LA-UR--82-2429

DE82 021891

TITLE: TRAC ANALYSIS OF THE SYSTEM PRESSURE EFFECTS TESTS IN THE SLAB
CORE TEST FACILITY

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MASTER

SUBMITTED TO: International Meeting on: Thermal Nuclear Reactor Safety,
Chicago, August 29 - September 2, 1982

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TRAC ANALYSIS OF THE SYSTEM PRESSURE EFFECTS TESTS IN THE SLAB CORE TEST FACILITY

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ABSTRACT

This paper describes the analysis, using the TRAC computer code, of three system pressure effects reflood tests performed during 1981 at the Slab Core Test Facility at the Japan Atomic Energy Research Institute in Tokai, Japan. Comparisons of the calculated results with the experimental data were very good, particularly for rod temperature histories, core differential pressures, mass inventories, liquid carryover, and fluid velocities in the loops. These comparisons indicate that the TRAC code can predict reasonably well the effects of pressure variations in test conditions. This and similar calculations demonstrate that TRAC is a useful tool for the design of nuclear reactor systems and the analysis of system response during postulated accident sequences.

INTRODUCTION

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

The SCTF is part of a multinational program to obtain data, develop improved correlations, and assess best-estimate computer codes 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.

TRAC CODE DESCRIPTION

The analysis tool used for these calculations was 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 capability for both bottom flood and falling film quench fronts; and consistent treatment of entire accident sequences, including the generation of consistent initial conditions. It has been assessed against data from a diverse range of experimental facilities.

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

DESCRIPTION OF THE SCTF

The SCTF is composed of the pressure vessel, primary coolant system, and emergency core cooling (ECC) system. Figure 1 shows an artist's view of the SCTF. The pressure vessel contains the slab core, downcomer, upper and lower plena, core baffle region, and upper head. The facility is full-scale in height 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.

Fig. 1. Artist's view of the Slab Core Test Facility.

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. The accumulator (ACC) and low-pressure coolant injection (LPCI) system is connected to the pressure vessel at the bottom of the lower plenum on the downcomer side.

TEST OPERATING PROCEDURE

The series of forced-flooding tests discussed herein included Runs 506, 507, and 508 (Run 507, the base case test, was the first of the main test series for this new facility). The initial system pressures were 3.97, 1.98, and 1.5 bars, respectively. For all these tests, the downcomer was blocked at the bottom, allowing flow only in the core and bypass regions. The lower-plenum ACC and LPCI systems were operational, but the cold-leg and upper-plenum ECC systems were not. The initial fluid and structure temperatures were the saturation temperature at the given system pressure. The initial power for all three tests was about 7 MW. The axial power distribution was a chopped cosine, and the radial power distribution was 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 of saturated water.

Operation of these tests began by heating the rods electrically until a specified 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 until quench. At 6 s, the power began to decrease according to the standard "ANS + actinides" decay curve.

THE COMPUTATIONAL MODEL

The TRAC computational model developed at Los Alamos for the SCTF uses 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. Figure 2 shows schematics of the SCTF pressure vessel and primary system. The hot leg and the lower-plenum ACC/LPCI injection nozzles were modeled as PIPE components, the intact cold leg with the cold-leg ECC injection nozzle as a TEE component, the two segments of the broken cold leg as VALVE components, the two containment tanks as BREAK components attached to each of the broken cold legs, and the cold-leg and lower-plenum ECC injection capability as FILL components. In total, there were 213 computational cells in the TRAC model.

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

COMPARISON OF THE CALCULATIONS AND THE TEST DATA

All three calculations showed good agreement with the test data, particularly for rod temperature histories, core differential pressures, mass inventories, liquid carryover, and fluid velocities in the loops. The calculations were in close agreement with the temperature data for the lower core regions, where the cooling mechanism for the rods resulted from flooding; quench was calculated earlier than the data for higher core regions, where the cooling mechanism resulted from steam flow with entrained droplets. Maximum cladding temperatures were consistently slightly higher in the calculations than in the data; for example, in the base case (Run 507), the largest difference in maximum midplane rod temperatures was in Bundle 4 (shown in Fig. 3), where the peak calculated temperature was 1005 K and the peak measured temperature was 995 K. The times to reach the maximum cladding temperatures agreed within a few seconds.

