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# TRAC ANALYSIS OF THE EFFECT OF INCREASED ECC SUBCOOLING ON THE REFLOOD TRANSIENT IN THE SLAB CORE TEST FACILITY*

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## ABSTRACT

A blind posttest calculation of Slab Core Test Facility (SCTF) Run 510, the high-subcooling test, was completed with TRAC-PD2/MOD1 using initial conditions provided by the Japan Atomic Energy Research Institute (JAERI), but without knowledge of the actual test results. There is good comparison between the calculation and the data for rod temperatures, turnaround times, core differential pressures, and mass inventories, and reasonable comparison for absolute pressures, upper plenum pool formation, and fluid temperatures and mass accumulation in the steam-water separator. Comparison of this calculation with the calculation of the base case test (Run 507) shows that the qualitative behavior during reflood is calculated correctly for both cases. In addition, from this comparison the following conclusions can be drawn: for the high-subcooling case, the peak rod temperature was lower, calculated quench times were earlier, there was more entrainment and liquid carryover from the core to the upper plenum, and the liquid mass accumulation in both the core and the upper plenum was greater.

*Work performed under the auspices of the United States Nuclear Regulatory Commission.

## I. INTRODUCTION

The TRAC[1] computer code was used to analyze two Slab Core Test Facility (SCTF) subcooling effects tests performed in 1981, at the Japan Atomic Energy Research Institute (JAERI) in Tokai, Japan. These calculations were performed knowing actual initial and boundary conditions but with no foreknowledge of the results.

In general, this calculation was in good agreement with the test data with regard to overall trends as well as to specific items of comparison. When this calculation is compared with the calculation of the base case test (Run 507), several conclusions can be drawn: in the high-subcooling test, the peak rod temperature was lower, the calculated quench times were earlier, there was more liquid entrainment and carryover from the core, and the liquid accumulation in both the core and the upper plenum was greater. The calculational model is reasonably accurate, and the TRAC code has produced good results.

### A. The 2D/3D Program

The SCTF is part of the 2D/3D Program, a multinational program to assess best-estimate thermal-hydraulic computer codes such as TRAC, and to obtain data and develop improved correlations for the analysis of loss-of-coolant accidents (LOCAs) in pressurized water reactors (PWRs) during the end-of-blowdown, refill, and reflood phases by means of experiments in large test facilities in Japan and Germany. The United States, with funding from the U. S. Nuclear Regulatory Commission, is providing analytical support for these test facilities.

### B. The Test Facility

The SCTF is composed of the pressure vessel, primary coolant system, and emergency core cooling (ECC) system. The pressure vessel contains the slab core, downcomer, upper and lower plena, core baffle region, and upper head. The facility is full-scale in the axial direction and half-scale in width. The slab core consists of eight bundles of electrically-heated rods in a 16 x 16 matrix. These eight bundles are arranged in a row numbered from the innermost bundle (1) representing the core center to the outermost bundle (8) on the downcomer side. There are blockage sleeves near the core midplane in Bundles 3 and 4 to simulate fuel rod ballooning; these represent 56 % coplanar blockages.

The primary coolant system comprises an intact loop, a broken loop with controllable valves simulating the breaks, a steam-water separator, and two containment tanks.

ECC water can be injected into the intact cold leg or directly into the plena. For the tests discussed in this report, the accumulator (ACC) and low-pressure coolant injection (LPCI) system are connected to the pressure vessel at the bottom of the lower plenum on the downcomer side. For these tests, the ACC and LPCI systems were operational, but the cold-leg and upper-plenum ECC systems were not; the ACC and LPCI systems provided forced injection directly into the lower plenum. Figure 1 shows a sketch of the SCTF.

### C. Test Description

The forced-flooding tests discussed herein are Runs 507[2,3] and 510.[4] These were discussed in detail in two unpublished informal reports by the author: "TRAC Analysis of the SCTF Base Case Test, Run 507" (Los Alamos 2D/3D Program Technical Note LA-2D/3D-TN-81-23, October, 1981), and "TRAC Analysis of the SCTF High-Subcooling Test, Run 510" (Los Alamos 2D/3D Program Technical Note LA-2D/3D-TN-81-24, December 1981). Run 507, the base case test, was the first of the main test series for this new facility; Run 510 was considered the

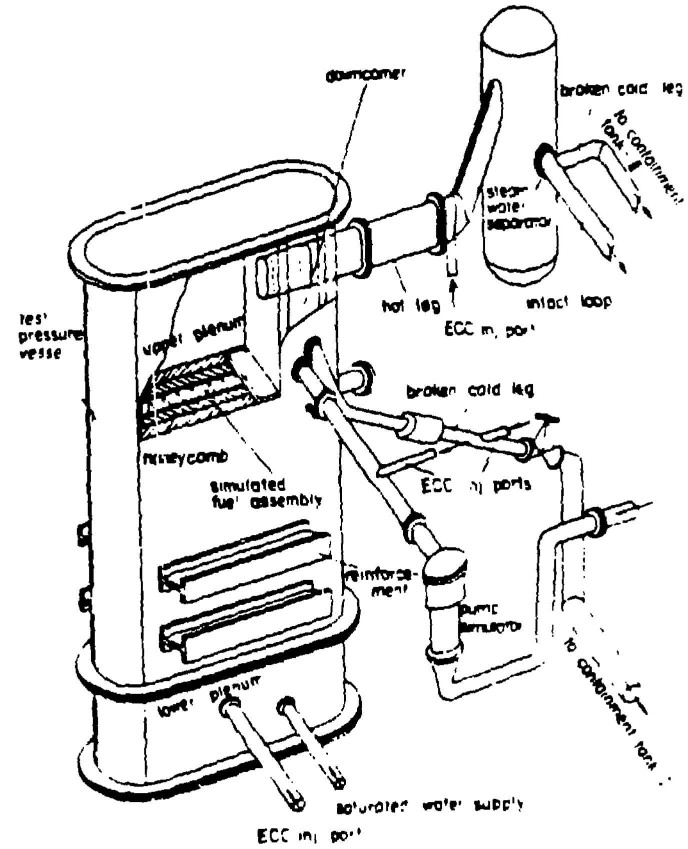


Fig. 1. Pictorial representation of the SCTF.

