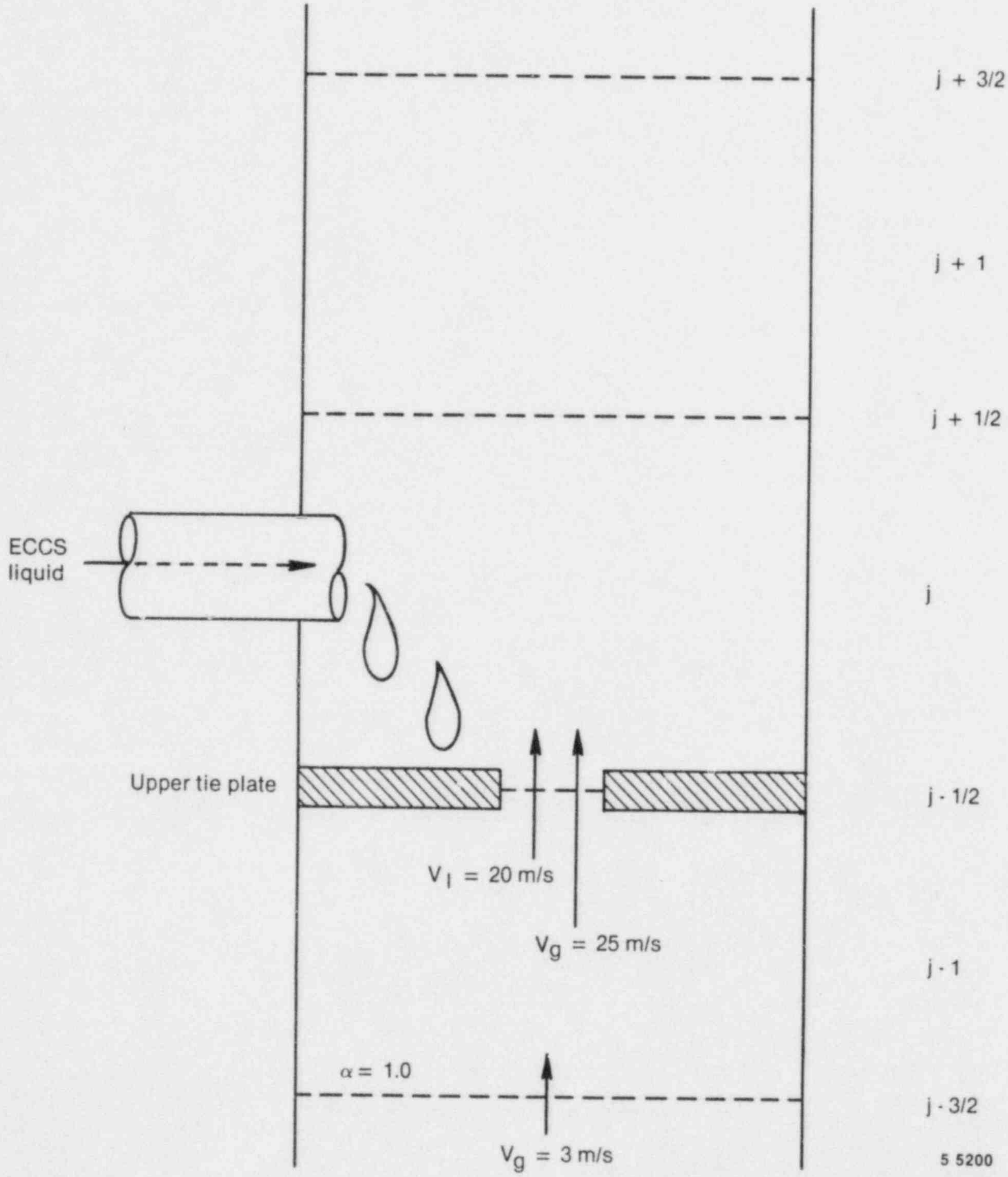

Figure 23. Schematic of upper region of vessel bypassed at upper tieplate.

reformulated in integral (conserving) form and, ideally, the mass, energy, and momentum equations should be applied to the same fluid volumes. Admittedly, this task is not trivial and would result in additional computational cost and perhaps computational stability problems. The resolution of the latter problem would probably require implicit evaluation of the spatial terms, which would further increase the costs of the calculation.

