PBF EXPERIMENTAL PROGRAM

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The Thermal Fuels Behavior Program of EG&G Idaho, Inc., conducts fuel behavior research in the Power Burst Facility (PBF) in support of the United States Nuclear Regulatory Commission's (NRC) Fuels Behavior Branch. The fuels behavior research in the PBF is directed toward providing a detailed understanding of the response of nuclear fuel assemblies to off-normal and hypothesized accident conditions. Single fuel rods and clusters of highly instrumented fuel rods are installed within a central test space of the PBF core for testing. The core can be operated in various modes to provide test conditions typical of accidents and off-normal conditions such as:

(1) Power-Cooling-Mismatch (PCM) accidents
(2) Loss-of-Coolant Accidents (LOCA)
(3) Reactivity-Initiated-Accidents (RIA)
(4) Operational Transients With and Without Scram (OPTRAN)
(5) Small Break LOCAs (SBL).

The in-pile experimental data obtained in the PBF experimental program are utilized for the development and assessment of fuel rod analysis codes and for confirming results obtained from out-of-pile and separate effects tests sponsored by the NRC.

There has been significant evolution of the Thermal Fuels Behavior Program (TFBP). In 1974, a 98-test program was planned. Because of funding limitations and changes in priorities for fuels research, the program was adjusted in 1975 to a 40-test program. Since 1975, there have been 11 subsequent adjustments in the PBF test program. These adjustments represent changes in priorities by the NRC, budgetary considerations, and the impact of the results of the TFBP and other fuels research programs. For example, in 1978, several loss-of-coolant accident bundle tests were deleted to minimize duplication with LOCA testing planned eleswhere in the world. Likewise, in January of 1979, an Operational Transient Test Series consisting of four tests was added to the program because of increased NRC priority in this area of fuels behavior research. The program now consists of 37 tests; 25 tests of which have been completed.

The following sections provide and overview of each of the PBF test series, with emphasis on what has been learned to date, and why we are conducting the remaining tests.

PCM Test Series

Fifteen in-pile PCM experiments have been performed in which thirty-one unirradiated fuel rods, nine irradiated fuel rods, and seven rods with irradiated cladding and fresh fuel have been tested. The results indicate that LWR rods can operate in film boiling and incur significant damage without failure. At temperatures below 920 K, cladding damage is minimal except for very long periods (hours) of operation. Cladding deformation (collapse and waisting at nominal PWR conditions) occurs above 920 K, but the cladding has sufficient ductility to accommodate the strains and preclude failure. The primary rod failure mechanism in previously unirradiated and previously irradiated fuel rods has been oxygen embrittlement of the cladding as a result of steam-zircaloy and UO_2-zircaloy reactions. Failure due to oxygen embrittlement during high temperature operation does not occur until the cladding has been nearly completely reacted to ZrO_2 and oxygen-stabilized alpha. Rod failure due to oxygen embrittlement during quench from film boiling operation or during posttest handling is predictable from out-of-pile isothermal steam tests. Room temperature cladding embrittlement from hydriding appears to occur with hydrogen concentrations in the prior beta material as low as 300 ppm. Hydriding has been an embrittlement mechanism only in rods that have failed prior to or during film boiling. Molten fuel-cladding contact, with the potential for cladding melting, has been observed. However, cladding melting has not occurred in the few

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PBF tests in which molten fuel has contacted the cladding. Fuel swelling has occurred in previously unirradiated rods due to thermal expansion, and to a larger extent in previously irradiated rods due to the rentention of fission gases. However, fuel swelling has not resulted in rod failure or significantly affected the behavior of rods with burnups ranging up to 17 000 MWd/t. Fuel grain separation or powdering has been observed in both fresh and previously irradiated test rods.

Results from the first nine-rod cluster PCM test indicate that rod-to-rod film boiling and fuel rod failure propagation did not occur. The departure from nucleate boiling (DNB) and post-DNB behavior of the entire nine-rod cluster was random in nature; however, the individual rod behavior was directly related to power-coolant variations. Two nonadjacent fuel rods within the cluster failed during the test, both due to extensive cladding oxidation and subsequent embrittlement. Most importantly, it was concluded that the center fuel rod in the test cluster behaved in an independent manner such as expected for a fuel rod contained in a separate coolant flow shroud. Therefore, the previously established DNB data base for individually shrouded fuel rods is considered applicable for assessing the DNB response of an intrabundle fuel rod. One more nine-rod cluster test, designated PCM-7, is planned to further substantiate the results obtained from the first cluster PCM test and to obtain additional cluster DNB and rewet (return to nucleate boiling) data.

During most of the four-rod PCM tests, the rods were thermally and hydraulically independent and DNB was induced by decreasing the coolant flow while maintaining constant test rod power. Return to nucleate boiling (rewet) of the test fuel rods was achieved by reestablishing the coolant flow to its original value while maintaining constant test rod power. In one such test, three of the test fuel rods rewet immediately upon flow increase, but one test fuel rod did not rewet until the power was significantly decreased and the coolant flow further increased. The reason for this unusual rewet behavior is not understood. A rewet test, designated PR-1, is planned for the PCM program to specifically investigate this behavior. This four-rod test will consist of a large number of DNB and subsequent rewet cycles. Results of Test PR-1 will yield further information on the thermal-hydraulic conditions at rod rewet, the potential for two-phase instabilities, the effect of fill gas on the onset of DNB and rewet, and additional data on effective fuel conductivity and gap conductance for helium- and argon-filled test rods.

LOCA Test Series

Two LOCA blowdown tests have been completed, and three additional tests are planned. The mechanical behavior of the pressurized fuel rods during a LOCA test primarily depends on the peak cladding per PEM temperatures and internal pressures. The maximum cladding temperature achieved during the first PBF-LOCA test was 1030 K. The resultant test rod deformation was small, but clearly defined the point of incipient deformation. Peak cladding temperatures of about 1070, 1190, 1350, and 1500 K were planned for the remaining four tests. These temperatures correspond to a maximum in ductility of the recyrstallized alpha-phase zircaloy, a minimum in ductility of the alpha plus beta two-phase zircaloy, a maximum in ductility of the beta phase, and an ever decreasing ductility due to oxidation embrittlement as the temperature increases above 1350 K. In each of these four-rod tests, the effects of rod internal pressure and prior irradiation will be evaluated along with the effect of temperature.

