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FOR CONSIDERATION FOR INCORPORATION IN THE
CSNI* STATE-OF-THE-ART REPORT (SOAR)
ON FUEL BEHAVIOR DURING A LOCA

FUEL BEHAVIOR DURING A LOCA -
LOFT EXPERIMENTS

By

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* CSNI - Committee on the Safety of Nuclear Installations

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FUEL BEHAVIOR DURING A LOCA - LOFT EXPERIMENTS

1. LOFT RESEARCH OBJECTIVES

The Loss-of-Fluid Test (LOFT) Program mission is presently, "To establish conditions characteristic of postulated pressurized water reactor accidents and to test and develop methods for analytical description, for accident recognition, and for manual and automatic plant stabilization and recovery." Derived from the mission statement are the following current principal points of program interest:

1. Creating an experimental data base which reflects a wide spectrum of reactor accident phenomena and plant states.
2. Developing, using, and evaluating methods to recognize, characterize, control, and recover from accident conditions.

2. FACILITY DESCRIPTION

The LOFT Facility¹ features an operating 50 MWth pressurized water reactor (PWR) system, an elaborate measurement system which records the response of the reactor to the transients, a satellite dual CYBERNET computer system for data process and evaluation and a satellite shielded facility for reactor autopsy and refurbishment. Fuel assemblies² are modeled after a typical commercial 15 x 15 fuel-rod-array fuel assembly; however, some differences do exist: (1) the core length is restricted to 1.68 m because of reactor size constraints, (2) the guide tubes are stainless steel instead of the conventional zircaloy for improved column strength during decompression loading, (3) the control rod deceleration is provided by a dashpot in the control rod drive mechanism, and (4) the initial core fuel rods are unpressurized to meet the requirement for

repeated loss-of-coolant experiments (LOCEs). Center fuel assemblies having prepressurized fuel rods and zircaloy guide tubes will be installed for later experiments.

LOFT is unique for a number of reasons:

1. The combination of facility and personnel permits intentional placement of an operating nuclear reactor into accident conditions, to measure responses, and to recover.
2. The reactor design allows development of the representative thermal-hydraulic conditions which control fuel behavior during loss-of-coolant accident (LOCA) transient events. The surprising core cooling behavior during the LOFT large break LOCA experiment emphasizes the value of integrated systems testing with representative geometrics and materials.

3. EXPERIMENTAL PROGRAM CONDUCTED

Since core loading in October 1977, the LOFT fuel has been exposed to the operation and test sequence shown on Table 1. The sequence has exposed the fuel to power range testing to 53 kW/m peak linear heat generation rate (LHGR), four operational transient tests, four small break LOCEs, and three large break LOCEs. The center fuel module was replaced after the third large break LOCE (L2-3) for examination after achieving a peak fuel burnup of 987 MWd/MTU. Peak fuel burnup is 1900 MWd/MTU (December 1, 1980) in the peripheral fuel bundle fuel rods which have been operated since LOCE L2-3 (May 1979) with annealed cladding conditions. No leaks are presently detectable in the fuel rods as determined by negligible iodine levels in the primary coolant. The non-destructive examination of the original center fuel bundle indicates the bundle is reuseable.

4. CORE MEASUREMENTS

The instrumentation features of presently installed fuel modules are provided in Table 2. The instrumentation is designed to provide fuel assembly structural response information as well as the reactor vessel thermal-hydraulic information. The arrangement of 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. Forty reed switches 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.

5. CORE THERMAL RESPONSE RESULTS

The two large-break experiments were the only tests that have caused the fuel rod cladding temperature to increase above steady state operation values. Figure 2 shows the measured fuel rod cladding temperature response which occurred in LOCE L2-3 compared against the EG&G best estimate and NRC evaluation model pretest predictions. The peak fuel rod cladding temperature measured during LOCE L2-3 was 914 K compared to the best-estimate prediction of 1000 K. The NRC evaluation model predicts a still increasing cladding temperature when the measured cladding temperature had quenched to the coolant temperature. The best-estimate prediction is based on models and material behavior correlations judged to be accurate and most of which have been based on independent verification experiments and analyses. The safety evaluation prediction is based on models and material behavior correlations that represent a conservative adverse accumulation of (a) manufacturing deviations from nominal design values and (b) uncertainties in fuel rod material behavior properties and thermal hydraulic computer models.

