OVERVIEW OF RECENT POWER BURST FACILITY (PBF) TEST RESULTS

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The Thermal Fuels Behavior Program (TFBP) of EG&G Idaho, Inc., conducts fuel behavior research in the PBF in support of the United States Nuclear Regulatory Commission's (NRC) Fuels Behavior Branch. The fuels behavior research in PBF is directed towards providing a detailed understanding of the response of nuclear fuel assemblies to off-normal and hypothesized accident conditions.

During the last twelve months, the TFBP has performed four test series in the PBF, Figure 1. Two of these test series are part of a thirty-seven test program to investigate light water reactor (LWR) fuels behavior. At this time, twenty-five tests in this program have been completed.

RIA Test Series

The first two PBF Reactivity Initiated Accident (RIA) Series 1 tests were performed with four light water reactor design fuel rods individually shrouded to provide thermal and mechanical independence. A summary of preliminary results for Test RIA 1-2, performed in November 1978, is presented in Figure 2.

The objectives for Test RIA 1-2 were to characterize the response of preirradiated fuel rods during an RIA event with a peak total pellet average energy of about 180 cal/g UO_2 and to evaluate the effect of internal rod pressure on preirradiated fuel rod response during an RIA event.

Test RIA 1-2 utilized four test rods which were preirradiated to burnups of about 5000 MWd/t. Only one of the four preirradiated rods failed following the power burst which produced a peak total pellet enthalpy increase of about 180 cal/g UO_2. This rod had not been opened prior to testing in the PBF, whereas the other three rods had been opened prior to testing to attach plenum pressure transducers. Opening of fuel rods to air, therefore, may alter the internal chemistry and thereby decrease the propensity for failure resulting

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from pellet-cladding interaction (PCI) during the RIA tests. The failed rod, Figure 3, was found to have 22 longitudinal cracks starting at about 18-cm and extending up to about 72-cm from the bottom of the 91-cm fuel stack. The cracks had the appearance of PCI stress corrosion type cracks. The average pellet energy deposited at the 18- and 72-cm locations was about 135 cal/g UO_2. Thus, the tentative conclusion is that the failure threshold during an RIA event for fuel rods with burnup of about 5000 MWd/t could be as low as 135 cal/g UO_2. The test rods are now undergoing extensive postirradiation examination and the full report of Test RIA 1-2 will be issued by January 1980.

Four more tests remain in the PBF RIA Series 1 program. These tests will evaluate fuel rod damage at the licensing criteria of 280 cal/g UO_2 for burnups of 15 000 MWd/t and investigate coolability of irradiated fuel rods in a bundle geometry subjected to an RIA transient.

LOFT Lead Rod Test Series

The PBF Loss-of-Fluid Test (LOFT) Lead Rod (PBF/LLR) Test Series, consisting of four loss-of-coolant accident (LOCA) blowdowns, was completed in May 1979. The test objectives for the PBF/LLR Test Series were to: (a) experimentally evaluate the extent of cladding collapse that is expected to occur during a LOFT Power Assention LOCA Test Series; (b) to evaluate the effects of mechanical deformation and PCI on the mechanical response of fuel rods subjected to subsequent power increases, long-term preconditioning, and LOCA conditions; and (c) to provide experimental data to benchmark the fuel rod analysis program (FRAP) that will be used to requalify the LOFT core.

Each test consisted of a fuel preconditioning and power calibration phase, a decay heat buildup phase, a blowdown phase, and reflood and quench cooling phases. The rods were subjected to a blowdown similar to that expected in LOFT during a 200% double-ended cold leg break. In each test, blowdown was initiated from an at-power condition with typical LOFT PWR coolant conditions.

Maximum cladding surface temperatures are indicated in Figure 4. Because of the low cladding surface temperatures, no mechanical deformation was expected to have occurred to the Test LLR-3 and LLR-5 fuel rods[1]. Cladding surface temperatures during Test LLR-4 ranged from 1060 to 1170 K. On the basis of comparisons of these indicated cladding surface temperatures with Olsen's data[2], two of the rods were expected to have reached the waisting regime of mechanical deformation, one rod the buckling regime.