The second PBF-LOCA test was conducted as planned with peak cladding per PEM temperatures of 1190 K. All four of the fuel rods in this test ballooned and failed. The preliminary results indicate that previously irradiated rods deform to a greater extent than previously unirradiated rods and that higher pressure rods deform to a greater extent than low pressure rods during a LOCA transient.

The remaining tests in the PBF-LOCA series are required to evaluate cladding ballooning in the alpha, alpha plus beta, and beta phases and to determine the influence on zircaloy deformation of rod internal pressure and prior irradiation. This data set will be used to assess and verify the existing out-of-pile data.

PBF-RIA Test Series

Six tests have been completed in the RIA Test Series, four single-rod tests with peak fuel enthalpies ranging from 185 to 565 cal/g UO_2, and two four-rod tests with peak fuel enthalpies of approximately 260 and 185 cal/g UO_2. Results of the tests indicate that while the failure thresholds for unirradiated and irradiated fuel rods of 225 and 140 cal/g UO_2 (peak fuel enthalpy) are consistent with previous Special Power Excursion Reactor Test (SPERT) and Japanese Nuclear Safety Research Reactor (NSRR) results, the consequences of fuel rod failure at BWR hot startup conditions are more severe than observed in either SPERT or NSRR. The mode of cladding failure for previously irradiated rods at a peak fuel enthalpy of 140 cal/g UO_2 appears to be pellet-cladding mechanical interaction. The mode of cladding failure for previously irraidated fuel rods tested at a peak fuel enthalpy of 260 cal/g UO_2 was rupture caused by fuel-melting-induced and fuel-swelling-induced cladding strain during fuel heatup. Rod failure occurred prior to significant oxidation. Fuel swelling of as much as 180% caused by fission gas release, combined with cladding fragmentation and fuel powdering, caused flow blockage around the separately shrouded, previously irradiated fuel rods. The mode of cladding failure for unirradiated fuel rods tested at peak fuel enthalpies of 245 to 250 cal/g UO_2 was mechanical overstraining of oxygen embrittled cladding during quench. The extensive cracking and crumbling of both previously irradiated and unirradiated fuel rods upon rewet was probably enhanced by thinning of the cladding wall that occurred during the power burst.

Two four-rod tests and two nine-rod bundle tests remain in the RIA Test Series. The next scheduled RIA test will provide data on bundle coolability and coolant channel integrity in a nine-rod cluster following an RIA of 280 cal/g UO_2 peak fuel enthalpy. The test rods will be PWR-type rods previously irradiated to 5300 MWd/t. Two four-rod tests will then provide data on previously irradiated, commercial BWR/6 fuel rods at peak fuel enthalpies of about 220 and 280 cal/g UO_2. These BWR/6 fuel rods were recently provided by the

General Electric Co. The results from these tests will be important because the previous RIA tests were performed using PWR-type fuel rods of lower burnup and because the largest postulated RIA peak enthalpy occurs for BWR rather than PWR plants.

The last RIA bundle test will also be performed using commercial BWR/6 fuel rods. The purpose of this test is to investigate coolability and coolant channel integrity in a nine-rod cluster of commercial BWR/6 fuel rods following an RIA of about 220 cal/g UO_2 peak fuel enthalpy.

OPTRAN Test Series

A new area of reserach for the TFBP is the OPTRAN Test Series. The initial OPTRAN research will focus on operational transients classified by the NRC as those that could potentially occur with moderate frequency, but should not result in fuel rod failure; as well as incidents with potential for infrequent occurrence, which may result in some rod failure.

Four tests are planned in the series, with the first scheduled for August 1980. Three of the tests will consist of four individually shrouded, previously irradiated test rods of BWR/6 design. One will be a 3 x 3 bundle test of the same rod type. The first three tests will simulate the calculated fuel and cladding temperature histories during a turbine trip without bypass transient in a boiling water reactor. Since the frequency of occurrence of these events, and other anticipated transients which induce similar rod power and coolant changes, is high in a commercial reactor, approximately 15 transients will be performed during each test to examine the potential for cumulative damage. Fuel rod damage mechanisms of particular interest include: (a) cladding collapse where dryout is predicted, and (b) pellet-cladding interaction (PCI) combined with stress corrosion cracking (SCC) for overpower transients which are not severe enough to trigger the boiling transition.

The fourth OPTRAN test will examine the effects of an anticipated transient with the postulated failure of the control rod scram system. The objectives of this anticipated transient without scram (ATWS) test will be to: (a) perform a single power transient simulating the expected power history and dryout conditions predicted by the General Electric Co., for a BWR/6 main steamline isolation valve (MSIV) closure without scram; (b) measure cladding temperatures in the dryout region; (c) measure permanent mechanical deformation of the cladding; and, (d) determine the extent of cladding oxidation in the dryout region. The probable damage mechanism is cladding collapse.

Small Break LOCA Test Series

A small break LOCA fuel behavior test program to study fuel behavior during nondesign basis LOCA accidents has been initiated for the PBF. The program will include bundle testing with slow system depressurizations and reductions of coolant flow similar to the Three Mile Island accident. The primary program objective is to characterize rod and core damage during a small break LOCA due to cladding oxidation and hydriding, zircaloy-UO_2 eutectic formation, and rod fragmentation. Secondary objectives include (a) an evaluation of the effects of heatup rate and prior oxidation on eutectic melting and rod fragmentation, (b) an evaluation of the effects of rod internal pressure and subsequent ballooning on cladding oxidation and eutectic melting, (c) measurement of fission product release and transport, and (d) measurement of framgented bundle heat transfer as a function of flow rate and pressure.

The initial phase of this program will include six to ten tests with peak cladding temperatures up to approximately 2300 K. The second phase of the program will continue the evaluation of fuel and cladding behavior, but at cladding temperatures from 2300 K up to that of molten UO_2. Data on the various material and structural problems unique to the containment of molten UO_2 will be obtained from the second phase of testing.

