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TITLE: Blowdown Hydraulic Influence on Core Thermal Response in LOFT Nuclear Experiment L2-3

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CALL FOR PAPERS

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1979 WINTER MEETING

November 11 - 16, 1979
San Francisco, California

Members of the SOCIETY are invited to submit papers for presentation at the 1979 Winter Meeting of the American Nuclear Society San Francisco, California Nov. 11 - 16, 1979 and to sponsor papers by nonmembers

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Content: Each contributed summary must present facts that are new and significant. A simple listing of material to be presented is not acceptable. *Abstracts are not acceptable.* The summary must contain not only the work that has been performed but also the results achieved. Proper reference should be made to all closely related information that has been published. **SUMMARIES SHOULD INCLUDE AN INTRODUCTORY STATEMENT INDICATING THE PURPOSE OF THE WORK AND A CLOSING STATEMENT SUMMARIZING THE SIGNIFICANT NEW RESULTS.**

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(a) Journal reference: G. E. Doe, *J. Appl. Phys.*, 24, 194 (1950).
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(c) Report reference: G. E. Doe, "How to Reference a Report," ANS, American Nuclear Society (1977).

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Neil Norman, ANS Technical Program Chairman
Attn: Ruth Farmakes
American Nuclear Society
555 North Kensington Ave.
LaGrange Park, IL 60525

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Name and full mailing address of author
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LOG # ____________

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Title of Summary Blowdown Hydraulic Influence on Core Thermal Response in LOFT Nuclear Experiment L2-3

This is to acknowledge receipt of your summary. Please use the log number above in future correspondence.

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Sincerely,

Neil Norman
ANS Technical Program Chairman
1979 Winter Meeting

CONF-791103--207 (Summ)

BLOWDOWN HYDRAULIC INFLUENCE ON CORE THERMAL RESPONSE IN LOFT NUCLEAR EXPERIMENT L2-3

by

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Experimental research into pressurized water reactor (PWR) loss-of-coolant phenomena conducted in the Loss-of-Fluid Test (LOFT) facility[1] has given results indicating that for very large pipe breaks the core thermal response is tightly coupled to the fluid hydraulic phenomena during the blowdown phase of the loss-of-coolant transient. This summary presents and discusses data supporting this conclusion.

LOFT Loss-of-Coolant Experiment (LOCE) L2-3 simulated a complete double-ended offset shear break of a primary coolant reactor vessel inlet pipe in a commercial PWR. The LOFT system conditions at experiment initiation were: fuel rod maximum linear heat generation rate (MLHGR) of 39.4 ± 3 kW/m, hot leg temperature of 593 ± 3 K, core ΔT of 32.2 ± 4 K, system pressure of 15.06 ± 0.03 MPa, and flow rate/system volume of 25.6 ± 0.8 kg/m^3. These conditions are typical of those in commercial PWR systems at normal operating conditions.

The thermal transient of the fuel cladding in the LOFT 1.68-m core was found to be limited in magnitude by the system hydraulics within the first 10 s of the blowdown. The peak fuel cladding temperature in this experiment was 914 K which occurred 4.95 s into the transient. This temperature is below fuel cladding damage thresholds.

The system depressurized to the saturation pressure of the fluid in the hot channels of the core in about 50 ms, followed by a core flow reversal at 0.3 s. The hotter fuel rods in the core began to experience departure from nucleate boiling at 0.94 s, followed by a rapid increase (~250 K/s) in fuel cladding temperature which began to moderate between 1.3 and 2 s with the reestablishment of low magnitude positive core flow. During this time, the hotter core fluid delivered to the lower plenum during the initial core flow reversal was migrating up the downcomer to the broken loop cold leg (BLCL), raising the temperature to saturation, and causing saturated choking to occur. The accompanying flow reduction reduced the magnitude of BLCL flow to below that of the intact loop cold leg (ILCL), allowing a portion of the ILCL flow to be diverted to the downcomer, lower plenum, and core regions, and causing a core-wide return to nucleate boiling (RNB) between 5 and 8.5 s. Following this RNB, the hotter fuel rods began to dry out at 9s and remained in a dryout condition until quenched by emergency core coolant (ECC).

ECC filled the lower plenum by 35 s, at which time core reflood started and proceeded at a rate of 100 mm/s, completely reflooding the core by 55 s. The accompanying quench occurred from the bottom up and the top down, quenching the hotter center region of the core last. Figure 1 shows a three dimensional axial temperature profile in the center fuel module.

The temperature-pressure history of the fuel rod experiencing the peak cladding temperature is shown in Figure 2 along with the thresholds for cladding deformation. These threshold conditions would have to have been maintained for 15 s for deformation to occur[2]. The data did not reach these conditions, and the period of time near the thresholds was about 3 s, indicating no cladding deformation occurred during LOCE L2-3.

The early RNB was directly related to the hydraulic effects of the saturated choking in the BLCL and subsequent diversion of some ILCL fluid into the core. These hydraulic effects were seen in LOCE

L2-2[3] and can be expected in other double-ended cold leg break loss-of-coolant transients in which a core ΔT is present and the pumps are operating. As the phenomena require the ILCL flow to be maintained after BLCL saturated choking occurs, the hydraulic influence may be diminished if power is lost to the primary coolant pumps coincident with pipe rupture.

The analysis of LOCE L2-3, initiated from nominal operating conditions, has revealed information pertinent to understanding the behavior of a reactor operating at normal power conditions that is subjected to a large break loss-of-coolant transient. Specifically:

(1) The hydraulic events early in the transient strongly influence and limit the severity of the core thermal response.