Since a major objective of performing the PBF/LLR Test Series was to evaluate the effect of cladding collapse and waisting on rod behavior during subsequent power ramps and LOCA transients, an additional LOCA test, LLR-4A, was performed[3]. Test LLR-4A was performed with the same test conditions as Test LLR-4. The thermal and mechanical response of Test LLR-4A was the same as for Test LLR-4, with cladding temperatures ranging from 1050 to 1205 K. No fuel rod failures were detected as a result of Test LLR-4A. A detailed postirradiation examination of the PBF/LLR test rods is in progress. The test results report will be available in June 1980.

LOCA Test Series

The major objective for the PBF LOCA program is to experimentally evaluate the extent of fuel rod cladding deformation during severe loss-of-coolant conditions. The key test variables in the PBF LOCA test program are presented in Figure 5.

Test LOC-3, the second test in the LOCA Test Series, was performed in June of 1979[4]. Test LOC-3 was to achieve a cladding temperature of approximately 1200 K and a cladding microstructure in the alpha plus beta region. One each of the previously irradiated and previously unirradiated rods was backfilled with helium to a pressure typical of beginning-of-life PWR fuel rods. The other rods were backfilled with helium to a pressure typical of fuel rods at the end-of-operational life.

The test was conducted in four phases: loop heatup, preconditioning, blowdown, and quench. To achieve the desired cladding temperatures, the PBF reactor power was controlled during the blowdown by a preprogrammed function generator. Power was reduced from a test rod power of 53 KW/m just prior to the blowdown to a reactor power of about 2 MW within 20 seconds after the blowdown.

The thermal-hydraulic response during the test was in good agreement with the predicted response. For both types of prepressurized rods, the cladding thermocouples indicated that the previously unirradiated rod cladding peak temperatures were lower than those of the previously irradiated rods. As indicated in Figure 6, the cladding peak temperature measured at the 0.625-m location was about 1070 K, which is below the predicted values; however, temperatures in the higher power flux region of the test rods are expected to be higher and closer to the predicted values. A reason for the higher predicted temperatures at critical heat flux (CHF) may be because CHF was predicted to occur at 1.5 seconds at the 0.625-m elevation, but was measured at 2.8 to 3 seconds. The delayed CHF resulted in the upper region of the fuel rod experiencing a longer period of nucleate boiling heat transfer early in the blowdown, thereby decreasing the internal energy of the rods and lowering the peak cladding temperature following CHF. The test instrumentation indicated, however, that CHF occurred within a few tenths of a second, probably at the lower elevation of the rods, then propagated up the rods toward the upper thermocouples. A detailed postirradiation examination, now in progress, will permit confirmation of these conclusions.

All four of the LOC-3 test rods ballooned and failed during the test. Posttest examination of the rods indicated that the previously irradiated low pressure rod ballooned near the top of the flux shaper, approximately 10 cm from the cladding surface thermocouples. The diameter increase was over a distance of less than 10-cm of the fuel rod length, whereas the other three rods increased in diameter over a significant length. Both the previously unirradiated (Figure 7) and previously unirradiated (Figure 8) high pressure rods experienced significant and symetrical diameter increases over a large portion of the test rods. Three of the rods also experienced some bowing. The previously irradiated high pressure rod experienced the largest diameter increase, extending from approximately 15- to 60-cm from the bottom of the rod.

Test LOC-5 was performed in September 1979. The expected results of Test LOC-5 were to have cladding deformation and ballooning at tempertures in the range of 1245 to 1350 K. For these temperatures, a maximum in ductility of the beta phase cladding is approached, and significant cladding ballooning is expected before rupture. The test did not meet its test objectives since only one of the four rods approached the expected cladding temperature.

Future Tests

TFBP experiments planned in the PBF for FY-1980 include two tests in the LOCA series, the first bundle Reactivity Initiated Accident test, the second bundle Power-Cooling-Mismatch test, a combined PCM/RIA test, and the first test in the Operational Transient series (OPTRAN 1-1).