An updated Program Requirements Document is planned to identify additional fuel behavior research needs for accidents that are beyond design basis accidents.

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PBF Experimental Program

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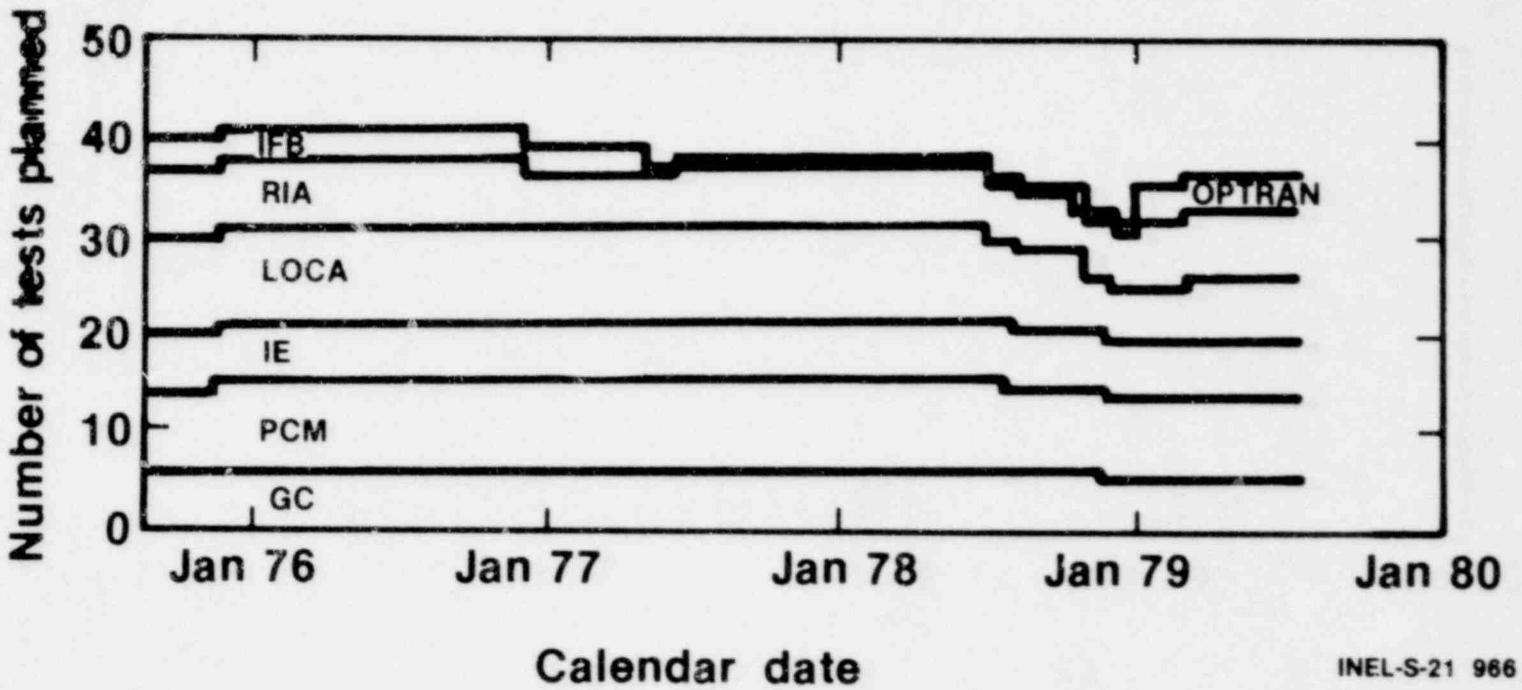
EG&G Thermal Fuels Behavior Test Program



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Single Rod PCM Test Results

- Cladding deformation (collapse or waisting) occurs between 870 to 920 K
- Zircaloy embrittled by Zr/water and Zr/O_2 reactions
- Posttest fracture due to embrittlment predictable by 95% O_2 saturation and 0.7 wt% criteria
- Greater embrittlement of failed rods upon cooldown due to hydrogen absorption

INEL-S-21 932

Single Rod PCM Test Results (cont'd)

- Fuel rods can operate in film boiling and incur significant damage without failure
- Clad failure due to MFCI unlikely unless molten fuel highly superheated
- Energetic molten fuel/coolant interaction did not occur as a result of failure at high test rod peak powers (78 kW/m)

INEL-S-21 929

Single Rod PCM Test Results (cont'd)

- At-power oxidation failure after complete reaction of cladding
- Fuel powdering may occur when fuel is quenched from temperatures above 1900 K
- Fission gas induced fuel swelling in moderate burnup rods

