UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

AUG 30 1982

MEMORANDUM FOR: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

FROM: Robert B. Minogue, Director
Office of Nuclear Regulatory Research

SUBJECT: RESEARCH INFORMATION LETTER - 132 - TRAC-BD1 COMPUTER PROGRAM

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I. INTRODUCTION

This RIL transmits the TRAC-BD1 computer program which is the first Nuclear Regulatory Commission (NRC) sponsored advanced best-estimate computer program developed at the Idaho Engineering Laboratory (INEL) to analyze large and small break Loss Of Coolant Accidents (LOCA's) in Boiling Water Reactors (BWR's). The code was released to the National Energy Software Center (NESC) in February 1981. A four volume manual containing (1) Model Description, (2) Users Guide, (3) Code Structure and Programming Information, and (4) Developmental Assessment was issued in October of 1981 (Reference 1).

Best-estimate codes such as TRAC-BD1 are developed and assessed in response to a number of requests in Reference 2 and to fulfill many needs recognized in References 2, 3, 4, and 5. Some of the licensing needs which were identified in Reference 2, are:

1. Quantification of margin of conservatism in licensing codes which are based on Appendix K.

2. Prediction and understanding of data from experimental facilities.

3. Analyses and prediction of consequences of postulated accidents and transients in full scale Light Water Reactors (LWR's) in order to resolve licensing and safety issues.

In addition to these needs, another licensing need was identified in Reference 6. Reference 6 requests the confirmation of the adequacy or conservatism of the one dimensional licensing models used to predict three dimensional phenomena in BWR's. TRAC-BD1 has the capability to predict three dimensional phenomena and will be used to address this concern in the independent assessment program.

The program for development of the TRAC-BWR code consists of three steps. The first step is the TRAC-BD1 code which has been released to the NESC and which is the subject of this Research Information Letter. The specific version which has been released is Version 8. This code provides a basic and best estimate capability for an analysis of a LOCA in a BWR. Since the release of the code some models for calculation of other Chapter 15 transients and additional user convenience features have been added. This led to the production of a new version of the code. The new version which provides also a limited capability to calculate Anticipated Transient Without Scram (ATWS) for which balance of plant modeling is not required, is Version 12. It is available to NRC and is being used by NRC contractors to perform audit calculations for NRR. An independent assessment is being conducted on this version. A limitation found in this version through the assessment process will generally also be a limitation in Version 8.

The second step is the TRAC-BD1/MOD1 code development. This code will extend the TRAC-BD1 capabilities to include all operational transients for which balance of plant modeling is required, but where spatial kinetics is not needed. It will also contain modeling improvements, the need for which is identified by the independent assessment program.

The third and final step will be the TRAC-BD2 code which will contain modeling of spatial kinetics in the core and all of the improvements, identified by the independent assessment program.

Development of TRAC-BWR codes (TRAC-BD1 and future versions) benefits from the development of TRAC-PWR codes at Los Alamos in that modeling of common modules is coordinated. In addition, TRAC-BWR codes are closely associated with joint NRC-EPRI-GE experimental programs such as BWR Refill Reflood, BWR Blowdown-Emergency Core Cooling System (ECCS) and Full Integral System Test (FIST) facility. There is a very close coordination between GE model development associated with these experimental programs and the code development at INEL.

The TRAC-BD1 code was developed from a Pressurized Water Reactor (PWR) version of the TRAC code (TRAC-P1A Version 22.8, Reference 7) which contained a full non-equilibrium and non-homogeneous two-fluid thermal hydraulics model of two-phase flow for the analysis of large and small break LOCA's. This version has been extensively changed to include (1) components necessary to model BWR plants, (2) thermal hydraulic phenomena encountered in BWR geometries, (3) models to increase the speed of calculations, and (4) user convenience features.

II. RESULTS

A. LOCA Calculations

TRAC-BD1 has been used to calculate large and small break LOCA's in its developmental assessment. These results are summarized in Appendix A.

B. Code Execution

The execution statistics of the TRAC-BD1 code represent an improvement over that of RELAP4/MOD6. Table I compares the running times for TRAC-BD1 and RELAP4/MOD6 codes. TRAC-BD1 had about three times as much modeling detail as RELAP4/MOD6 but used less computer time. Table II summarizes the execution statistics of the TRAC-BD1 code for some of the developmental assessment cases. It should be noted that the detailed nodalization used in Cases 1 and 2 in Table II is not necessary for all LOCA calculations. Hence, some LOCA calculations can be executed faster.

C. Developmental Assessment

The models developed have been subjected to "Developmental Assessment" in order to ascertain that these models have been correctly implemented in the code and the results obtained from the exercise of these models are in reasonable agreement with the test data. The developmental assessment test cases can be divided into three groups: (1) separate effects, (2) separate effects heat transfer tests, and (3) integral system effects tests. Separate effects test cases were chosen to exercise a specific hydraulic or heat transfer model in the code while the integral system effects tests were chosen to exercise the code as a whole. Table III discusses results of the developmental assessment.

III. EVALUATION

The code provides a basic capability to analyze the entire large or small break LOCA sequence, beginning with the blowdown, through heatup, reflood with both top and bottom quenching, and finally with refill of the entire core region in one continuous calculation. The code, Version 8, can be used to analyze large and small break LOCA sequences provided that the control system and balance of plant are appropriately modeled using input boundary conditions. An interim version, Version 12, permits the user to model the control system. As stated, capability to model balance of the plant will be provided in TRAC-BD1/MOD1.

Analysis of the LOCA sequence involves modeling of many physical phenomena and specific BWR components. The code uses a fully nonequilibrium and nonhomogeneous two-fluid thermal hydraulics model of the two-phase flow in all positions of the BWR system. The vessel module permits the modeling of three-dimensional thermal hydraulic phenomena in the upper and lower plenums as well as in the bypass region. The constitutive relations for treatment of mass, energy and momentum interchanges between the phases are based on flow regime maps.

The models important in BWR LOCA analysis using TRAC-BD1 (Version 8) are identified, described and evaluated in Table IV. Table V presents specific BWR components used in LOCA analysis. Additional component models which are necessary for the analysis of transients involving balance of plant are identified in Table VI.

The evaluations in Tables IV and V show that the models used in LOCA analysis are believed to be adequate; however, in some cases further improvements are needed. These evaluations are preliminary and complete evaluations will be performed in the independent assessment program. Pending the independent assessment, some important models are discussed in Appendix B.

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Enclosures: As stated

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References

1. J. W. Spore, et al., "TRAC-BD1: An Advanced Best Estimate Computer Program for Boiling Water Reactor Loss-of-Coolant Accident Analysis," NUREG/CR-2178, EGG-2109, October 1981.

2. E. G. Case, "NRR Requirements for Loss-of-Coolant Accident Analysis Computer Programs (RR-NRR-77-5)," NRC Memo to S. Levine, June 23, 1977.

3. "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," NUREG-0578, July 1979.

4. ACRS, "Review and Evaluation of the NRC Safety Research Program," NUREG-0392 (1977), NUREG-0496 (1978), NUREG-0603 (1979).

5. Report to the American Physical Society by the Study Group on LWR Safety, "Reviews of Modern Physics," v.47 (Supp.1), 1975.

6. Memo from R. J. Mattson to O. E. Bassett, "Audit Analysis of a BWR-3 LOCA Model," May 22, 1981

7. "TRAC-P1A: An Advanced Best-Estimate Computer Program for PWR LOCA Analysis," Los Alamos Scientific Laboratory, LA-7777-MS, NUREG/CR-0665, May 1979.

TABLE I

RELAP4 vs TRAC - BDI PERFORMANCE

BENCHMARK CASE: BWR/6 (218-624)
PLANT - 200% PUMP SUCTION BREAK

MODEL DESCRIPTION and EXECUTION STATISTICS	RELAP4/ MOD6	TRAC-BDI
TOTAL CELLS	34	111
VESSEL CELLS	8	32
HEAT STRUCTURES	45	129
REAL TIME	40s	45s
$\frac{\text{CPU}}{\text{REAL}}$	50.5	37.2
$\frac{\text{CPU}}{\text{REAL CELLS}}$	1.49	0.34
$\frac{\text{CPU}}{\text{REAL CELLS HEAT STRUCTURES}}$	0.033	0.0026

TABLE II

TRAC-BD1 EXECUTION STATISTICS

Case	Comments	Transient Duration (s)	Total No. Cells	No. of VESSEL Cells	CPU Time/ Transient Time	CPU Time Transient Time/ No. Cells
1. BWR6 Large Break Analysis	Extremely fine core noding (6 CHAN with 8 levels). Multi-dimensional representation of annulus and bypass. Executed until CCFL breakdown had occurred and lower plenum filling began.	120	204	64	112	0.55
2. BWR6 Large Break Analysis	Coarser code noding (3 CHAN with 8 levels) than Case 1. One-dimensional representation of annulus and bypass. Executed until CCFL breakdown had occurred and lower plenum filling began.	90	114	32	44.9	0.44
3. BWR6 Small Break	Same noding as Case 2. Executed Until CCFL breakdown had occurred and the lower plenum was full and the hot rods quenched.	400	114	32	28.9	0.25
4. TLTA 6422-3	Executed until rods quenched because of ECCS water.	80	81	26	50.8	0.62

TABLE III

DEVELOPMENTAL ASSESSMENT

Test	Comments
INEL Jet Pump	Comparisons of TRAC-BD1 jet pump model with data is quite good. Comparison of TRAC-BD1 predictions with data is shown in Figure 1. It should be noted that this data is all single phase steady state data.
Edward's Pipe	Short time behavior is predicted well for the Edward's Pipe experiment (see Figure 6); however, the calculation tends to depressurize too rapidly later in the transient (see Figure 3). Improvements in this model are being investigated based on RELAP5 critical flow model improvements.
Semiscale Nozzle S-07-6	The comparison with this data is not good. This comparison indicates that the subcooled blowdown model in TRAC-BD1 is inaccurate.
GE 8 x 8 CCFL Data	Comparisons are quite good. TRAC predicts the saturated CCFL data (see Figure 4). For subcooled CCFL, breakdown is predicted to occur at the same conditions as observed in the tests.
GÖTA Radiation Test 27	Comparison is quite good (see Figure 5).

TABLE III (Continued)

Test	Comments
GöTA Spray only Test 78	General trends are predicted well. The rod heatup rate is consistent with the data. However, the calculated peak clad temperature peaks too early and results in a predicted peak clad temperature of 250K less than measured. Comparisons with previous calculations of this test indicate that the interfacial heat transfer rate is too high which results in a desuperheating of the steam which causes the rods to cool down relative to the data. Improvements in interfacial heat transfer models should improve this behavior.
TLTA Blowdown	The overall trends of the calculation for TLTA Test 6422-3 agree well with the data. Events which were not predicted properly were jet pump uncovery and core dryout.
FLECHT Test 9077	In general, the results are adequate demonstrating fair agreement with the data. Figure 6 gives an indication of TRAC-BD1's strengths and weaknesses in simulating bottom reflood phenomena. The weakness is due to over-estimation of the entrainment during the first half of the transient when the flow is dispersed droplet. As the quench front approaches, this inaccuracy diminishes and the period immediately preceding and following the arrival of the quench front shows the strength of the predictive capability.

TABLE III (Continued)

Test	Comments
GE Level Swell	Shown in Figures 7 and 8 are examples of the ability of the code to accurately track a rising and falling level. Test 8-21-1 contains a perforated plate at the 2.44m elevation. This is what causes the secondary level shown by TRAC-BD1. The coarseness of the data nodes does not allow for demonstration of this second level.
CISE Steady State Void Fraction	The results of simulation of the adiabatic CISE tests are quite good. Sample results are shown in Figure 9. However, these comparisons indicate that improvements in the interfacial friction can still be made.
TLTA 4904	The TLTA core was modeled using TRAC-BD1. Measured core inlet parameters were used to drive the model to observe whether or not TRAC correctly calculated transition from nucleate boiling. The calculation does not agree well with the data when the Biasi local critical heat flux relation is used as received in TRAC-P1A. Figure 10 shows a significant improvement when the newly implemented critical quality boiling length correlation (CISE-GE) is used.
GE 16 Rod DNB Tests	Figure 11 is an example of the excellent agreement obtained in comparing with these tests. Of particular note is the ability to accurately predict not only the nucleate boiling transition but the rewet as well. Both the CISE-GE and the improved Biasi critical quality correlations are available in TRAC-BD1.
BWR6 Full Scale Jet Pump	Comparisons are quite good (see Figure 12). Data is only for steady state single phase flow in normal operating range.

