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FUEL CLADDING TEMPERATURE PREDICTIONS FOR LOFT LOCE L1-5

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SUMMARY

Fuel cladding temperature predictions were performed for loss-of-fluid test[1] (LOFT) loss-of-coolant experiment (LOCE) L1-5[2] using the RELAP4 computer code. Two versions of this code were used, RELAP4/MOD5[3] and RELAP4/MOD6[4]. Measured thermocouple data has been compared to these predictions to determine the relative accuracy of the two RELAP4 computer code versions and their respective blowdown heat transfer correlations selected to describe the fuel cladding surface temperature response during the subcooled and saturated blowdown phase of the LOCE.

LOFT LOCE L1-5 simulated a 200% double-ended offset shear break in the cold leg of a four-loop large pressurized water reactor (LPWR). The initial conditions for the LOCE were: zero power with a nuclear core installed, primary coolant (PC) temperature of 541 K, PC pressure of 15.6 MPa, and PC flow of 176 Kg/sec. The PC pumps were running until the end of the blowdown phase of the LOCE, and cold leg emergency core coolant (ECC) injection initiated at 19 seconds.

The RELAP4 fuel rod model for LOCE L1-5 describes a single fuel rod and the fluid conditions surrounding it. This system is modeled as 14 control volumes and 12 heat slabs (Figure 1). Each heat slab consists of three materials, simulating the fuel pellet, gap, and cladding. The fluid conditions for this model were generated by a detailed RELAP4 system model and input as time dependent upper and lower plenum conditions (control volumes 1 and 14).

A comparison of cladding surface temperature predictions calculated by RELAP4/MOD5 and MOD6, with measured temperature data for a position 28 cm above the bottom of the fuel rod is presented in Figure 2. During the saturated blowdown phase (0.08 - ∿ 25 seconds) of LOCE L1-5 both of the RELAP4 calculations, as well as the measured thermocouple data show that the fuel cladding temperature follow the PC system saturation temperature. The heat transfer (HT) correlations used during this time period, namely, the Thom(3) correlation for RELAP4/MOD5, and the Chen(4) correlation for MOD6, calcualted HT coefficients with relative differences as great as 47%. However, the two REALP4 fuel cladding temperature calculations are nearly identical to each other, and are in good agreement with the measured data.

The calculated heat fluxes for the same position on the fuel rod as the measured temperature data is shown in Figure 3. It is observed on this figure that the RELAP4/MOD6 calculated heat fluxes dropped sharply to zero at 25 seconds and remained near zero (except for some spikes caused by core flow reversals) until the end of the saturated blowdown phase. After 25 seconds the void fraction is calculated to be greater than 0.96, which is the fuel rod dryout criteria used by the RELAP4/MOD6 selected critical heat flux (CHF) correlation, namely, the W-3, Hsu-Berner and Modified Zuber Correlation(4). That is, when the void fraction is greater than 0.96 the correlation calculates CHF to occur by taking into account the liquid deficiency on the fuel rod.

RELAP4/MOD5 calculated heat fluxes exhibited an oscillatory behavior of departure from nucleate boiling (DNB) and return to nucleate boiling (RNB), with the initial occurrence of DNB at 26.5 seconds. This oscillatory behavior stops at approximately 33 seconds and stabilizes into one of the pre-CHF HT modes. The General Electric (GE) Company CHF Correlation(3) was used in RELAP4/MOD5 calculations. This correlation is a function of quality and mass flow rate, and does not take into account the void fraction (a dryout criterion). Consequently, at approximately 33 seconds the RELAP4/MOD5 CHF correlation did not calculate CHF to occur, and selected a high void fraction nucleate boiling HT correlation, which over-predicted heat transfer.

In conclusion, based on data presented, explicit dryout criterions in the CHF correlations used by RELAP4 will result in a better prediction of cladding temperature than CHF correlations without such dryout criterions. In addition, the RELAP4 calculated pre-DNB fuel cladding temperature is not strongly effected by the calculated HT coefficients for LOCE L1-5.

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1. D. L. Reeder, LOFT System and Test Description (5.5 Foot nuclear Core I Loss-of-Coolant Experiments), TREE-NUREG-1208 (July 1978).

