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EXPERIMENT PREDICTIONS OF LOFT REFLOOD BEHAVIOR USING THE RELAP4/MOD6 CODE

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SUMMARY

The RELAP4/MOD6 computer code was used to predict the thermal-hydraulic transient for Loss-of-Fluid Test (LOFT) Loss-of-Coolant Accident (LOCA) experiments L2-2, L2-3, and L2-4. This analysis will aid in the development and assessment of analytical models used to analyze the LOCA performance of commercial power reactors. Prior to performing experiments in the LOFT facility, the experiments are modeled in counterpart tests performed in the nonnuclear Semiscale MOD 1 facility. A comparison of the analytical results with Semiscale data will verify the analytical capability of the RELAP4 code to predict the thermal-hydraulic behavior of the Semiscale LOFT counterpart tests. This paper describes the analytical model and the results of analyses for the reflood portion of the LOFT LOCA experiments and compares these results with the data from Semiscale.

RELAP4 is a computer program developed to describe the thermal-hydraulic transient behavior of water-cooled nuclear reactors subjected to postulated accidents such as those resulting from loss of coolant, pump failure, or nuclear power excursions. Fundamental assumptions inherent in the thermal-hydraulic equations are that a two-phase fluid homogeneous and that the phases are in thermal equilibrium. Models are available in the code to modify these homogeneous assumptions. The program is sufficiently general to be applied to experimental water reactor simulators and many other hydrodynamic experiments.

RELAP4/MOD6, the latest code version in RELAP4 series, extends the calculational capabilities of RELAP4 from blowdown and refill through core reflood for pressurized water reactor (PWR) systems. Major improvements in RELAP4/MOD6 are as follows:

1. Core superheat model for reflood
2. Moving heat transfer mesh model
3. Implicit and explicit entrainment models
4. Reflood heat transfer correlation set
5. Steam generator natural convection heat transfer correlation
6. Best estimate fuel models

The LOFT facility[1] has been designed to simulate the major components and system responses of a PWR during a LOCA. The test assembly is comprised of five major subsystems which have been instrumented such that desirable system parameters can be measured and recorded during a loss-of-coolant experiment (LOCE). The subsystems are described as follows:

1. The LOFT reactor vessel simulates the reactor vessel of a PWR. It has an annular downcomer, lower plenum, lower core support plates, a nuclear core and an upper plenum. The LOFT core contains 1300 rods arranged in five square and four triangular fuel modules. The fuel rods have an active length of 167.64 cm and an outside diameter of 1.07 cm.

2. The intact loop simulates the three unbroken loops of a large PWR. This loop contains a steam generator, two circulating coolant pumps connected in parallel, a pressurizer, a venturi flowmeter, and connecting piping.

3. The broken loop simulates the broken loop of a large PWR. It consists basically of hot and cold legs that are connected to the reactor vessel and the blowdown suppression tank header. Each leg consists of a break plane orifice which determines the break size to be simulated, a quick-opening blowdown valve which simulates a pipe break, a recirculation line, an isolation valve, and connecting piping.

4. The blowdown suppression system simulates the containment back pressure of a large PWR.

5. The LOFT emergency core cooling system (ECCS) simulates the ECCS of a large PWR. The accumulator, the high and low pressure injection systems were used during the L2 series tests. Each system was configured to inject scaled flow rates of emergency core coolant directly into the primary coolant system cold leg.

LOFT test series L2 is a series of LOCE's conducted at gradually increasing power levels to determine the nuclear core and reactor system's response to the depressurization, blowdown, and reflood transients. LOCE's L2-2, L2-3 and L2-4 are the subject of this experiment prediction analysis. Each LOCE will simulate a 200% (100% of the break area in each leg) double-ended offset shear in the cold leg of a four-loop PWR. The plant operating conditions will be essentially the same for each of these LOCE's except for the reactor maximum linear heat generation rates which will be 26.2, 39.4, and 52.5 kW/m for LOCE's L2-2, L2-3 and L2-4, respectively.

The RELAP4/MOD6 model of the LOFT thermal-hydraulic system is defined with close correspondence to the actual LOFT system. A schematic of the LOFT reflood system model for LOCE L2 series is shown in Figure 1. The model consists of 18 control volume and 31 junctions. The actual geometric data were used for the control volumes. Measured pump performance data were input for the pump model. The calculated loss coefficients were used for junctions.