TABLE IV

MODELS IMPORTANT FOR LOCA ANALYSIS

Model	Description	Preliminary Evaluation Pending Independent Assessment
Critical Flow	TRAC-BD1/MOD1 has adopted the RELAP5/MOD1 choking model. The model limits the mixture velocity to the homogeneous equilibrium sound speed. This removes the need to use fine nodalization in the vicinity of breaks and speeds up the calculation by avoiding the restrictive Courant time step limits of fine nodalization.	Developmental assessment of this model has indicated that the two-phase break flow rates are overpredicted. Preliminary results of independent assessment indicate that the two-phase critical flow rates are predicted accurately but that the subcooled critical flow rates are underpredicted.
Countercurrent Flow Limiting (CCFL) and its breakdown	TRAC-BD1 uses a correlation to compute the maximum liquid downflow rate in counter-current flow. The constants in the correlation were determined from tests performed by GE using prototypical BWR hardware. TRAC-BD1 calculates CCFL breakdown using the normal hydrodynamic solution scheme.	Developmental Assessment calculations of a LOCA transient in a BWR/6 show qualitative agreement with data obtained in the Steam Sector Test Facility.

TABLE IV (Continued)

Model	Description	Preliminary Evaluation Pending Independent Assessment
Upper Plenum	TRAC-BD1 has the capability to model phenomena in the upper plenum of a BWR reactor vessel although it does not contain a distinct upper plenum component as such. This capability consists of (a) the normal multi-dimensional hydrodynamic solution scheme in the vessel component, (b) a stratification model to equalize the liquid "levels" in the several cells representing the upper plenum, (c) the CCFL model described previously, and (d) FILL components to represent ECCS sprays located on the outer periphery of the plenum.	TRAC-BD1 has no explicit mechanistic model for the ECCS spray jets in the upper plenum. The spray distribution may play a role in determining the distribution of liquid subcooling in the upper plenum. The distribution of liquid subcooling across the tie-plates of the fuel bundles influences the distribution of CCFL breakdown. Developmental assessment indicates that the present capability should be adequate for LOCA, since the upper plenum phenomena predicted for a BWR/6 DBLOCA are in qualitative agreement with preliminary data from the Steam Sector Facility and the GE 16° Sector Facility. Independent assessment of the upper plenum modeling capability will be performed using the results of tests from multi-dimensional test facilities.
Interfacial Shear Model	TRAC-BD1 has the same interfacial shear model as TRAC-PD2. The interfacial shear determines the system void distribution which in turn affects the system mass distribution.	The shear package in TRAC-BD1 (i.e., the TRAC-PD2 shear package) was found to be adequate for LOCA during the developmental assessment of TRAC-BD1/MOD1. A code version intended for transients as well as LOCA, will implement the Andersen-Ishii interfacial shear model.

TABLE IV (Continued)

Model	Description	Preliminary Evaluation Pending Independent Assessment
Entrainment	TRAC-BD1 has the same entrainment correlation as TRAC-PD2. The effect of entrainment on interfacial heat transfer is different because of the updates to the heat transfer package developed during the developmental assessment of TRAC-BD1.	The amount of liquid entrainment is important in computing clad temperatures during reflood. Independent assessment of TRAC-PD2 indicates that the correlation overpredicts the entrainment rate. TRAC-BD1/MOD1 will implement the new Ishii entrainment correlation. Independent assessment will determine the adequacy of this correlation for entrainment in fuel bundles.
Quench Front Propagation	TRAC-BD1 contains the same quenching model as TRAC-P1A. The quench front position is calculated using a quench front velocity correlation. The model was extended to compute quench front propagation on the inner surface of the fuel bundle cannister wall.	The moving fine mesh quench front model developed for TRAC-PD2 is being incorporated into TRAC-BD1/MOD1. Independent assessment of TRAC-PD2 has indicated that the moving fine mesh technique improves the prediction of the peak clad temperatures.
Level Tracking Within a Node	TRAC-BD1 has no capability to track a liquid or "froth" level within a computational cell. Tracking of a liquid or "froth" level is accomplished by using nodalization, the finer the nodalization, the better the definition of the liquid or "froth" level through finer resolution of the void distribution.	The mesh refinement needed to adequately track the liquid level may be very expensive. An explicit level tracking model would provide the same capability using a coarse nodalization at reduced computational cost. A level tracking model will be investigated and incorporated into TRAC-BD2.

TABLE IV (Continued)

Model	Description	Preliminary Evaluation Pending Independent Assessment
Heat Transfer	The heat transfer packages, both wall-fluid and interfacial, are based on TRAC-PD2. However, several new models including a radiation heat transfer model and a critical-quality boiling transition correlation were implemented.	Developmental assessment of radiation heat transfer model and critical-quality boiling transition correlation produce excellent agreement with available data. Several modifications were developed during the developmental assessment of TRAC-BD1 to improve the agreement with DSF-P1 spray cooling tests. The heat transfer packages as well as the radiation heat transfer model will be assessed during independent assessment.
Fuel Gap Conductance	TRAC-BD1 has the same constant gap conductance model as TRAC-P1A.	The gap conductance affects the fuel temperatures and hence the peak clad temperature for LOCA and other severe transients. Independent assessment will provide feedback on the adequacy of the present model using NRU tests. In addition, a simplified version of the FRAP-T5 fuel model will be implemented in TRAC-BD1/MOD1.

TABLE IV (Continued)

Model	Description	Preliminary Evaluation Pending Independent Assessment
Heat Conduction Calculations	TRAC-BD1 has the same heat conduction solution routines as TRAC-PD2. The conduction models are separated according geometric shape of the solids in which they are computing the temperature distribution. They model heat conduction in cylindrical structures (pipes, control rod guide tubes and vessel and downcomer walls), slabs (support plates) and fuel rods. Both distributed and lumped parameter solutions are available. The basic TRAC package has been improved to allow calculation of heat transfer to fluids on both the inside and outside of cylindrical structures.	Independent assessment using NRU data will provide guidance as to the adequacy of the heat conduction models used for fuel rods. In addition, user definable material properties will be added to allow the use of materials not presently included in TRAC-BD1.
Decay Heat	TRAC-BD1 incorporates the decay heat model based on the ANSI/ANS-5.1 decay heat standard. A number of options have been incorporated: (1) specification of fissile nuclides U^{235}, U^{238}, U^{239}, (2) specification of plant operating history, (3) incorporation of heavy element decay heat, and (4) inclusion of effect of fission product neutron capture.	TRAC-BD1 contains the most recent decay heat standard.

TABLE V

MODELING OF BWR COMPONENTS

Model	Description	Preliminary Evaluation Pending Independent Assessment
Fuel Channel Model	TRAC-BD1 contains a new component model called the CHAN to describe a BWR fuel bundle. The model includes the following capabilities: (1) multiple fuel rods or fuel rod groups (2) radiation heat transfer between fuel rod groups, fluid and the fuel bundle cannister wall, (3) falling film and bottom flood quench fronts on all rod groups and the interior channel wall, (4) a leakage path between channel and core bypass region, and (5) heat transfer between exterior of channel wall and core bypass fluid.	The critical quality boiling transition model, the radiation heat transfer model, and the quench front models were evaluated during developmental assessment of TRAC-BD1. All but the quench front model were judged satisfactory. Further comparisons will be made during independent assessment. In addition, the fine moving mesh quench front model, similar to that in TRAC-PD2, is being corporated into TRAC-BD1/MOD1. A more generalized heat transfer coupling between fuel bundles and the core bypass is being developed for TRAC-BD1/MOD1 to allow more than one CHAN component to within a single VESSEL component cell.
Separator-Dryer	TRAC-BD1 contains a perfect separator-dryer with no carryunder.	Developmental assessment of TRAC-BD1 has shown that this model is adequate for large break LOCA transients. However, carryover and carryunder affect the enthalpy distribution in the reactor vessel for long transients such as small break LOCA's and operational transients. An improved separator-dryer model is being developed by GE.
Jet Pump	TRAC-BD1 provides a jet pump component based on the TRAC TEE component.	Developmental assessment of this model has been shown in excellent agreement with INEL 1/6 scale single phase jet pump data.

TABLE VI

MODELS NEEDED TO CALCULATE SEVERE TRANSIENTS

Model	Status	Plans for Improvement
Reactor power, Reactivity Feedback and Multi-D Neutron Kinetics	Reactor power is calculated using point kinetics.	The reactivity feedback model will be included in TRAC-BD1/MOD1.
Boron Tracking Model	There is no boron tracking model.	The boron tracking model will be included in TRAC-BD1/MOD1.
Subcooled Boiling	Void fraction calculations are performed the same way as in TRAC-PD2.	Lahey's mechanistic subcooled boiling model will be implemented in the TRAC-BD1/MOD1 code.
Control Systems	There is no capability to model control systems. The code has capability to model only the trips.	Capability to model control systems will be included in TRAC-BD1/MOD1.
Balance of Plant (Turbine Condenser, Feedwater Heater)	There is no capability to model balance of plant.	The capability to model balance of plant will be included in TRAC-BD1/MOD1.
Containment	There is no containment model.	A simple containment model will be developed and implemented in future versions.

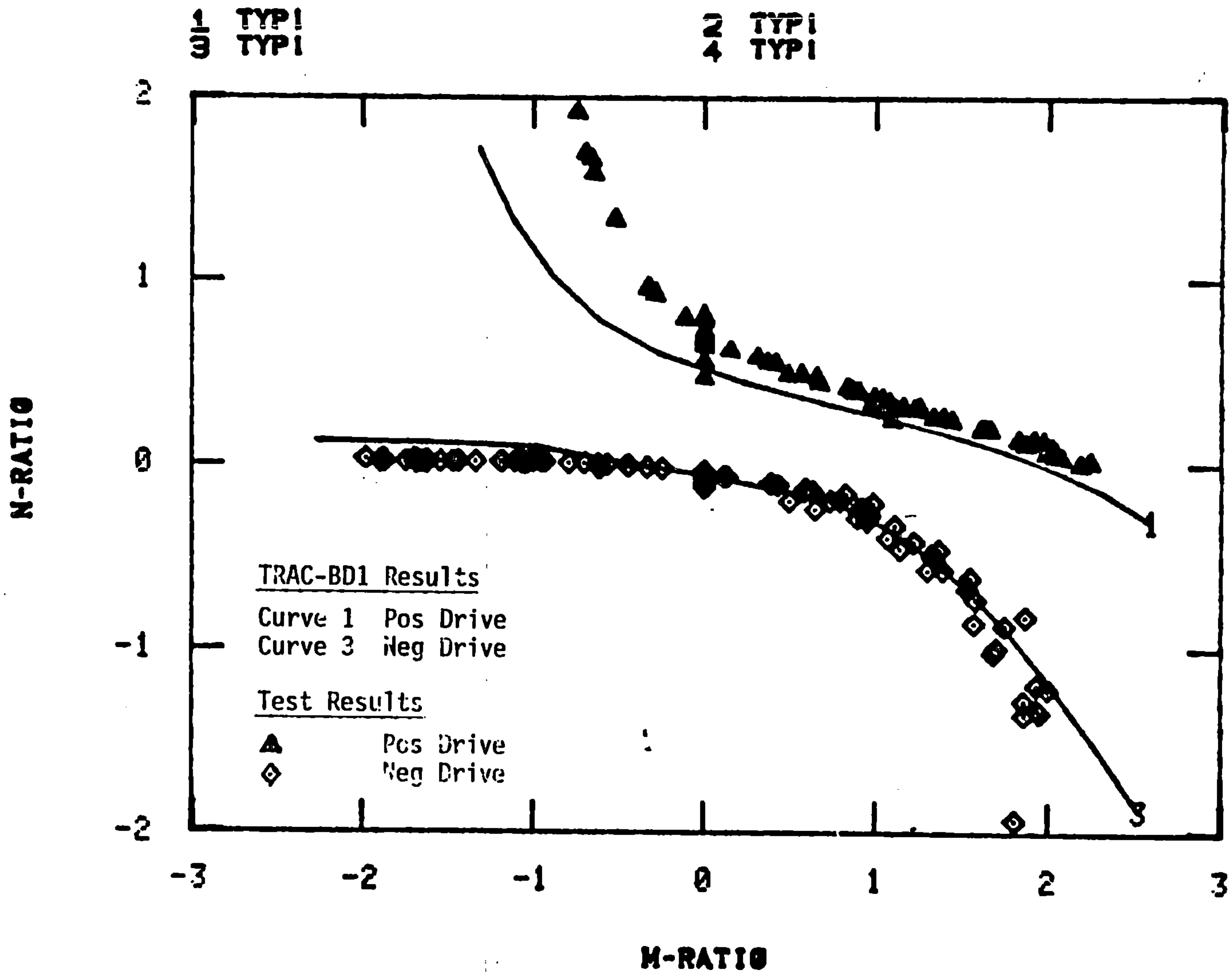

Figure 1. Comparison of TRAC-BD1 Predictions with INEL 1/6 Scale Jet Pump Data.

