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# LOFT FUEL DESIGN AND OPERATING EXPERIENCE

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## Purpose

The objective of the LOFT fuel design and fabrication effort was to provide a pressurized water reactor core that has (1) materials and geometric features to ensure that heat transfer, hydraulic, mechanical, chemical, metallurgical and nuclear behaviors are typical of large pressurized water reactors (PWR) during the loss-of-coolant accident (LOCA) sequence and (2) test instrumentation for measurement of core conditions. The LOFT core is unique because it is designed for exposure to several LOCAs without loss of function. This paper summarizes the design effort and the extent to which the design objectives have been achieved.

## Physical Description

A typical instrumented LOFT fuel module is a 5.3-m-long, 800-kg assembly composed of the fuel bundle (core section), upper support structure, and instrumentation penetration subassembly. Six instrumented and three non-instrumented fuel bundles are arranged as shown in Figure 1 to compose the LOFT core.

The fuel bundle is modeled after a typical commercial 15x15 fuel-rod-array fuel assembly, however, some compromise was necessary. The core length is 1.68 m because of reactor size constraints. Guide tubes are stainless steel instead of the conventional zircaloy for improved column strength during blowdown loading. Control rod deceleration is provided by a dashpot in the control rod drive mechanism. The fuel rods are not prepressurized to meet the requirement for exposure to repeated LOCAs. One replacement center fuel module will feature prepressurized fuel rods and zircaloy guide tubes.

Instrumentation attached to the fuel bundles in the core region is shown in Figure 1. Instruments attached to the fuel modules in the upper plenum include 33 coolant thermocouples, 10 upper structure thermocouples, 3 drag disc turbine flowmeters, 2 free field pressure detectors, a conductivity liquid level detector and 2 axial motion detectors.

All LOFT fuel modules were designed and fabricated using a comprehensive quality assurance program to assure complete characterization and documentation[1]. The fuel characterization program included determination of critical heat flux (CHF), interchannel thermal mixing coefficients, column strength, flow distribution, control rod drop time, pressure drop, fretting corrosion resistance and fuel pellet densification stability. Significant features of the characterization program include:

1. A 5 to 28% CHF penalty is attributed to the presence of the fuel rod surface thermocouples[2].

2. The time to CHF during a LOCA is not significantly affected by the fuel rod surface thermocouples[3].

3. The room temperature column load capability is 10,000-kg load for the skeleton (guide tubes, spacer grids, and fuel rods) and 4,500-kg load for the upper end box.

4. Maximum misalignment and flow conditions only slightly affect control rod drop times.

5. Lower tie plate induced flow maldistributions are essentially dissipated after passing through the first two spacer grids.

6. The fuel densification characteristic is moderately unstable.

## Operating Experience

Since core loading in October 1977, the fuel has been exposed to primary system operation, critical experiments, a loss-of-coolant experiment (LOCE L1-5) without nuclear heat[4], power range testing to 53 kW/m peak fuel rod linear heat rate and a loss-of-coolant experiment (LOCE L2-2) from a 26.4 kW/m peak fuel rod linear heat rate operation condition. The loss-of-coolant experiments were double-ended inlet line break simulations and included dropping the control rods during the LOCE as a natural function of the plant protection system. The peak fuel rod cladding temperature measured during LOCE L2-2 was 787K. The control rods dropped without hesitation in both LOCE L1-5 and LOCE L2-2 and the vertical loadings on the fuel module during the subcooled decompression were insufficient to induce any residual deformation. Evaluation of data from the loss-of-coolant experiments and posttest core thermal-hydraulic and neutronics recharacterization tests indicates the fuel has not been structurally damaged. No leaks occurred in the fuel rod cladding or instrument cable pressure boundary penetrations. At the completion of LOCE L2-2, 319 of the original 342 measurements on the fuel modules were still functioning.

## Conclusion

The data obtained to date from the instrumented LOFT fuel is providing encouraging indications that fuel design objectives will be achieved and significant PWR LOCA fuel information will be obtained from the LOFT test program.

References

1. M. L. Russell, LOFT Fuel Modules Design, Characterization and Fabrication Program, TREE-NUREG-1131, EG&G Idaho, Inc. (June 1977).

2. S. A. Eide and R. C. Gottula, Evaluation and Results of LOFT Steady State Departure from Nucleate Boiling Tests, TREE-NUREG-1043 EG&G Idaho, Inc. (April 1977).