The chronology of core thermal-hydraulic events during LOCE L2-3 indicate an unexpected core cooling event occurred. Initial departure from nucleate boiling (DNB) occurred between 1 to 2 seconds and coincided with a general flow stagnation after which the measured cladding temperature rapidly increased for 1 to 2 seconds. At approximately 3 seconds, measured upward core flow was established which tended to cool the core and reduce the rate of cladding temperature increase. At about 4 seconds the flow in the reactor vessel was increased as a result of flow reduction out the cold leg break and resulted in increased flow through the reactor core at velocities estimated from 150 to 200 cm/s which resulted in the measured cladding quench from approximately 5.5 to 7.5 seconds. The cladding quench was maintained for several seconds, but eventually as the reactor vessel coolant was depleted, a second DNB or dryout occurred at about 10 to 12 seconds after test initiation. After this time the cladding temperatures in the peak power location increase slowly until ECC water reflooded the core. Fuel rod stored energy calculations based on the measured temperatures show the early quench period and precursory cooling is effective in removing approximately 80 percent of the stored energy in the fuel rods with surface thermocouples.

The effectiveness of this cooling on the nuclear fuel rods has been questioned due to uncertainty in the influence of the cladding surface thermocouples during the LOFT nuclear tests. The thermocouples have been hypothesized to reduce the LOFT peak cladding temperature by (1) delaying the initial departure from nucleate boiling (DNB), (2) increasing the fuel rod heat transfer by effectively increasing the cladding surface area (fin effect), and (3) selective cooling of the entire fuel rod, and in particular the surface thermocouples, during cooling periods characterized by low quality, two-phase coolant flow.

Several separate effects experiments have been conducted with both nuclear (INEL Power Burst Facility)^{17,18} and electrically, (INEL¹⁹ and Columbia University²⁰ thermal-hydraulic test facilities) heated rods to

determine the surface thermocouple accuracy during the LOCA events and the effects on fuel rod cooling. Figure 3 compares the measured LOCE L2-3 cladding temperature to the upper bound of cladding temperature of uninstrumented fuel rods which was calculated using the results of the separate effects experiments. The graph indicates uninstrumented fuel rods may have achieved peak cladding temperatures nearly 100 K higher than the measured values and the reflood cooling may have lasted 25 seconds longer than indicated by the thermocouple data. This peak temperature difference significantly changes the risks of prepressurized fuel rod balloon and burst during a large break LOCA.

Spacer grid effects, superheated vapor migration and upper plenum liquid fallback also appear to be beneficial to core cooling during a PWR large break LOCA. Fuel rod surface thermocouple and conductivity liquid level detector measurements in the vicinity of spacer grids the LOFT during large break LOCEs indicate a tendency for formation of superheated vapor regions beneath spacer grids and liquid phase enriched (lower quality) regions downstream of (immediately above) the spacer grids. The liquid retention is beneficial to cooling the core. Superheated vapor penetrates the upper tie plate approximately 4 seconds after initial fuel rod DNB and is condensed within nine inches above the tie plate. This event indicates a counter flow condition in the core which could be delaying core liquid loss through the core inlet. Later in the decompression event as the core and upper plenum are enveloped in superheated vapor, indications of liquid are obtained at the tie plate thermocouples which is attributed to liquid fall back from the upper plenum towards the core region.

6. FUEL ASSEMBLY STRUCTURAL RESPONSE RESULTS

The structural response of the fuel modules was similar in all three large break experiments (LOCE's L1-5, L2-2 and L2-3) and was less severe than predicted. Figure 4 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 calculated using the SHOCK computer code¹³ lumped-mass-spring model of the LOFT fuel modules and support structures. The data from LOCE L2-2 and L2-3 are similar, but less severe because of the cushioning effect of earlier formation of steam in the reactor vessel. 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 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 calculated 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. 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 code¹⁴. The calculated stresses did not exceed 60% of those allowed by the American Society of Mechanical Engineers (ASME) code. Comparison of the pre- and posttest LVDT signals and the center fuel module examinations indicates that no residual deformation of the center fuel module occurred during LOCEs L1-5, L2-2 and L2-3.

7. CONTROL ROD DROP RESPONSE RESULTS

The control rods were released during the large-break LOCEs 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 5 shows the LOFT control rod drop time data for LOCEs L1-5 and L2-2. LOCE L2-3 drop times are similar. The control rod drop is measured by four magnetic-reed switches located above the control rod drive motors.

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 and core temperature measurements indicate 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 produced by the necked-down lower section of guide tube which occurs in some conventional PWRs.