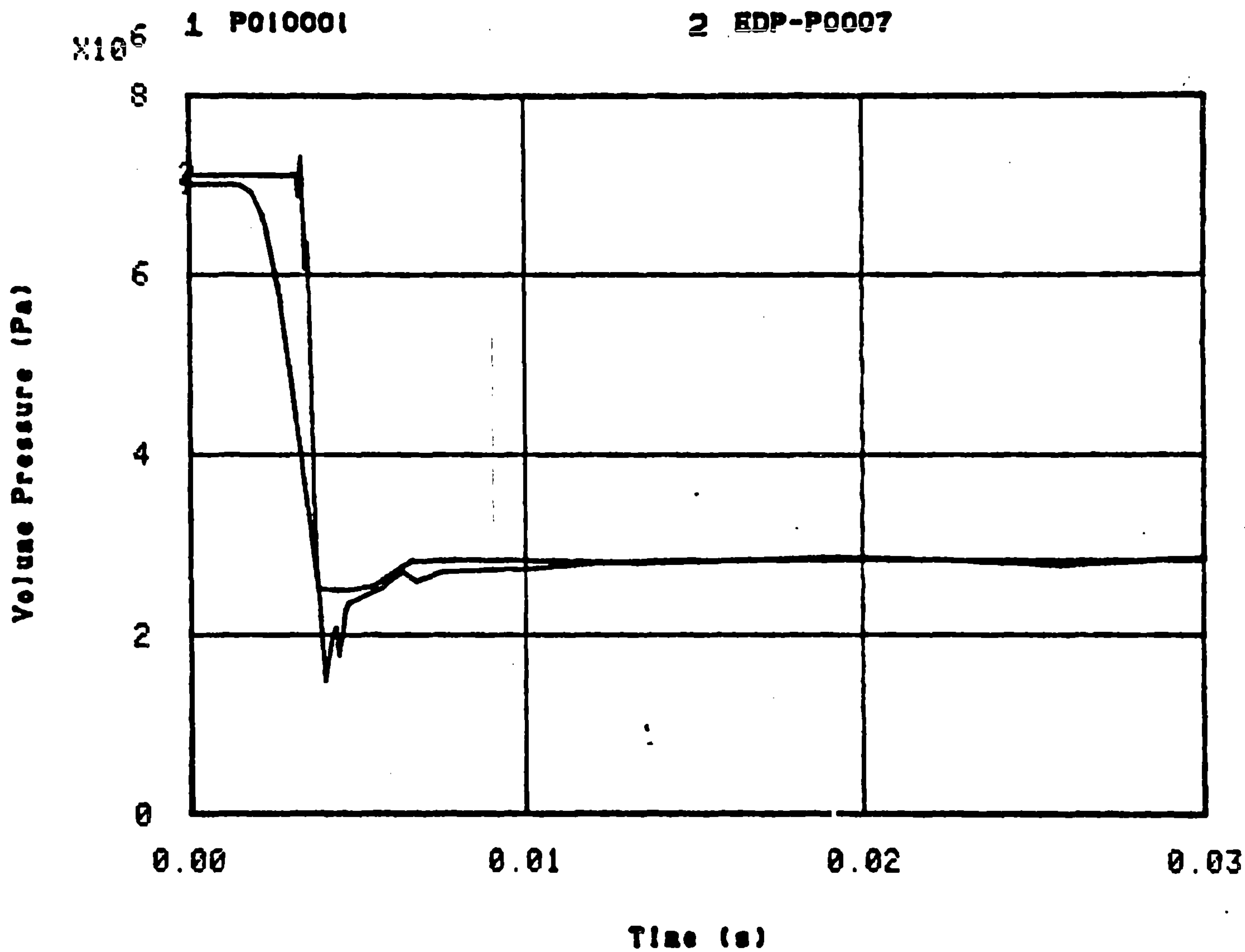

Figure 2. Comparison of TRAC-BD1 Predictions with Edwards Pipe Blowdown Data at the Closed End of the Pipe.

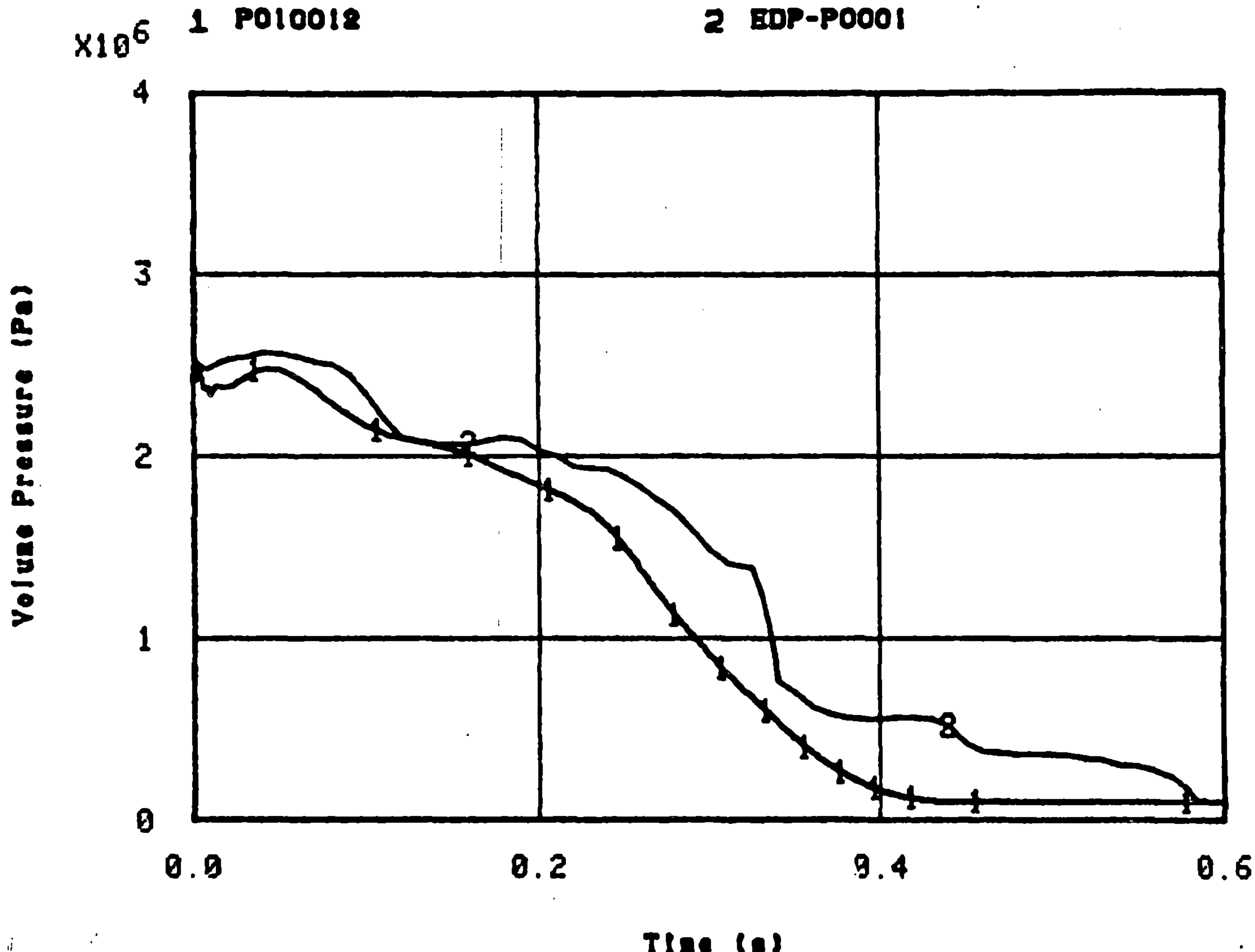

Figure 3. Comparison of TRAC-BD1 Predictions with Edwards Pipe Blowdown Data at the Open End of the Pipe

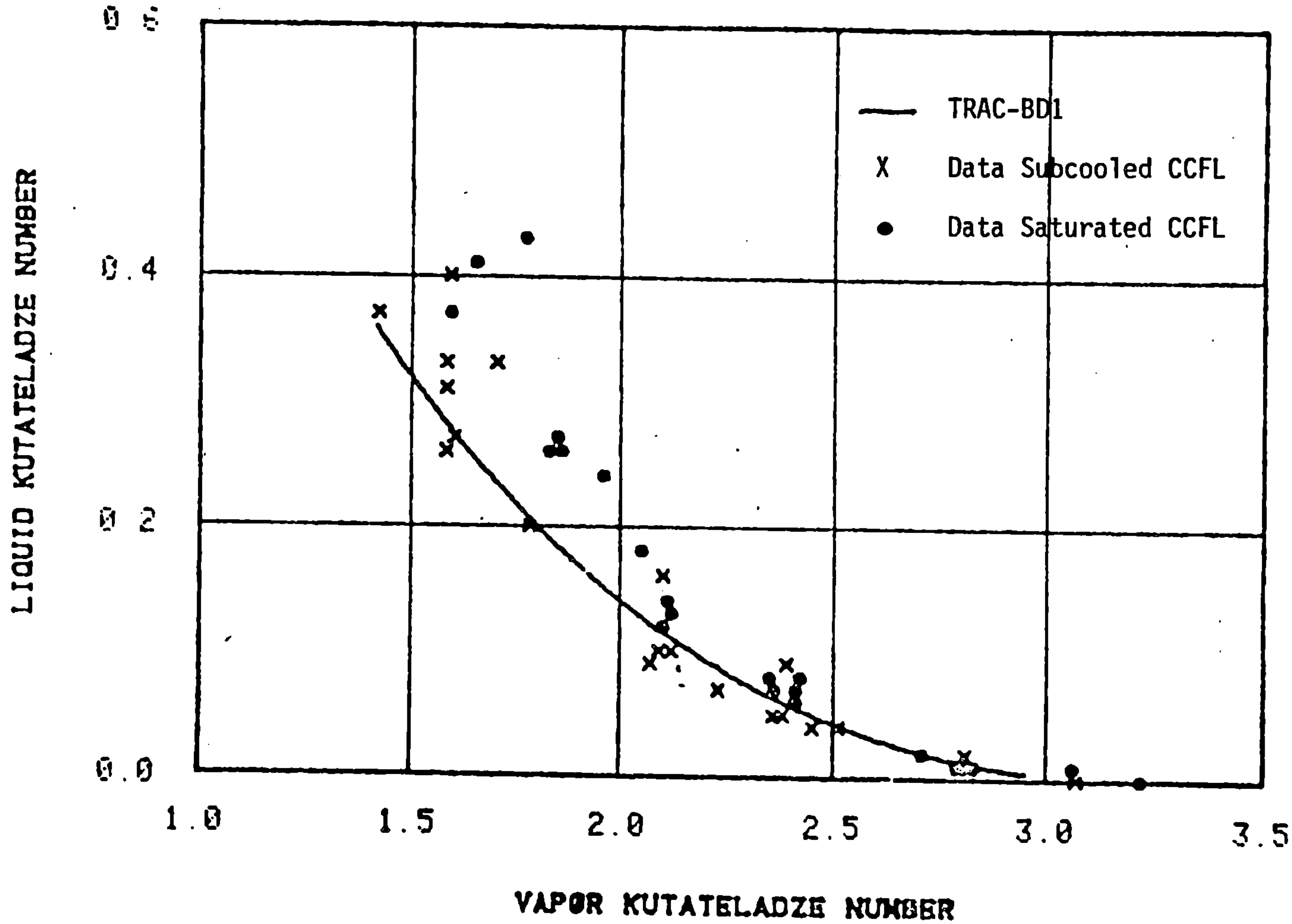

Figure 4. Data obtained from NEDG-NUREG-23549.