high-subcooling test. For all these tests, the downcomer was blocked at the bottom, allowing flow only in the core and bypass regions. The initial system pressures were 2.0 bars. The initial vapor and structure temperatures were the saturation temperature at the given system pressure. An initial temperature profile was specified for the rods. The initial power for these tests was about 7 MW. The axial distribution of the power was a chopped cosine, with the radial ratios of bundle power specified as:

| | |
|---|---|
| Bundles 1 and 2 | 0.940 |
| Bundles 3 and 4 | 1.0 |
| Bundles 5 and 6 | 0.953 |
| Bundles 7 and 8 | 0.863 . |

Initially, the lower plenum was half full; this water was at saturation for Run 507, but was subcooled by about 25 K for Run 510.

Operation of these tests began by heating the rods electrically until a specified maximum cladding temperature (926 K) was reached (t = 0 s). After a 2 s delay, a 20 kg/s ACC flow was initiated and held constant until 17 s, when a 10 kg/s LPCI flow was actuated and held constant for the remainder of the test. For Run 507, the ACC water was subcooled by about 30 K and the LPCI water was about at saturation; for Run 510, the ACC water was subcooled by about 50 K and the LPCI water was subcooled by about 35 K. At 6 s, the power began to decrease according to the American Nuclear Society standard decay curve.

D. The Computational Model

The TRAC computational model developed at Los Alamos, reported in an informal report by the author, "Revision of the TRAC Calculational Model for the Slab Core Test Facility" (Los Alamos 2D/3D Program Technical Note LA-2D/3D-TN-81-17, October, 1981), used a two-dimensional VESSEL component for the pressure vessel (154 cells) and a three-dimensional VESSEL component for the steam-water separator (8 cells). One-dimensional components comprising 51 computational cells were used for the rest of the primary system. There was a total of 213 computational cells in the TRAC model. Figure 2 shows schematic diagrams of the pressure vessel and primary system.

E. TRAC Code Description

The analysis tool used for these calculations is the Transient Reactor Analysis Code (TRAC),[1] which has been developed at Los Alamos to provide an advanced best-estimate predictive capability for the analysis of postulated accidents in light water reactors. TRAC provides this analysis capability for light water reactors and for a wide variety of thermal-hydraulic experimental facilities. It features a three-dimensional treatment of the pressure vessel and associated internals; two-phase nonequilibrium hydrodynamics models; flow-regime-dependent constitutive equation treatment; reflood tracking

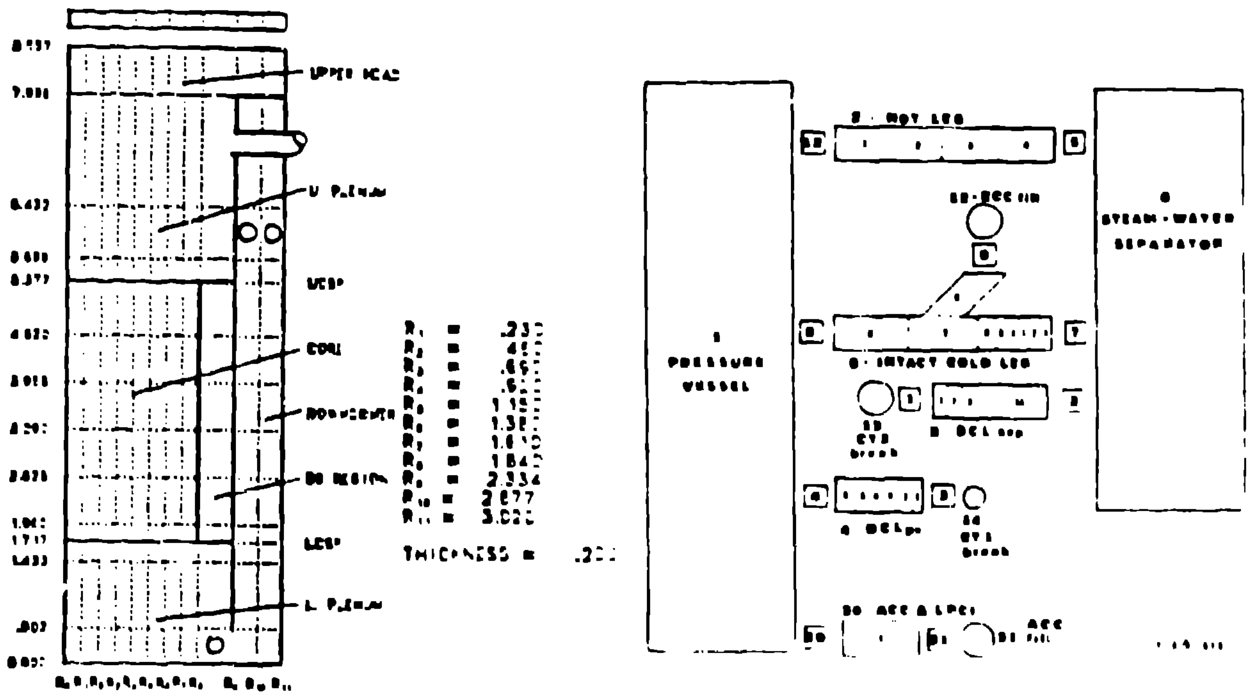

Fig. 2. Schematics of the SCTF pressure vessel and primary system.

capability for both bottom flood and falling film quench fronts; and consistent treatment of entire accident sequences, including the generation of consistent initial conditions.

