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TRAC ANALYSES FOR CCTF AND SCTF TESTS
AND UPTF DESIGN/OPERATION*

by

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ABSTRACT

The analytical support in 1985 for Cylindrical Core Test Facility (CCTF), Slab Core Test Facility (SCTF), and Upper Plenum Test Facility (UPTF) tests involves the posttest analysis of 16 tests that have already been run in the CCTF and the SCTF and the pretest analysis of 3 tests to be performed in the UPTF. Posttest analysis is used to provide insight into the detailed thermal-hydraulic phenomena occurring during the refill and reflood tests performed in CCTF and SCTF. Pretest analysis is used to ensure that the test facility is operated in a manner consistent with the expected behavior of an operating full-scale plant during an accident. To obtain expected behavior of a plant during an accident, two plant loss-of-coolant-accident (LOCA) calculations were performed: a 200% cold-leg-break LOCA calculation for a 2772 MW_t Babcock and Wilcox plant and a 200% cold-leg-break LOCA calculation for a 3315 MW_t Westinghouse plant. Detailed results will be presented for several CCTF UPI tests and the Westinghouse plant analysis.

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*Work performed under the auspices of the US Nuclear Regulatory Commission

INTRODUCTION

The 2D/3D Program is a multinational (Germany, Japan, and the United States) experimental and analytical nuclear reactor safety research program. Its main purpose is the investigation of multidimensional thermal-hydraulic behavior in large-scale experimental test facilities having hardware prototypical of pressurized water reactors (PWRs). The Japanese are operating two large-scale test facilities as part of this program: the Cylindrical Core Test Facility (CCTF), which completed its testing program this year, and the Slab Core Test Facility (SCTF), which will begin its third phase of testing in 1986. The CCTF is a 2000-electrically-heated-rod, cylindrical-core, four-loop facility with active steam generators primarily used for investigating integral system reflood behavior. The SCTF is a 2000-electrically-heated-rod, slab-core (one fuel assembly wide, eight across, and full height), separate-effects reflood facility. Both facilities have prototypical power-to-volume ratios preserving full-scale elevations, and both are much larger than any existing facilities in the United States. The German contribution to the program is the Upper Plenum Test Facility (UPTF) in Mannheim, West Germany, a full-scale facility with vessel, four loops, and a steam-water core simulator. All these facilities have more instruments than any other existing facilities: each has more than 1500 conventional instrumentation data channels, alone. As its contribution to the program, the United States is providing advanced two-phase flow instrumentation and analytical support.

The Los Alamos National Laboratory is the prime contractor to the US Nuclear Regulatory Commission (NRC) in the latter activity. The main analytical tool in this program is the Transient Reactor Analysis Code (TRAC), a best-estimate, multidimensional, nonequilibrium, thermal-hydraulics computer code developed for the US NRC at Los Alamos. Through code predictions of experimental results and calculations of PWR transients, TRAC provides analytical coupling among the facilities and extends the results to predict actual PWR behavior.

During FY 1985, TRAC-PF1/MOD1 analyses were completed for seven CCTF-II experiments. Predictions of upper-plenum injection (UPI) tests 57, 72, 76, and 78 demonstrated that TRAC can predict correctly when UPI flows enhance core cooling and when they contribute to steam binding and degraded core cooling. In addition, TRAC was used to analyze nine SCTF experiments: the base case for Core-II (Run 604), the flat power and initial rod temperature profile (Run 605), the steep power and initial rod temperature profile (Run 611), the FLECHT-SET coupling test (Run 613), the best-estimate base case (Run 614), the separate-effects countercurrent flow-limiting (CCFL) tests (Runs 606 and 610), and others. The analyses of these tests demonstrated that in general TRAC-PF1/MOD1 accurately simulates the reflood thermal-hydraulic behavior of the SCTF tests.

In support of the UPTF, three pretest predictions were performed with TRAC-PF1/MOD1: downcomer separate-effects analyses, a German PWR base case analysis, and a hot-leg small-break test analysis. From these analyses, initial and boundary conditions for the tests can be determined to ensure proper operation of the test facility.

A fine-node 2004 cold-leg-break loss-of-coolant-accident (LOCA) calculation of a Babcock and Wilcox (B&W) 2772 MW_t PWR, assuming licensing-type boundary and initial conditions, was completed. This calculation predicted a peak cladding temperature (PCT) of 995 K to occur in the average rod during blowdown.

In addition, a fine-node 200% cold-leg break LOCA calculation of a Westinghouse 3315 MW_t PWR, assuming licensing-type boundary and initial conditions, was completed. This calculation predicted a PCT of 897 K to occur in the average rod during blowdown.

MAJOR PHENOMENA DURING A LARGE BREAK LOCA IN A UPI PLANT

For a Westinghouse two-loop PWR with a 200% cold-leg break, the sequence of events may vary slightly from plant to plant because of geometry differences and operating assumptions; however, a "typical" sequence of events can be specified (Ref. 1) and is given in Table I.

The blowdown transient is typically less than 20 s, because of the large break area to primary-fluid-volume ratio. During the blowdown transient as the core voids, the core heats up significantly. LOFT experiments and TRAC calculations[1] indicate that the heating during blowdown is terminated when choked-flow conditions at the break restrict the outflow and allow the remaining fluid in the intact cold legs and downcomer to reflood the core partially. The extent of this core recovery during blowdown is dependent upon the number of intact loops, whether or not the reactor-coolant system (RCS) pumps are tripped, and upon the subcooling in the lower plenum and upper head.

For the "typical" sequence of events given in Table I, the refill period is between 18 and 28 s. During the refill period, the core will heat up until core recovery begins. The degree of heating during refill is dependent upon the amount of stored energy retained in the core at the end of blowdown, core power level, and core steam-flow rates. Most of the accumulator flow injected into the cold legs bypasses the downcomer and lower plenum and exits the break during blowdown. However, during refill, most of the accumulator flow in the intact loops ends up in the downcomer and lower plenum. For the "typical" sequence of events given in Table I, both accumulators are empty at the end of the refill phase of the transient.

TABLE I

TYPICAL EVENT SEQUENCE FOR A 200% COLD-LEG BREAK IN A WESTINGHOUSE TWO-LOOP PWR

Event	Times(s)
200% cold leg break	0.0
Reactor scram & feedwater trip	0.1-1.0
High-pressure injection	1.0
Accumulator check valves open:	
Loop A (Intact)	6.0-7.0
Loop B (Broken)	3.0
Low-pressure injection	13.0
Pressurizer empty	15.0
End of blowdown	18.0
Accumulators empty:	
Loop A (Intact)	28.0
Loop B (Broken)	25.0
Beginning of reflood	28.0
Core quenched	300.0-500.0

Of most interest in UPI plants is the reflood phase of the transient, when the water level in the lower plenum reaches the bottom of the core. During reflood, the low-pressure injection (LPI) flow is injected into the upper plenum. Typical LPI flow rate assuming single failure, is ~120 kg/s. The high-pressure injection (HPI) flow into the cold leg is at a rate of ~19 kg/s. During the later stages of refill and the early stages of reflood, the UPI water entering the upper plenum forms a pool in the upper plenum. Small-scale experiments[3,4] and large-scale experiments[5] indicate that subcooled CCFL breakdown requires penetration of subcooled water into the core. Once subcooled water penetrates the core, the steam flow upward is reduced because of condensation and more subcooled water is allowed into the core, which results in more condensation. This is the process that initiates the dumping of UPI water from the upper plenum into the core region. The rods below this region of UPI-water dumping begin to quench, producing additional steam. The steam can either flow up and interact with the subcooled water falling back into the core or it can flow radially over and then up. The latter case is illustrated in Fig. 1.