8. FUEL ROD BURST AND FLOW CHANNEL BLOCKAGE INFORMATION

The fuel rod cladding temperatures achieved during the experiments were insufficient to cause prepressurized fuel rod ballooning or bursting during large break LOCAs. An experiment (LOC-11)¹⁶ in the Idaho National Engineering Laboratory (INEL) Power Burst Facility (PBF) confirms this conclusion. Examination of the LOFT fuel rod surface thermocouple data indicates that: (1) 9 x 9 fuel rod arrays by 28 cm long (axially) regions of fuel bundles maintain uniform temperatures during periods when fuel rod burst could occur, and (2) control rod guide tubes do not cause large azimuthal temperature variations in the cladding of adjacent fuel rods. Since uniform temperature conditions are conducive to developing flow blockades from fuel rod ballooning the results should be considered in assessment of flow blockages.

LOCE L2-3 probably caused annealing at fuel rod cladding in the peripheral fuel bundle hotter regions. Subsequent operation has included three small break LOCEs which require 43 kW/m peak fuel rod LHGR in the peripheral fuel rods at initiation and four operational transient tests which require 32 kW/m peak fuel rod LHGR at test initiation. One of the transients (L6-3) caused an increase in peak LHGR from 32 to 36 kW/m in 8 seconds; however, fuel preconditioning at a higher value had been

established prior to the test to minimize fuel rod cladding mechanical interaction during the rapid reactivity power insertion during the test, and no evidence of pellet cladding interaction (PCI) damage has been discovered. Power escalation recommendations from the LOFT fuel supplier are currently for no restrictions which indicates unprepressurized fuel with annealed cladding could be used in load-follow maneuvers with appropriate fuel preconditioning.

9. SUMMARY OF EXPERIMENTAL RESULTS

The LOFT experiments have provided the following fuel behavior information which appears to be valuable for improving the safety of PWR operation and resolving PWR licensing issues:

- (1) A generic unassisted core cooling event occurs during large-break LOCAs that dominates the cooling of the core before ECC reflood commences and potentially eliminates the possibility of flow channel blockage from prepressurized fuel rod swelling.
- (2) The large-break LOCA decompression forces do not disturb the normal control rod gravity drop and may not structurally damage the fuel assemblies.
- (3) Large-break LOCA core cooling may also be enhanced by spacer grid and core counter flow delay of liquid escape from the core boundaries and liquid fallback from the upper plenum into the core region.
- (4) Lower fuel rod prepressurization may be possible in PWR fuel rods which would reduce flow channel blockage complications during LOCA's.

- (5) Uniform fuel rod cladding temperature indications during the large break LOCE's do not confirm expectations for the fuel rod cladding temperature variations that would inhibit development of flow channel blockages by ballooning of prepressurized fuel rods.

10. FUTURE TEST PLANS

The LOFT future includes improved core instruments, and experiments with prepressurized fuel rods that will cause fuel damage to occur.

LOFT fuel instrumentation improvements which are planned to increase the ability of LOFT to provide safety-related PWR fuel behavior information, include expanded measurement capabilities and improved durability and precision. Core inlet coolant density and flow rate, centerline fuel rod temperature, and fuel rod length measurements are examples of the expanded capabilities. Fuel rod internal gas pressure and temperature will be measured when prepressurized fuel rods are installed. Turbine flow meter bearings and zircaloy-sheathed fuel rod surface thermocouples are examples of durability improvements. Small zircaloy-sheathed thermocouples embedded in the fuel rod cladding inside surface will provide a less obtrusive more accurate measurement of cladding temperature and quantify the inherent inaccuracy of the fuel rod surface thermocouple.

Figure 6 is a chart showing the most recently developed 6 year test program and fuel utilization and destructive examination plans. Additional test plan description is provided in Table 3. The test program shown is unapproved, and subject to change as the heuristic LOFT test program develops better definitions of safety information needs. LOFT is a potential site for PWR core melt experiments and safety experiments with other type nuclear reactors (see Table 4). The six year test plan shown in Figure 6 and Table 3 is based on the L1 through L10 test series. The L11

through L26 test series listed in Table 4 would be follow-on activities extending beyond FY-1986 and are classified as "additional LOFT testing possibilities."