TRAC-PD2/MOD1 (version 26.2 with updates) was the code version used for these calculations.

## II. RESULTS OF THE CALCULATION

This section is subdivided into two parts. First, the calculated results and the experimental data for the base case (Run 507) and the high-subcooling case (Run 510) will be compared and discussed; and, second, a comparison of the calculations of the two tests will be discussed.

### A. Comparison of the Test Data with the Calculated Results

The calculated core differential pressures agreed reasonably well with the data, as shown in Fig. 3. For these tests, the calculation underpredicted the differential pressure in the bottom half of the core, but agreed fairly well with the data for the top half of the core. The calculation and the data were in closer agreement for the differential pressures for the full height of the core, particularly later in the transient, although there was a tendency toward underprediction in the earlier portion of the transient. The data for the core lower half may be a bit too high; the differential pressure measurement indicates that the water level is nearly 2 m, and the distance between the two ΔP cells is about 1.97 m. This would indicate that there is no boiling taking place at all in the lower regions of the core after quench in the experiment, an unlikely event. Because of this, the calculated water level of about 1.8 m in the lower half of the core does not seem unreasonable at all.

Comparison of the experimental and calculated rod temperatures is given for several thermocouple elevations in Bundle 4 in Fig. 4. This rod is in the high-powered assembly, having a power of about 930 kW. The quench times and temperatures at corresponding thermocouple locations in all the bundles are almost identical for both the experiments and the calculations at lower core elevations where the cooling results from the direct contact of the flooding water with the rods. In the upper core regions where cooling comes about from steam flow with liquid droplet entrainment and from liquid fallback into the core from the upper plenum, the calculations consistently lead the data in quench times. The maximum temperatures reached at the lower core elevations agree within a few degrees for both the calculations and the data, but at higher core elevations the spread increases to about 50 K. Neither the calculations nor these forced-flood experiments seemed to be affected in quench times and temperatures by the 56% blockages at the midplane in Bundles 3 and 4.

The code slightly underpredicts the hot leg mass flow rate but predicts fairly accurately the two cold leg mass flow rates. This indicates that the correct amount of liquid is not calculated for the entrainment and the carryover from the vessel through the hot leg, but that the steam generation and its flow out of the cold legs is correct. The agreement between the calculation and the experiment in most other parameters was good for both tests.

### B. Comparison of This Calculation with the Base Case Calculation

For most of the figures in this section, the base case is shown with a solid line, and the high-subcooling case is shown with a dashed line; the curves are identified on the plot in cases where this is not true.