For the case in which the steam flows radially out and then up, the low-power bundles in that region of the core will quench much earlier than the rest of the core, which then allows dumping of the UPI water directly into the pool of water that is quenching the core from the bottom.

If the additional steam produced from quenching bundles directly below UPI nozzles flows up and interacts with the UPI water falling into the core, then dumping in that region of core will be stopped. Once dumping is stopped, then steam production is reduced and subcooled water begins to penetrate again and the cycle repeats. Therefore, the difference between the two cases is that one results in continuous dumping of (ECC) water from the upper plenum into the core, while the other results in intermittent dumping. The continuous-dumping case tends to result in lower PCTs and faster core quenches. Calculations and data tend to support the continuous dumping case, if sufficient subcooling is available in the upper plenum. It should be noted that even if the core radial power profile is flat, dumping in the outer bundles still occurs, since the largest amount of subcooling of the UPI water will still be directly below the UPI nozzles. The UPI water interacts very quickly with the upper plenum structure and tends to fall to the upper core support plate (UCSP) and to form a pool.

The outer bundles directly under the region of the core dumping will quench in 100 to 200 s. The rest of the core will quench in 300 to 500 s depending upon core-stored energy and ECC flows and temperatures.

UPI Test (CCTF)

Experimental data from the Japanese Atomic Energy Research Institute (JAERI) CCTF UPI tests listed in Table II and the TRAC analyses of these tests showed the following phenomena to be significant.

1. Pooling in the upper plenum.
2. Subcooling in the upper plenum.
3. Entrainment of water from the upper plenum into the hot legs.
4. Dumping or channeling of water in the low-power region of the core.
5. Condensation in the upper plenum.

In the Run 57 experiment and in the posttest calculation,[6] significant core heating was observed after beginning of core recovery (BOCREC). As this was a high-power, high-stored-energy experiment, this heating was expected. In the calculation, significant amounts of water were entrained into the hot legs

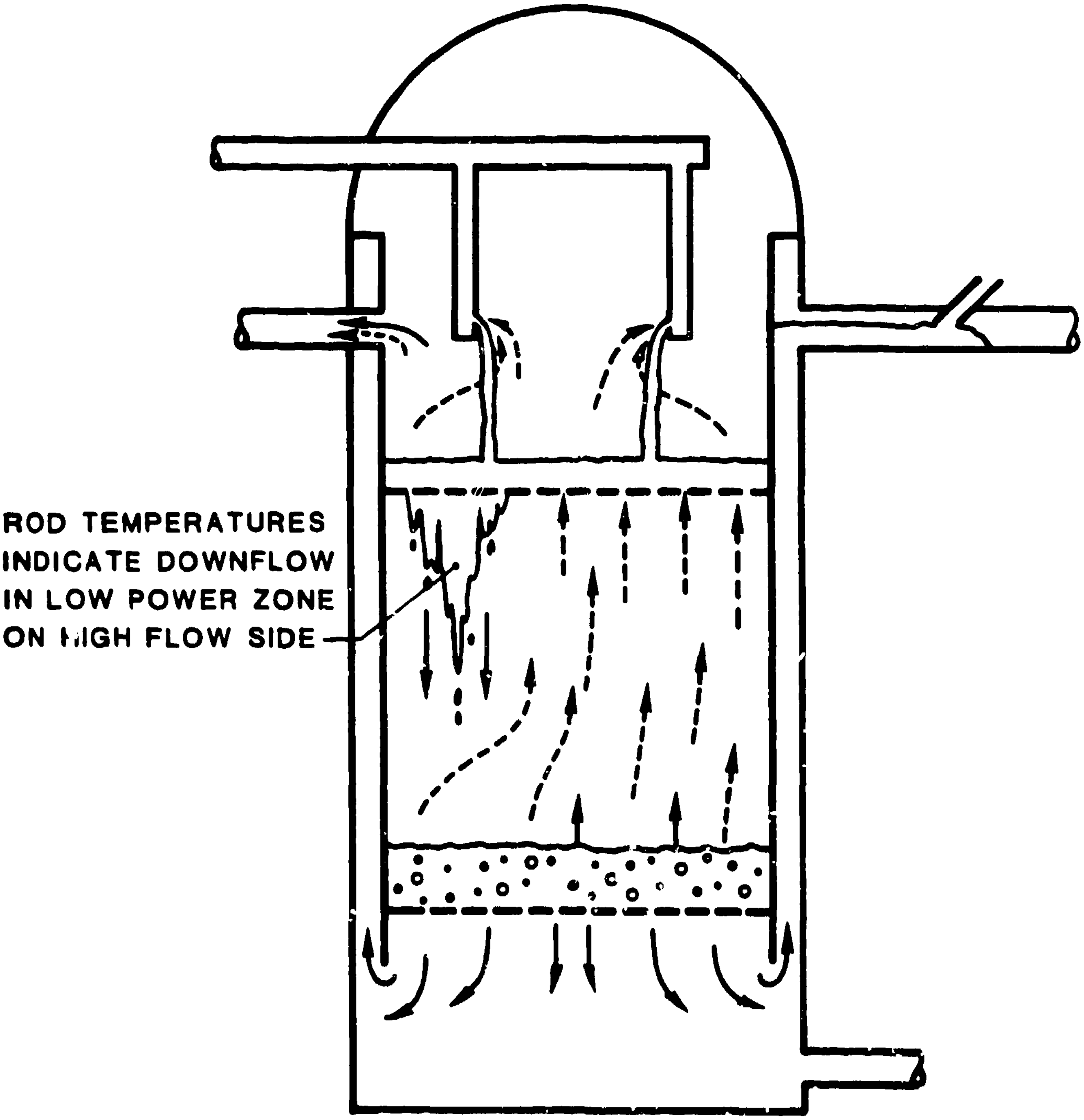


Fig. 1.
Observed experimental behavior in the CCTF.

TABLE II

UPI TEST ANALYZED WITH TRAC

Run #	Description and Comments	Power Level	ECC FLOWS: CL	ECC FLOWS: UPI	UPI Flow Split	PCT at BOCREC (k)	PCT (K)
57	High Power, High CL ECC Flow, Low UPI Flow, High Stored Energy	(1.2*ANS + Actinides) @ 30 s after scram	3/4 (ACC + 10*HPCI)	3/8 LPCI	1.6/1.0	1085	1242
59	Single Failure UPI, High Stored Energy	(1.03*ANS + Actinides) @ 30 s after scram	3/4 (ACC + HPCI)	1/2 LPCI	1.6/1.0	1074	1110
72	No Failure UPI, High Stored Energy	(ANS + Actinides) @ 30 s after scram	3/4 (ACC + HPCI)	Full LPCI	1.0/1.0	1057	1070
76	Asymmetric Injection High Stored Energy	1.07 * 1.02 * (ANS + 1.1 * Actinides) @ 30 s after scram	3/4 (ACC + HPCI)	1/2 LPCI	0.0/1.0	1073	1100
78	Refill-BE-Reflood, Low Stored Energy	1.02 (ANS + Actinides) @ 40 s after scram	3/4 (ACC + HPCI)	1/2 LPCI	0.0/1.0	692	722

LPCI ~15 ℓ/s

HPCI ~3.7 ℓ/s

ACC ~100 ℓ/s

from the upper plenum. As the water flashed in the steam generator tubes, the resulting pressure increase in the steam generator caused the core quench front propagation to slow down. In the experiment, the power in the high power bundles was tripped at 200 s to protect the electrical rods from damage. The calculation at this point was stopped. Both the calculation and the data indicate that UPI water was penetrating into the core. However, in comparison to the data, TRAC predicted too much steam and entrained UPI water flowing into the hot legs. It is anticipated that a higher UPI flow with more condensation in the upper plenum would have reduced the steam flow and entrained UPI water into the hot legs; an earlier turnaround of the rod temperatures would have been the result.