The L8, L12 and L17 are a test series set that would carefully ascend into the most severe core decomposition and migration conditions. The L8 series would allow center fuel bundle damage to occur yet permit the removal of fuel with the existing fuel removal equipment. The L12 series would allow entire core damage and decomposition to occur within the core boundaries and the L17 test series would allow the decomposed core to migrate and erode the core support structures, reactor vessel, containment structure and containment substratum. Important objectives of the experiments would be to: (1) prove that the massive heat sink characteristics of PWR power generation structures cause solidification of the migrating core before the containment structure is breached, and (2) provide data for developing systems which would minimize damage to a power generation facility in the event core decomposition and migration occurs.

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TABLE 1. LOFT FUEL OPERATING MODES SINCE CORE LOAD

September 1977 Core Load.

LOCE L1-5 (Large Break) Without Core Heat³.

Power Range Test Sequence to 53 kw/m.

LOCE L2-2 (Large Break) from 26 kw/m⁴.

LOCE L2-3 (Large Break) from 39 kw/m⁵.

LOCE L3-1 (Small Break) from 53 kw/m⁶.

LOCE L3-2 (Small Break) from 53 kw/m⁷.

LOCE L3-7 (Small Break) from 53 kw/m⁸.

OPTRAN L6-5 (Steam Generator Feedwater Loss) from 39 KW/m⁹.

LOCE L3-5/5A (Small Break) from 53 kw/m¹⁰.

OPTRAN L6-2 (PC Pump Trip) from 39 kw/m¹¹.

OPTRAN L6-1 (Turbine Trip) from 39 kw/m¹¹.

OPTRAN L6-3 (Turbine Load Increase) from 39 kw/m to 45 kw/m¹¹.

TABLE 2. 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)	<u>2</u>
Total Measurement Locations		345*

* Six pressure taps to external detectors.

TABLE 3. PLANNED LOFT TEST SEQUENCE AND TARGET DATES AS OF SEPTEMBER 1980

(The Current Schedule Is Being Reassessed Using
Criteria Developed For The Long Term Planning Strategy.
While The Short Term Test Schedule Is Not Expected To Change,
Long Term Test Plans May Be Modified As A Result Of This Assessment.)

Test ID	Date	Initial Power Level (MW)	Initial Core ΔT °F	Comments
L3-5	09-29-80C	50	35	Small break (2.5%) intact loop cold leg--pumps off.
L3-5A	09-29-80C	Add on to L3-5		Investigate primary system recovery utilizing steam generator.
L6-2	10-07-80C	37	25	Loss-of-power to primary coolant pumps.
L6-1	10-08-80C	37	25	Loss-of-steam load (closure of MSIV's).
L6-3	10-09-80C	37	25	Excess load increase (cooldown transient).
L3-6	12-10-80	50	35	Small break (2.5%) intact loop cold leg--pumps on. Pumps tripped at end of experiment to measure water remaining.
L8-1	12-10-80	Add on to L3-6		Core uncover without ECC at low decay heat level.
L9-1	04-01-81	50	35	Loss of all feedwater (multiple failures) with scram on high pressure; PPS set-points representative of LPWR (PORV challenged) Mild ATWS.
L3-3	04-01-81	50	35	Small hot leg break (PORV Flow of 105 lb/hr/MW) equal to break flow. Dry steam generator secondary. Needs further justification.

TABLE 3. (Continued)

Test ID	Date	Initial Power Level (MW)	Initial Core ΔT °F	Comments
CV Leak Test	06/81			Required test of containment leak integrity.
L5-7	07/81	50	65	LOFT typicality to Arkansas Nuclear One startup test.
L9-2	07/81	Add on to L6-7		Rapid cold water accident, upper plenum voiding.
L5-1	08/81	50	65	Intermediate size break (accumulator line). Determine if large break and small break models continue to predict intermediate break results. Also check out liquid level device.
L8-2	08/81	Add on to L5-1		Core uncover at high decay heat level. Reflood with degraded ECC capability. May be the same as L5-1.
Replace CB A2 with F1	09/81			F1 center bundle at 350 psi (BOL).
L2-5	12/81	16 kw/ft	65	Worst prototypic hydraulic conditions in core. Investigate fuel behavior at BOL fuel pressure (no fuel damage expected).
Replace CB F1 with F2	03/82C			F2 will be pressurized to 700 psi.
L2-6	04/82	16 kw/ft	65	Same as L2-5 with 700 psi fuel pressure (EOL). Fuel damage and fission product release expected.