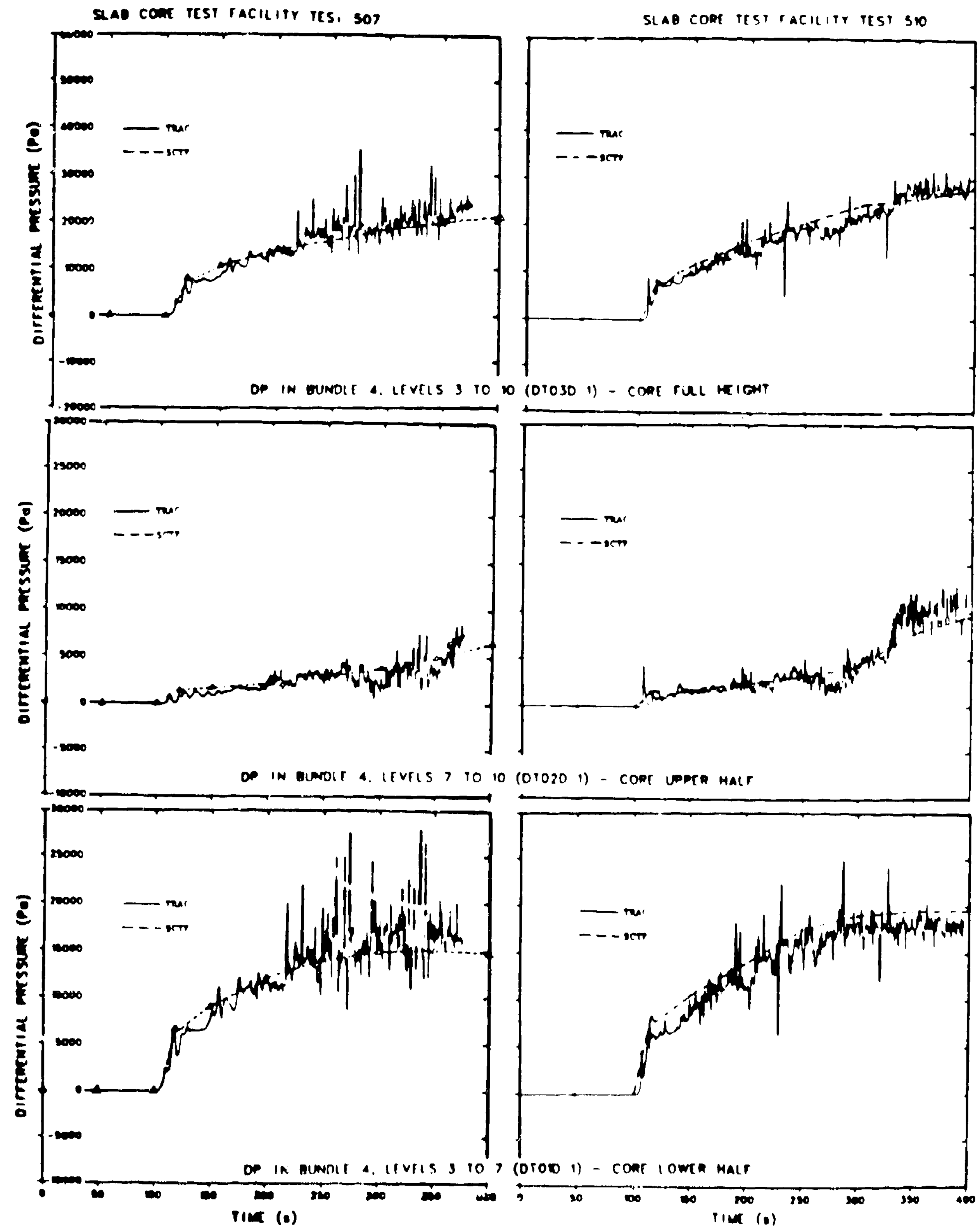


Fig. 3. Comparison of core differential pressures.

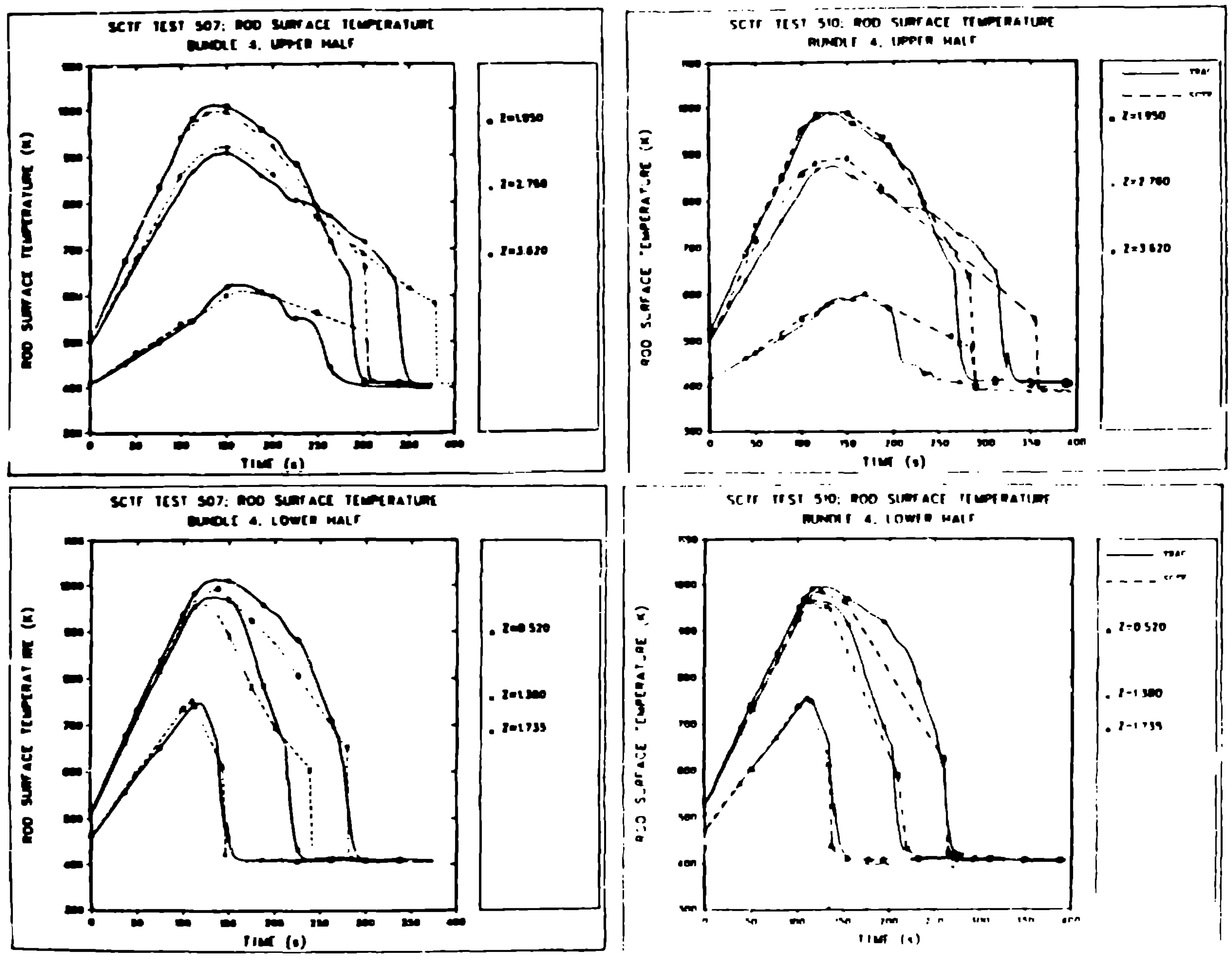


Fig. 4. Comparison of rod surface temperatures.

A comparison of the containment pressure for these two tests is shown in Fig. 5; Fig. 6 compares their ECC mass flow rates. Later reference will be made to the lower containment pressure in Run 510 and to Run 510's higher ECC mass flow. The ECC subcooling and average subcooling in the lower plenum are compared for these two runs in Figs. 7 and 8, respectively.