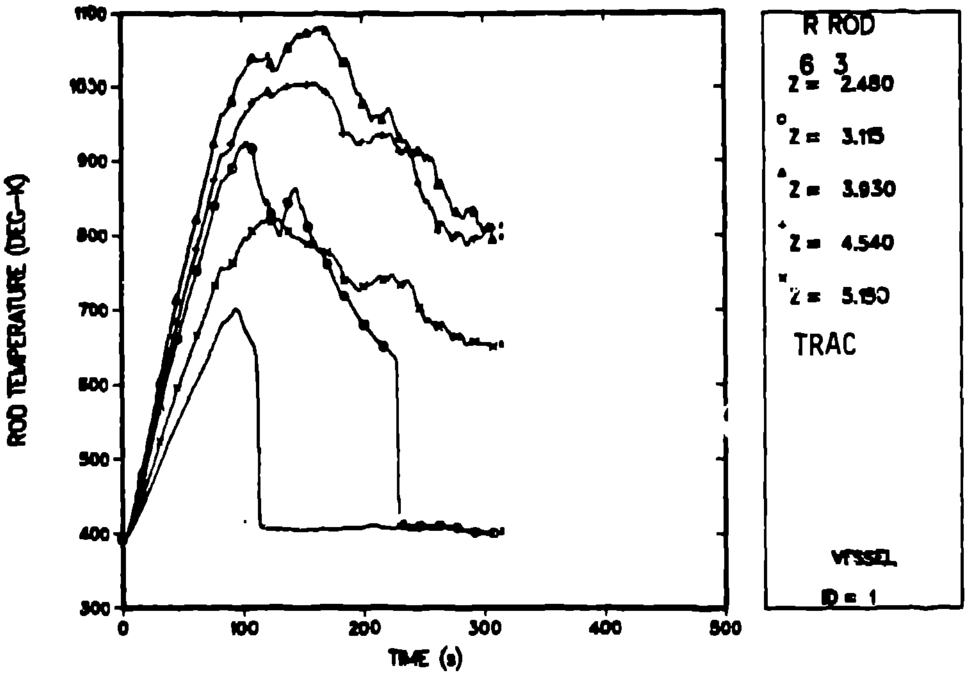
For Run 59 the UPI flow rate was increased and the core power level was decreased compared to Run 57. With the higher UPI flow, more condensation occurred in the upper plenum, resulting in less steam flow and fewer entrained droplets into the hot legs. Both the TRAC calculation[7] and the experimental data indicate lower PCTs for Run 59 as compared to Run 57.

For Run 72, the UPI was increased again, and the power and stored energy were reduced slightly as compared to Run 59. In Run 72, significant channeling was observed in both the experiment and the calculation.[8] This channeling or dumping of ECCS water occurred in the low-power region of the core and was observed to occur on only one side of the core underneath one of the injection nozzles, even though the UPI flow is the same in both UPI nozzles.

Input errors were found in the original TRAC calculation for Run 72; therefore, the calculation is being repeated with the errors corrected. The repeat calculation is in progress and preliminary results are available. In Fig. 2, TRAC results are compared with experimental data for the high-power region of the core. TRAC is overpredicting the PCT by ~70 K because of core heating that was calculated by TRAC to occur from 120 to 200 s. This core heating was not observed in the data. The difference may be caused by TRAC's overestimating the amount of UPI water entrained into the hot legs; however, it is still being investigated at this time. For the rest of the transient, the comparison is quite good and the overall trends are being predicted. As illustrated in Figs. 3 and 4, dumping was correctly predicted by TRAC. Rods 9 and 12 are TRAC-simulated rods in the low-power region of the CCTF core. As illustrated in Fig. 3, neither TRAC nor the data indicate significant dumping in the region around rod 12. In Fig. 4, TRAC and the data both indicate significant dumping in the region around rod 9. It should be noted that the TRAC rod 9 simulates all of the rods in CCTF bundles 5, 6, 7, and 8; therefore, exact comparison with a single measurement is not expected.

For Run 76, the initial stored energy and transient power level were both increased as compared to Run 72. The UPI flow was reduced. In Run 76, only one injection nozzle was used; therefore, asymmetric quenching was expected. Again, both in the data and in the TRAC calculation, channeling and dumping of ECC water were observed. Comparisons to TRAC for Run 76 are illustrated in Figs. 5 and 6. TRAC overpredicted the PCT by ~100 K and calculated heating in the upper portion of the rods that was not observed in the data. Overprediction of the entrainment of UPI water into the hot leg as was mentioned for Run 72 and overprediction of the boiloff of water in the downcomer are two explanations currently being considered.

For the UPI transients, negative core inlet flow is established at or soon after BOCREC. The water flowing out the bottom of the core is saturated liquid or a low void fraction bubbly mixture. This saturated liquid mixes with the cold water in the downcomer and lower plenum, causing a temperature rise. Wall heat transfer from the hot vessel walls also contributes to the heating of the



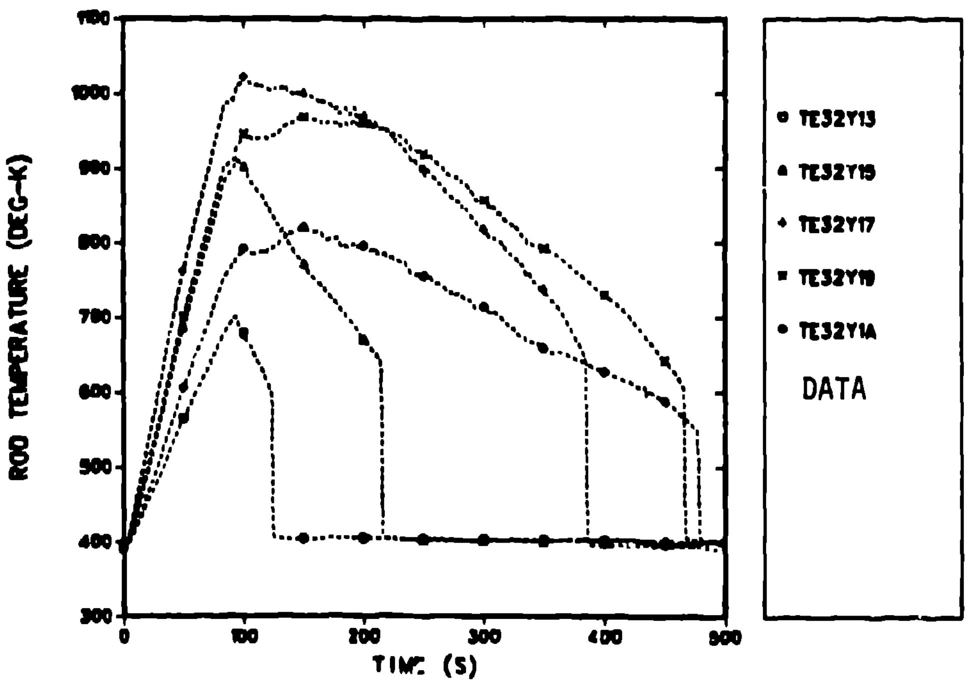


Fig. 2.
TRAC comparison to rod temperature data in the high-power region of the core for Run 72.