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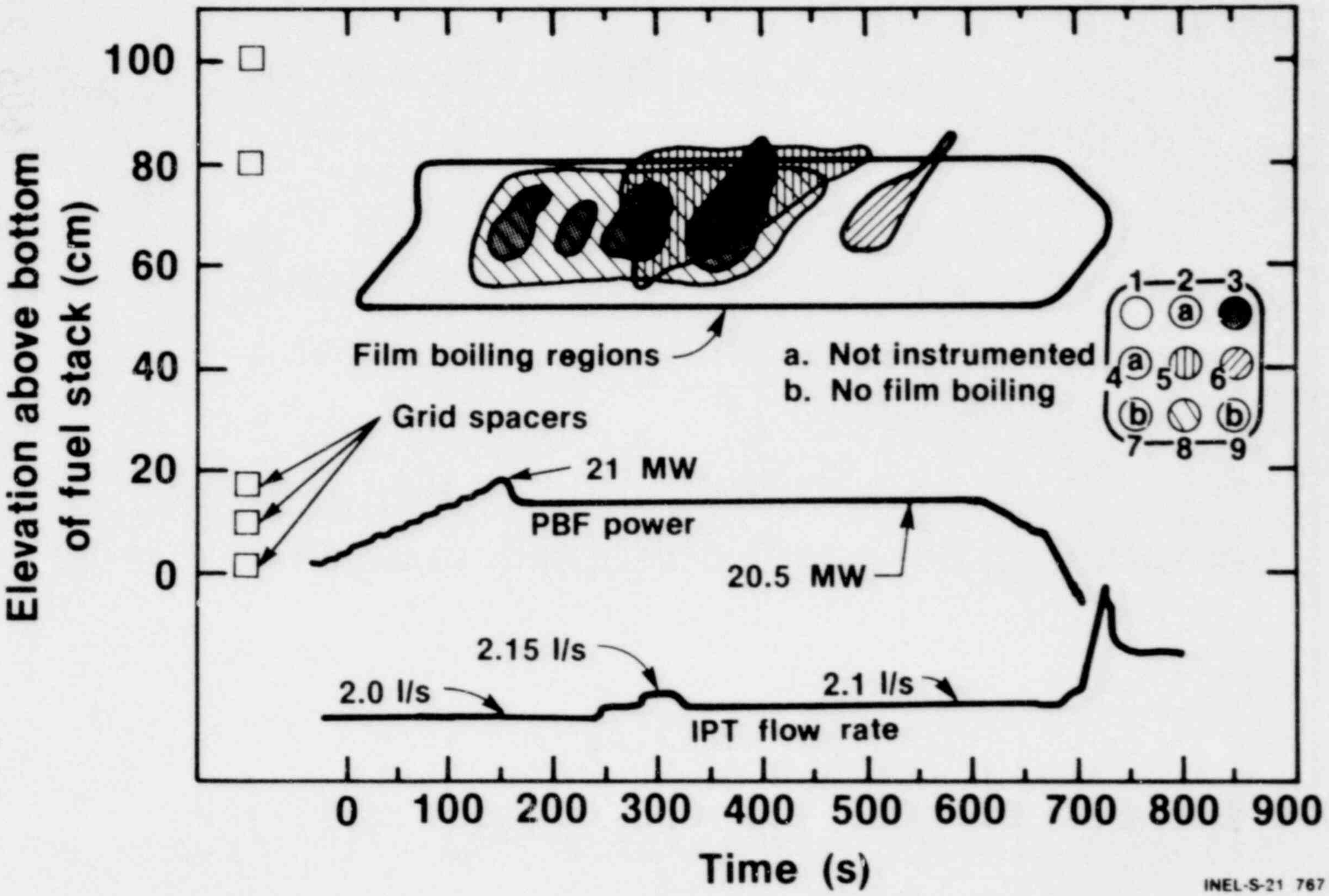
PCM-5 Film Boiling Scenario



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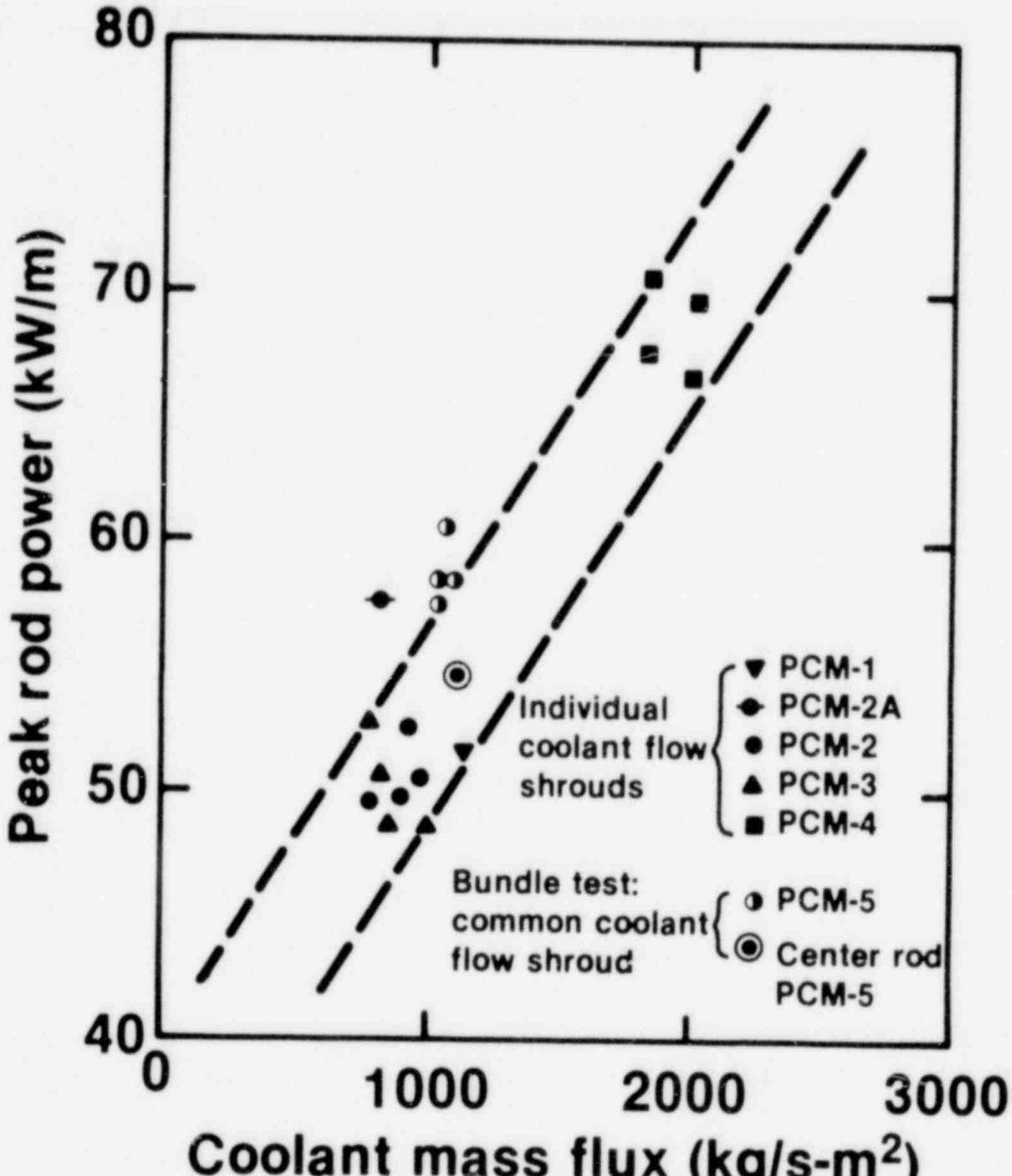
Comparison of CHF Conditions Bundle Rod vs Single Rod