TABLE 3. (Continued)

Test ID	Date	Initial Power Level (MW)	Initial Core ΔT °F	Comments
Whole Core Changeout C.F.M.	08/82			Only minimal fuel damage experiments can be done until F1 is examined for damages.
L5-2	09/82	16 kw/ft	65	Intermediate size break on hot leg. Pressurizer surge line. Needs further justification based on L5-1.
L6-4	09/82	16 kw/ft	65	Uncontrolled rod withdrawal at power. Investigate worst case moderate frequency accident.
L9-3	12/82	16 kw/ft	65	ATWS. Loss-of-Feedwater is initiating event (multiple failures).
L9-4	03/83	16 kw/ft	65	ATWS. Loss of offsite power as initiating event (multiple failures).
Put F1 Bundle back in				F1 inspection completed and fuel is assumed not damaged.
L8-3	08/83	16 kw/ft	65	Small break with slow core heatup (10°F/min). Investigate potential initiating events (candidate: loss-of-feedwater.)
Replace F1 with A3				
L7-1	12/83	16 kw/ft	65	Large break with S.G. tube ruptures at start of reflood/refill (> 25 tubes ruptures). Critical number of tube ruptures resulting in extreme core temperatures expected to be between 10 and 25 based on Semiscale results.

TABLE 3. (Continued)

Test ID	Date	Initial Power Level (MW)	Initial Core ΔT °F	Comments
L7-2	02/84	16 kw/ft	65	Large break with S.G. tube ruptures at start of reflood/refill (< 10 tubes ruptured). Provides a lower bound of envelope on effect of ruptures. L7-3 should be inserted if possible which has critical number of ruptures.
L4-1	05/84	16 kw/ft	65	200% C. L. break. Accumulator injection into U.P. Investigate topdown core quench. Applicability to UHI plants.
L4-2	08/84	16 kw/ft	65	200% cold leg break. U.P. LPIS injection. Investigate <u>W</u> two loop plant phenomena.
Replace A3 with press F3	12/84			
L8-4	03/85	16 kw/ft	65	Severe core damage. Investigate potential initiating events (candidate: loss-of-offsite power).
Whole core changeout	04/85			F4 Center bundle.
L10-1	07/85	16 kw/ft	65	Override test. Override of L8-3 transient.
L10-2	09/85	16 kw/ft	65	Override test. Override of L8-4 transient.
L8-5	11/85	16 kw/ft	65	Severe core damage. Investigate potential initiating events (candidate: steam line rupture).
Decommission	12/86			

TABLE 4. LOFT EXTENDED TEST SERIES

	L1	Non Nuclear
	L2	Power Ascension Series
	L3	Small Break Series
	L4	Alternative ECC
	L5	Intermediate Breaks
	L6	Anticipated Transients - Operational Transients
	L7	Steam Generator Tube Failure
	L8	Limited Damage Core Transients - Core Uncovery Fuel Ballooning and Bursting, etc.
	L9	Anticipated Transients with Multiple Failures (ATWS, TMI)
	L10	Override Plant Protection Mode
	L11	Reactor State Series - Determine common states that reactor can get to and need to recover from Pot Boiler mode of recover.
	L12	Extensive Core Damage Experiments
	L13	Plant Availability
	L14	Man Machine Interface Experiments - Testing EOP not the man
	L15	Rubble Bed Heat Transfer Like TMI
	L16	TMI Verification Experiments
MTA2	L17	Core Melt and Containment Integrity
MTA3	L18	LOFT Redesign with 12 ft. core, Secondary Active, Second Active Loop, full height steam generator
MTA4	L19	Alternate Design (B&W, Combustion)

TABLE 4. (Continued)

MTA5	L20	BWR LOFT
MTA6	L21	LMFBR LOFT
MTA	L22	CANDU or HTGR LOFT
	L23	New NRC Requirement Checkout Series
	L24	Commercial PWR Instrumentation Test Series
	L25	Seismic and Structural Test Series
	L26	Containment (Structural) Testing

INEL-A-7862

Fig. 1 LOFT core configuration and instrumentation location.

FIGURE 2

L2-3 Peak Cladding Temperatures

Figure 3. Comparison of L2-3 Peak Cladding Temperature Envelope to RELAP4/MOD6 Pretest Prediction

Fig. 4 Center fuel module holdown spring displacement during LOCE L1-5 subcooled blowdown (uncertainty analysis for displacement data is contained in Reference 15).

Fig. 5 LOFT control rod drop times during LOCE scrams (rod position uncertainty of ± 0.006 m based on data from reed switch instrument supplier).

LOFT Six Year Test & Fuel Plan

☒ Prepressurized fuel rods