Until such changes are effected, however, it is recommended that code users be aware of the current deficiency and take measures to mitigate the effects. Specifically, the following guidelines are suggested:

- The level tracking model should be used with discretion, and level tracking should probably not be performed in regions where a liquid-vapor froth may be assumed to exist.

- When possible, regions of discontinuity of different field variables should be separated by appropriate nodalization measures. For example, a cell in which a steep void gradient exists should not contain a severe flow area change. Rather, the cell should be split into two, with the void discontinuity in one cell and the area reduction in the other.

- Fluid velocities in regions with steep void gradients should be minimized by use of large flow areas. Where this approach may bias the calculation unreasonably, the computational grid should be refined and smaller cells should be employed in the regions of steep gradients.

5. CODE UNCERTAINTY

The USNRC is currently in the process of developing a statistically based methodology to quantify the uncertainty of their thermal-hydraulic safety codes. In support of the USNRC, the TRAC-BD1/MOD1 simulation of the ROSA-III small break test, RUN 912, was analyzed using two proposed methods: variance and time series analysis. The purpose of the study was to aid the development of the uncertainty methodology; however, while performing this task, preliminary information was determined with regard to the code uncertainty of TRAC-BD1/MOD1.

The study was limited to the examination of the cladding temperature at the core hot spot, evaluating the differences in the temperature between the code and the actual of averaged data. Appendix H describes in more detail the two analysis methods and the results from the application to RUN 912.

The relative merits of each analysis method were evaluated by comparing the maximum and minimum values of the cladding temperature difference for each of the different phases of the test (i.e., blowdown, core heatup, core reflood). Table 5 presents the uncertainty range derived from local temperatures (actual thermocouple measurements) and volume-averaged temperatures (averaged from all thermocouple measurements) with code-calculated volume-averaged temperatures. The uncertainty is defined as the experimental data value minus the code-calculated value; therefore, the uncertain range of -20 to 6 K indicates that the experimental values fell into a range of 20 K below to 6 K above the code value. There is some risk in attaching significant importance to the results of this single example of code uncertainty characterization. However, pending further and more complete analyses, the current results are of preliminary interest, and are provided in that spirit.

In Table 5, the temperature ranges under "Local Temperature" indicate how the code-calculated volume-averaged temperature may differ from any single local measured temperature. The temperature ranges under "Volume-Averaged Temperature" indicate how the code volume-averaged temperature may differ from an experimental volume-averaged temperature calculated as the arithmetic average of all thermocouples in the volume. In both cases, the table provides temperature ranges by method and by transient phase. The preliminary conclusions made with reference to how the code represented the data during each phase of the transient are as follows:

- Both methods show the code tended to be conservative, nonconservative, and best estimate during the blowdown, heatup, and reflood phases, respectively. For this discussion, "conservative" means the code-calculated value was higher than the experimental value, and "best estimate" indicates the code-calculated value tended to represent a mean experimental value.

- The uncertainty in the code-calculated values tended to increase as the code simulation progressed through each phase of the transient. Note that the increasing uncertainty is strongly influenced by timing differences, as opposed to outright inaccuracies in calculating the temperature. The separation of the timing contribution from the temperature level effects remains to be resolved in the methodology. Such a separation is considered to be of more interest to the code developer than to the regulator.

- Both methods show that the more fair comparison of volume average temperature in the code, to volume average temperature in the experiment, is less uncertain than determining how a local temperature might vary from the code-calculated volume average temperature.