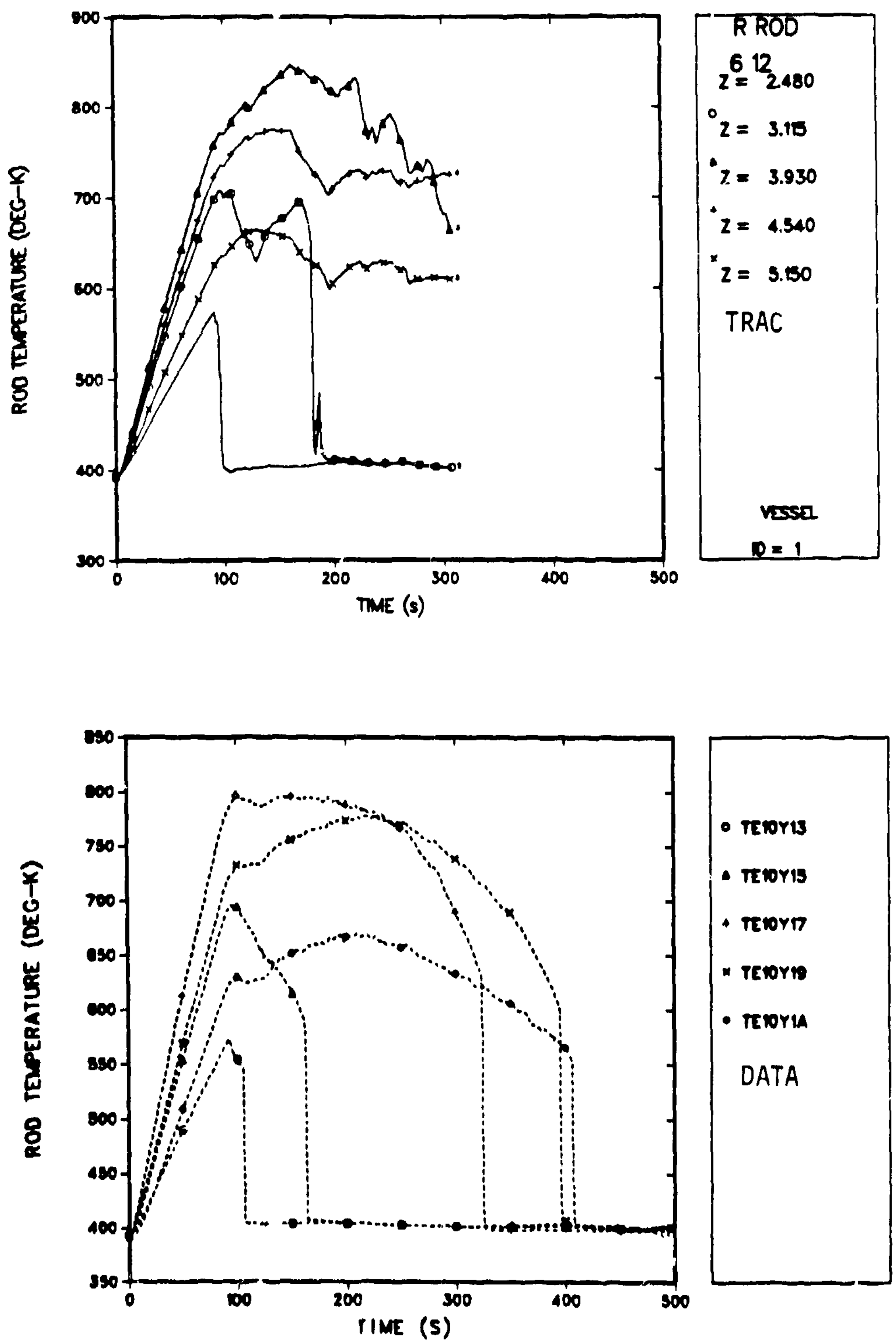


Fig. 3.
TRAC comparison to rod temperature data in the low-power region of the core for Run 72.

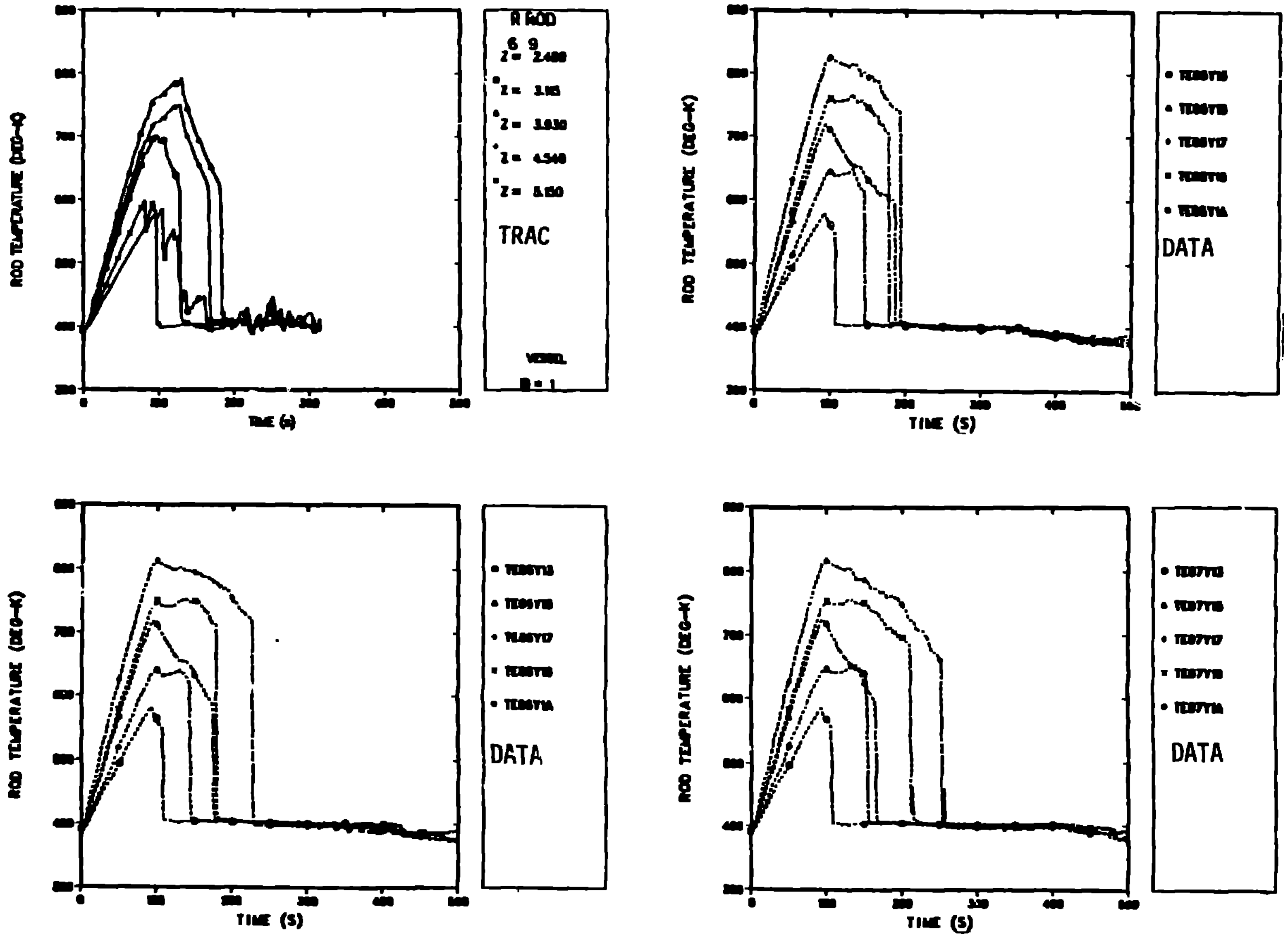
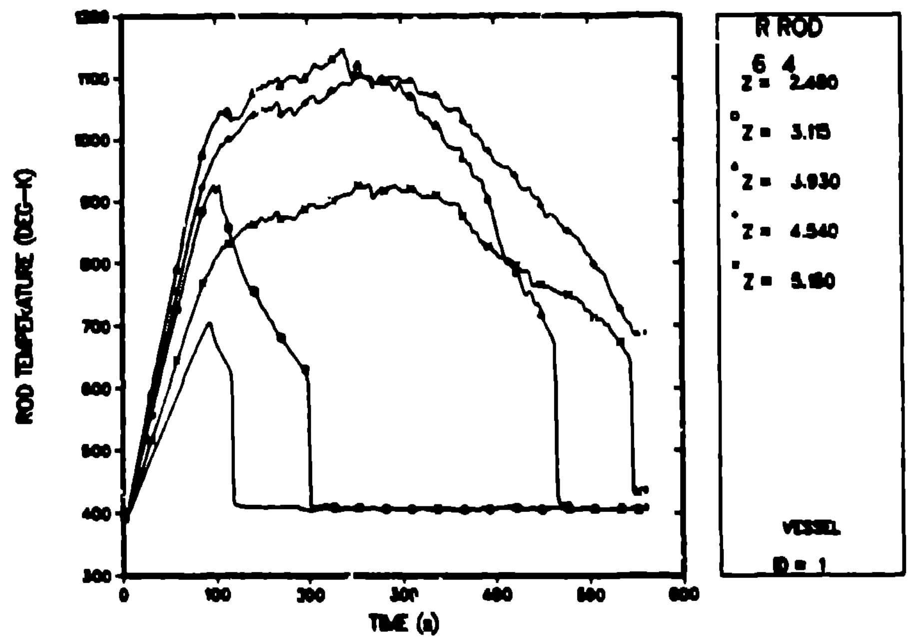