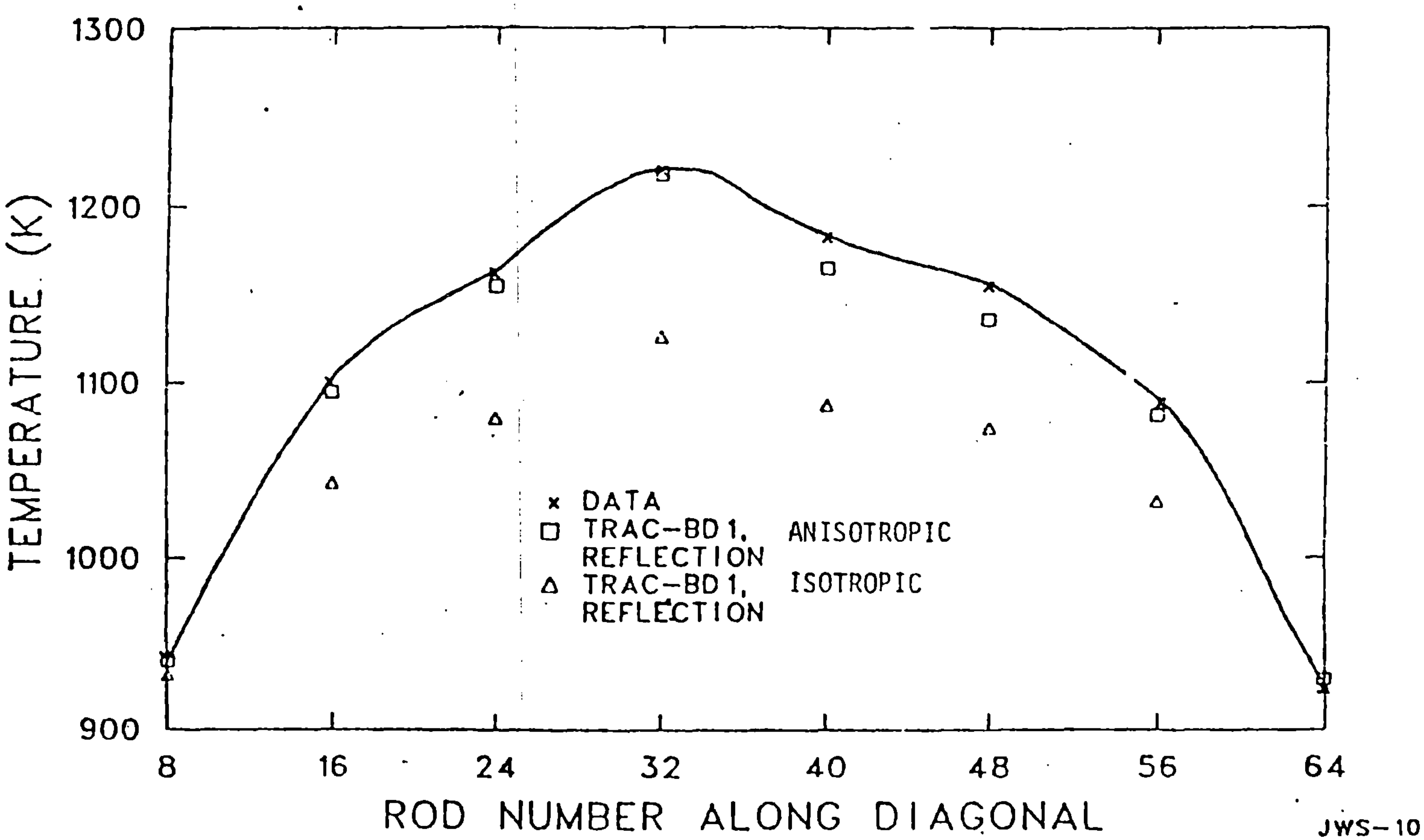

Figure 5. TRAC-BD1 results Göta radiation test data.
Data obtained from C. Nilssan, etal,
Experimental Investigation of Cooling by Top Spray and Bottom Flooding of a Simulated 64-Rod Bundle for a BWR, Part 2., Studsvik IRL-78/59, Norhav S-046, June 30, 1978.

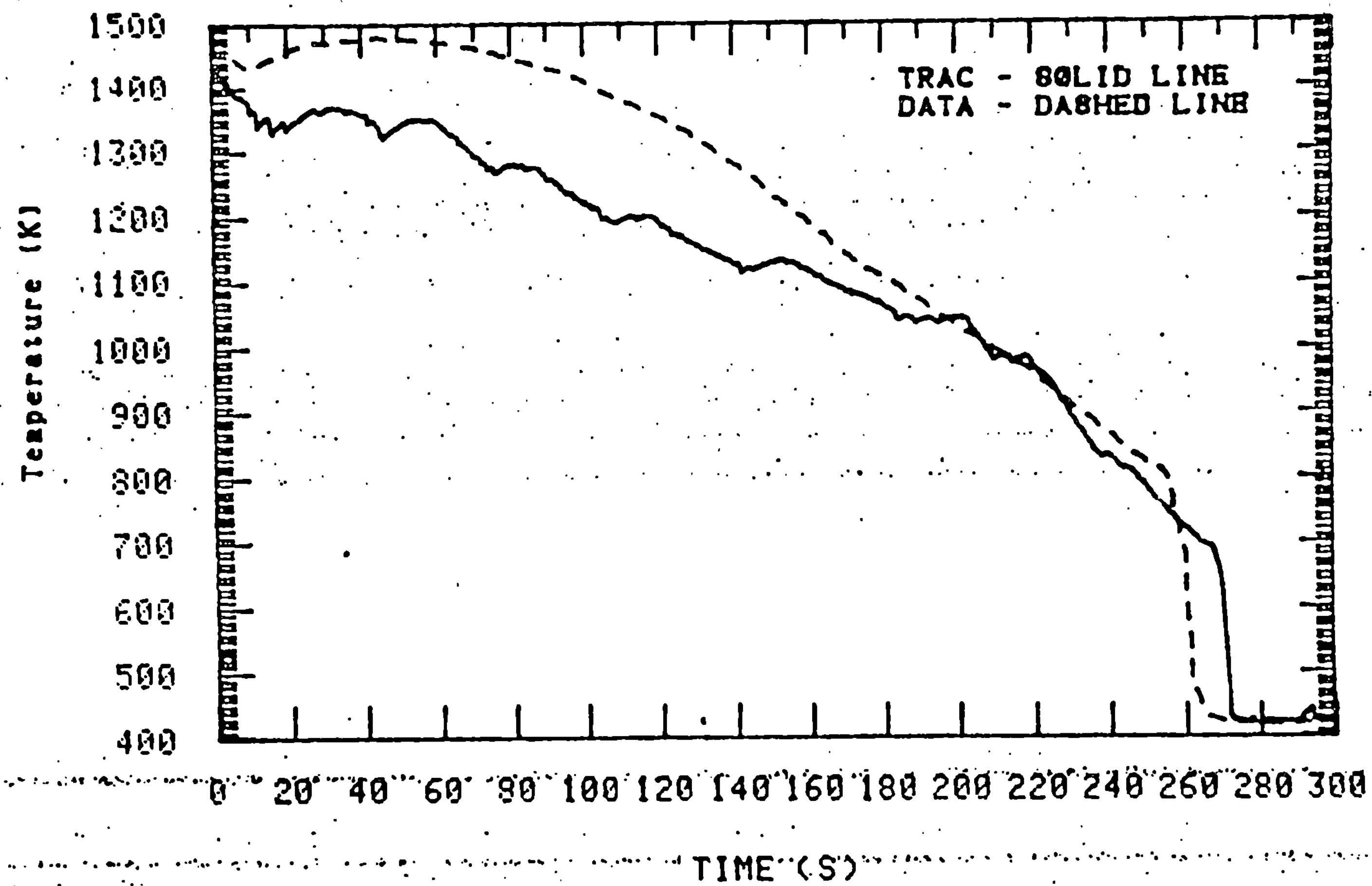

FIGURE 6. FLECHT TEST 9077
ROD TEMPERATURES
6 FT ELEVATION (MID-PLANE)

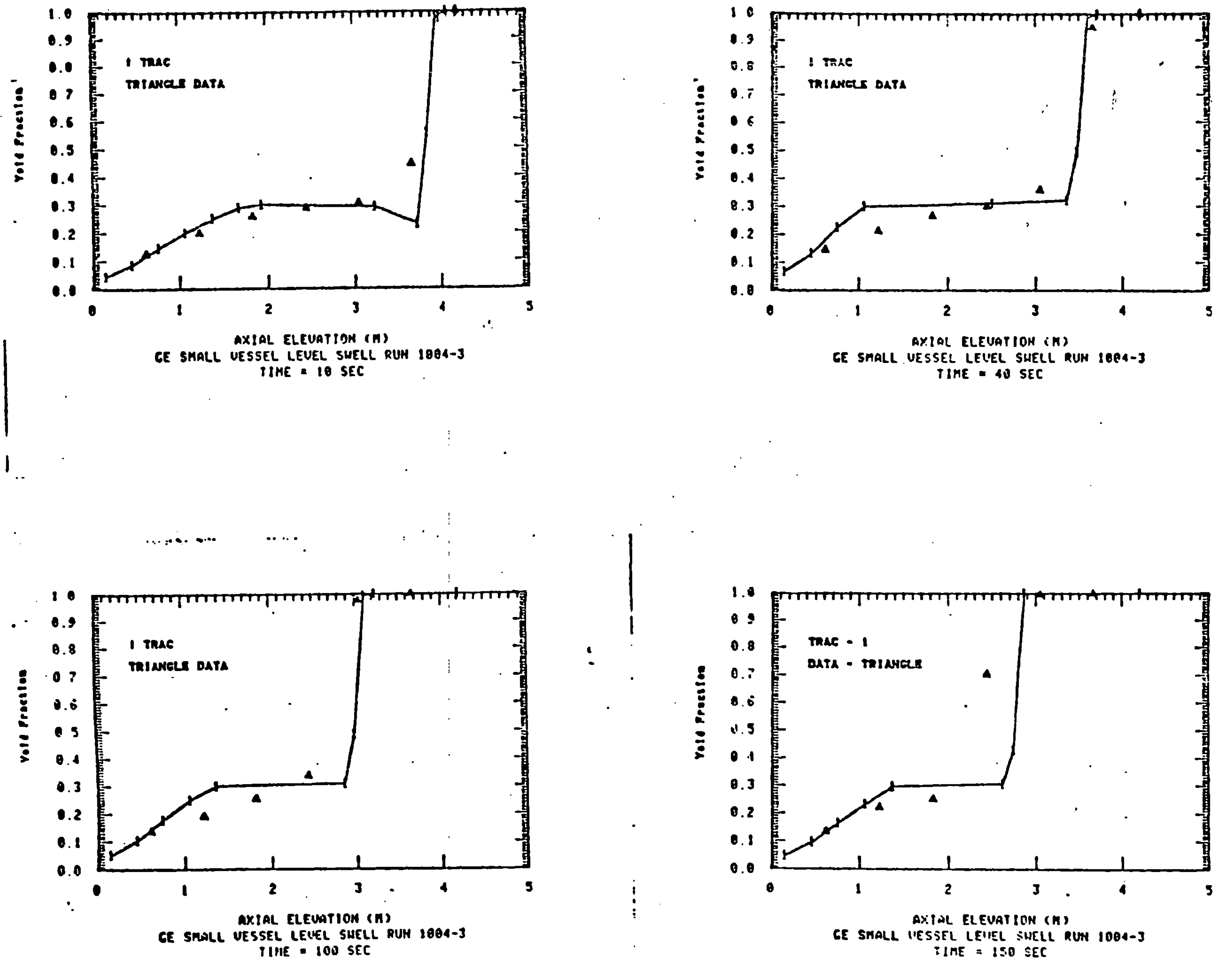

Figure 7. Comparison of TRAC-BD1 Predictions with GE Small Vessel Level Swell Data (1004-3).

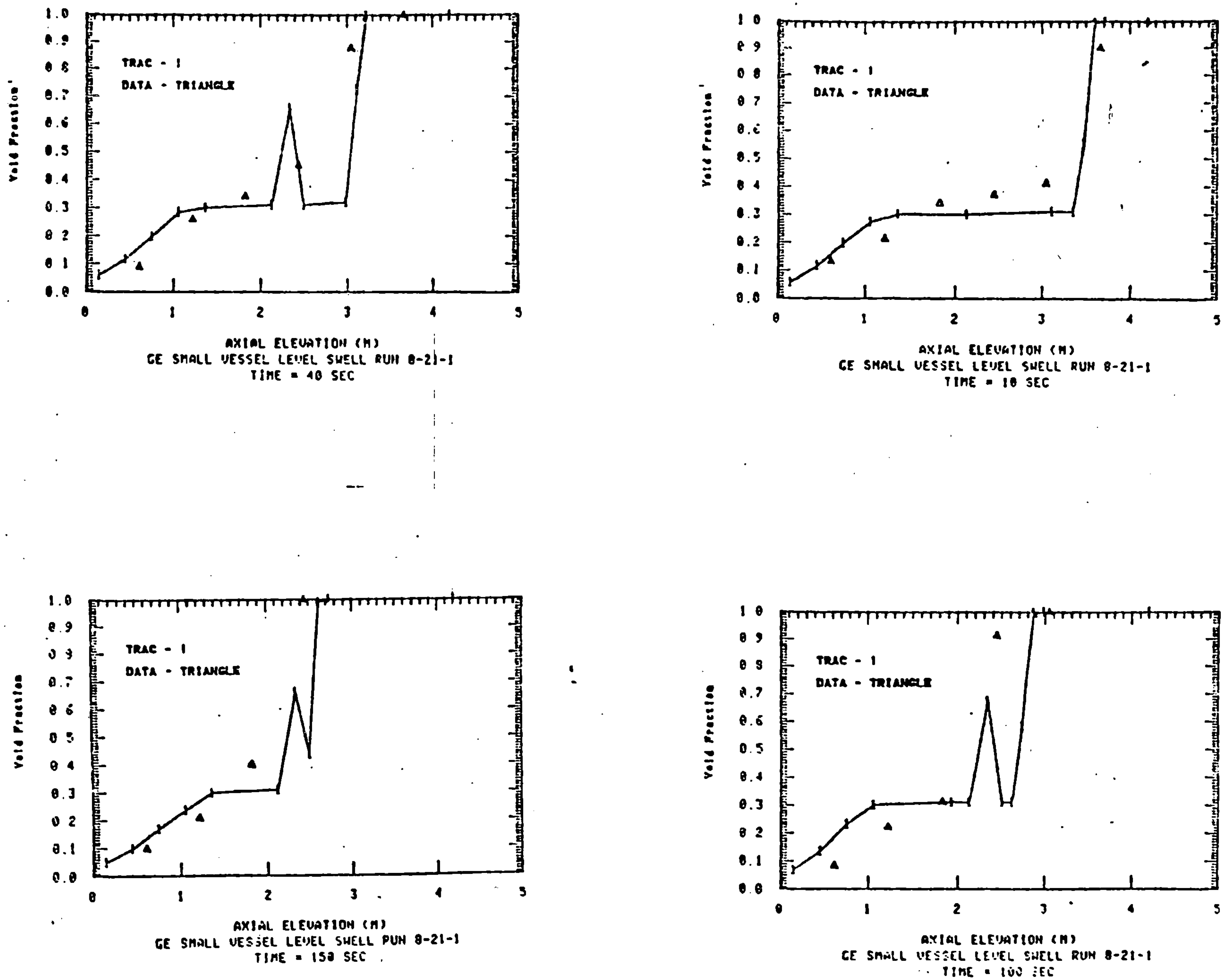

Figure 8. Comparison of TRAC-BD1 Predictions with GE Small Vessel Level Swell Data (8-21-1).

