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LOFT INSTRUMENTED FUEL DESIGN AND OPERATING EXPERIENCE

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ABSTRACT

A summary description of the Loss-of-Fluid Test (LOFT) system instrumented core construction details and operating experience through reactor startup and loss-of-coolant experiment (LOCE) operations performed to date are discussed. The discussion includes details of the test instrumentation attachment to the fuel assembly, the structural response of the fuel modules to the forces generated by a double-ended break of a pressurized water reactor (PWR) coolant pipe at the inlet to the reactor vessel, the durability of the LOFT fuel and test instrumentation, and the plans for incorporation of improved fuel assembly test instrumentation features in the LOFT core.

THE OBJECTIVE OF THE LOFT fuel design and fabrication effort is to provide an experimental pressurized water reactor core that has (a) materials and geometric features that ensure heat transfer, hydraulic, mechanical, chemical, metallurgical, and nuclear behavior are typical of those of large PWRs during the hypothesized loss-of-coolant accident (LOCA) sequence and (b) test instrumentation for measurement of core conditions. The LOFT core is unique because it is designed to be subjected to several LOCAs without loss of function. The program for designing and fabricating replacement fuel modules for LOFT provides a method for incorporating improvements in contemporary PWR fuel design and in-reactor measurement techniques. This paper summarizes the design effort and the extent to which the design objectives have been achieved.

FUEL MODULES AND ATTACHED INSTRUMENTATION DESIGN FEATURES

A typical instrumented LOFT fuel module, shown in Figure 1, 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 noninstrumented fuel bundles are arranged as shown in Figure 2 to compose the LOFT core.

The LOFT fuel bundle was modeled after a typical commercial 15 x 15 fuel-rod-array fuel assembly; however, some compromise was necessary. The core length is restricted at 1.68 m because of reactor size constraints. For improved column strength during blowdown loading, guide tubes are stainless steel instead of the conventional zircaloy.

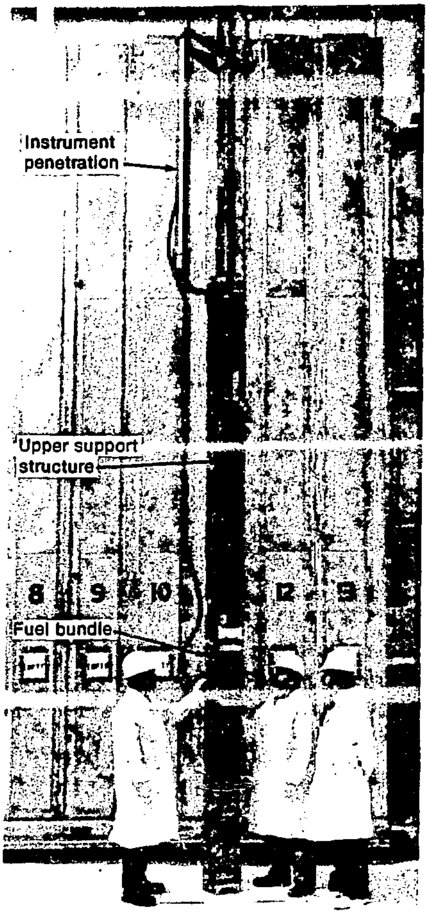

Fig. 1. - Typical instrumented LOFT fuel module

Control rod deceleration is provided by a dashpot in the control rod drive mechanism. To meet the requirement for exposure to repeated LOCAs, the initial core fuel rods installed for the first series of experiments are not prepressurized. However, one replacement center fuel module featuring prepressurized fuel rods and zircaloy guide tubes will be installed for a later experiment.

The instrumentation features of presently installed fuel modules are provided in Table 1. The instrumentation is designed to provide fuel assembly structural response information as well as the more emphasized thermal-hydraulic information. The arrangement of instrumentation attached to the fuel bundles in the core region is shown in Figure 2. 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. Reed switches (40) attached to the control rod drive mechanism housings provide data for determining the time of the gravity drop of the control rods during the LOCE.