Fig. 4.
TRAC comparison to rod temperatures in the low-power region of the core where dumping occurs for Run 72.



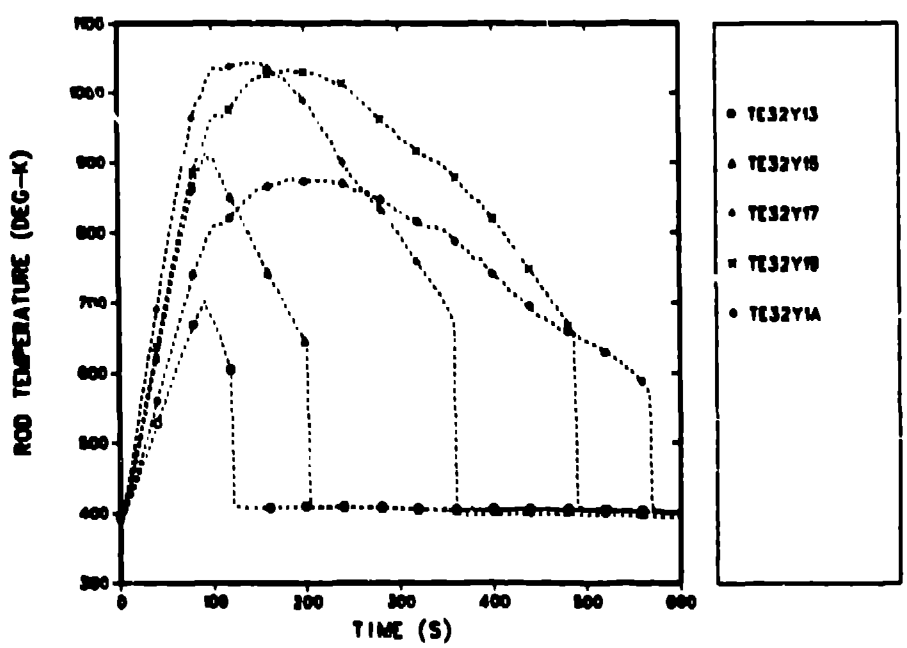


Fig. 5.
TRAC comparison to rod temperature data in the high-power region of the core for Run 76.

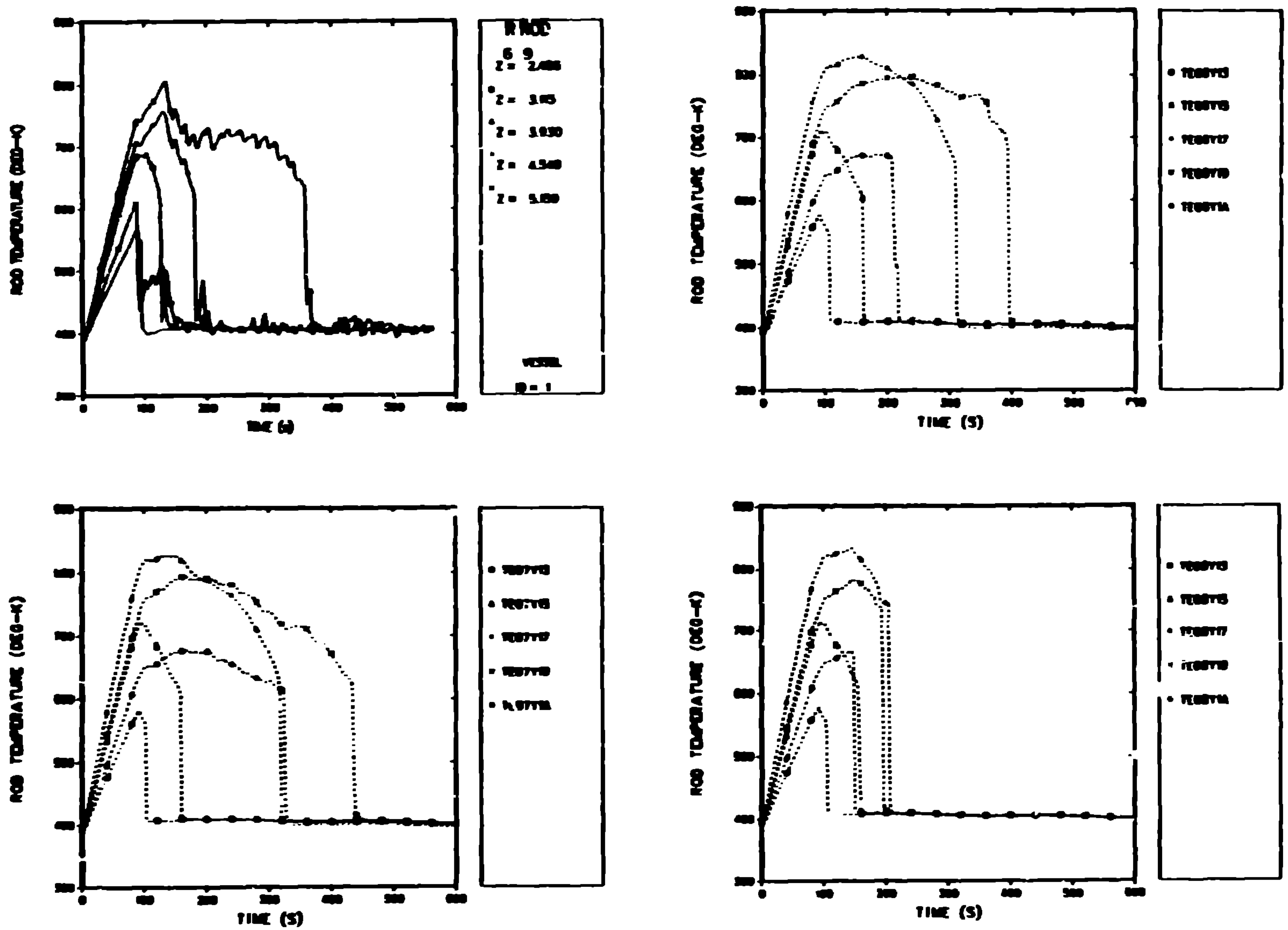


Fig. 6.
TRAC comparison to rod temperature data in the low-power region of the core for Run 76.

fluid. TRAC, in all the UPI calculations, tends to overpredict this heating and subsequent boiling of fluid in the downcomer.