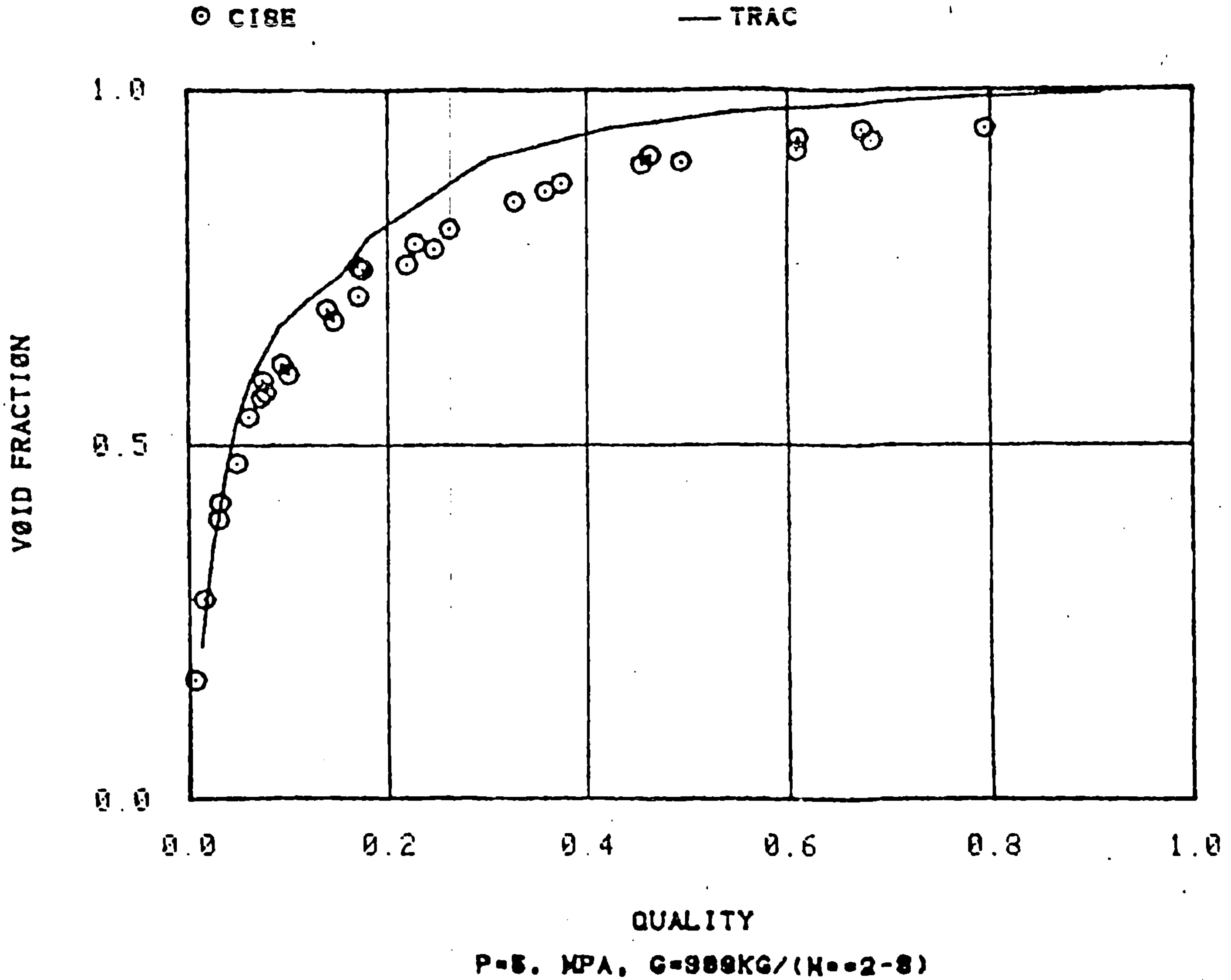

Figure 9. TRAC-BD1 Steady State Void Fraction Predictions Compared with Steady State Void Fraction Data Obtained from CISE-R-291.

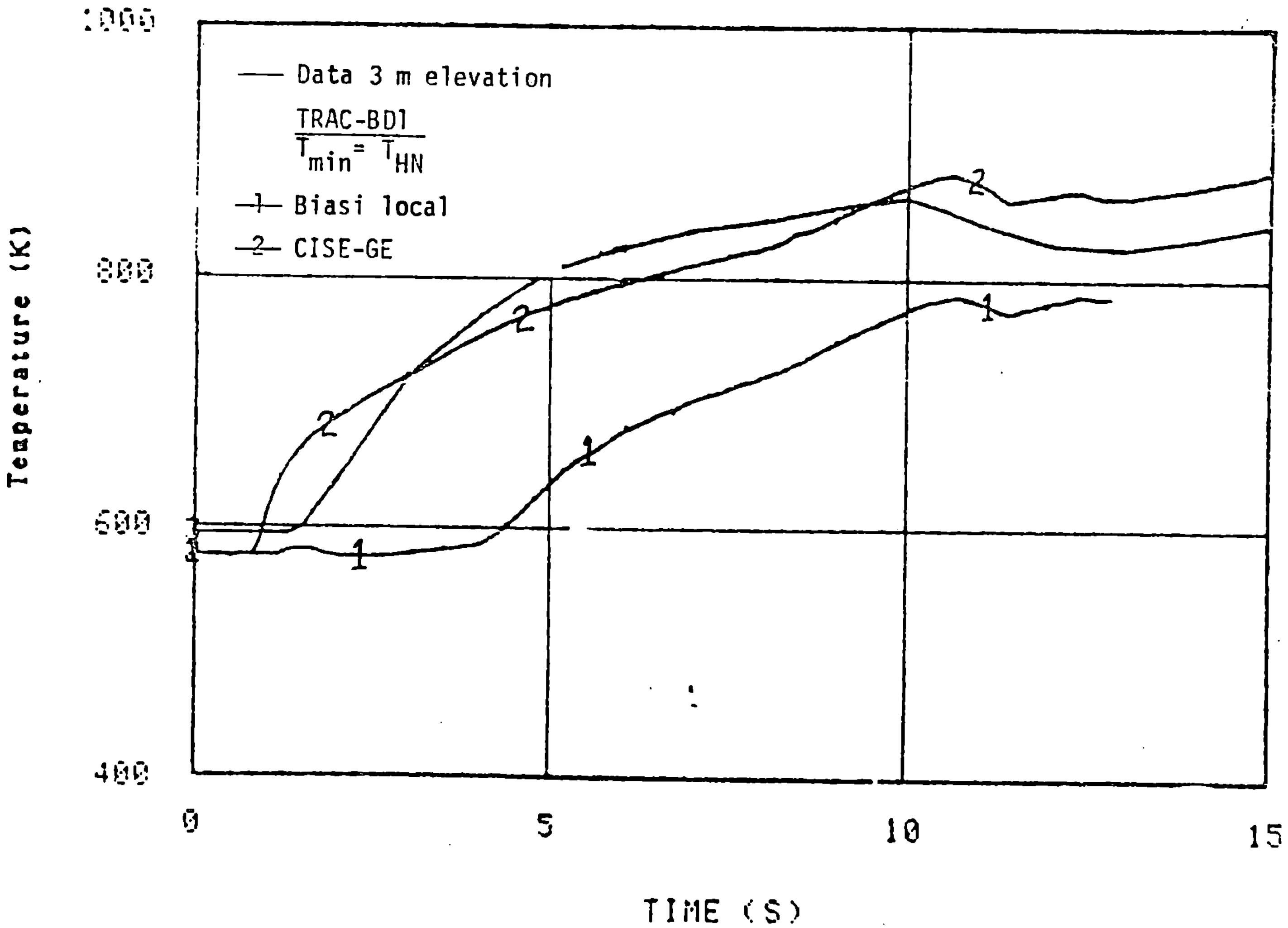

Figure 10. Data Obtained from TLTA Data, CISE-GE and Biasi Local.

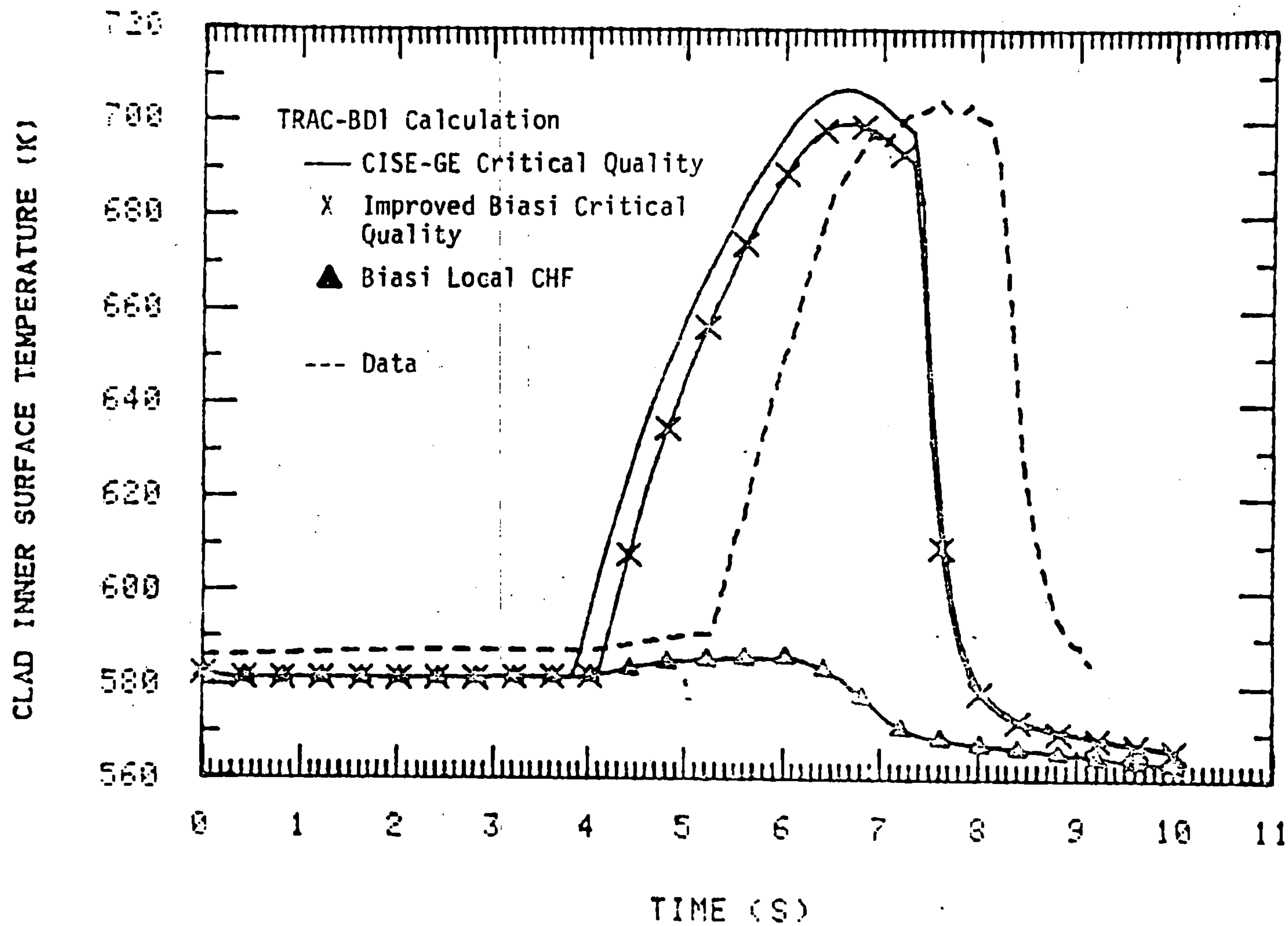

Figure 11. Comparison of TRAC-BD1 with GE Boiling Transition Data Obtained From Elevation 2 Rod Temperature Transient T_{CHF} Run 110.