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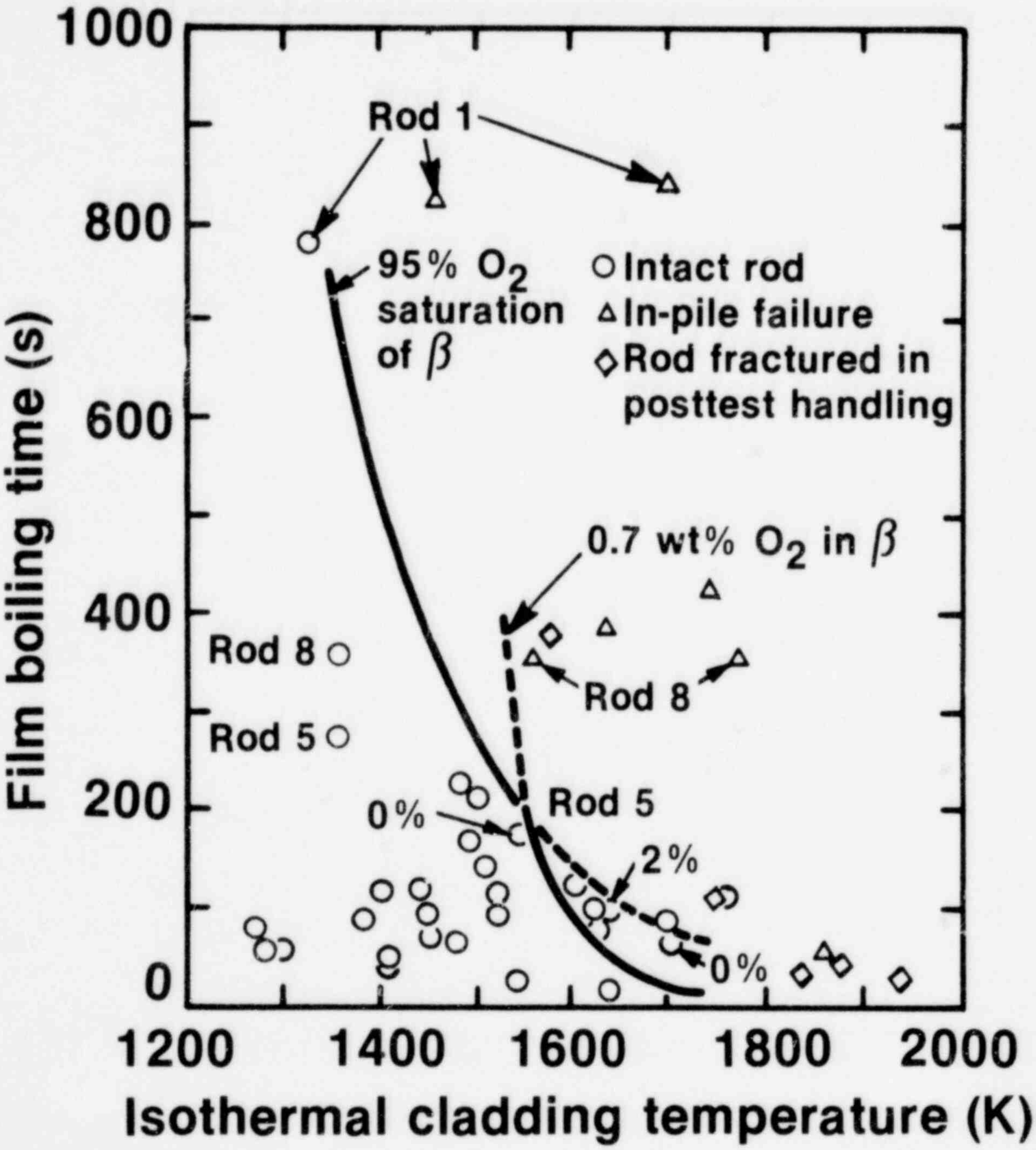
Fracture Map of Rods Tested Under PCM Conditions



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Results of Small Cluster Test (PCM-5)

- No evidence of film boiling or rod failure propagation
- Film boiling sequence was random relative to position within the bundle
- Individual rod DNB and RNB behavior coincident with power/coolant changes
- Film boiling behavior of an interior rod within a small cluster is predictable from single rod data base
- The embrittlement and fracture behavior was consistent with previous single rod test results