Channeling of the UPI flow into the core was observed in both the experiment and the data. As shown in Fig. 6, core assembly 8 experiences a very early quench. This assembly is located very near the injection point. Nearby assemblies 6 and 7 do not exhibit such a strong effect and quench somewhat later. The TRAC calculation for this region shows a somewhat average behavior of the data.

For Run 78, both the power and the core initial stored energy were reduced. In addition, the radial power distribution was flat in Run 78 as opposed to the steep radial power profile in Run 76. However, the UPI rate was the same. Figures 7 and 8 illustrate that TRAC did a good job of predicting the overall PCT and the channeling in the outer bundles, although TRAC-calculated quench occurs too early. This is believed to be caused by TRAC's allowing too much UPI water to fall back into the core. This is consistent with comparison of TRAC to small-scale CCFL data.[9] At low steam-flow rates, TRAC tends to let too much water fall down as compared with the data.

Westinghouse 3315 MW_t Plant Analysis

The TRAC model uses 950 cells to model a Westinghouse 3315 MW_t plant with 15 × 15 fuel-rod assemblies. All the loop components such as the hot leg, steam generator, loop seal, circulating pump, cold leg, and emergency core-cooling system (ECCS) were modeled as physically complete as possible. A schematic of the vessel component is shown in Fig. 9. The vessel has been subdivided into 17 axial levels, 4 radial rings, and 8 azimuthal sectors for a total of 544 hydrodynamic cells. The core region consists of the two inner radial rings and the five axial levels extending between levels 4 to 9. The barrel baffle region extends from levels 4 to 10 and occupies the third radial ring within these levels. The fourth radial ring represents the downcomer region from levels 3 to 15. At the top of level 15 in the fourth radial ring and in each azimuthal sector are open flow area passages that model the upper head spray nozzles. Flow paths between the upper head and upper plenum were represented at the top of level 15 and in the three inner rings by modeling the appropriate reduced flow area and flow losses to simulate the flow through the control-rod penetrations in the upper support plate.

This PWR analysis simulates a 200% guillotine break of a cold leg between the cold-leg nozzle and the ECC injection port immediately outside of the biological shield. ECC flows were based on the single failure assumptions. Accumulators contained the minimum volume allowed, and the core power was 2% over the design limit. The core-power peaking was based on beginning of life; however, the power-decay curve assumed an infinite operating period.

The maximum average rod temperature is shown in Fig. 10. At the beginning of the blowdown phase the core voids rapidly and the fuel rod cladding heats up quickly. The PCT occurs during this early portion of the blowdown. However, as can be seen from Fig. 11, the core fills to ~75% full within the first 10 s after the first dryout. This is because the core flow turns positive as the three intact loop flows exceed the two-phase choked flow out the broken loop. This positive flow into the core from the lower plenum terminates the early heating of the fuel rod cladding. As the blowdown transients continue, the core dries out again; however, steam flow rates through the core are high enough such that no significant heat up occurs until refill begins at about 25 s.

At the end of blowdown and at the beginning of refill steam flows through the core are insufficient to cool the core; therefore, heating occurs again from about 20 to 40 s. This second period of core heating is terminated by the

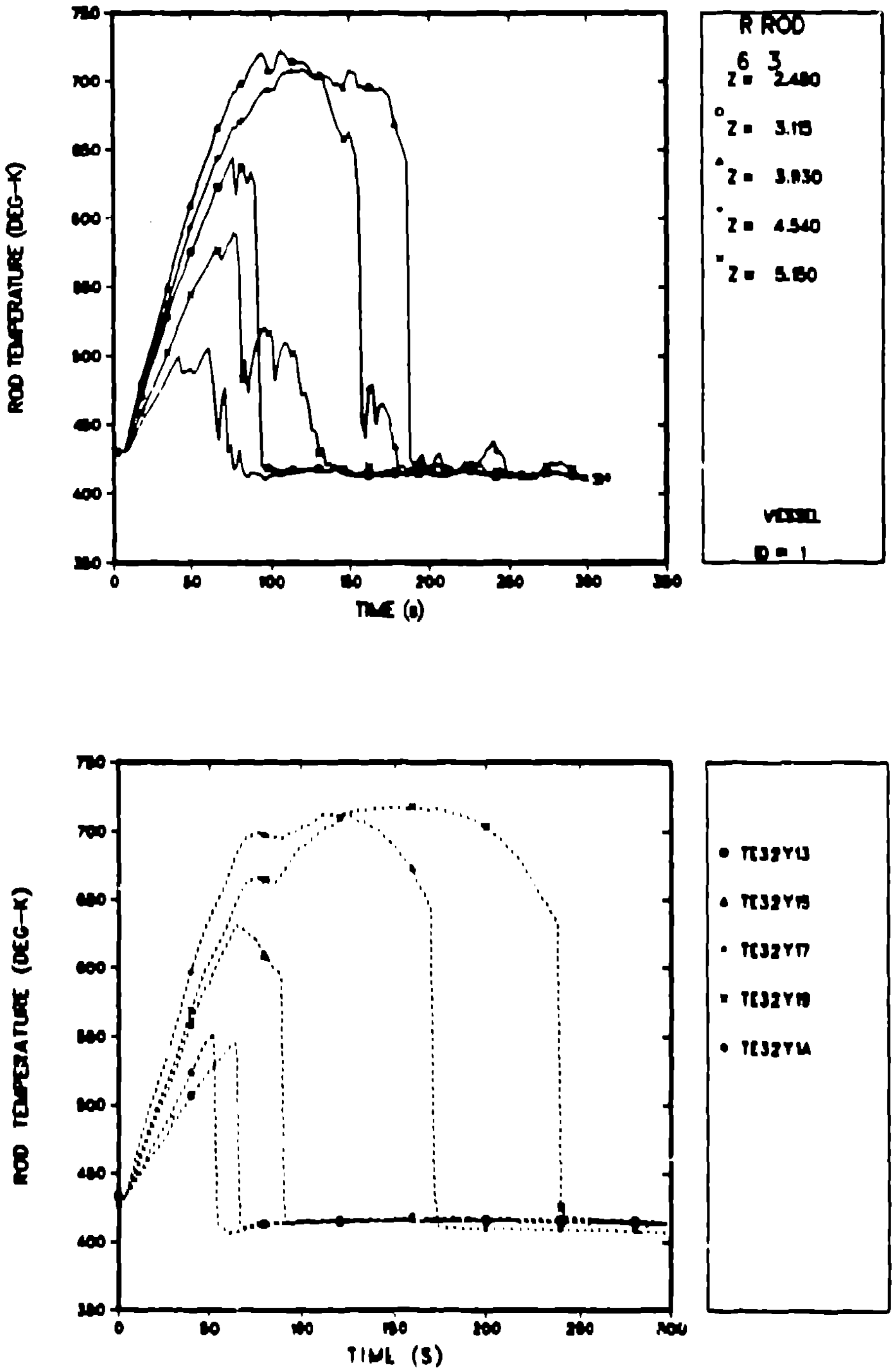


Fig. 7.
TRAC comparison to rod temperature data in the high-power region of the core for Run 78.