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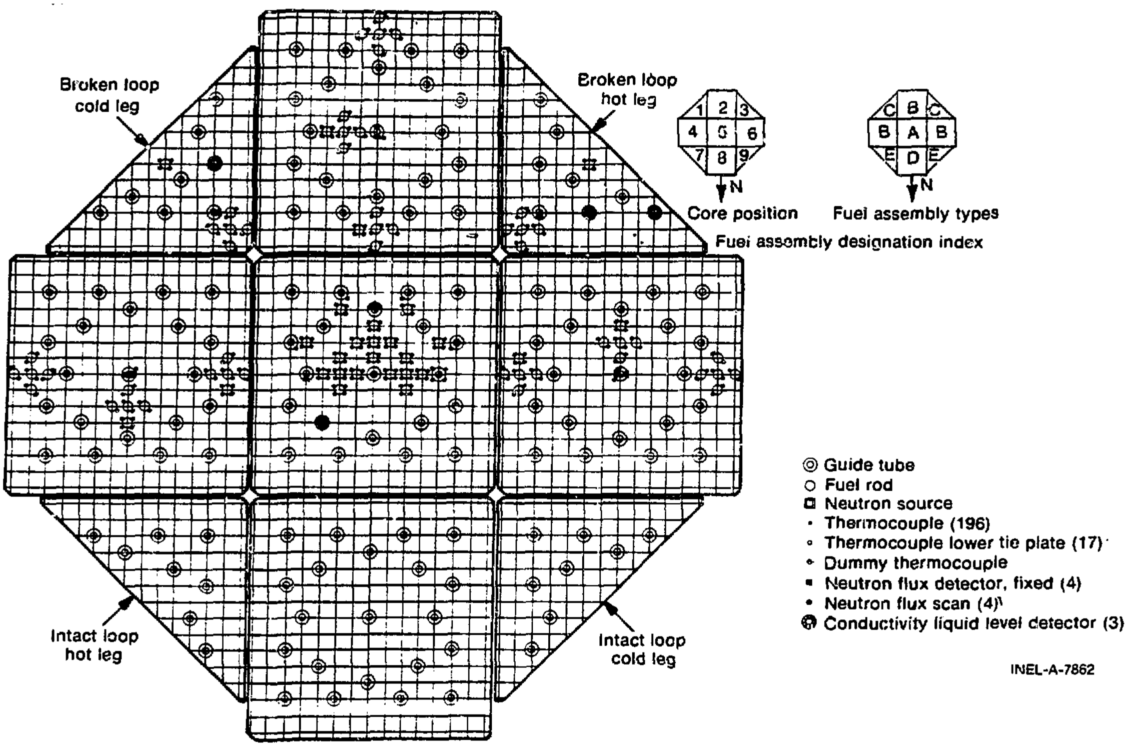

Fig. 2. - LOFT core configuration and instrumentation location

FUEL MODULE DESIGN CHARACTERIZATION

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 results 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 load capability is 10 000-kg load for the skeleton (guide tubes, spacer grids, and fuel rods) and 4500-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) Flow induced vibration damage will not occur in the fuel bundles during steady state operating conditions.
(7) Fuel bundle exit modifications were required to obtain a radially uniform flow rate at the core outlet.
(8) The fuel densification characteristic is moderately unstable.

The densification stability characteristic of the LOFT fuel is different from that of the fuel being fabricated using current fabrication techniques. The LOFT fuel pellets, which were fabricated in 1972, densified approximately three times as much as modern PWR fuel in the standard resintering test. The effect of the difference in densification characteristics during a LOCE will be to cause the peak fuel rod cladding temperatures in LOFT to be slightly higher than would be expected using a modern, more stable fuel pellet.

LOFT CORE RESPONSES DURING LOCE TRANSIENTS

Since core loading in October 1977, the LOFT fuel has been exposed to primary system operation, critical experiments, a LOCE (LOCE L1-5) without nuclear heat(4), power range testing to 53-kW/m peak fuel rod linear heat generation rate, and a LOCE (LOCE L2-2) with a 26.4-kW/m peak fuel rod linear heat generation rate operation condition at test initiation(5). At present the LOFT fuel has achieved a peak fuel burnup of 833 MWd/MTU. No leaks are presently detectable in the fuel rods as determined by negligible iodine levels in the primary coolant, and 323 of 346 fuel module experimental measurement installations are still functioning.

* Numbers in parentheses designate References at end of paper.