3. R. C. Gottula, LOFT Transient (Blowdown) Critical Heat Flux Tests, TREE-NUREG-1240, EG&G Idaho, Inc. (April 1978).

4. M. S. Jacoby, Experimental Data Report for LOFT Nonnuclear Test L1-5 (Isothermal Test with Core 1 Installed) TREE-NUREG-1215, EG&G Idaho, Inc. (June 1978).

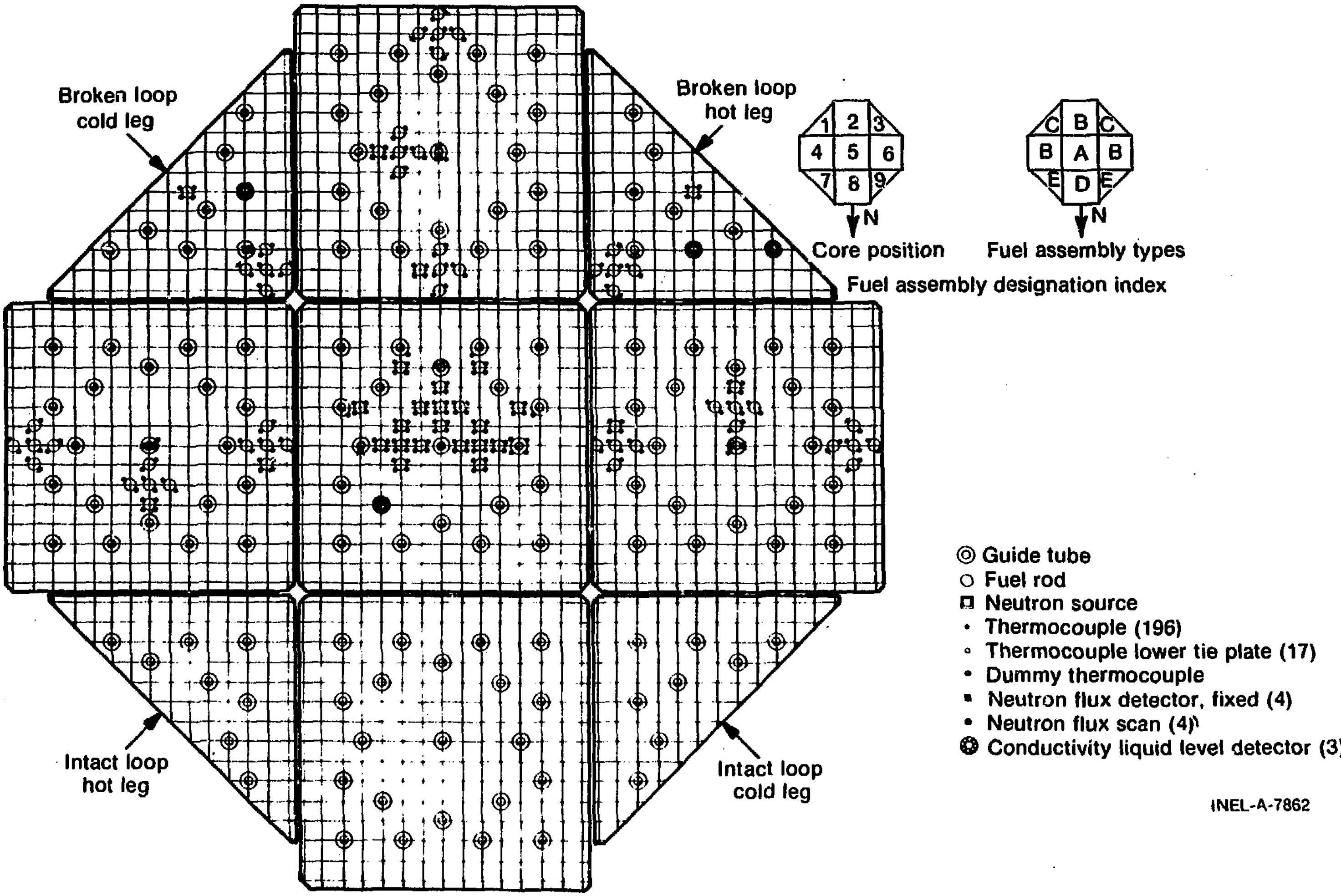


FIG 1 - LOFT CORE CONFIGURATION AND INSTRUMENTATION LOCATION