Table 5. Comparison of uncertainty ranges between variance and time series methods

Phase	Uncertainty Range: From Variance (K)	Uncertainty Range: From Time Series Maximum/Minimum Points (K)
Blowdown		
Local temperature	-20 to 6	-31 to 40
Volume-averaged temperature	-19 to 5	-20 to 29
Core heatup		
Local temperature	-2 to 81	-20 to 107
Volume-averaged temperature	27 to 52	31 to 54
Core reflood		
Local temperature	-256 to 290	-412 to 464
Volume-averaged temperature	-118 to 152	-122 to 175

6. RUN TIME STATISTICS

The run time statistics for the assessment calculation are presented in Table 6. The central processor unit (CPU) time divided by the real time (RT), or transient time, is a common indicator of how fast a calculation is running, i.e., a CPU/RT ratio of 2 indicates that the calculation compared to the actual transient required twice as much time to run as the actual transient. The (RT)(1.E3)/(DT) ratio is the average transient time step expressed in milliseconds. Several other ratios are also presented in Table 6.

It is important to consider many factors when evaluating the efficiency of a code using run-time statistics. Some of the more important factors are the size of the model (number of one-dimensional and three-dimensional components, number of heat transfer surfaces, etc.), the type and durations of events being simulated, the type of computer system the assessment calculations are performed on, and the user-specified maximum time step

It is evident by comparing the CPU/RT ratio, that there were complications in the GLR Test (Browns Ferry Plant) that significantly slowed down the simulation. The time step was reduced to eliminate convergence problems occurring during the scram. The CPU/RT ratio also indicates that the simulations which did not include a significant period of high velocity blowdown ran significantly faster. The natural circulation simulations ran faster because the system events were occurring more slowly.

The material Courant limit is usually a primary factor in restricting the time step size. During most transients in which two-phase break flow occurs, the Courant limit at the break will significantly reduce the time step size. When the Courant limit is not occurring at the break, it is usually occurring at the jet pump discharge or in the core region.

Currently, there is no way to mitigate the Courant limit problem except to increase the size of the cells where the Courant limit is restricting the time step size. Nevertheless, TRAC does allow the time step size to be maximized when the Courant limit reduces it from the maximum time step, specified on input. TRAC will attempt to set the time step to 95% of the calculated Courant limit time step.

Table 6. TRAC-BD1/MOD1 code assessment run-time statistics (all calculations were performed on a CDC 176 computer)

Facility	Type of Test	Test Number	Real Time in s (RT)	Number of Cells 1-D	Number of Cells 3-D	Total Cells (C)	Number of Heat Transfer Surfaces	Number of Time Steps (DT)	CPU Time in s (CPU)	(CPU)/(RT)	(RT)(1.E3)/(DT)	(CPU)(10)/(RT)(C)	(CPU)(1.E6)/(RT)(C)(DT)	(CPU)(1.E3)/(C)(DT)	Notes
Marviken	Blowdown/ containment	BLOWDOWN 18	136.5	31	0	31	42	26520	6100	44.7	5.1	14.4	54.4	7.4	a
MARK I	Blowdown/ containment	SEQUENCE 483	2200	7	0	7	14	44950	1095	0.5	48.9	0.1	1.6	3.5	a
FIX-II	Medium break	3025													
		Blind	90	86	30	116	136	9173	3948	33.9	9.8	2.9	31.8	2.9	b
		Posttest	76	86	30	116	136	8308	2721	35.8	9.1	3.1	37.2	2.8	b
FIST	Power transient	6PMC2	800	85	58	143	200	32838	19175	24.4	24.0	1.7	5.1	4.1	c
	Natural circulation	6PNC1-2b	110	74	58	132	159	2325	1226	11.1	47.0	0.8	36.3	4.0	d
		6PNC1-4	300	74	58	132	159	6895	3381	11.3	44.0	0.9	12.4	3.7	d
		6PNC1-6	100	74	58	132	159	2645	1350	13.5	37.0	1.0	38.9	3.9	d
SSTF	Large break reflood	EA3.1													
		Base case	25	56	40	96	102	3947	2079	83.0	6.3	8.7	219.5	5.5	e
		Sensitivity	50	56	40	96	102	8286	4612	92.0	6.0	9.6	115.9	5.8	e
Browns Ferry	One pump trip test	PT	20	117	40	157	254	2046	1919	96.0	9.8	6.1	298.7	6.0	f
	Two pump trip test	TPT	20	117	40	157	254	2046	1963	98.2	9.8	6.3	305.6	6.1	f
	Feedwater turbine trip	FTT	40	117	40	157	254	4046	3430	85.8	9.9	5.5	135.0	5.4	f
	Generator load rejection	GLR	14.3	117	40	157	254	4816	4773	333.8	3.0	21.3	441.4	6.3	g