Table 1 - LOFT Core 1 Fuel Module Instrumentation Features

Test Information	Measurement Device	Quantity
Fuel rod surface temperature	Thermocouple	185
Neutron flux	Self-powered neutron detector	4
Neutron flux axial profile	Traversing fission chamber	4
Guide tube inside surface temperature	Thermocouple	11
Core coolant density/water level	Electrical conductivity probes	57/3
Core inlet coolant temperature	Thermocouple	17
Core outlet coolant temperature	Thermocouple	24
Core outlet coolant velocity	Turbine/eddy current sensing coil	3
Core outlet coolant density	Drag disc/torsion bar/diff. motion transformer	3
Core to upper plenum pressure drop*	Bonded strain gage transducer, external	1
Upper plenum coolant pressure*	Bonded strain gage transducer, external	2
	Bellows, strain post, and strain gage transducer, internal	2
Upper plenum coolant temperature	Thermocouple	9
Upper plenum coolant density/water level	Electrical conductivity probes	9/1
Upper support structure wall temperature	Thermocouple	10
Fuel module length and acceleration	Linear variable differential transformer (LVDT)	2
	Total Measurement Locations	346*

* Six pressure taps to external detectors.

FUEL ROD TEMPERATURE RESPONSE - The peak fuel rod cladding temperatures measured during LOCE L2-2 was 787 K compared to the best-estimate prediction of 1020 K, which was calculated by the LOFT Experimental Program Division, and the safety evaluation prediction of 1377 K. The best-estimate prediction is based on models and material behavior correlations judged to be the most accurate and most of which have been verified by experimental results. The safety evaluation prediction is based on models and material behavior correlations that represent an adverse accumulation of (a) manufacturing deviations that would occur from nominal design values and (b) uncertainties in experimentally derived material behavior properties. The unexpected lower peak cladding temperature measured during LOCE L2-2 is believed to be caused by the presence of more liquid within the core boundaries than was expected during the initial 12 seconds of the LOCE. Events in rates of coolant delivery to and ejection from the reactor vessel caused a partial replenishment of liquid in the core region which appears to have caused a core wide rewetting of fuel rod surfaces about 8 seconds after break initiation. The greater-than-anticipated transfer of fuel rod stored energy to the core coolant during the initial 12 seconds is not yet fully explained. Accuracy tests of the LOFT fuel rod cladding surface thermocouples and LOCE L2-2 evidence that almost no stored energy existed in the fuel rods at commencement of the reflood phase indicate that the measured fuel rod cladding temperatures during LOCE L2-2 are accurate.

FUEL MODULE STRUCTURAL RESPONSE - The structural response of the fuel modules during the LOCE L1-5 and LOCE L2-2 events was similarly less severe than predicted. Figure 3 compares the LOCE L1-5 displacement data obtained from the linear variable differential transformer (LVDT) with the compression of the center fuel module holddown spring as predicted using the SHOCK computer code(6) lumped-mass-spring model of the LOFT fuel modules and support structures. The data from LOCE L2-2 are similar. An experimentally determined pressure correction factor is applied to the SHOCK prediction to account for the actual relaxation of the reactor vessel head components during the rapid LOCE

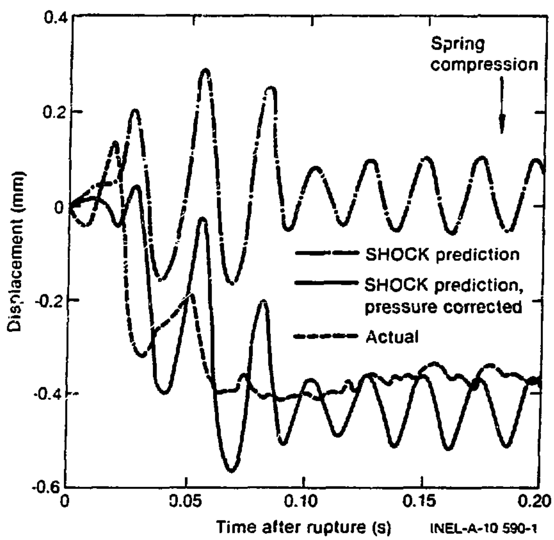

Fig. 3. - Center fuel module holddown spring displacement during LOCE L1-5 subcooled blowdown (uncertainty analysis for displacement data is contained in Reference 8)

depressurization, since the SHOCK model is based on the assumption that the reactor vessel head is stationary. The experimental data indicate that the SHOCK code reasonably predicts the displacement frequency and peak-to-peak amplitude of the initial response cycle which occurs during the subcooled blowdown phase of the LOCE. The more rapid attenuation of the cyclic motion is due to either an underestimation of the damping that occurs in the actual structural system or a similar attenuation of the hydraulic forcing function. Comparison of the pre- and posttest LVDT signals indicate that no residual deformation of the center fuel module occurred during LOCE L2-2.