One noticeable difference between the calculations of Run 510 and Run 507, illustrated in Fig. 9, is that the rods quench earlier in the high-subcooling case than in the base case, particularly above midplane. In the higher regions, the time difference can be as large as 35 - 40 s in both the experiment and the calculation, whereas in the lower regions the difference is smaller, more on the order of 5 - 10 s. This is an effect directly attributable to the high subcooling, in part because the liquid mass in the core is consistently higher throughout the transient for Run 510, and in part because this case has a consistently larger total core outlet liquid mass flow, implying a higher rate of entrainment that results in greater precooling.

The total core inlet liquid mass flow, compared in Fig. 10, the core liquid mass, compared in Fig. 11, and pressure vessel liquid mass, compared in Fig. 12, are greater for Run 510; this is a result of the slightly higher ECC mass flow rate for this test.

The water level in the core was about the same in the two calculations; however, in the experiment the ultimate water level in the base case was less than in the high-subcooling case, as pointed out in the previous section. The calculational result probably comes about from the way in which TRAC approximates the bubbles resulting from the boiling when the code is calculating

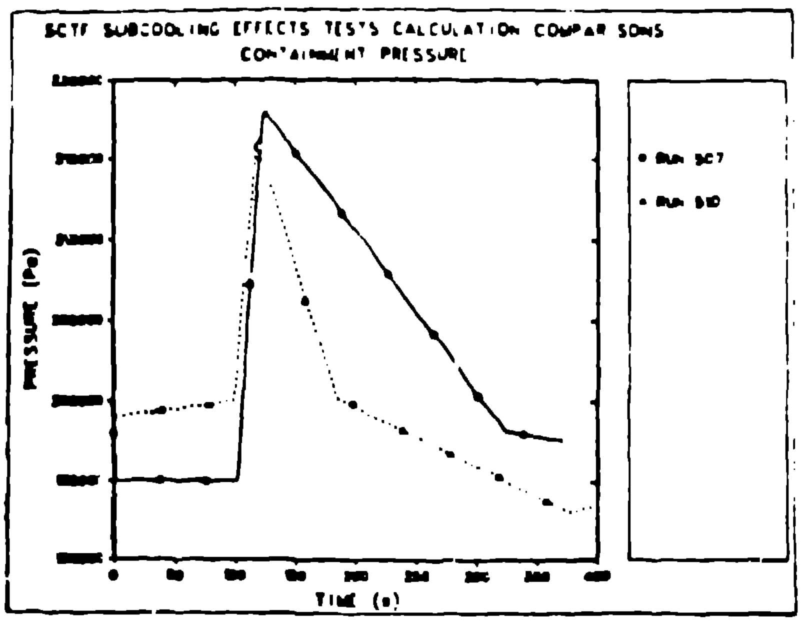

Fig. 5. Comparison of the containment pressure for Runs 507 and 510.

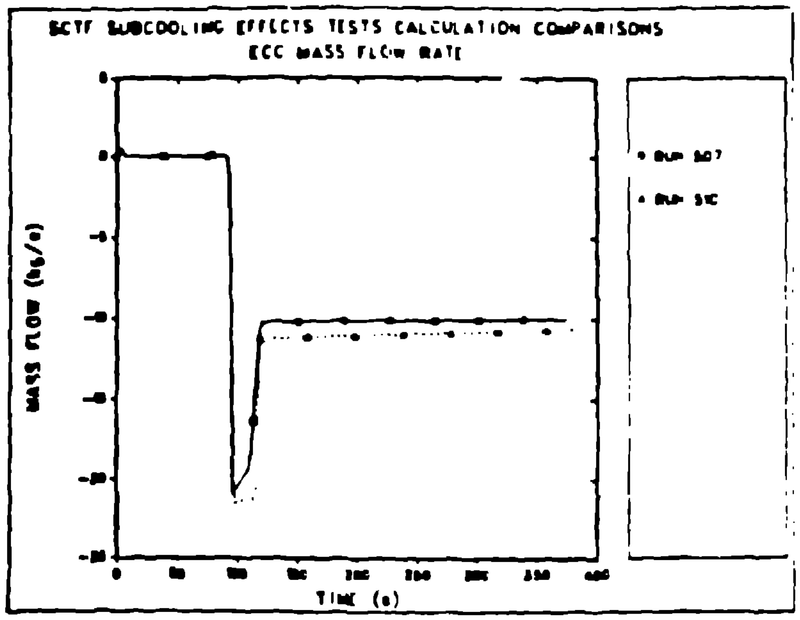


Fig. 6. Comparison of the ECC mass flow rates for Runs 507 and 510.

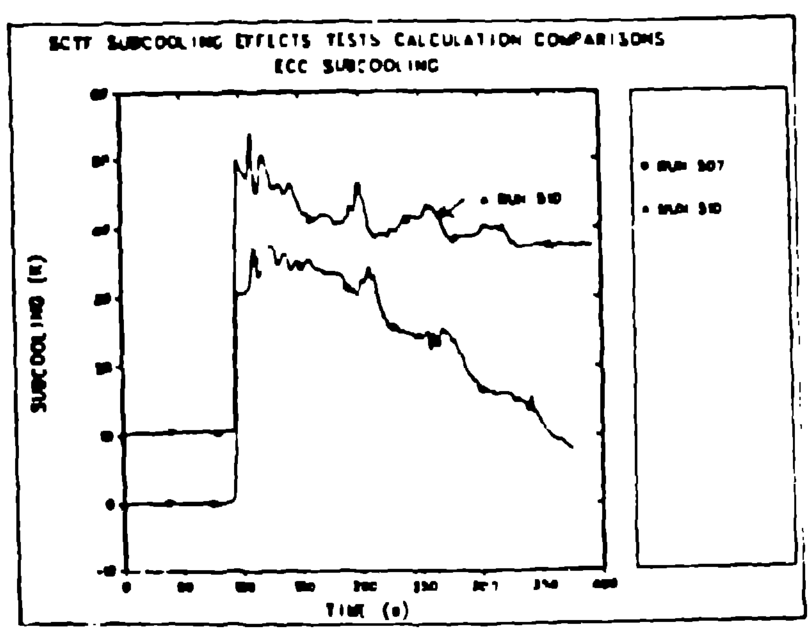


Fig. 7. Comparison of the ECC subcooling for Runs 507 and 510.