A one-dimensional radial heat conductor (heat slab) model is used in RELAP4 to calculate heat transfer from metal walls and fuel rods to the fluid. To model the fuel rod, twelve heat slabs were used in the system model.

The fuel rod temperatures at the end of refill (the total liquid mass in the system equals to the total mass required to fill the lower plenum, downcomer of the reactor vessel, and the intact loop cold leg pipe) were used as initial fuel temperatures. The system pressure was set to containment simulator (suppression vessel) pressure.

After the system response is computed, the cladding temperature response of the highest powered fuel rods in the various assemblies was predicted using separate fuel rod models. Figure 2 presents a schematic of the fuel rod model which illustrates the control volumes, junctions, and heat slabs representing the single fuel rod and its adjacent fluid volume. Time dependent conditions (fluid density, pressure and temperature) conditions for the upper and lower reactor vessel plena generated with a RELAP4 system run were used as input to this model.

The predicted results for LOFT LOCE's L2-2, L2-3, and L2-4 are in good agreement with the data from the LOFT counterpart tests performed in Semiscale. The LOFT L2 series tests will provide a final check of the analytical model.

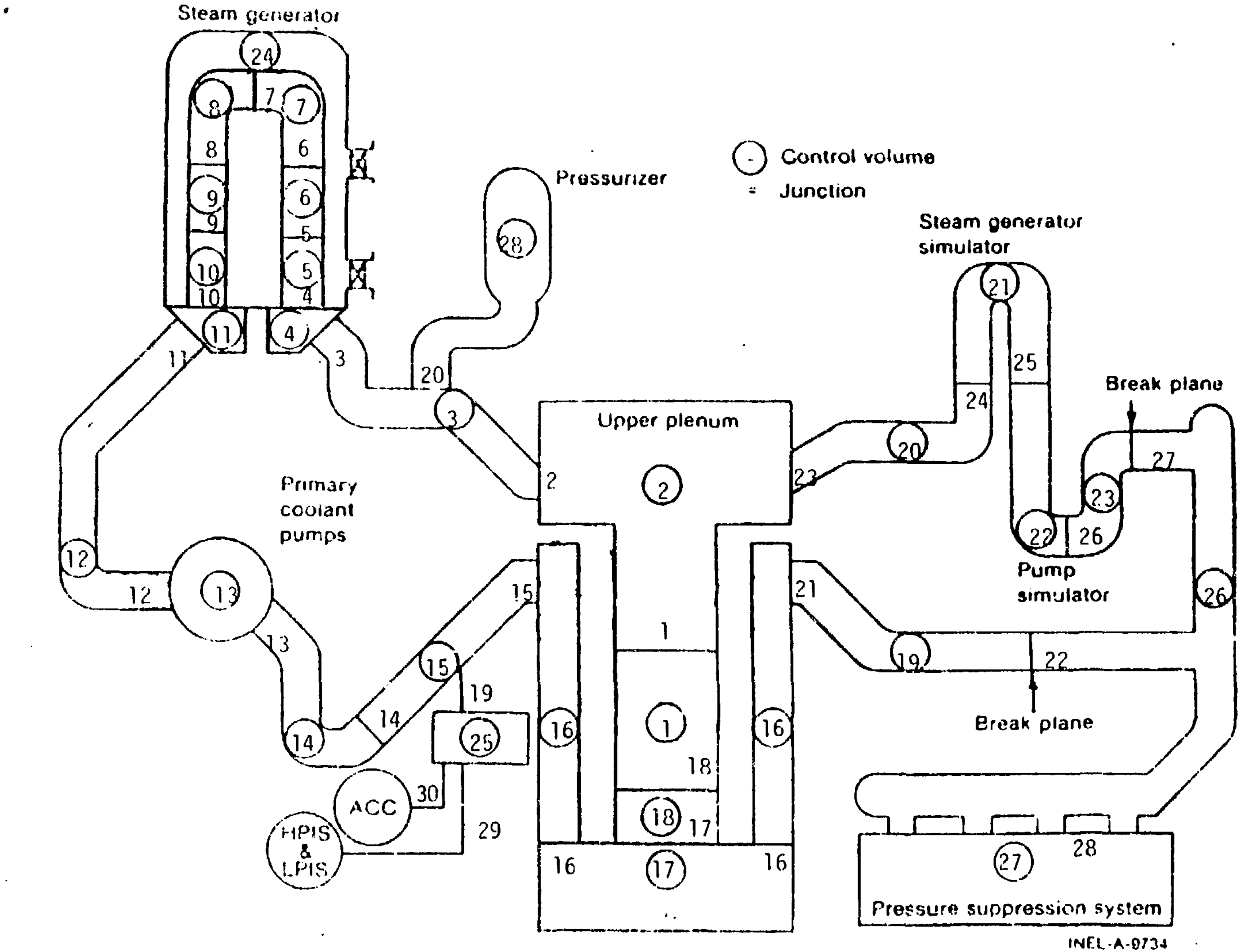

Figure 1 RELAP4 System Model Schematic

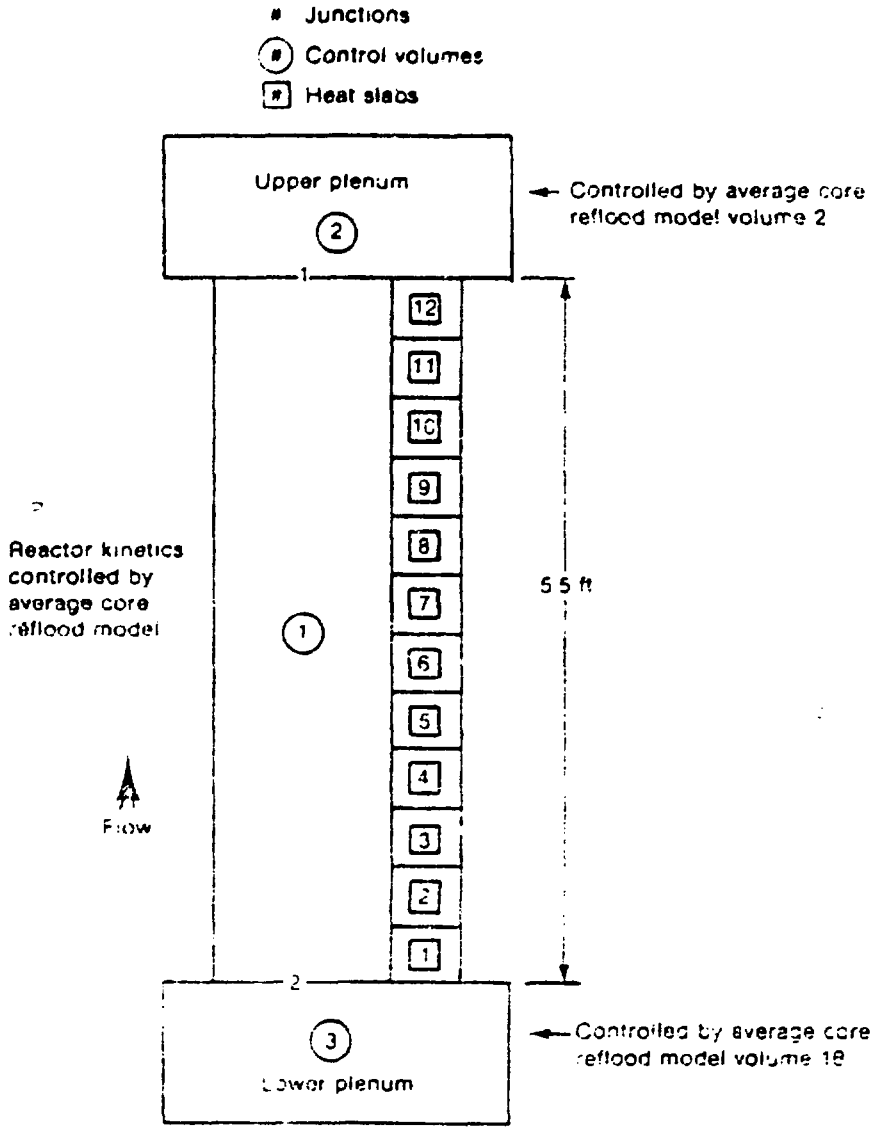

Figure 2 RELAP4 Fuel Rod Reflood Model Schematic

REFERENCES

1. D. L. Reeder, LOFT System and Test Description (5.5 ft Nonnuclear Core 1 LOCE's), NUREG/CR-0247, TREE-1208 (July 1978)