Table 6. (continued)

Facility	Type of Test	Test Number	Real Time in s (RT)	Number of Cells 1-D	Number of Cells 3-D	Total Cells (C)	Number of Heat Transfer Surfaces	Number of Time Steps (DT)	CPU Time in s (CPU)	(CPU)/(RT)	(RT)(1.E3)/(DT)	(CPU)(10)/(RT)(C)	(CPU)(1.E6)/(RT)(C)(DT)	(CPU)(1.E3)/(C)(DT)	Notes
ROSA-III	Small break	RUN 912	425	104	27	131	186	28546	14922	35.0	14.8	2.7	9.3	3.9	h

a. Courant limited at the break in the primary loop.

b. The maximum time step was set at 0.010 s, which could have been increased by a factor of two or three during the first 60 s. After 60 s, the calculation was Courant limited by the high velocities at the break. Then the time step stayed between 0.005 and 0.010 s.

c. The maximum user-specified time step was 0.025 s until 600 s, and 0.035 s after 600 s. Figure 24 shows the time step size and Courant limit during the calculation. The Courant limit was approximately 0.025 s until the core power was terminated at 325 s. After this time, the Courant limit increased dramatically.

d. The Courant limiting cell was in the core region and reduced the time step to about 0.15 s. The time steps during the calculation were about 95% of the Courant limit. To ensure this would be the case, the maximum time step was set at 0.100 s. Figure 25 is a typical plot of the time step size and Courant limit.

e. The time step was Courant limited during most of the calculation. The minimum time steps were occurring in the volumes with the two-phase break flow. The time step was also significantly reduced whenever water packing occurred. During most of the calculation, the time step was between 0.005 and 0.010 s.

f. This calculation had time steps close to 0.010 s, and the Courant limit stayed close to 0.013 s. The volume containing the jet pump discharge was limiting the time step during most of the calculation.

g. The time step was reduced to 0.003 s to eliminate convergence problems occurring during the scram.

h. The maximum time step set was 0.025 s, as recommended in the user guide. The simulation ran at about 95% of the calculated Courant limit. The VESSEL and CHAN components were modified since the previous calculation. These modifications allowed the model to run faster (use less CPU seconds per calculational second).