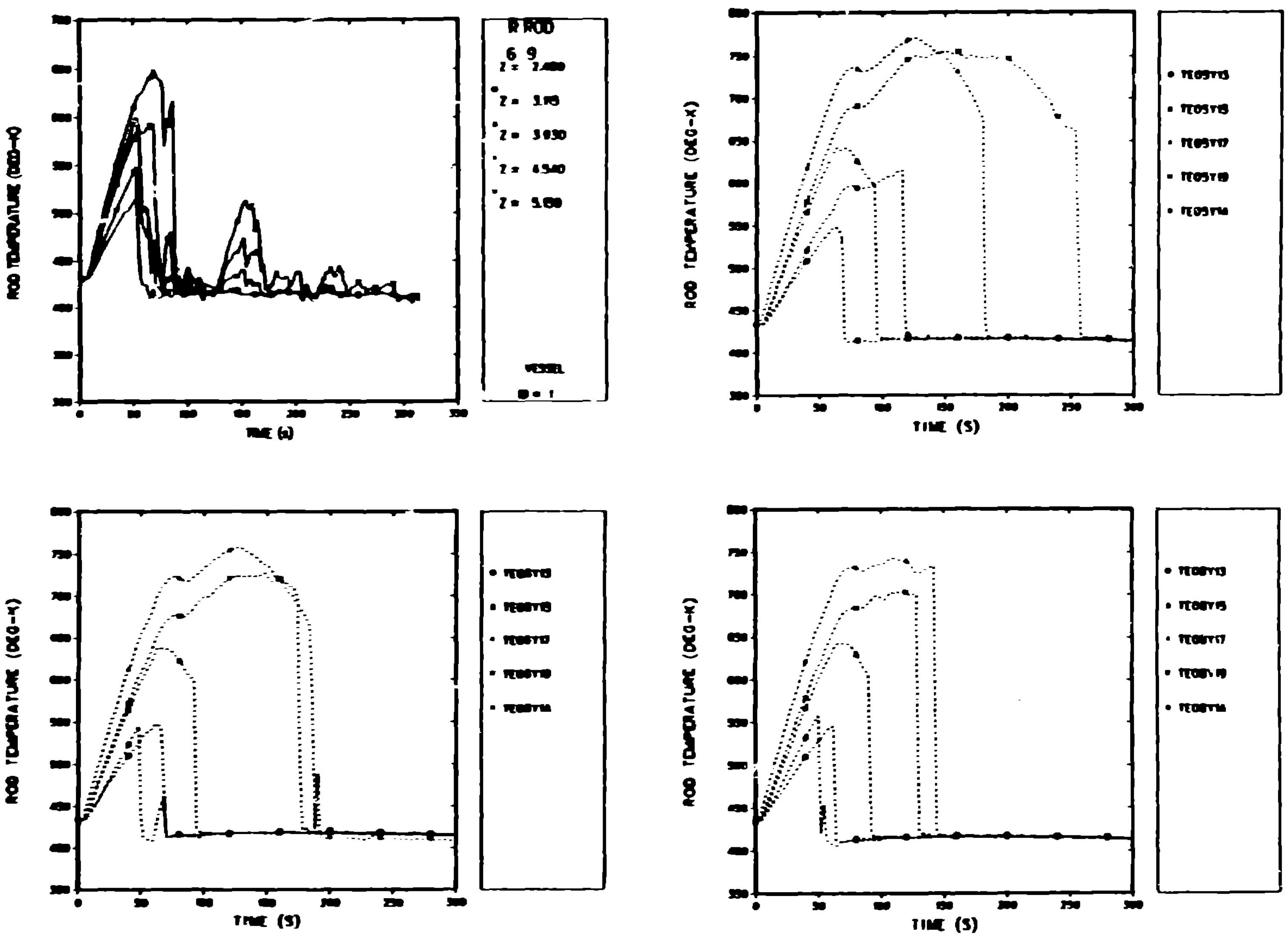


Fig. 8.
TRAC comparison in the low-power region of the core for Run 78.

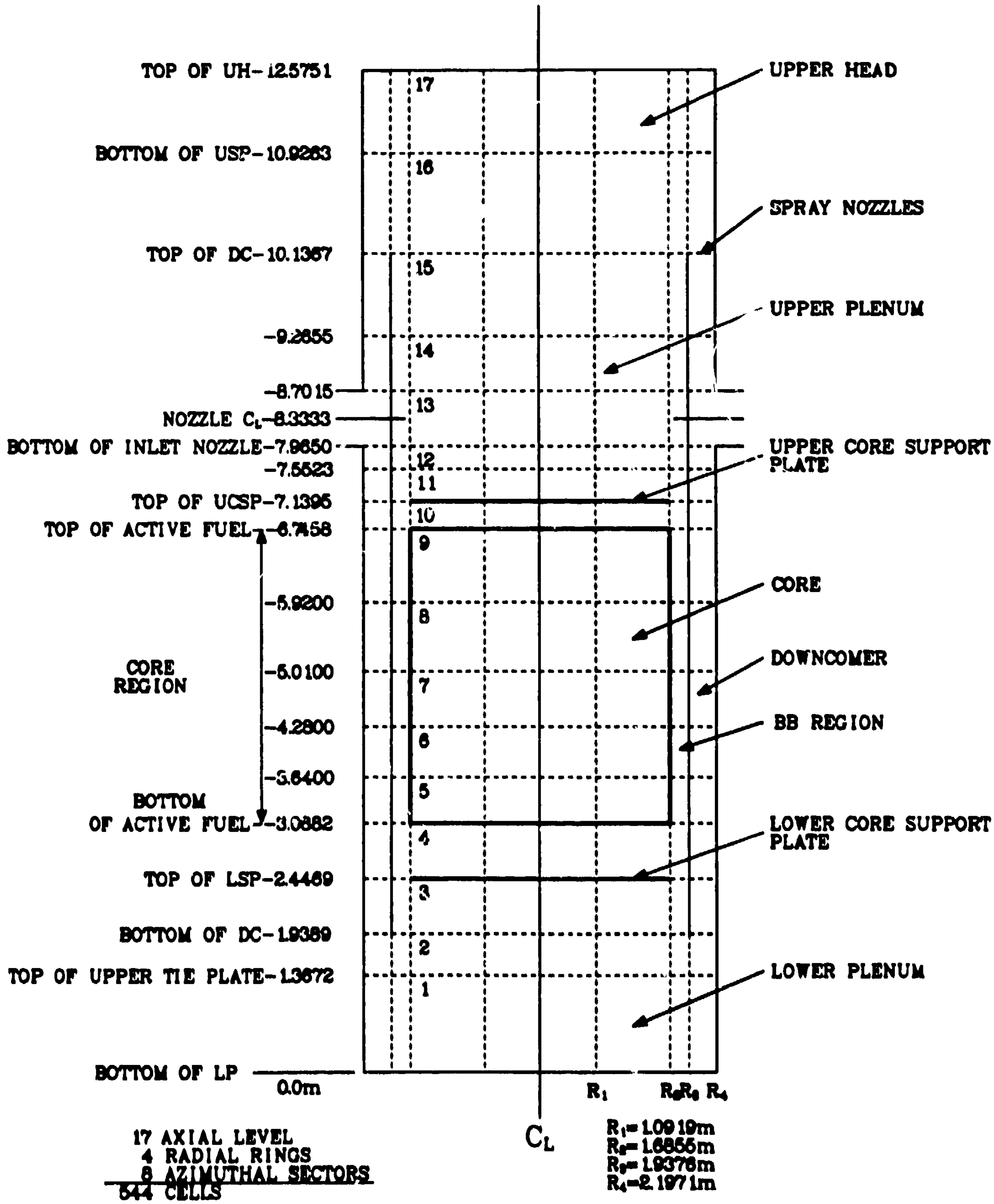


Fig. 9.
Vessel component for Westinghouse PWR.