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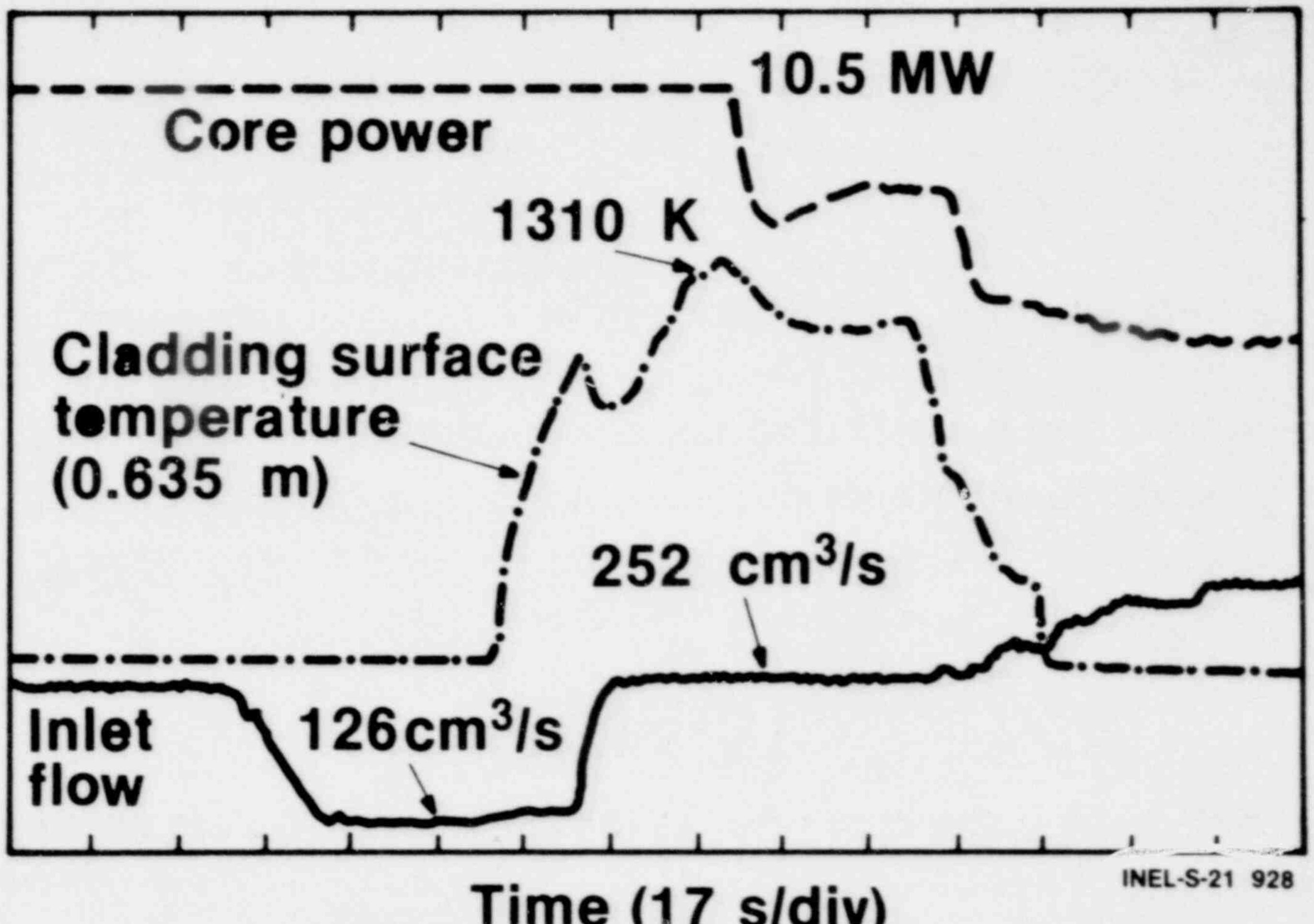
Thermal Fuels Behavior Program Results of PCM-2 Test



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Information Expected from Test PR-1

(December 1979)

- T-H conditions and cladding temperatures upon rewet

- Potential for two-phase instabilities (Leninegg and density wave)

- Effect of fill gas on onset of DNB and rewet

- Effective fuel conductivity and gap conductance data

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PBF LOCA Program Objectives

- Obtain in-pile zircaloy burst temperature and pressure data for verification of out-of- pile data

- Obtain in-pile data on the magnitude and axial extent of cladding ballooning

- Evaluate irradiation effects on cladding deformation behavior

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PBF LOCA Test Program

Test Variable	LOC-3	LOC-5	LOC-6	LOC-7
Unirradiated rods [1]	2	2	2	2
Irradiated rods	2	2	2	2
Cold internal pressure (MPa)	2.41 & 4.83	2.41 & 4.83	2.41 & 4.83	2.41 & 4.83
Axial power profile	Flat	Flat	Flat	Flat
Peak cladding temperature (K)	1190	1350	1070	1500
Cladding structure	$\alpha + \beta$	Low β	High α	High β
Time in CHF (s)	50	50	50	50
Test date	June, 79	Sept, 79	April, 80	June, 80

[1] Rod and bundle dimensions typical of a PWR 15×15 configuration.

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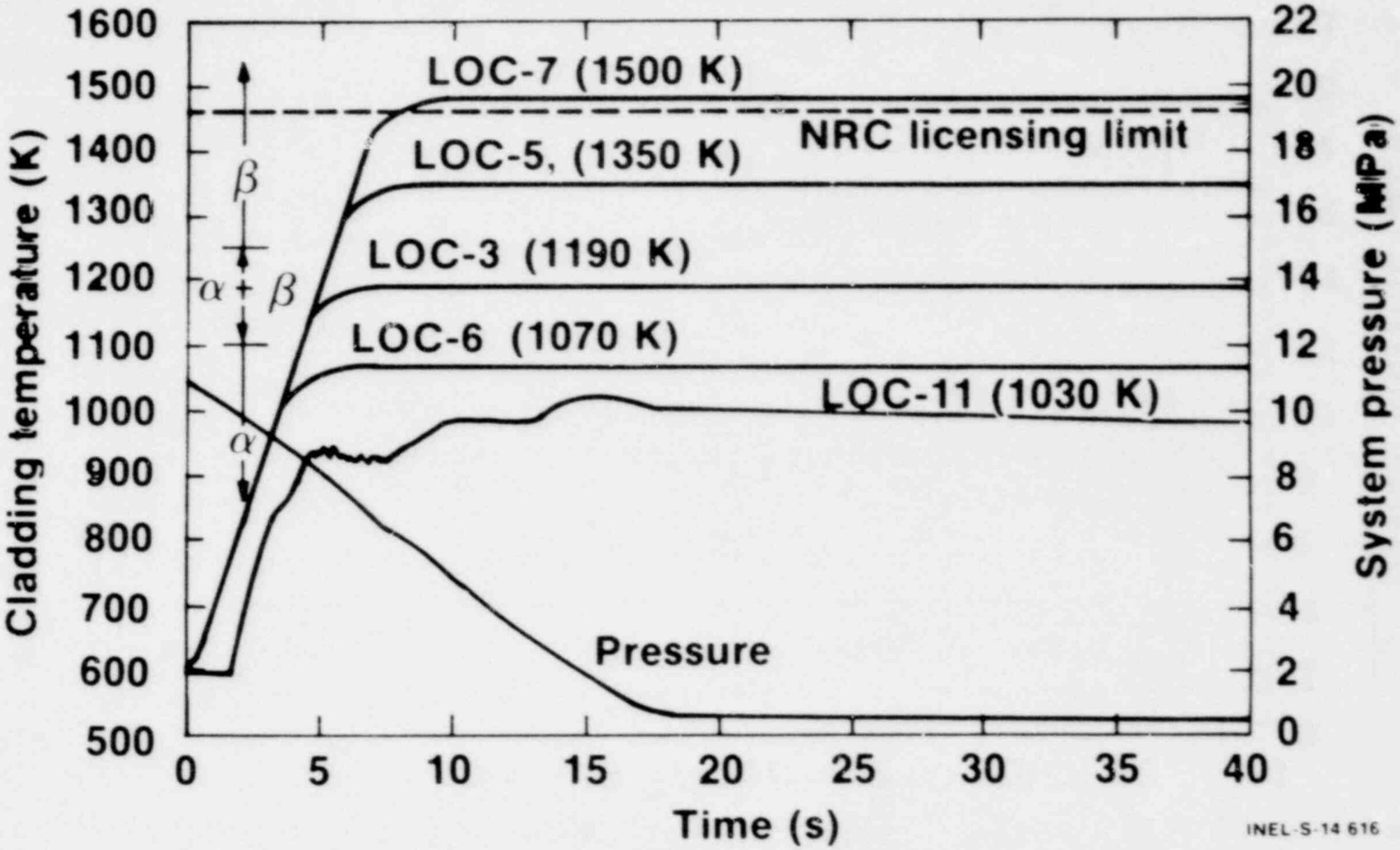
PBF LOCA Test Clad Temp Histories