The loads predicted by the SHOCK code were converted to stresses using a finite element model of the fuel bundle in the SAP-IV computer code(7). The predicted stresses did not exceed 60% of those allowed by the American Society of Mechanical Engineers (ASME) code.

CONTROL ROD DROP RESPONSE - The control rods were released during LOCE L2-2 as a natural function of the plant protection system. The inherent system time delay causes the holding current to be broken after the subcooled-blowdown hydraulic forces have subsided. Figure 4 shows the LOFT control rod drop time data for LOCEs L1-5 and L2-2. The control rod drop is measured by four magnetic-reed switches located above the control rod drive motors.

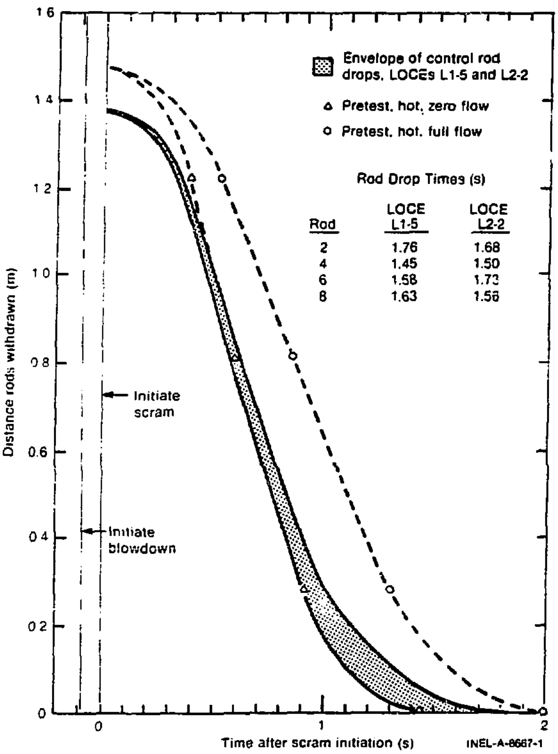

Fig. 4. - LOFT control rod drop times during LOCE scrams (rod position uncertainty of ±0.006 m based on data from reed switch instrument supplier)

The drop time measured is consistent with that expected under zero flow conditions. No indications were observed of mechanical interaction caused by guide tube deformation, lateral fuel assembly motion during the saturated phase of the blowdown, or less-than-expected drag forces, which would be an effect of loss of liquid from the guide tubes before entry of the control rods. A computer code analysis indicates that liquid will remain in the guide tube during the control rod drop period. This result is of significance when deceleration of the control rods is by the necked-down lower section of guide tube which occurs in some conventional PWRs.

FUTURE INSTRUMENTATION IMPROVEMENTS

A comprehensive program is currently underway to develop, fabricate, and install measurement devices for the following: (a) core inlet and outlet coolant void fraction, (b) core inlet coolant velocity, (c) fuel centerline temperature, (d) fuel rod length, and (e) fuel rod internal plenum gas pressure and temperature. Westinghouse Hanford Co. is developing the fuel rod measurement devices, EG&G Idaho, Inc. is developing the coolant measurement devices, and Exxon Nuclear Company is developing the instrument attachment to the fuel bundles. The improved measurement features will be first included in the LOFT tests conducted in 1980.

CONCLUSIONS

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. The LOFT test data indicate that conventional fuel bundle structural analysis techniques are valid for predicting effects of a LOCE and that the LOCA hydraulic forces do not cause residual deformation of the fuel bundle or disturb the normal gravity drop of the control rods. Also, the LOFT replacement fuel program has developed promising devices for measuring core inlet coolant conditions, centerline fuel temperature, fuel rod length, and fuel rod internal gas conditions during a LOCE.

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