2. W. H. Grush, et al., Experiment Prediction for LOFT Nonnuclear Experiment L1-5, TREE-NUREG-1209 (March 1978).

3. Aerojet Nuclear Company, RELAP4/MOD5 - A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems - User's Manual, ANCR-NUREG-1335 (September 1976).

4. EG&G Idaho, Inc., RELAP4/MOD6 - A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactors and Related Systems - User's Manual, CDAP-TR-003 (January 1978).

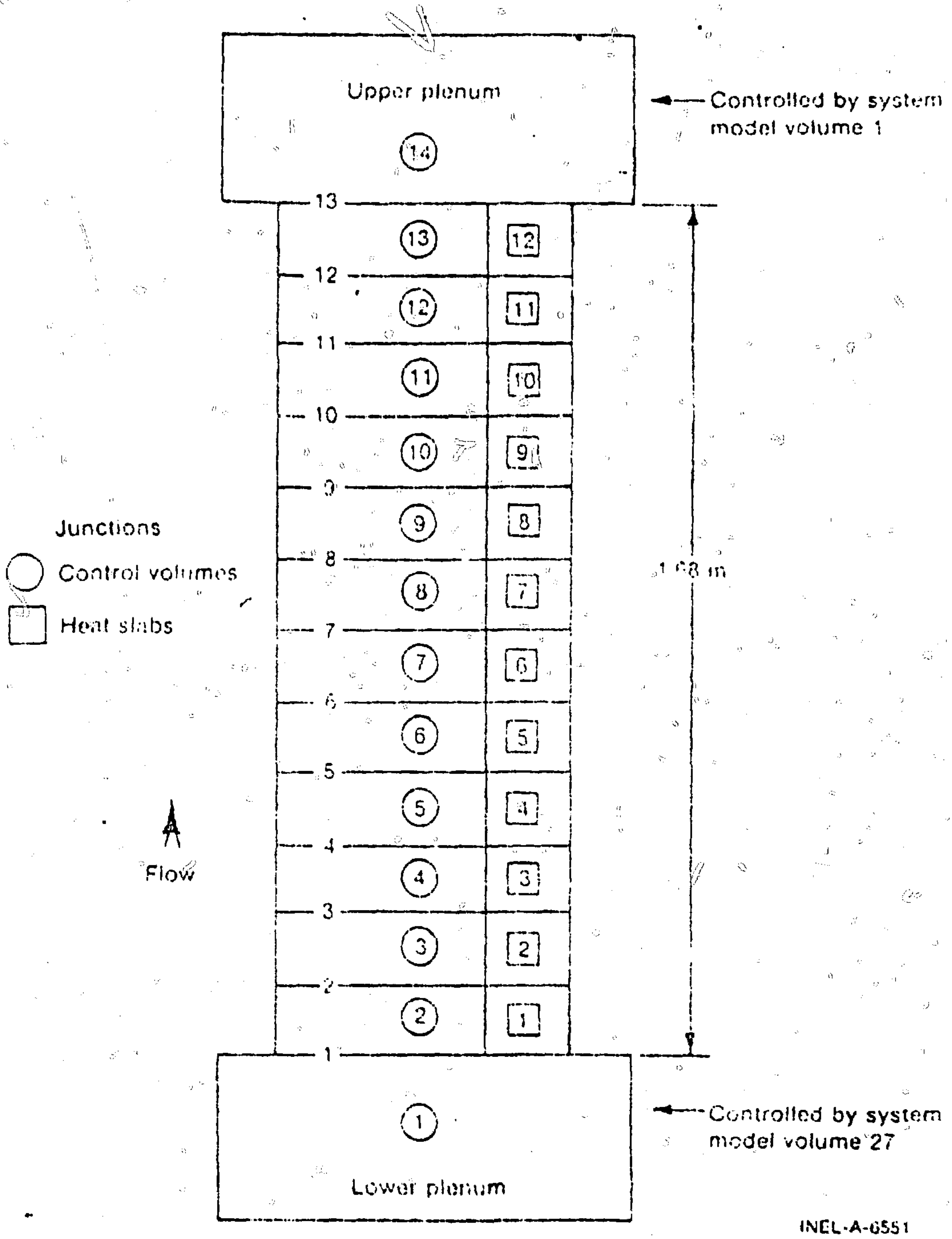

Figure 1. RELAP4 Fuel Rod Model

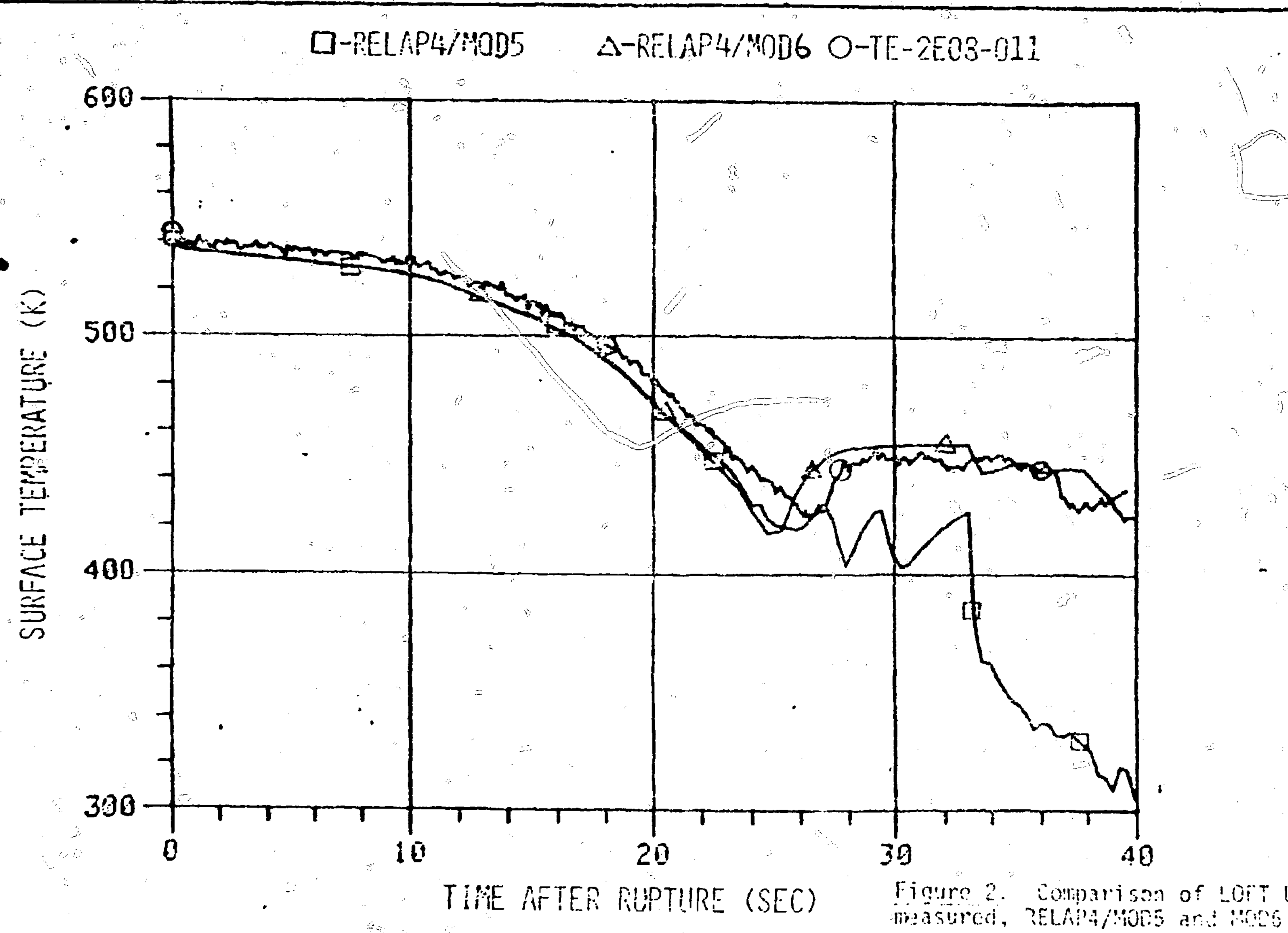

Figure 2. Comparison of LOFT L1-5 measured, RELAP4/MOD5 and MOD6 calculated fuel cladding temperature.