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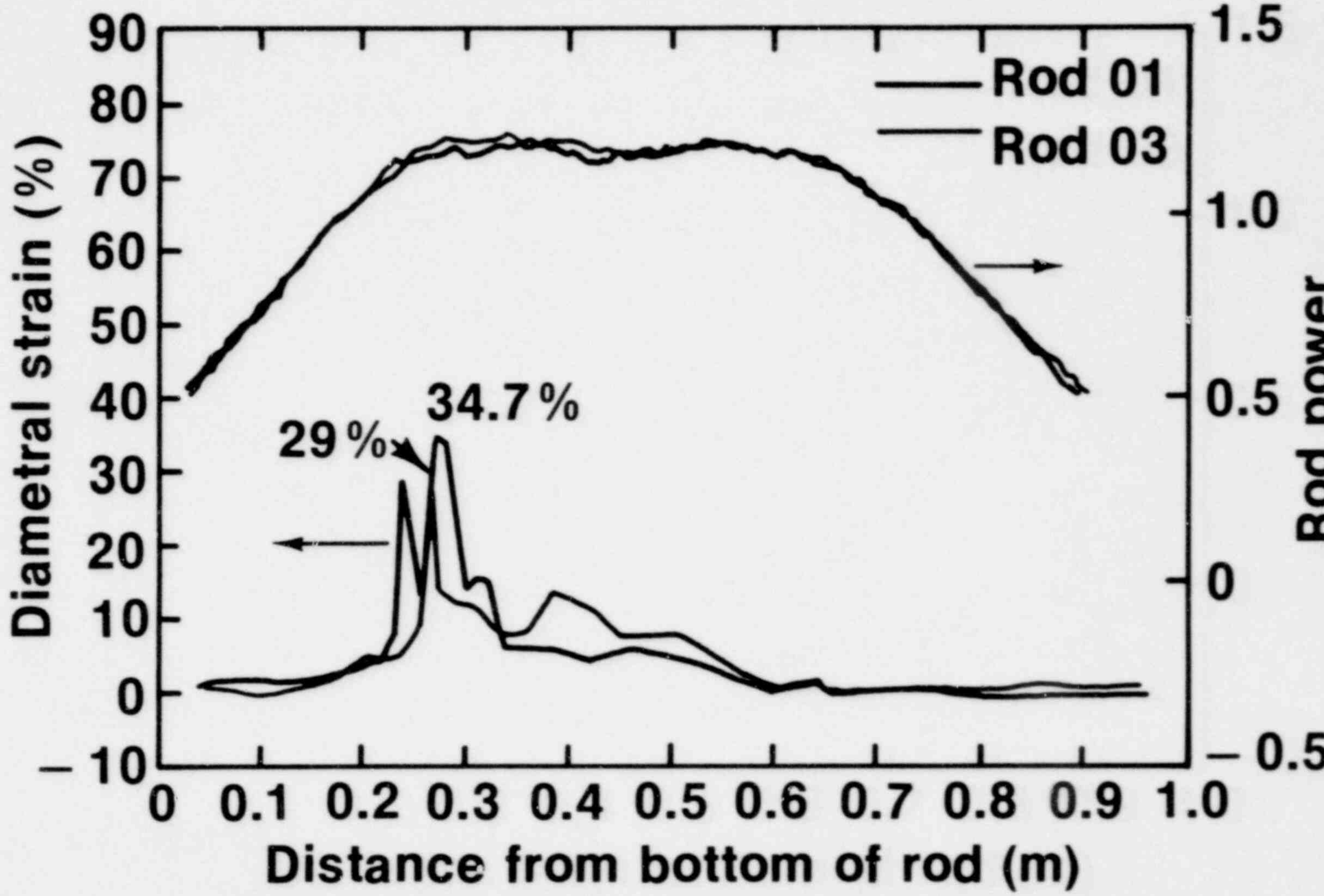
LOC-3 Unirradiated Rods 01 and 03