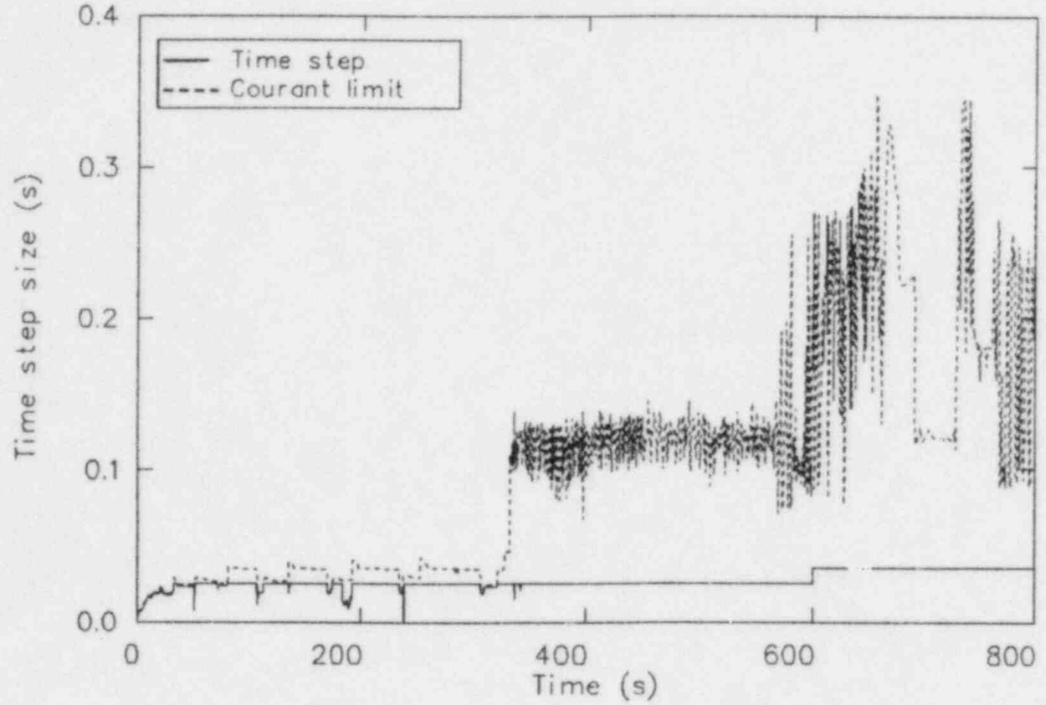

Figure 24. Calculated time step size and Courant limit for power transient Test 6PMC2 (FIST).

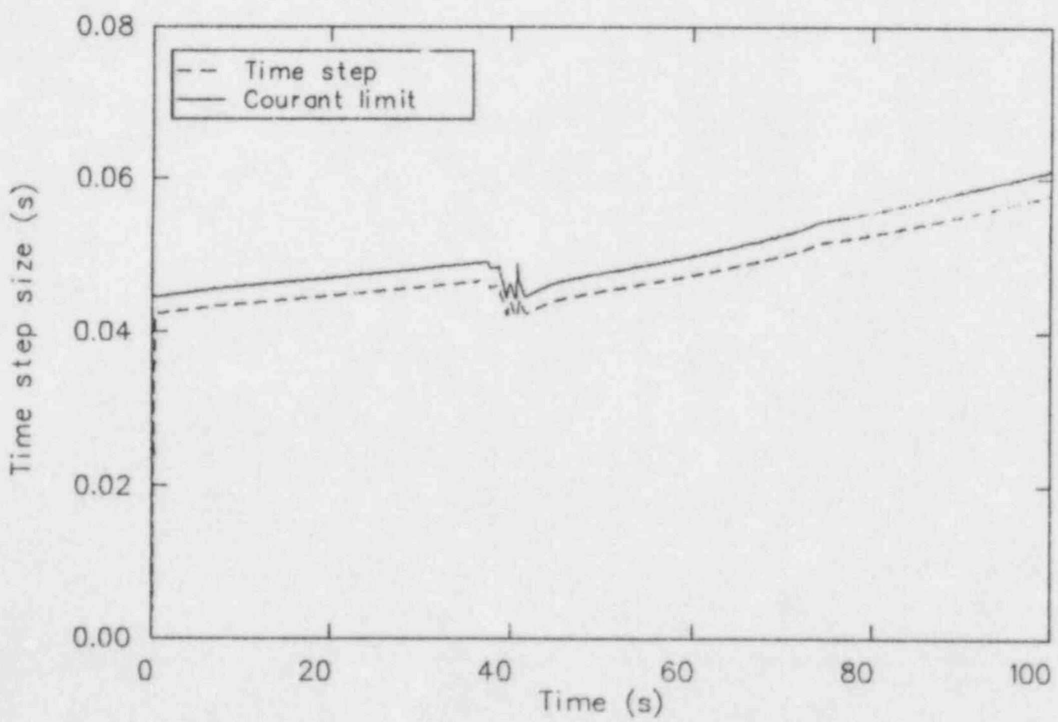

Figure 25. Calculated time step and Courant limit for natural circulation Test 6PNC1-2b (FIST).

7. CONCLUSIONS

The conclusions below all refer to TRAC-BD1/MOD1 unless otherwise stated. The conclusions are organized to correspond with the sections of the previous discussion that support them.

System Pressure Response

1. System pressure was well-predicted by TRAC during the LOCA simulations, natural circulation transients, power transients, containment assessments, and the operational transients. Most discrepancies between measured data and calculated results are attributed to modeling sensitivities rather than code deficiencies.