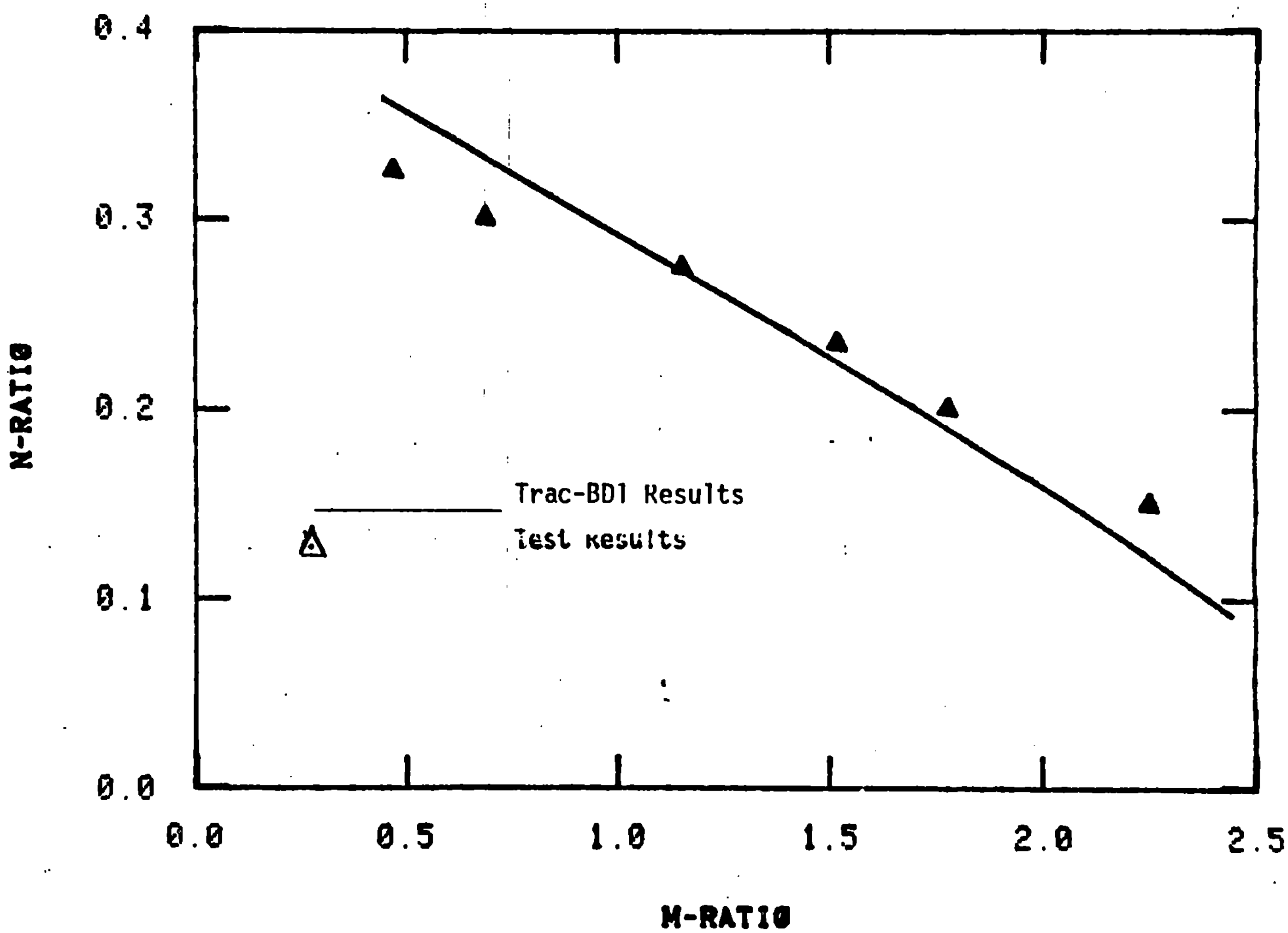

Figure 12. Full Scale BWR Jet Pump-Positive Drive Flow.

APPENDIX A - LOCA CALCULATIONS

Two LOCA analyses, one for a large break and the other for a small break (10%) break, have been performed for a generic BWR/6 (218) plant as part of developmental assessment. The breaks were at the suction side of the recirculation loop. The plant was modeled with 22 TRAC-BD1 components and 111 fluid cells.

The code qualitatively predicted many important phenomena in sequence occurring during a LOCA which were observed in tests performed in SSTF and TLTA facilities and reported in References A-1 and A-2. Both the data and calculations showed occurrence of counter current flow limitation (CCFL) phenomenon and its breakdown at the bypass outlet, bundle side entry orifice (SEO) and bundle upper tie plate (UTP). The CCFL phenomenon and its breakdown are very important in determining the bundle heatup, maximum peak clad temperatures and the final quench of the bundle. The breakdown of CCFL depends on local properties such as subcooling in the upper plenum.

Figures A-1 through A-4 present the results of the small break LOCA calculations. The sequence of events can be followed from the figures. Core spray induces CCFL breakdown at 223s because of local subcooling of the fluid in the upper plenum above the peripheral bundles. Local subcooling and CCFL breakdown were observed in tests performed in the SSTF facility (Reference A-2) and in 18° Toshiba test facility (Reference A-3). The CCFL breakdown leads to a sudden quench of the rods at 223s in peripheral bundles. Slight changes in subcooling delay the complete quench which occurs at 241s. Accurate prediction of local subcooling in the upper plenum is necessary in order to accurately predict the time for the rod quench. This requires multidimensional calculational capability for the code.

Quenching of the hot bundle is influenced by water leaving through side entry orifices of peripheral bundles and Low Pressure Core Injection (LPCI) at 245s in the bypass. The mixture of LPCI water and water leaving peripheral bundles enters the hot bundle through the side orifice and quenches the bundle from the bottom up. The LPCI water is injected into the outer ring at Level 4. Modeling of the movement of the LPCI water from this location to the core center where the hot bundle is located, and its mixing with peripheral bundle water again requires multidimensional capability for the code. The bypass gap between the fuel canisters is small and exerts considerable resistance to flow. The radial void fraction gradient in the bypass influences the rate at which the hot bundle receives the LPCI and peripheral bundle water. This mechanism leads to quenching of hot rod and terminates the transient.

In summary, there are complex interactions between the LPCI, upper plenum behavior and CCFL breakdown. In this particular calculation the rod temperatures in peripheral bundles are readily influenced by the upper plenum behavior while the temperatures in the hot bundle are sensitive to LPCI injection. Tests in Reference A-3 showed that LPCI would promote CCFL breakdown by increasing the region of subcooling in the upper plenum. The accurate modeling of the phenomena occurring in the upper plenum and bypass and their interactions require very accurate multidimensional capability for the code. It is planned to assess and improve present models in our ongoing research program using the Toshiba, SSTF and other test results which may be available to us through our international cooperative agreements.

References:

A-1. "BWR Blowdown Emergency Core Cooling Eighteenth Quarterly Progress Report," April 1-June 30, 1980, NUREG/CR-1154, April 1981.

A-2. "BWR Refill-Reflood Program Task 4.4 - CCFL/Refill System Effects Tests (30° Sector) Experimental Task Plan," NUREG/CR-1846, July 1981.

A-3. "CCFL and CCFL Breakdown Phenomena at BWR Refill-Reflood Phase," H. Nagasaka, M. Katoh, I. Onodesa, paper presented at Japan Atomic Energy Society Meeting, October 1981.