Similarly, the differential pressures calculated in the lower half of the core agreed almost exactly with the data; in the core's upper half, a slightly lower differential pressure was calculated than shown by the data. However, the calculated full-height differential pressures agreed well with the data. Figure 4 shows a comparison of the core full-height differential pressures in the experimental data and the calculation for Run 507.

The calculated core liquid mass was slightly less than the data indicated, and the calculated mass in the upper plenum exceeded the data because of the higher calculated entrainment rate, which also contributed to the early calculated quench of the top half of the core. Comparisons of these liquid masses for Run 507 are shown in Fig. 5. In addition, the calculations consistently showed a larger mass

Fig. 3. Comparison of the calculated and experimental cladding temperatures at the ten thermocouple locations for Run 507.

Fig. 4. Comparison of the calculated and experimental core full-height differential pressures for Run 507.

Fig. 5. Comparisons of calculated and experimental liquid mass in the core and upper plenum for Run 507.

accumulation in the separator than the data because of more carryover into the hot leg of the entrained liquid in the calculations.

EFFECTS OF THE SYSTEM PRESSURE VARIATION

The effects of the pressure variation were well-predicted by TRAC, agreeing closely with the trends seen in the tests. The general trend was that the higher the system pressure, the more rapidly the core cooled and quenched.

Maximum cladding temperatures were lower at higher pressure and quench occurred sooner, demonstrated by comparing the quench envelopes for selected bundles for the three runs shown in Fig. 6. The heat-transfer processes were more sedate at lower pressures, with less sputtering steam generation as the quench front progressed through the core. In addition, the lower pressure allowed more liquid fallback into the core from the upper plenum, resulting in more top-down quenching. The higher system pressure produced less liquid carryover into the steam-water separator, as well as less liquid accumulation in the upper plenum, than did the lower pressures. The reason is that, even though higher pressures require lower steam velocities for entrainment, the high-pressure test generated steam more rapidly at the quench front, leaving less liquid for entrainment in the region of the quench front. The calculation of the low-pressure test showed larger steam velocities (because of a lower steam density) with more entrainment than the high-pressure test, so that the core flooding was slower, but still the calculation showed that the core cooled more rapidly than was observed in the experiment. There were some wide-band pressure pulses observed in the calculation that were not seen in the data; the period of the pulses was about the same (50 - 60 s) for all three runs, but the amplitude was considerably less at lower pressure because, for film and transition boiling, the differences in the heat-transfer coefficients are higher at higher pressures [2], indicating more volatile heating processes.

Fig. 6. Comparison of quench times after bottom-of-core-recovery (BOCREC) for Bundles 2, 4, and 8 in Runs 506, 507, and 508.

There was little noticeable effect of the 50% blockages in Bundles 3 and 4 in either the calculations or the data for any of the experiments when the blocked bundles were compared with the unblocked bundles; this was anticipated because of previous experiments [3] at Karlsruhe. However, this may change under gravity-reflood conditions.

CONCLUSIONS

The most important result arising from this study is that both the calculations and the data show that the core quenches sooner at higher pressure. More liquid accumulation in the steam-water separator and more upper plenum liquid pool formation and fallback into the core were seen clearly in both the calculations and the data at lower pressure. In most other areas, the calculations were close to the experimental data, particularly in predicting the trends.

These calculations show that TRAC can predict both the qualitative and quantitative effects of parametric variations in test conditions quite well. This and similar calculations demonstrate that TRAC is a useful tool for the design of nuclear reactor systems and the analysis of system response during postulated accident sequences.

ACKNOWLEDGMENTS

This work was performed under the auspices of the United States Nuclear Regulatory Commission.

Special assistance in plotting the experimental results over our calculated results came from Dr. Yukio Sudo (JAERI) [4], who visited us at Los Alamos during September, 1981, bringing with him the data plots from the first few tests.

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