2. The system pressure response is very sensitive to SRV flow, break mass flow, core thermal response, and flashing/condensation. The calculated response of each parameter had a direct and significant influence on the system pressure during the calculations.

3. Excessive vapor condensation in the jet pump is suspected of causing accelerated depressurization rates. It is recommended that the TRAC interfacial heat transfer model be reviewed with regard to this problem. Finer nodalization is one modeling technique that may reduce the effect of this problem.

4. The containment pressures were well-simulated until the suppression pool reached saturated conditions. After this point, an accurate pressure response is impossible, since TRAC does not include a containment pool boiling model.

Core Thermal Response

5. TRAC demonstrated the capability to accurately simulate the core thermal response provided the mass inventory in the bundle is correct.

6. Improvements in TRAC since the TRAC-BD1/Version 12 assessments, provided for a better simulation of the data. Heater rod response and core fluid behavior were in good agreement with the data.

7. The fine-mesh model in the CHAN component worked well during the reflood portion of the transients. The fine-mesh model provided a good simulation of the rod quench behavior and averted a pressure spike found in earlier versions of the code when large, coarse nodes quenched.

8. Upper plenum drainage trends were difficult to calculate with TRAC because of its nonconserving form of the momentum equation.

Break Flow Response

9. Calculated break flows adequately simulate the data.

10. Uncertainties remain in the use of the TRAC critical flow model. Adjustment of the break discharge coefficient is required to correctly simulate the break mass flow. Break discharge coefficients producing good simulation range from 0.7 to 1.0. Development of more specific user guidelines is considered to be a worthy effort.

11. The subcooled critical flow model is very sensitive to the amount of fluid subcooling. A sensitivity study showed that a 5-K change in break flow subcooling can cause a 15% decrease in break mass flow.

Containment Model

12. TRAC demonstrated the capability to calculate BWR containment behavior during both large break and ATWS transients. Results from transients in which the pool temperature nears saturation should be used with caution.

13. The multi-room capability demonstrated by the Marviken calculation worked very well. The multi-room capability may also have application to the secondary containment of BWRs.

Natural Circulation Assessments

14. The key parameters were well-calculated by TRAC during the natural circulation

assessments. The calculated core flow, downcomer level, bypass flow, and vessel temperature distribution were in adequate agreement with the data.

15. A secondary natural circulation path between the core and bypass was calculated correctly; liquid drained from the upper plenum into the bypass and back into the core.

16. Incorrect pressure distributions caused by the level tracking model created step changes in the natural circulation flow rates. Nevertheless, these fluctuations did not severely affect the transients. The influence of this problem was less with a rapidly changing level than with a slowly changing level.

17. The leak path calculation should be made implicit. In the FIST model, the core bypass flow was modeled using an explicit leak path junction. The explicit nature of the leak flow calculation caused oscillations in the bypass junction. At low flow rate, these oscillations might adversely affect the core flow.

18. The code needs to be assessed for natural circulation with subcooled conditions and varying downcomer levels. The available experimental data base does not include tests with these conditions. It is recommended that future experimental test programs consider including tests with these conditions.

19. The inclusion of a source momentum term for a one-dimensional component connection to the vessel should be investigated. The absence of a momentum term calculation at the jet pump discharge connection to the vessel inhibited the code from correctly calculating the lower plenum flows.

Sensitivity Calculational Results and User Guidelines

20. The use of the level tracking model significantly improved the calculational results of the simulations. It allows a coarser nodalization while improving the calculated response of the systems.

21. Three limitations have been identified in the liquid level tracking model: (a) oscillating levels occur at cell boundaries, (b) conflicting levels may occur near flow restrictions, and (c) pressure disturbances occur as cell boundaries are passed.

22. The TRAC extract feature needs to have an increased number of significant digits taken from the RESTART file. Extracted components used in sensitivity runs were found to adversely affect the transient. TRAC has been updated to correct this problem.