BWR6 Small Break, TRAC-BD1
Steam Dome Pressure

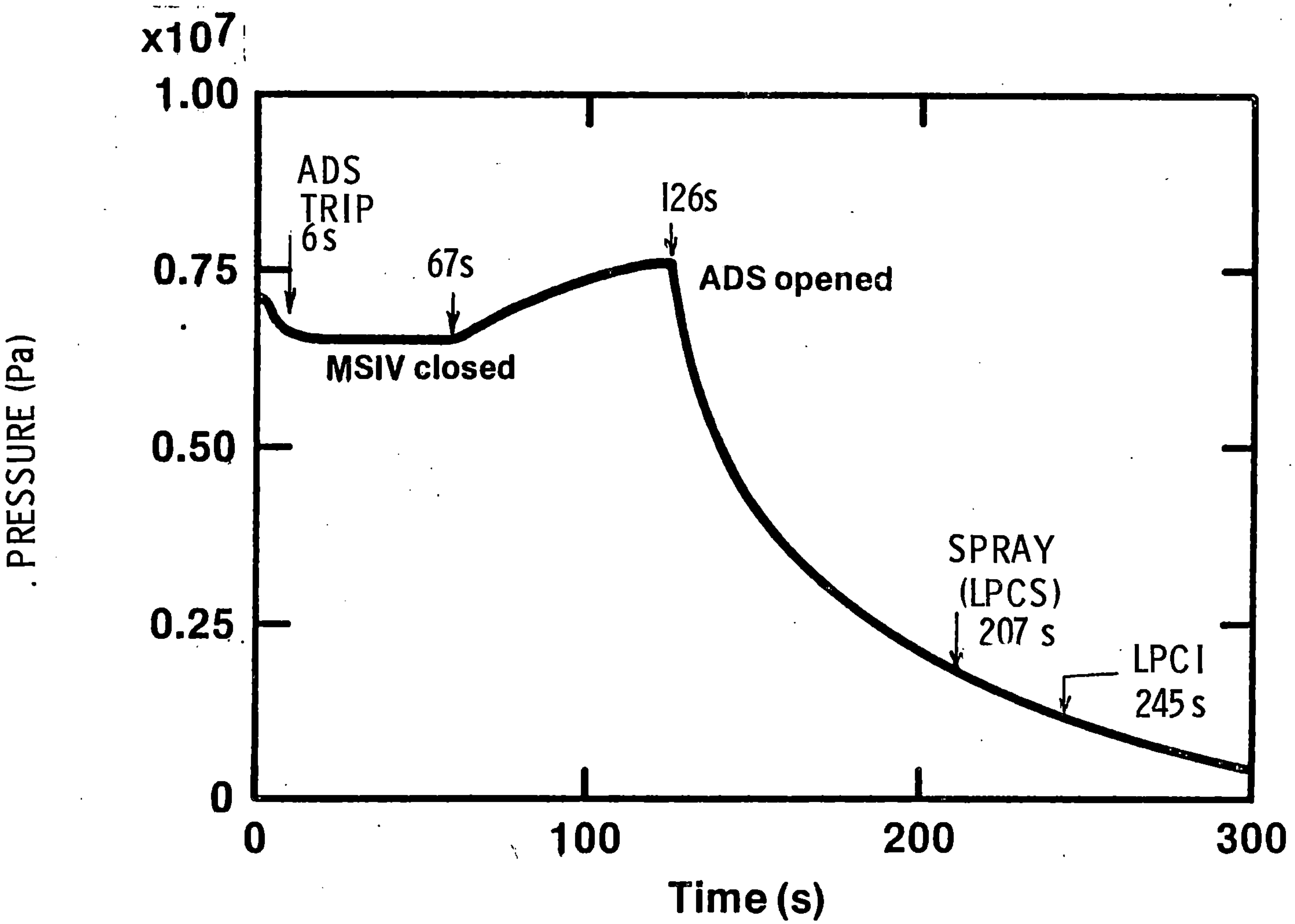

FIGURE A-1. SMALL BREAK CALCULATIONS

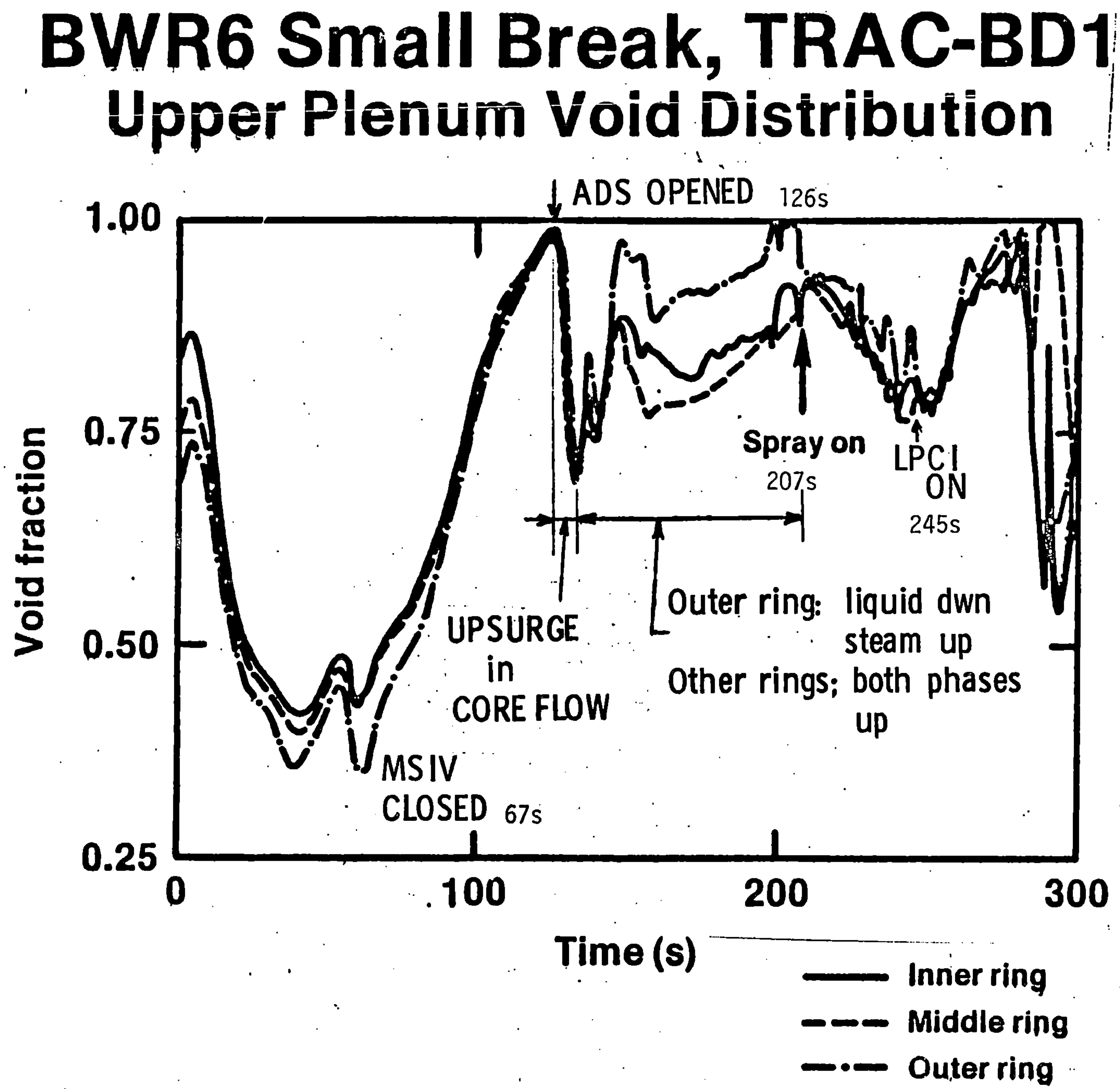

FIGURE A-2. SMALL BREAK CALCULATIONS

BWR6 Small Break, TRAC-BD1
Upper Plenum Liquid Temperature Ring 1 2 3 and Sat

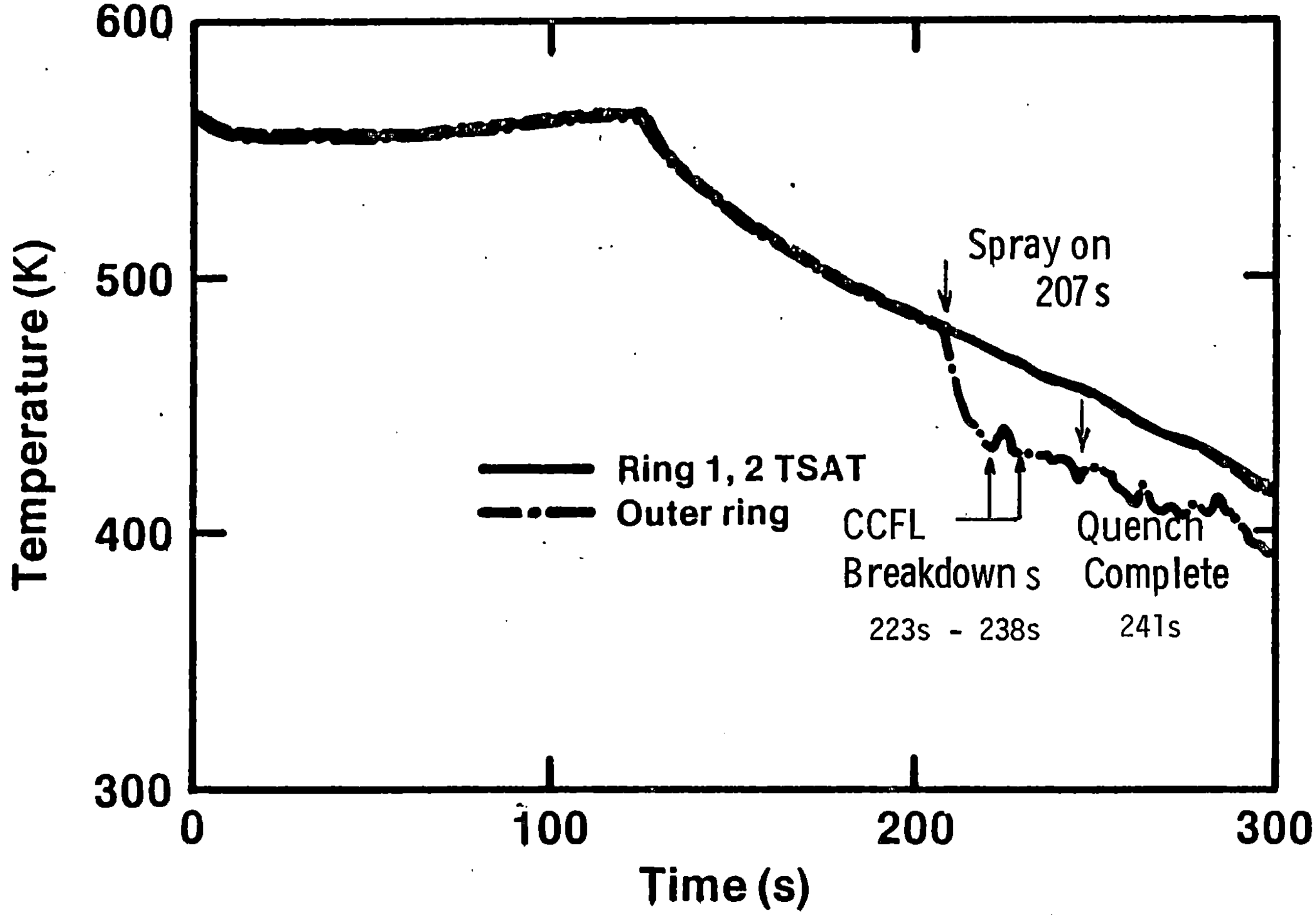

FIGURE A-3. SMALL BREAK CALCULATIONS

BWR6 Small Break, TRAC-BD1
Midplane Rod Temperatures

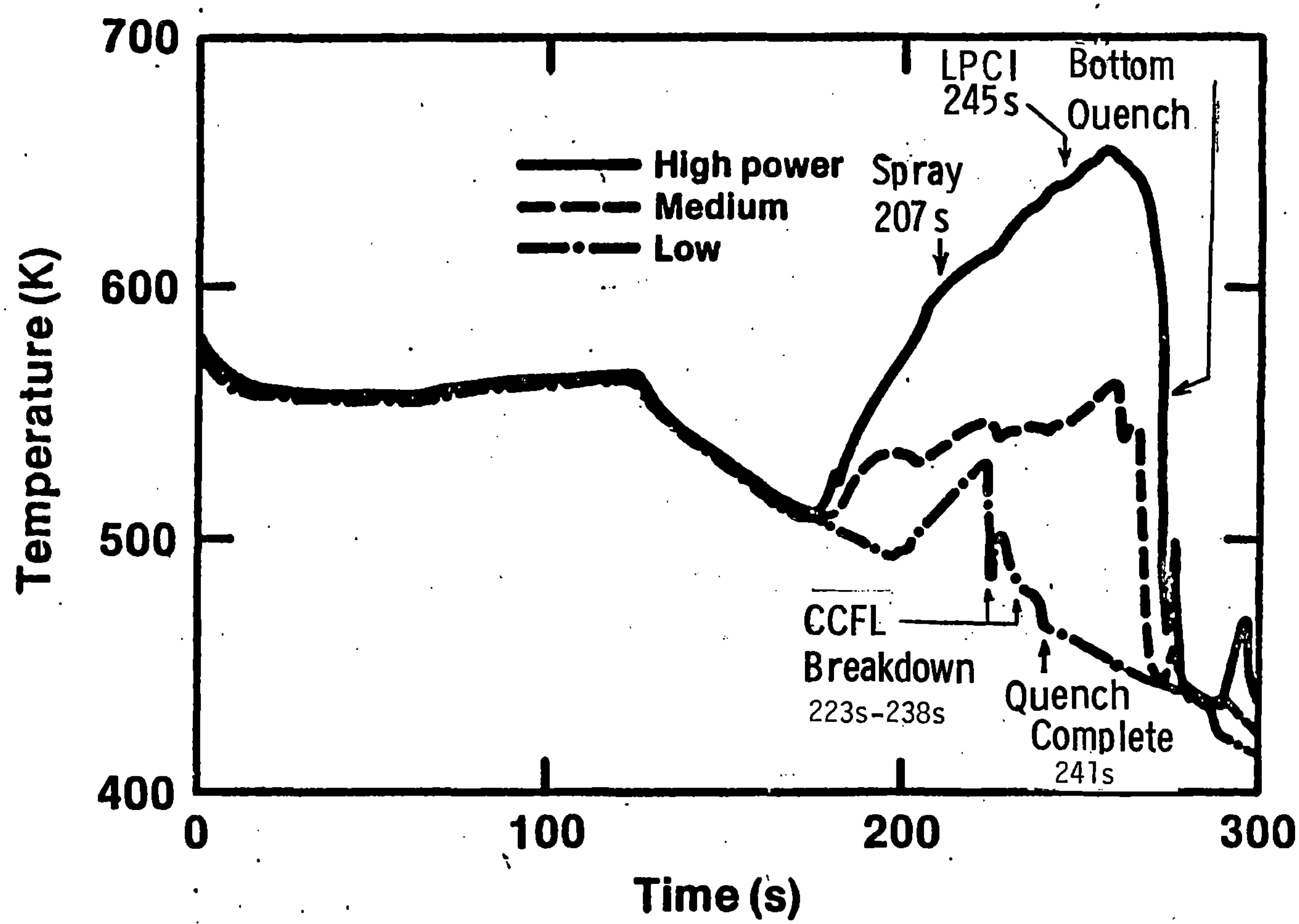

FIGURE A-4. SMALL BREAK CALCULATIONS

APPENDIX B - IMPORTANT MODELING AND KEY FEATURES OF TRAC-BD1

Upper Plenum Modeling: In TRAC-BD1 the upper plenum is modeled as an integral part of the vessel. The vessel contains upper and lower plenums, the downcomer and bypass regions which can be modeled multidimensionally and the BWR fuel canisters which are modeled using one dimensional flow components, called CHANs. The radial and tangential momentum equations contain radial pressure gradient terms accounting for hydrostatic and stratification effects. This permits a realistic determination of the ECC mixture distribution both in the upper plenum and bypass.

TRAC-BD1 does not contain an explicit spray distribution or a jet stream model. In any node, the code calculates average properties of the liquid and steam phases. Tests in Reference B-1 showed that if the spray nozzle is located above the two phase mixture level, the temperature of the subcooled spray liquid rises to that of the saturated steam in approximately 4≈5 nozzle diameter downstream of the injection point. The condensation of steam on liquid surfaces increases the temperature of the jet stream. Since the main contribution of the upper plenum modeling is the calculation of local subcooling over the upper plenum tie plates, for this particular case, calculation of one set of liquid properties over the entire node may be sufficient. Hence, the TRAC model may be satisfactory.

However, if the spray nozzle is located below the two phase liquid level, the jet is covered with saturated water. There is turbulent mixing as well as heat transfer via liquid entrainment between subcooled and saturated regions within a node. The present model in the TRAC-BD1 code will only calculate an average liquid temperature in the node.

The significance of this deficiency as well as the capability of the code to calculate horizontal stratification will be assessed through the assessment program using experimental data obtained from the SSTF and 18° sector (Toshiba) test facilities. A cursory review of calculations performed by General Electric with TRAC-BD1 of their 16° Upper Plenum Facility reveals no serious consequences for this deficiency. Examination of calculations of large and small break LOCA reveals that the predicted upper plenum appear qualitatively correct and are consistent with expected behavior.

CCFL Phenomenon: TRAC-BD1 uses a correlation to compute the maximum liquid downflow in countercurrent flow. The constants in the correlation were chosen such that the data in Reference 12 are predicted conservatively. The constant K for upper tieplates is 3.2 - see Eq. (34) of Reference 1. This permits less liquid downflow than the best estimate value would indicate. The best estimate value for K varies from 3.55 to 4.40 depending on the mode of steam injection - see Reference B-2. In order to assess the best estimate behavior of the CCFL phenomenon, Version 12 provides a capability for the user to choose a coefficient for the upper plenum tieplate. For safety analyses the conservative value of K=3.2 can be chosen. However, this does not provide a large margin. Sensitivity studies which are being performed at INEL, indicate only 15K conservatism in the hot bundle peak clad temperature for a large break LOCA, Reference B-3.

Steady State Initialization: The basic TRAC code calculates the steady state as a transient with given boundary conditions. Convergence to pressure and velocity distribution is relatively fast; however, it is, in general, slow for void and temperature distribution. In general, in order to obtain a satisfactory steady state, the transient may have to be calculated long enough for fluid particles to circulate several times (about 200s) through the system. However, it is recommended that the user make several short runs in order to ascertain the correct flow splits between the bundles and the bypass and the correct flow leakages from bundles to the bypass. In many reactors 97% of the jet flow goes through the bundle inlet orifices and 3% goes through the bypass at the inlet elevation.