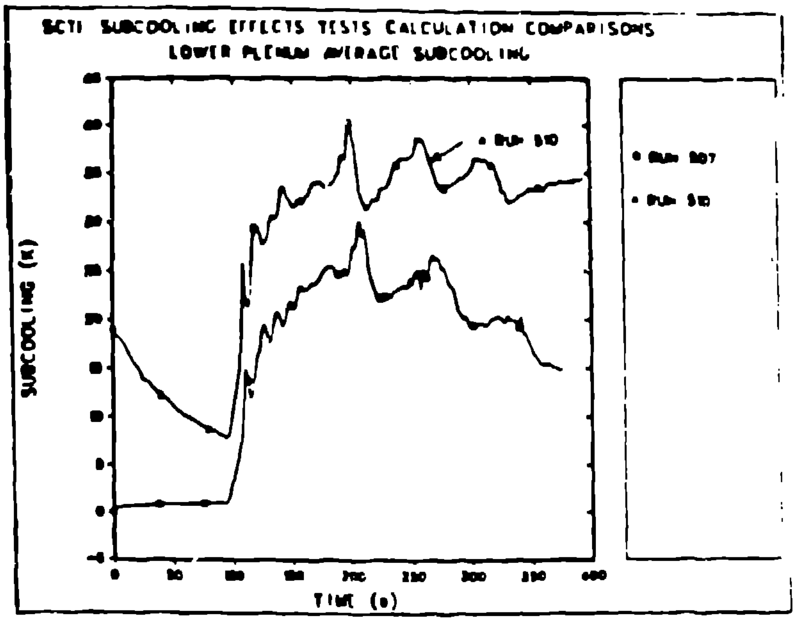


Fig. 8. Comparison of the lower plenum average subcooling for Runs 507 and 510.

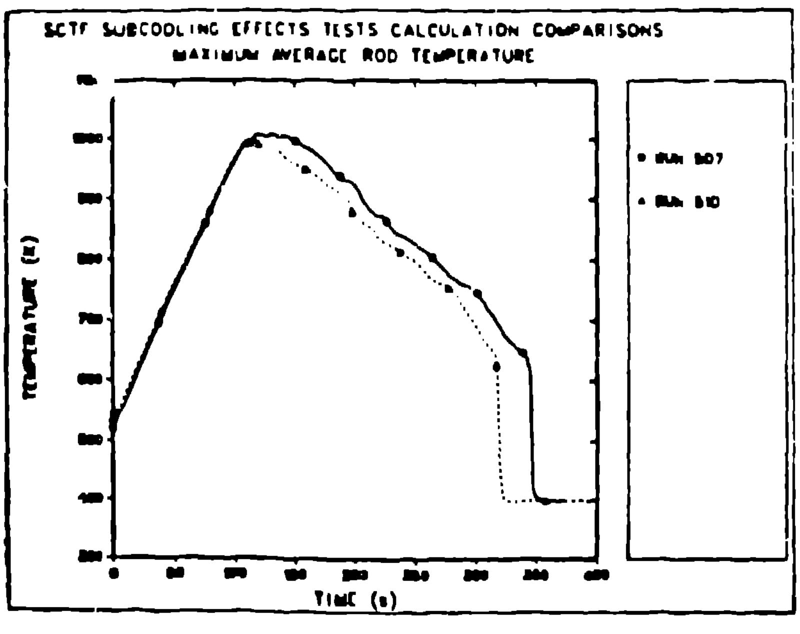


Fig. 9. Comparison of maximum average rod temperatures for Runs 507 and 510.

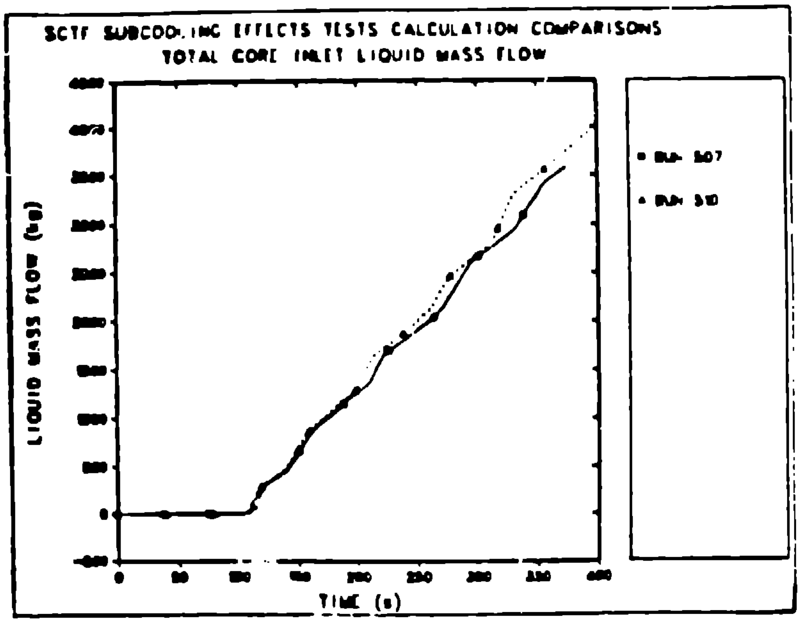


Fig. 10. Comparison of total core inlet mass flow for Runs 507 and 510.