23. The simulations indicate the default coefficients used in the Counter Current Flow Limiting (CCFL) model were insufficiently general for all geometric configurations present in the test facilities. TRAC has been updated to allow additional CCFL coefficients to be entered.

24. The TRAC pipe separator models were inadequate. The explicit nature of the liquid separation calculation caused excessive numerical instability. The pipe separator models have been replaced with models based on a tee component to allow for a more implicit leak path calculation. Independent assessment of these models has not yet been completed.

25. It is recommended that cell-to-cell heat transfer in the theta direction be implemented in TRAC. This would greatly increase the modeling capabilities of the code. The priority of this need is currently under review by the TRAC-BWR Developmental Group.

26. When pressure drop information is unavailable, flow restriction should be modeled using the reference upstream flow areas and their corresponding loss coefficients. If the actual minimum flow areas and their corresponding loss coefficients are modeled, TRAC may overpredict the corresponding pressure drops. This problem should be eliminated when the central differencing scheme is implemented.

27. It is recommended that hydraulic diameters be added to the MBSRV component input to allow the modeling of different sizes of SRVs.

28. Water packing is a significant problem with the code. It is recommended the water packing option be improved.

29. Numerical water hammering oscillations are either handled incorrectly or are ignored by the water packing model. The water packing model needs to be improved to prevent nonphysical water hammering problems between one-dimensional components and the VESSEL component.

30. It is recommended the separator be modeled so that the bottom edge of the separator cell is at the same elevation as the lowest pickoff ring.

31. When modeling a containment compartment, the user must be careful that the spill option is activated or that the compartment has sufficient volume for the effluent or the condensation. If the break outlet compartment fills with liquid from the break, the code calculation will stop.

32. The level tracking model should be used with discretion, and level tracking should probably not be performed in regions where a liquid-vapor froth may be assumed to exist.

33. Where possible, regions of discontinuity of different field variables should be separated by appropriate nodalization measures. For example, a cell in which a steep void gradient exists should not contain a severe flow area change. Rather, the cell should be split into two, with the void discontinuity in one cell and the area reduction in the other.

34. Fluid velocities in regions with steep void gradients should be minimized by use of large flow areas. Where this approach may bias the calculation unreasonably, the computational/grid should be refined and smaller cells should be employed in the regions of steep gradients.

Code Uncertainty

The following conclusions relating to the code's uncertainty must be considered only preliminary, since they are based on a single example of code uncertainty characterization. As preliminary, there is some risk in attaching significant importance as to their meaning.

35. Both methods show the code tended to be conservative, nonconservative, and best estimate during the blowdown, heatup, and reflood phases, respectively. For this discussion, "conservative" means the code-calculated value was higher than the experimental value, and "best estimate" indicates the code-calculated value tended to represent a mean experimental value.

36. The uncertainty in the code-calculated values tended to increase as the code simulation progressed through each phase of the transient. Note that the increasing uncertainty is strongly influenced by timing differences, as opposed to outright inaccuracies in calculating the temperature. The separation of the timing contribution from the temperature level effects remains to be resolved in the methodology.

37. Both methods show that the more fair comparison of volume average temperature in the code, to volume average temperature in the experiment, is less uncertain than determining how a local temperature might vary from the code calculated volume average temperature.

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13. ABSTRACT (200 words or less)

This report summarizes a series of computer simulations sponsored by the United States Nuclear Regulatory Commission (USNRC) performed at the Idaho National Engineering Laboratory (INEL) to continue the advancement of boiling water reactor (BWR) safety research. The simulations were performed to evaluate the analysis capabilities of the Transient Reactor Analysis Code BWR version (TRAC-BD1/MOD1) to calculate operational transients, including anticipated transients without scram (ATWS) and loss-of-coolant accidents (LOCAs). The assessment simulations were performed for a broad range of scenarios, to encompass as many different phenomena as possible. Comparisons are made between the measured and calculated data. Conclusions are made with respect to the calculated system pressure response, thermal response, and break flow response, as well as the capabilities to model containment and natural circulation conditions. Recommendations are made with respect to user guidelines.

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