REFERENCES

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Thermal Fuels Behavior Program

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FIGURE 1

TFBP PBF Tests

November 1978 to November 1979

PBF RIA Series 1	RIA 1-2
PBF/LLR Test Series	LLR-3 LLR-5 LLR-4 LLR-4A
PBF LOCA Series	LOC-3 LOC-5
Thermocouple Response Test Series	TC-1 (4 blowdowns)

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FIGURE 2

RIA 1-2 Energy Data

Test/Rod	Burnup (MWd/t)	Peak Pellet Average Energy (cal/g UO_2)	Failure
RIA 1-2/1	5200	180	Intact
2	5100		20%swelling
3	4400		PCI cracks
4	4500		20%swelling

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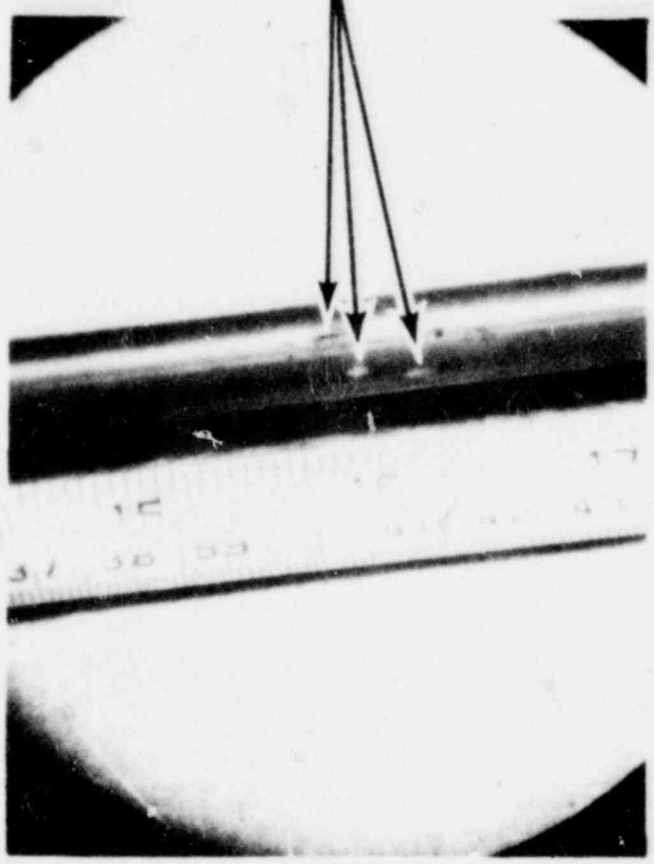
FIGURE 3

Test RIA 1-2, Rod 2-3
Cladding Cracks

Cracks



90°

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FIGURE 4

PBF/LLR Test Results

Test	Rod Power (kW/m)	Rod Cladding Temperature (K)
LLR-3	40.5	870 to 1005
LLR-5	46.2	995 to 1015
LLR-4	56.6	1060 to 1170
LLR-4A	56.0	1150 to 1205

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FIGURE 5

PBF LOCA Test Program

Test Variable	LOC-3	LOC-5	LOC-6	LOC-7
Test rods	2 irradiated / 2 unirradiated			
Cold internal pressure (MPa)	2.41 and 4.83			
Peak cladding temperature (K)	1190	1350	1070	1500
Cladding structure	$\alpha + \beta$	Low β	High α	High β