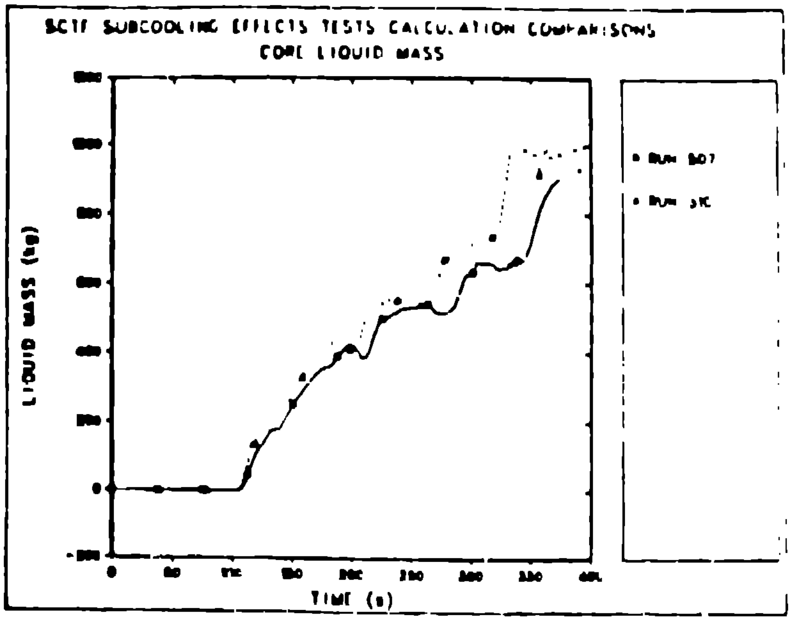


Fig. 11. Comparison of core liquid mass for Runs 507 and 510.

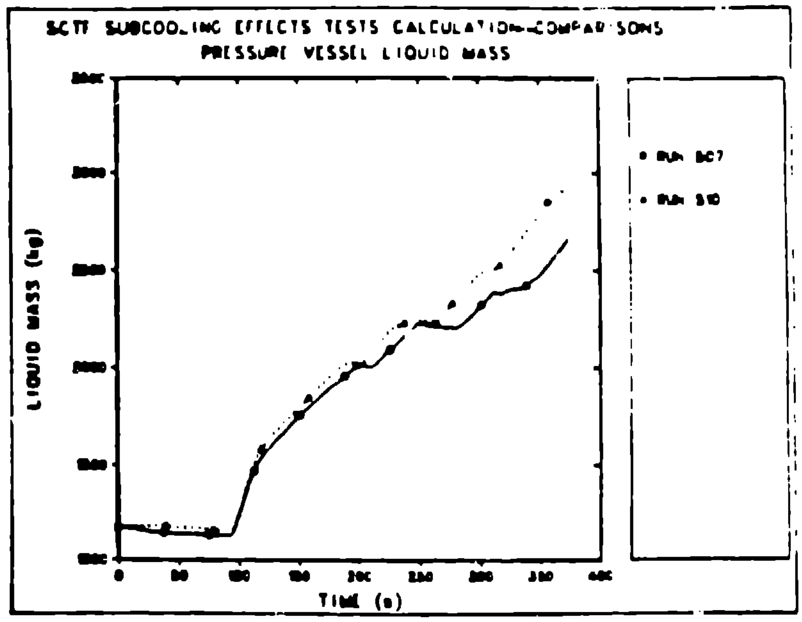


Fig. 12. Comparison of pressure vessel liquid mass for Runs 507 and 510.

collapsed water levels, and the experimental result probably arises from the initial bias of the instruments.

The higher subcooling resulted in subcooled water in the bottom two levels of the core, whereas in the base case the lower plenum liquid remained subcooled, but as soon as this liquid passed into the core, it heated up to saturation. The higher subcooling resulted in more entrainment and more effluent from the core, as shown in Fig. 13; hence, more liquid passed through the hot leg into the separator, as illustrated by the carryover fraction and the separator liquid mass shown in Figs. 14 and 15, respectively. This is in part also attributable to the slightly higher system pressure in Run 510, as the system pressure effects study demonstrated earlier.[3]

Another look at Fig. 14 also shows the greater amount of liquid fallback into the core from the upper plenum in Run 510. The calculation of Run 507 shows a small amount of liquid entering the core at just past 300 s; the calculation of Run 510, however, shows nearly 100 kg of liquid draining back into the core starting at about 325 s.

## III. CONCLUSIONS

The comparison between the calculated results and the data from SCTF Run 510 is very good, particularly for rod temperatures, turnaround times, core differential pressures, mass inventories, and fluid temperatures.