(2) These hydraulic events can be expected to occur in double-ended cold leg break loss-of-coolant transients at higher power densities.

(3) The fuel rod cladding was not damaged in the experiment.

(4) The LOFT ECC system was very effective in reflooding the core and quenching the fuel rods.

REFERENCES

1. D. L. Reeder, LOFT System and Test Description (5.5-ft Nuclear Core 1 LOCEs), NUREG/CR-0247, TREE-1208 (July 1978).
2. C. S. Olsen, Zircaloy Cladding Collapse Under Off-Normal Temperature and Pressure Conditions, TREE-NUREG-1239 (April 1978).
3. M. McCormick-Barger, Experiment Data Report for LOFT Power Ascension Test L2-2, NUREG/CR-0492, TREE-1322 (February 1979).

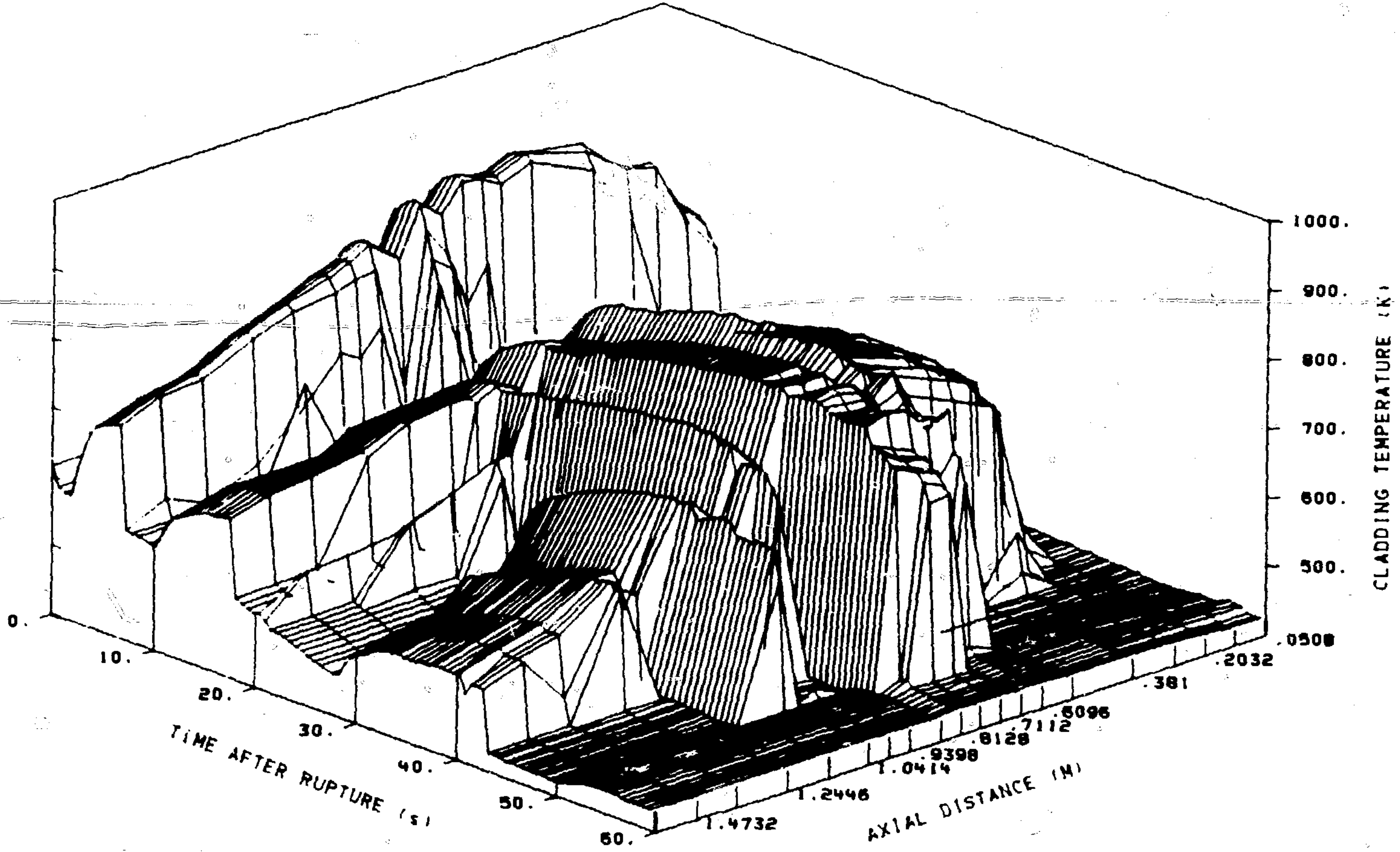

Fig. 1 Three-dimensional axial profile of cladding temperature in center fuel module.

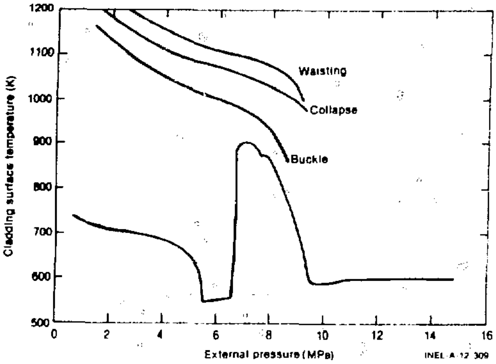

Fig. 2 Temperature and pressure history of fuel rod experiencing peak cladding temperature during LOCE L2-3 compared with modes of cladding deformation.