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FIGURE 6

LOC-3 Thermal Response

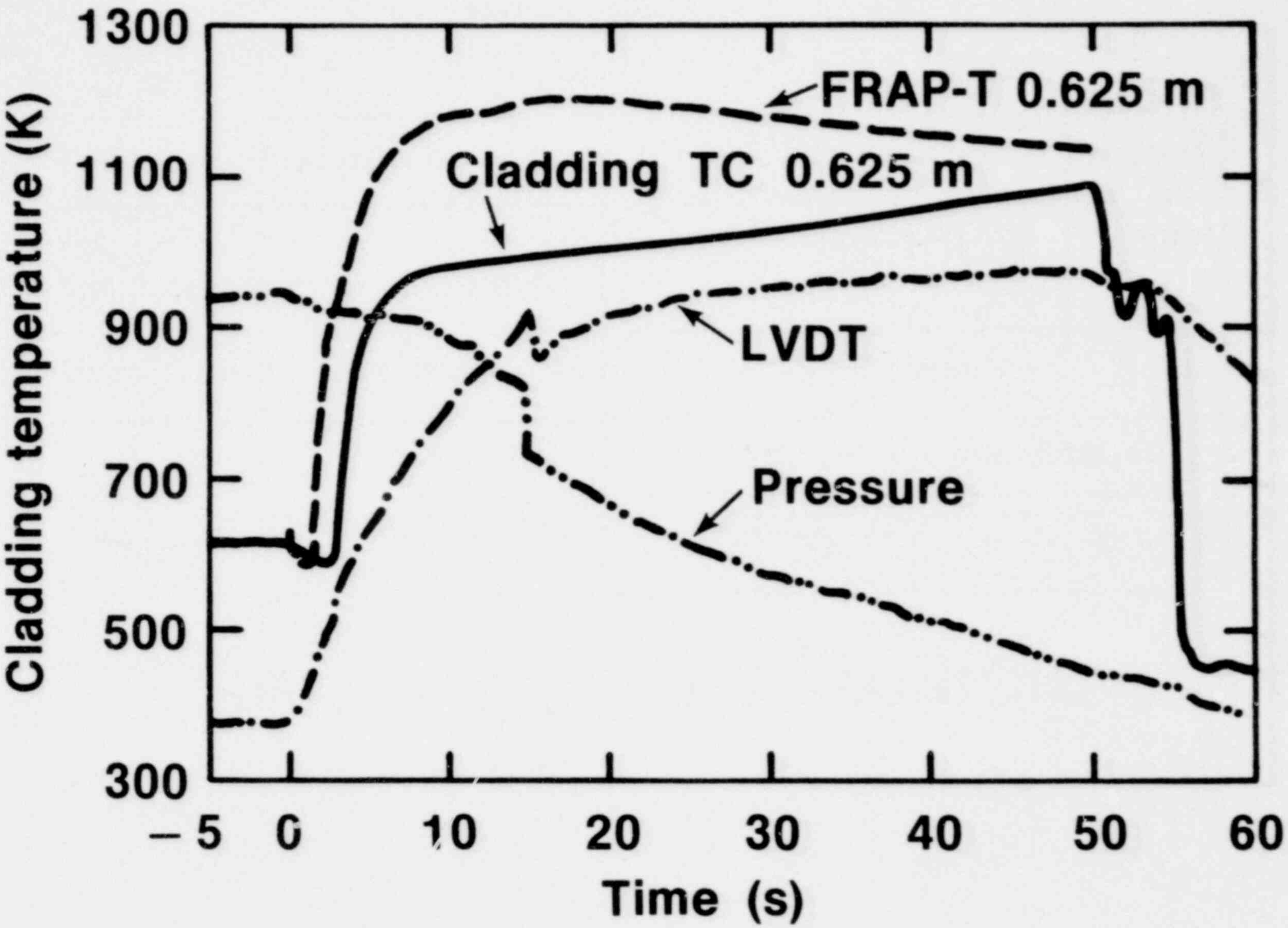

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FIGURE 7

LOC-3 High Pressure Unirradiated Rod

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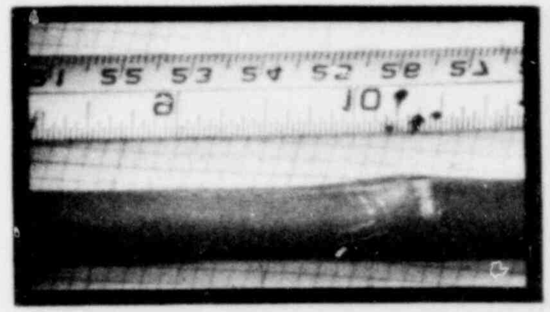