Comparison of this calculation with the base case test (Run 507) shows that the qualitative behavior during reflood is calculated correctly for both cases. The high subcooling had its greatest effect upon quench times when compared with the base case. There was a slight difference in the core differential pressures

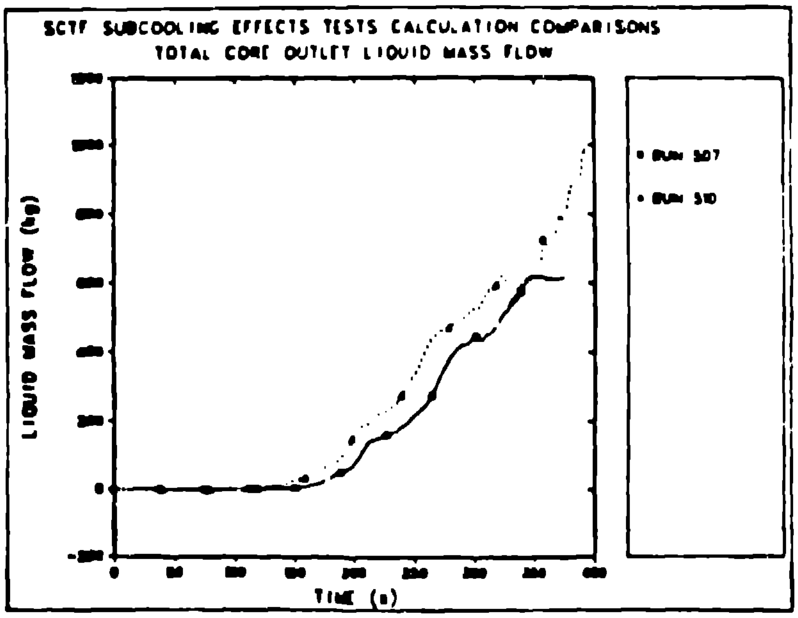


Fig. 13. Comparison of total core outlet liquid flow for Runs 507 and 510.

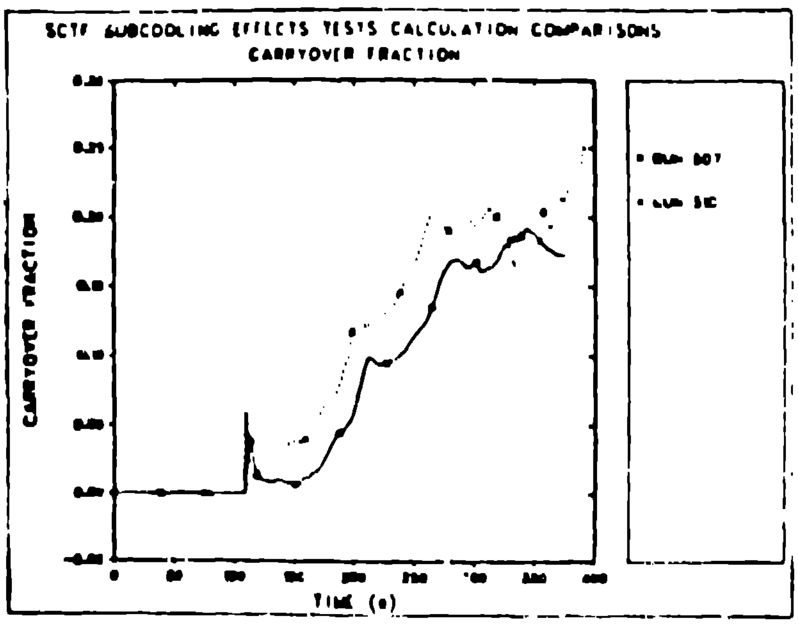


Fig. 14. Comparison of the carryover fraction for Runs 507 and 510.

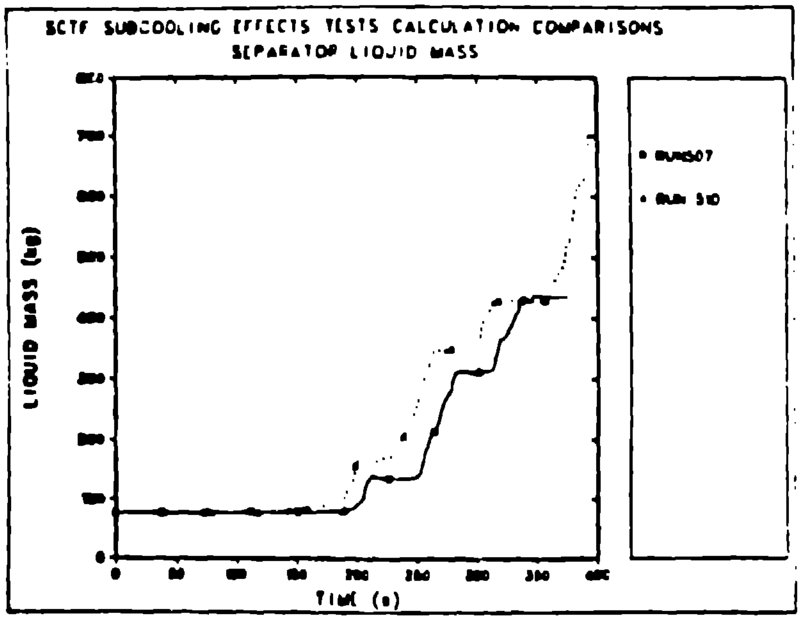


Fig. 15. Comparison of the separator liquid mass for Runs 507 and 510.

when comparing the two runs; the high-subcooling case had more water in the bottom half of the core than did the base case. Also, there was more liquid fallback into the core in the high-subcooling case. Most other parameters behaved in roughly the same way in both Run 510 and Run 507.

In addition, from the comparison of the high-subcooling test with the base-case test, the following conclusions can be drawn: for the high-subcooling case, the peak rod temperature was lower, calculated quench times were earlier, there was more entrainment from and liquid carryover from the core to the upper plenum, and the liquid mass accumulation in both the core and the upper plenum was greater. These are directly attributable to the higher subcooling throughout the system.

From the comparison of the calculation of Run 510 with the experimental data, one concludes that the calculational model and TRAC-PD2/MOD1 can predict well the system responses arising from parametric variations in subcooling conditions.

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