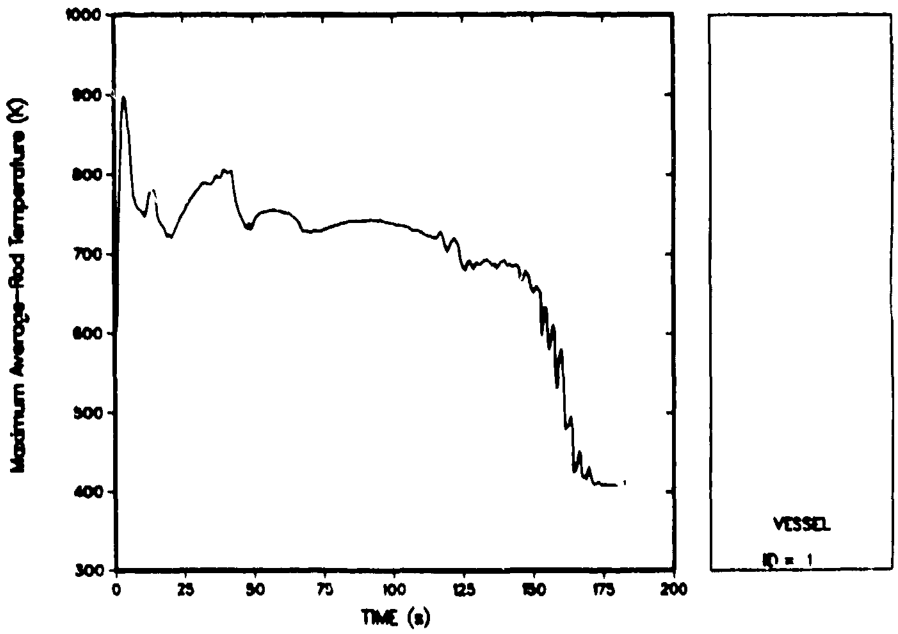


Fig. 10.
Maximum average rod temperature.

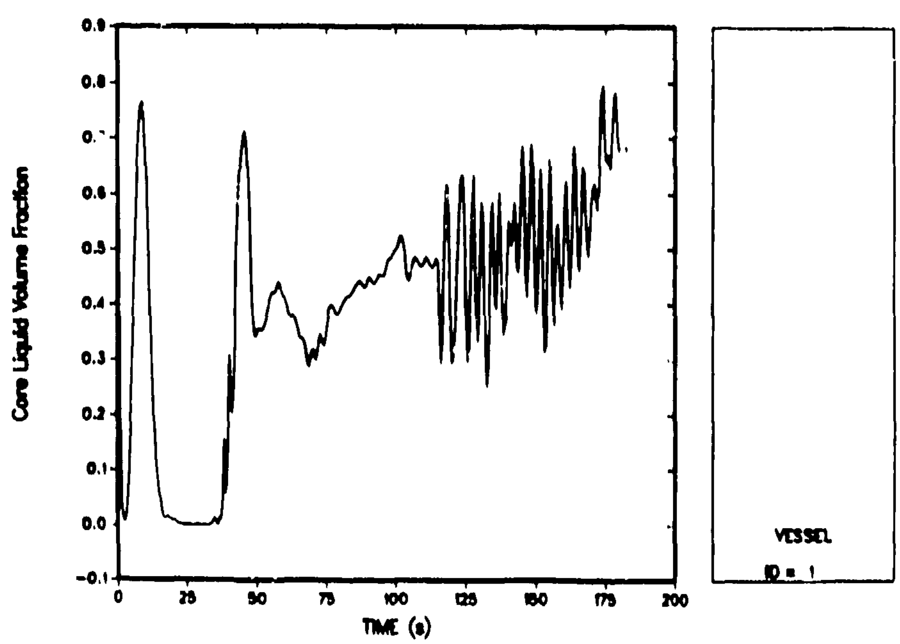


Fig. 11.
Core liquid volume fraction.

BOCREC that occurs at ~39 s. A very rapid core cooldown occurs from ~45 to ~55 s as the intact accumulators empty and nitrogen gas from the accumulators enters the cold legs and top of the downcomer. This nitrogen gas has the effect of reducing the condensation rate in the intact cold legs and pressurizing the intact cold legs and downcomer. As can be seen from Fig. 11, this results in a core refill to ~70% liquid full just before 50 s.

From ~55 s to ~170 s, the core slowly cools and quenches with no other significant heating in the average rods. Late in the reflood transient, manometer-like oscillations between the downcomer and core occur (Figs. 11-12); however, the core continues to cool.

CONCLUSIONS

Both the TRAC calculations and the CCTF UPI experimental data indicated channeling of ECC water from the upper plenum into the core. The experimental data indicated asymmetric behavior in the core and upper plenum, even when the power profile was flat or when UPI flows were symmetric; therefore, multidimensional analysis capability was required to simulate the test behavior accurately. TRAC correctly predicted when UPI flows enhance core cooling and when they contribute to steam binding and degraded core cooling. TRAC tended to overpredict the steam binding effect at high power and overpredict the water fallback rate at low power.

The Los Alamos analysis effort is functioning as a vital part of the 2D/3D Program. Results from this program have already addressed, and will continue to address, key licensing issues including scaling, multidimensional effects, downcomer bypass and refill, reflood steam binding, core blockage, alternate ECCS, and code assessment. The CCTF analyses have demonstrated that TRAC-PF1/MOD1 can correctly predict multidimensional, nonequilibrium behavior in large-scale facilities prototypical of actual PWRs. Through these and future TRAC analyses, the experimental findings can be related from facility to facility; more important, the results of this multinational research program can be related directly to licensing concerns affecting actual PWRs.

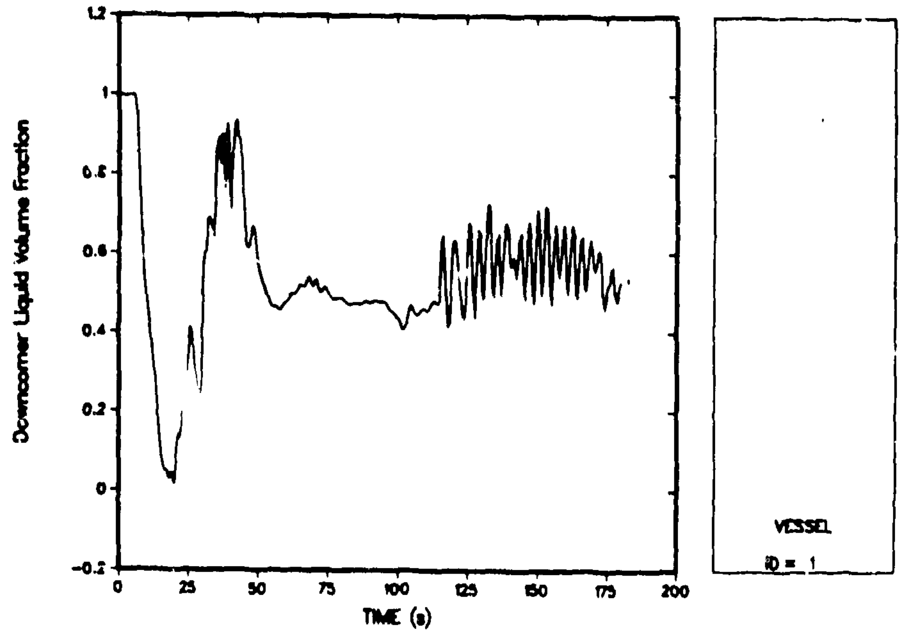


Fig. 12.
Downcomer liquid volume fraction.

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