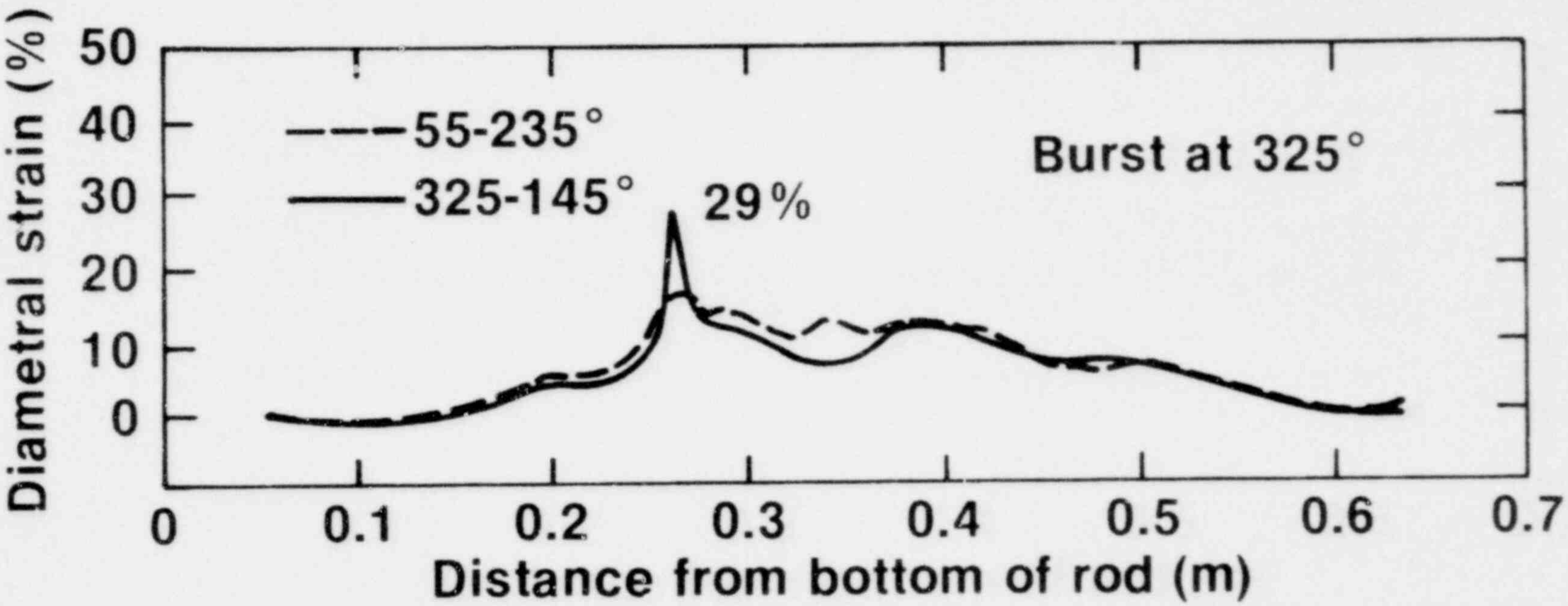

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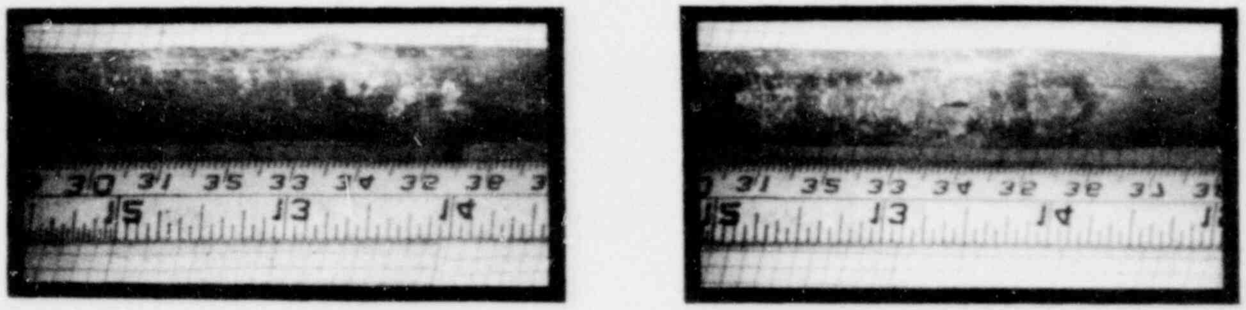
