CONF-810803--1
CONF-8010103--7

LOFT/UTILITY TECHNOLOGY TRANSFER MEETING

October 16-17, 1980

PWR FUEL BEHAVIOR - LESSONS LEARNED FROM LOFT

by

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A summary of the experience with the Loss-of-Fluid Test (LOFT) fuel during loss-of-coolant experiments (LOCEs), operational and overpower transient tests and steady-state operation is presented. LOFT provides unique capabilities for obtaining pressurized water reactor (PWR) fuel behavior information because it features the representative thermal-hydraulic conditions which control fuel behavior during transient conditions and an elaborate measurement system to record the history of the fuel behavior. The LOFT experience has provided the following fuel behavior information which appears to be valuable for improving the safety of PWR operation and resolving PWR licensing issues:

(a) During large-break loss-of-coolant accidents (LOCAs) a generic unassisted core cooling event occurs, the decompression forces do not disturb the normal control rod gravity drop or structurally damage the fuel bundles, and core cooling is also enhanced by spacer grids, core counter flows and upper plenum liquid fallback,

(b) Operation with unprepressurized fuel rods supports the possibility for PWR power generation with lower fuel rod prepressurization.

a. Also being submitted to ANS Topical Meeting on Reactor Safety Aspects of Fuel Behavior at Sun Valley, Idaho, August 2-6, 1981.

(c) fuel replacement in a PWR may cause flow reductions in the longer-residence fuel bundles.

The LOFT 50-MW(t) reactor core is composed of six instrumented and three noninstrumented fuel assemblies. The six instrumented fuel assemblies provide 346 locations for measurement of reactor conditions during steady-state power operation and transient experiments. The LOFT fuel assemblies 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 exposure to repeated LOCEs. Replacement center fuel assemblies, featuring prepressurized fuel rods and zircaloy guide tubes, will be installed for later experiments.

Since core loading in October 1977, the LOFT fuel has been exposed 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 1542 MWd/MTU in the peripheral fuel bundle fuel rods which have been operated since LOCE L2-3 (May 1979) with annealled 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.

An unexpected cooling of the core was measured by the fuel rod surface thermocouples before emergency core cooling (ECC) injection in the large break decompression event. A generic system behavior replenishes the core liquid inventory early in the decompression event to improve the core cooling. Analysis, comparing behavior of LOFT and large electric generating plants, indicates the liquid inventory replenishment in the large core would be even more beneficial.

Investigation of the thermocouple performance indicates the surface-mounted thermocouples are probably inaccurate when the local liquid inventory increases and also enhance the cooling of the fuel rod. However, the thermocouple indication during the subsequent depletion of liquid from the core and the rapid fuel rod surface rewetting during ECC reflooding coincides with analytical predictions that during the decompression event the core cooling removed approximately 40% of the initial fuel rod stored energy. The enhanced cooling effect of the surface thermocouples is judged to be inconsequential in causing premature cooling of the entire core because only 76 of 1300 fuel rods are equipped with surface thermocouples. The rod to rod spacing measurements on the removed center fuel bundle indicate fuel rod diameter and bowing are the same in both instrumented and uninstrumented core regions which is further evidence that instrumented and uninstrumented core regions behave similarly.

Control rod drop times measured during the large break LOCEs indicate the decompression hydraulic forces do not disturb the normal gravity drop of the control rods. The drop times are consistent with that expected under zero flow conditions. Measurement of coolant temperature inside the control rod guide tubes and analytical predictions indicate that a liquid inventory is maintained in the guide tubes long enough to provide control rod deceleration during a LOCE in PWRs that use necked down guide tubes for control rod deceleration.

No residual deformation of the fuel bundle structure has occurred during the large break LOCE decompression events. This conclusion is based on examination of the removed center fuel bundle periscope photographs and closed circuit television (CCTV) scans and measurements of fuel module axial motion during the large break experiment decompressions, extraction forces during center fuel bundle withdrawal from the tight fitting core space, and rod to rod spacing of the removed center fuel bundle. Comparison of conventional fuel bundle structural analysis results with the axial motion measurements demonstrated the validity of the analysis techniques for predicting fuel bundle structural response during the decompression event.

Spacer grid effects, superheated vapor migration and upper plenum liquid fallback appear to be beneficial to core cooling during the large break LOCA. Fuel rod surface thermocouple and conductivity liquid level detector measurements in the vicinity of spacer grids during large break LOCEs indicate a tendency for formation of superheated vapor regions beneath spacer grids and liquid phase enriched 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 departure from nucleate boiling (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.

The utilization of unpressurized fuel rods in LOFT is developing information that supports the possibility for PWR power generation with lower prepressurization fuel rods which would enhance flow blockage resistance during LOCA's. 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) 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 (a) large (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 (b) 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 swelling the results should be considered in assessment of flow blockages. The operating experience with LOFT unprepressurized fuel rods indicates annealed cladding in the peripheral fuel bundle hotter regions as a result of LOCE L2-3. 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 clad mechanical interaction during the rapid reactivity power insertion during the test, and no evidence of pellet cladding interaction (PCI) damage has been discovered. The LOFT fuel supplier power escalation recommendations all currently for no restrictions which indicates unprepressurized fuel with annealed cladding could be used in load-follow manuevers with appropriate fuel preconditioning.

A coolant temperature rise decrease corresponding to a flow rate increase of approximately 8% (5% reduction in temperature rise plus 3% calculated increase in fuel rod power peaking) was measured after installation of the new center fuel module. The core coolant temperature rise measured by the tie plate thermocouples during steady-state operation correlates well to the temperature rise of the core flow channel which the thermocouple is positioned over. Since the replacement and removed fuel bundles were identical, the preliminary conclusion is that crud deposits in the resident peripheral fuel bundles caused flow to redistribute to the clean center fuel bundle. The removed center fuel bundle photographs have been judged to show representative PWR fuel crud deposits including preferential deposits in the vicinity of spacer grids. This information should be considered in PWR fuel replacement planning because of the potential for flow reductions in the longer residence fuel bundles.

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. 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.

The data obtained from the LOFT fuel instruments, the fuel performance, and the planned measurement improvements provide indications that (a) significant PWR fuel behavior information can be obtained from the LOFT test program and (b) the information will be especially valuable because LOFT provides more of the representative PWR system thermal-hydraulic conditions which control fuel behavior than any other test facility in the world.