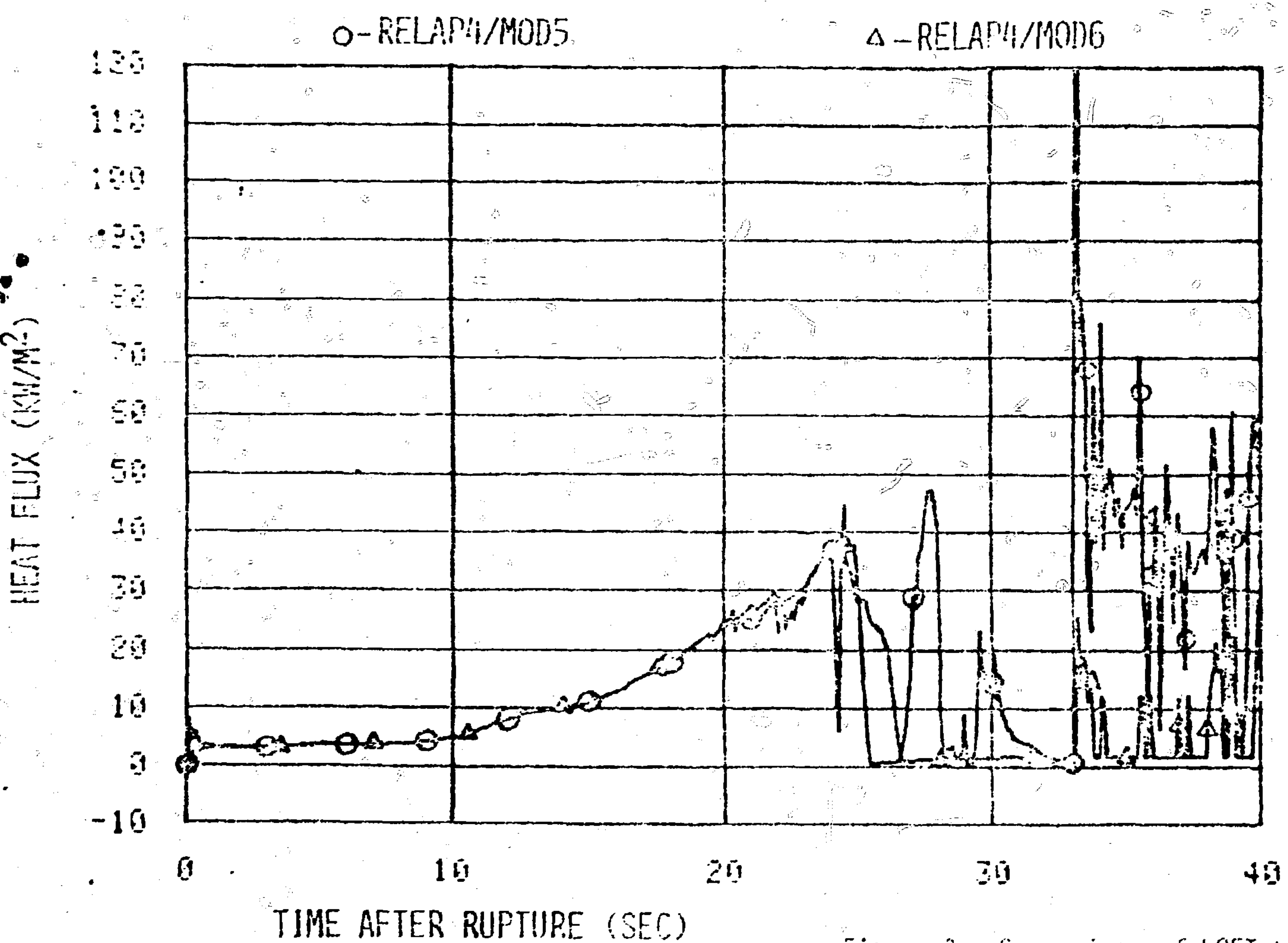

Figure 3. Comparison of LOFT L1-5 RELAP4/MOD5 and MOD6 calculated heat fluxes.