Above the lower core plate fluid leaks from bundle holes to the bypass increasing the bypass flow up to 10% of the total flow. Making several short runs with different loss coefficients to obtain correct flow splits may be necessary.

In addition, the steady state obtained may not be the desired steady state in terms of steam dome pressure, core flow rate and downcomer water level. These parameters are not boundary conditions. The user may have to rerun several cases changing the boundary conditions until desired values of these parameters are obtained. All these calculations for a BWR may take a long time in a computer. In Version 12, some improvements in steady state initialization have been made. The control system can be used to automatically make adjustments to give the desired values for the boundary conditions. The feedwater flow rate, recirculation pump speed and turbine control valve position are modified to obtain the desired values of downcomer level, core flow rate and steam dowm pressure respectively. The boundary conditions are changed as the operator would change them in the reactor. Hence, using Version 12, the user substantially decreases the number of the steady state runs to obtain the desired parameters.

Heat Transfer: The TRAC-P1A, Version 22.8 (Reference 7), heat transfer package was used as the starting point to develop a BWR heat transfer package. The package was expanded to include radiation heat transfer and critical quality-boiling length correlation for nucleate boiling transition in the released version. Developmental assessment using the DSF-P1 test indicated that predicitons of post-CHF heat transfer were not accurate. In order to improve the accuracy, some updates on the logic of selecting film boiling correlations were made and reported in Reference 1. The updates were not included in the released tape; however, later versions of the code contain these updates.

Version 12 includes, also, Lahey's mechanistic subcooled boiling model, Reference B-4, and an improved condensation model based on Unal correlation for bubble collapse rates, Reference B-5. Developmental Assessment performed using the Bennet Post CHF tests, Reference B-6, shows that although there are some improvements in predicting post CHF heat transfer in high pressure and high mass flux regions, the code has computational difficulties at low pressure and low mass flux regions, Reference B-7. Further improvements will be made in future versions.

User Convenience Features: Table B-1 presents the status of the user convenience features. This table should be of particular interest to the contractors and NRR staff planning to perform audit calculation.

Configuration Control Systems: TRAC-BD1 configuration control system is unprecedented and represents a conscientious effort to enhance the quality of the code. This configuration control system permits maintenance of a high degree of software reliability. Each change to the code, as simple as a one-line error correction or as complicated as a major component model addition, is given a unique identifier (a Program Change Label or PCL), which appears not only on the FORTRAN source cards but also on the documentation that was supplied before the incorporation of a proposed change. The minimum documentation identifies the engineer and/or programmer making the change as well as a short description of the reasons for the change. This minimum documentation also appears in the FORTRAN source of the code. For large or extensive changes to the code, design and/or completion reports are written which describe the theoretical basis for the change to the code, identify test cases and expected results to qualify the change, and give programming information such as the names of new input or output variables and a list of the subroutines affected by the change. The configuration control system permits a fully documented working version of TRAC-BD1 to be available at all times. Hence, other contractors or code users can more easily understand what has been done, can more easily incorporate additional changes, and can more easily track down any errors discovered.

References: Appendix B.

B-1. "A Basic Study of Mixture Type Steam Condensers," Y. Takahashi, M. Masuda, K. Aikawa and M. Tahasa, Mitsubishi Heavy Industries Technical Report, Vol. 9, No. 1, pp. 15-20, 1972.

B-2. "Flooding Correlation for BWR Bundle Upper Tieplates and Bottom Side-Entry Orifices," K. H. Sun, paper presented at Second Multi-Phase Flow and Heat Transfer Symposium Workshop, April 16-19, 1979, Miami Beach.

B-3. "TRAC-BD1, Version 12, BWR-6 Large Break LOCA Calculations," R. Shumway, EG&G-CDD-5995, (to be published late 1982).

B-4. "A Mechanistic Subcooled Boiling Model," Proceedings of Sixth International Heat Transfer Conference, Vol. 1, Toronto, Canada, 1978.

B-5. "Maximum Bubble Diameter, Maximum Bubble Growth Time, and Bubble Growth Rate During the Subcooled Nucleate Flow Boiling of Water up to 17.7 mPa," Int. J. of Heat and Mass Transfer, V. 17, 1976, pp 643-649.

B-6. "Heat Transfer to Steam-Water Mixtures Flowing in Uniformly Heated Tubes in Which the Critical Heat Flux has been Exceeded," AERE-R-5373, 1967.

B-7. "TRAC-BWR Heat Transfer: Model Description and Steady State Experiment Assessment," Internal Technical Report, EG&G Idaho, Inc., WR-CD-82-064, May 1982.

TABLE B-I

USER CONVENIENCE FEATURES

Feature	Status	Planned Improvements and Evaluation
Free Format Input	The input is prepared based on the fixed format.	Preparation of input based on the fixed format is time consuming. Free format input processor will be provided in TRAC-BD1/MOD1.
Steady State Initialization	The user requires several runs to produce the desired steady state.	Data decks for simple controllers have been developed for use with the control systems model in TRAC-BD1/MOD1 to generate the desired steady state in a single run. A further improvement would be the development of an input processor to set up these simple controllers using a minimum of user input.
Execution Time	The execution times are, on the average, 30:1 for a detailed 3-D analysis of a small or large break LOCA.	Methods to improve computational efficiency are being investigated. These include: (1) simpler nodalization of the plants which is undertaken by the independent code assessment and application program; (2) two-step method developed at Los Alamos, and (3) other improvements such as smoothing of transition regions of correlations and flow regime maps.
Multiple Sources	TRAC-BD1 contains multiple source capability where more than one 1-D component can be connected to the same cell in the 3-D vessel component. This capability reduces the number of 3-D vessel cells required to model a BWR which speeds up the calculations.	

(Continuation)

TABLE B-I. USER CONVENIENCE FEATURES

Feature	Status	Planned Improvements and Evaluation
Documentation	The documentation of TRAC-BD1 is very extensive and complete. The Q/A procedure generates documentation for each code change. Model design and completion reports are written for each model. TRAC-BD1 manual (NUREG/CR-2178) describes the code.	The documentation is extensive and complete. The configuration control system will be very useful to the code users.
Error Diagnostics	TRAC-BD1 performs many range and consistency checks on the input data to identify input errors.	Improvements will be made in TRAC-BD1/MOD1 and future versions. All transient runs performed in developmental and independent assessment programs will be documented in detail to identify errors and calculational difficulties.

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MEMORANDUM FOR: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

FROM: Robert B. Minogue, Director
Office of Nuclear Regulatory Research

SUBJECT: RESEARCH INFORMATION LETTER - 132 - TRAC-BD1 COMPUTER PROGRAM

I. INTRODUCTION

This RIL transmits the TRAC-BD1 computer program which is the first Nuclear Regulatory Commission (NRC) sponsored advanced best-estimate computer program developed at the Idaho Engineering Laboratory (INEL) to analyze large and small break Loss Of Coolant Accidents (LOCA's) in Boiling Water Reactors (BWR's). The code was released to the National Energy Software Center (NESC) in February 1981. A four volume manual containing (1) Model Description, (2) Users Guide, (3) Code Structure and Programming Information, and (4) Developmental Assessment was issued in October of 1981 (Reference 1).

Best-estimate codes such as TRAC-BD1 are developed and assessed in response to a number of requests in Reference 2 and to fulfill many needs recognized in References 2, 3, 4, and 5. Some of the licensing needs which were identified in Reference 2, are:

1. Quantification of margin of conservatism in licensing codes which are based on Appendix K.

2. Prediction and understanding of data from experimental facilities.

3. Analyses and prediction of consequences of postulated accidents and transients in full scale Light Water Reactors (LWR's) in order to resolve licensing and safety issues.

In addition to these needs, another licensing need was identified in Reference 6. Reference 6 requests the confirmation of the adequacy or conservatism of the one dimensional licensing models used to predict three dimensional phenomena in BWR's. TRAC-BD1 has the capability to predict three dimensional phenomena and will be used to address this concern in the independent assessment program.

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The program for development of the TRAC-BWR code consists of three steps. The first step is the TRAC-BD1 code which has been released to the NESC and which is the subject of this Research Information Letter. The specific version which has been released is Version 8. This code provides a basic and best estimate capability for an analysis of a LOCA in a BWR. Since the release of the code some models for calculation of other Chapter 15 transients and additional user convenience features have been added. This led to the production of a new version of the code. The new version which provides also a limited capability to calculate Anticipated Transient Without Scram (ATWS) for which balance of plant modeling is not required, is Version 12. It is available to NRC and is being used by NRC contractors to perform audit calculations for NRR. An independent assessment is being conducted on this version. A limitation found in this version through the assessment process will generally also be a limitation in Version 8.

The second step is the TRAC-BD1/MOD1 code development. This code will extend the TRAC-BD1 capabilities to include all operational transients for which balance of plant modeling is required, but where spatial kinetics is not needed. It will also contain modeling improvements, the need for which is identified by the independent assessment program.

The third and final step will be the TRAC-BD2 code which will contain modeling of spatial kinetics in the core and all of the improvements, identified by the independent assessment program.

Development of TRAC-BWR codes (TRAC-BD1 and future versions) benefits from the development of TRAC-PWR codes at Los Alamos in that modeling of common modules is coordinated. In addition, TRAC-BWR codes are closely associated with joint NRC-EPRI-GE experimental programs such as BWR Refill Reflood, BWR Blowdown-Emergency Core Cooling System (ECCS) and Full Integral System Test (FIST) facility. There is a very close coordination between GE model development associated with these experimental programs and the code development at INEL.

The TRAC-BD1 code was developed from a Pressurized Water Reactor (PWR) version of the TRAC code (TRAC-P1A Version 22.8, Reference 7) which contained a full non-equilibrium and non-homogeneous two-fluid thermal hydraulics model of two-phase flow for the analysis of large and small break LOCA's. This version has been extensively changed to include (1) components necessary to model BWR plants, (2) thermal hydraulic phenomena encountered in BWR geometries, (3) models to increase the speed of calculations, and (4) user convenience features.

II. RESULTS

A. LOCA Calculations

TRAC-BD1 has been used to calculate large and small break LOCA's in its developmental assessment. These results are summarized in Appendix A.

B. Code Execution

The execution statistics of the TRAC-BD1 code represent an improvement over that of RELAP4/MOD6. Table I compares the running times for TRAC-BD1 and RELAP4/MOD6 codes. TRAC-BD1 had about three times as much modeling detail as RELAP4/MOD6 but used less computer time. Table II summarizes the execution statistics of the TRAC-BD1 code for some of the developmental assessment cases. It should be noted that the detailed nodalization used in Cases 1 and 2 in Table II is not necessary for all LOCA calculations. Hence, some LOCA calculations can be executed faster.

C. Developmental Assessment

The models developed have been subjected to "Developmental Assessment" in order to ascertain that these models have been correctly implemented in the code and the results obtained from the exercise of these models are in reasonable agreement with the test data. The developmental assessment test cases can be divided into three groups: (1) separate effects, (2) separate effects heat transfer tests, and (3) integral system effects tests. Separate effects test cases were chosen to exercise a specific hydraulic or heat transfer model in the code while the integral system effects tests were chosen to exercise the code as a whole. Table III discusses results of the developmental assessment.

III. EVALUATION

The code provides a basic capability to analyze the entire large or small break LOCA sequence, beginning with the blowdown, through heatup, reflood with both top and bottom quenching, and finally with refill of the entire core region in one continuous calculation. The code, Version 8, can be used to analyze large and small break LOCA sequences provided that the control system and balance of plant are appropriately modeled using input boundary conditions. An interim version, Version 12, permits the user to model the control system. As stated, capability to model balance of the plant will be provided in TRAC-BD1/MOD1.

Analysis of the LOCA sequence involves modeling of many physical phenomena and specific BWR components. The code uses a fully nonequilibrium and nonhomogeneous two-fluid thermal hydraulics model of the two-phase flow in all positions of the BWR system. The vessel module permits the modeling of three-dimensional thermal hydraulic phenomena in the upper and lower plenums as well as in the bypass region. The constitutive relations for treatment of mass, energy and momentum interchanges between the phases are based on flow regime maps.

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The models important in BWR LOCA analysis using TRAC-BD1 (Version 8) are identified, described and evaluated in Table IV. Table V presents specific BWR components used in LOCA analysis. Additional component models which are necessary for the analysis of transients involving balance of plant are identified in Table VI.

The evaluations in Tables IV and V show that the models used in LOCA analysis are believed to be adequate; however, in some cases further improvements are needed. These evaluations are preliminary and complete evaluations will be performed in the independent assessment program. Pending the independent assessment, some important models are discussed in Appendix B.

Original Signed by

Denwood F. Ross, Jr.

for Robert B. Minogue, Director
Office of Nuclear Regulatory Research

Enclosures: As stated

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RECORD NOTE: Memo from F. Odar to G. Knighton, dated April 8, 1982 requested comments from NRR on this RIL. The RIL was revised extensively based on informal comments by W. Hodges of NRR who sought inclusion of more technical information.

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