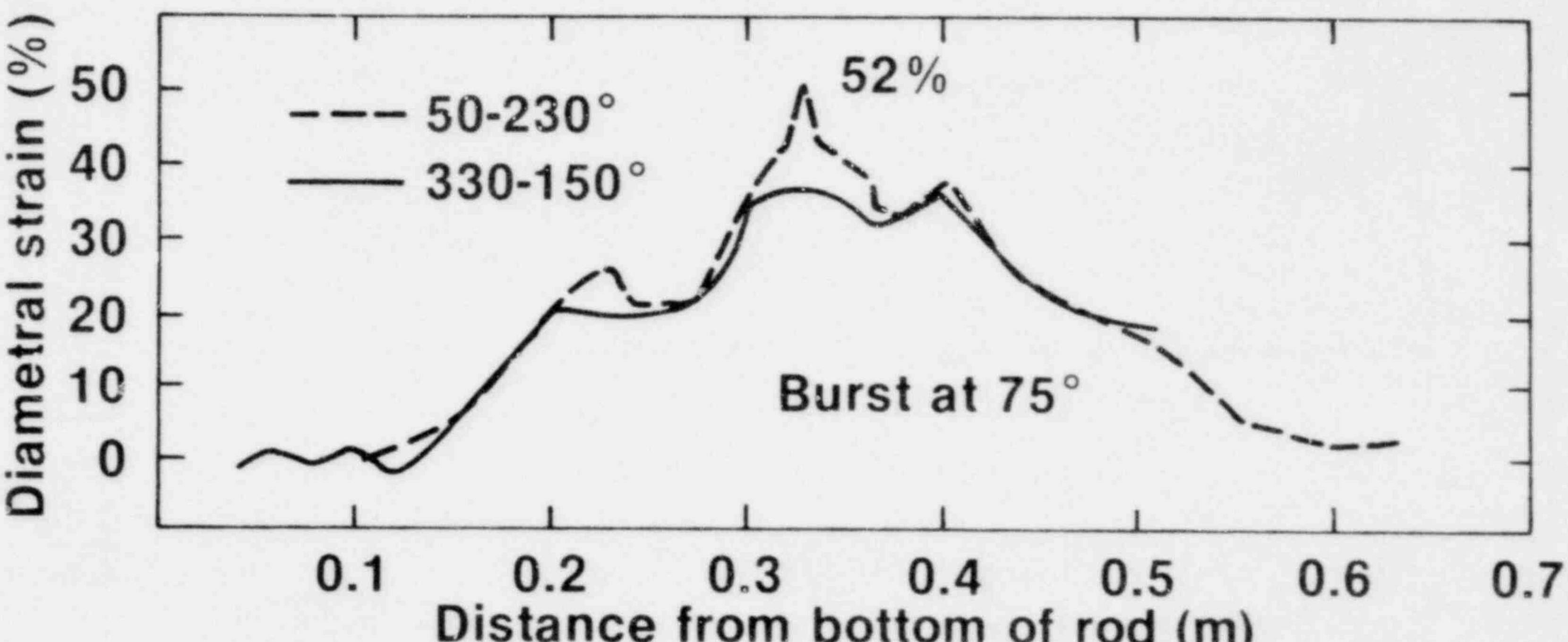
FIGURE 8

LOC-3 High Pressure Irradiated Rod





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