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POSTTEST ANALYSIS OF INTERNATIONAL STANDARD PROBLEM 10 USING RELAP4/MOD7

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INTRODUCTION

RELAP4/MOD7,[1] a best estimate computer code for the calculation of thermal and hydraulic phenomena in a nuclear reactor or related system, is the latest version in the RELAP4 code development series. This paper evaluates the capability of RELAP4/MOD7 to calculate refill/reflood phenomena. This evaluation uses the data of International Standard Problem 10, which is based on West Germany's KWU PKL refill/reflood experiment K9A.[2] The PKL test facility represents a typical West German four-loop, 1300 MW pressurized water reactor (PWR) in reduced scale while maintaining prototypical volume-to-power ratio. The PKL facility was designed to specifically simulate the refill/reflood phase of a hypothetical loss-of-coolant accident (LOCA).

The initial conditions for experiment K9A were that the system contained saturated steam at a pressure of 0.451 MPa and the core power was 1.327 MW. Subcooled emergency core cooling (ECC) water was injected at initation of the experiment at an average rate of 16.74 Kg/s for the first 26 s and was decreased to 1.9 Kg/s thereafter.

In the RELAP4 system modelling, a nodalization of 37 control volumes, 40 junctions, and 50 heat slabs was used to represent the PKL test facility. To closely simulate the core behavior, 36 heat slabs with moving mesh models were used.

RESULTS

Core Thermal Behavior

The temperature of the fuel bundles is of primary concern in the refill/reflood experiment and Figure 1 presents a comparison of calculated and measured maximum cladding surface temperature as a function of core elevation. Clearly shown is the effect of the cosine axial power distribution on the maximum cladding surface temperature. The maximum temperatures were calculated relatively accurately in the lower and middle core regions but the calculated temperature was about 60 K higher than measured data near the top of the core. The turnaround time (time of occurrence

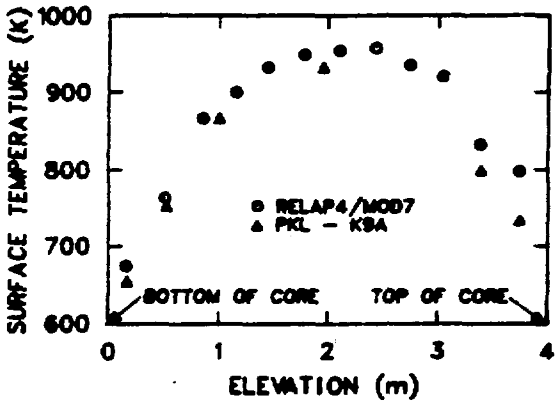

Figure 1. Axial distribution of maximum cladding surface temperature.

of maximum temperature) as a function of core elevation is shown in Figure 2. The calculated times shown were within 10 s over the entire core, except for the top node. The code calculated the quench time well in the lower core region, but the calculated quench time was too early in the middle core region and too late in the upper core region. The early quench in the middle core region was caused by too large a calculated cooldown rate due to oscillations in the core inlet flow. The failure to calculate the top-down quench in the upper core was due to the lack of a top-down quench model in RFLAP4/MOD7.

System Hydraulic Behavior

Figure 3 shows the mass flow rate in the broken loop. The calculated results agreed well in magnitude and trend with the data, except high amplitude and high frequency oscillations were calculated for the first 70 s. Similar oscillations were seen in the intact loop mass flow rate and in the core collapsed liquid level. A major driving force of the oscillations was the steam generation in the core. The calculation of oscillation in the steam generation was partially due to the dispersed flow heat transfer calculation that was influenced by the Steen-Wallis entrainment model. The Steen-Wallis entrainment model calculates entrainment of liquid in the core only when the core inlet flow is positive. Therefore, the entrainment changed instantaneously from on to off, when the core inlet flow changed from inflow to outflow, contributing to the oscillation. Except for these oscillations, the overall system hydraulic response agreed well with data.

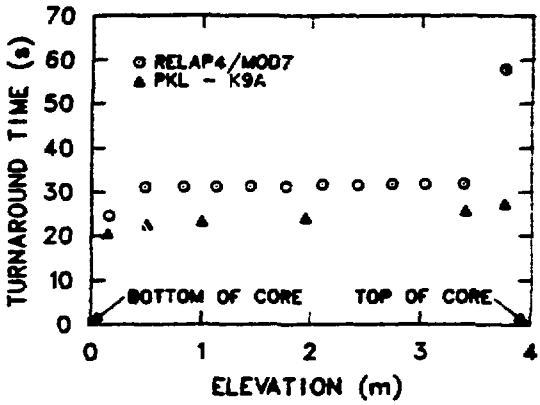

Figure 2. Axial distribution of turnaround time.

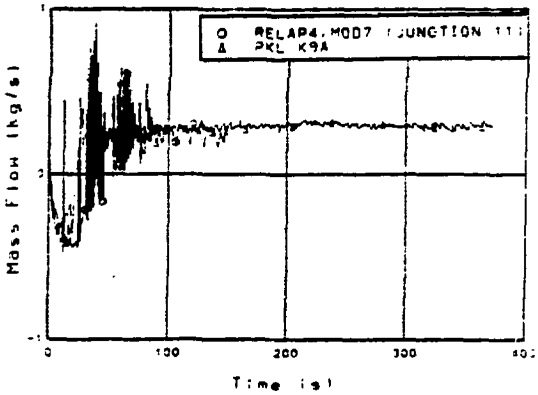

Figure 3. Mass flow rates in the broken loop.

CONCLUSIONS

Conclusions about the capability of RELAP4/MOD7 to calculate the refill/reflood phenomena, derived from this study are as follows:

1. The maximum cladding surface temperatures were accurately calculated (<4% error) except at the top of the core where calculated temperature was 60 K (~8% error) higher than measured data.

2. Turnaround times were calculated accurately (<10 s error) in the entire core except for the top 0.25 m.

3. Quench times were calculated poorly at the core midplane and above.

4. The calculation did not adequately represent the top-down quench observed in the experiment.

5. The calculation adequately represented the general hydraulic trends observed in the experiment.

REFERENCES

1. EG&G Idaho, Inc., "RELAP4/MOD7, a Best Estimate Computer Program to Calculate Thermal and Hydraulic Phenomena in a Nuclear Reactor or Related System" (to be published).

2. B. Brand et al., "Refill Experiment in a Simulated PWR Primary System (PKL)," Specification OECD-CSNI LOCA Standard Problem No. 10: GRS, December 1979.