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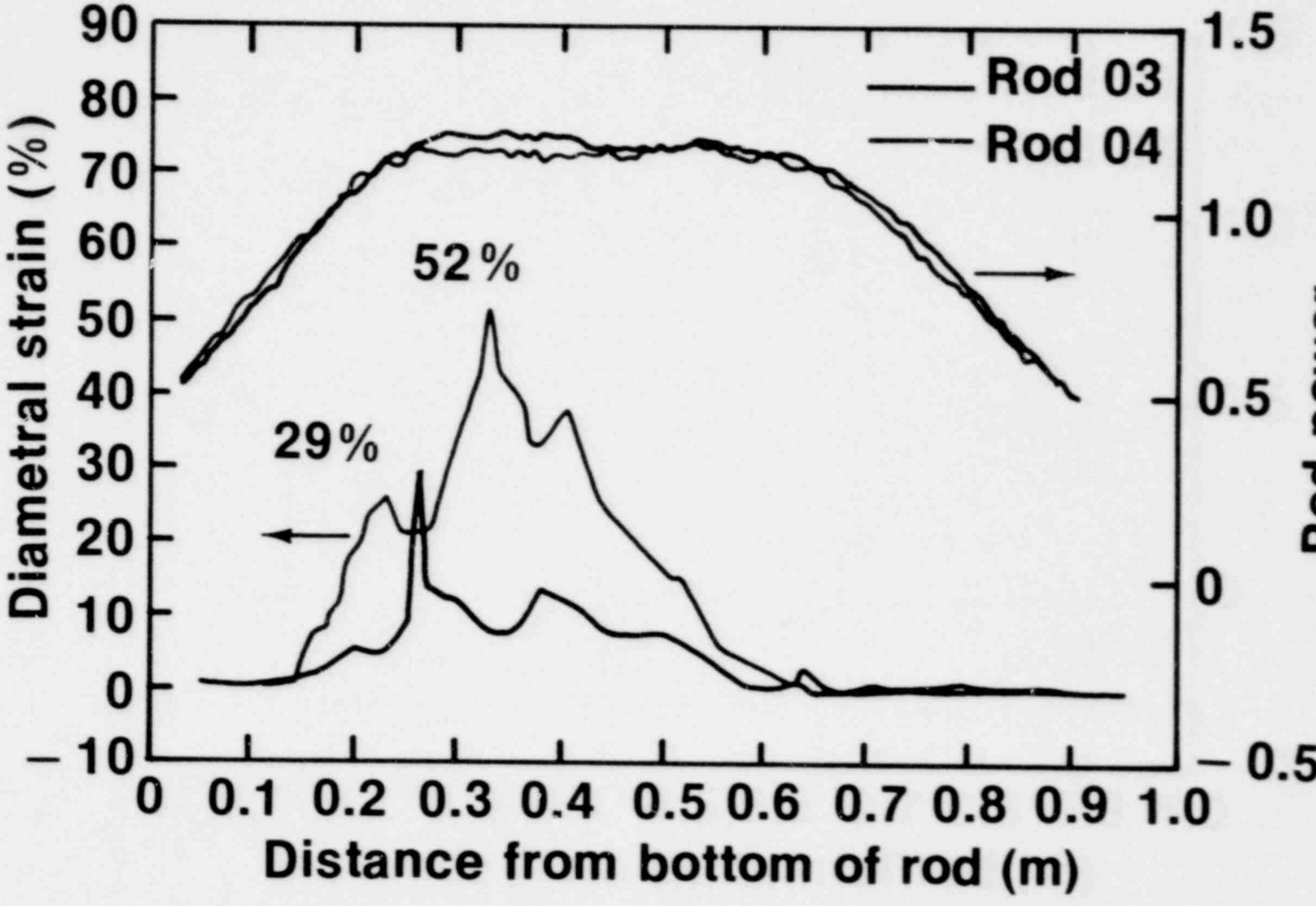
LOC-3 High Pressure Rods 03 and 04



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Expected LOC-5, LOC-6 and LOC-7 Results

- Fuel deformation data at 1070 K, 1350 K and 1500 K
- Evaluation of rod internal pressure and prior irradiation on rod behavior during a LOCA

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RIA Test Program Objectives

- Define energy depositon failure threshold for unirradiated and irradiated fuel rods
- Characterize fuel rod damage at energy depositions near the NRC limit of 280 cal/g
- Define energy deposition threshold for loss of coolable geometry
- Obtain detailed measurements of fuel behavior during an RIA at BWR hot startup conditions

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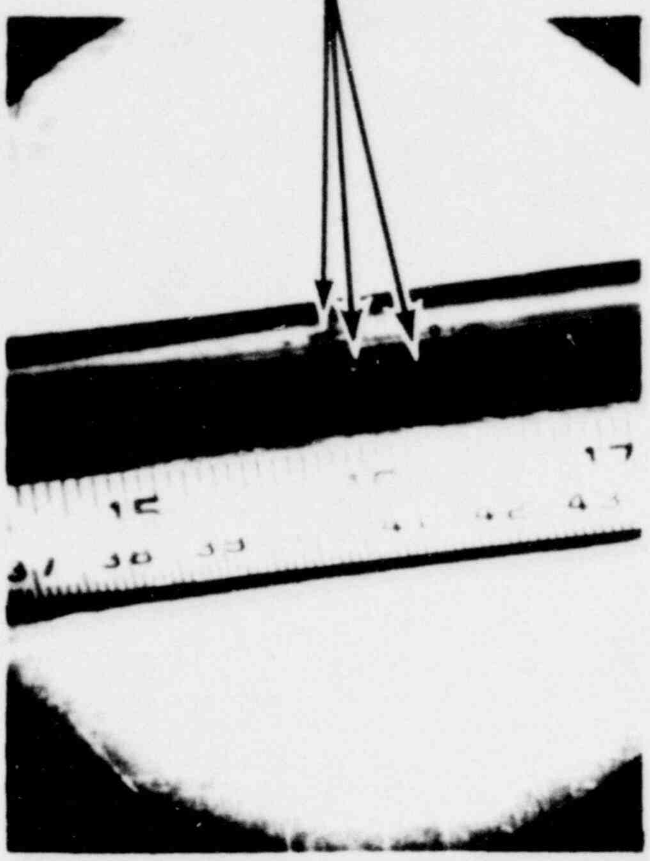
Test RIA 1-2, Rod 2-3
Cladding Cracks

Cracks



90° A-487

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Test RIA 1-1 Rod 801-1

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RIA Scoping Test 1 - Photomicrograph Cladding at Peak Flux Location, 0.35-m Elevation

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Future RIA Test Objectives

Test	No. of Rods	Peak fuel enthalpy (cal/g UO_2)	Objective
1-4	9	280	Investigate the coolability of a bundle of pre-irradiated rods
1-3	4	220	Test commercially irradiated BWR/6 fuel at alternate licensing criteria
1-7	9	220	Investigate coolability of a bundle of commercially irradiated BWR/6 fuel rods
1-6	4	200	Investigate alternate licensing criteria

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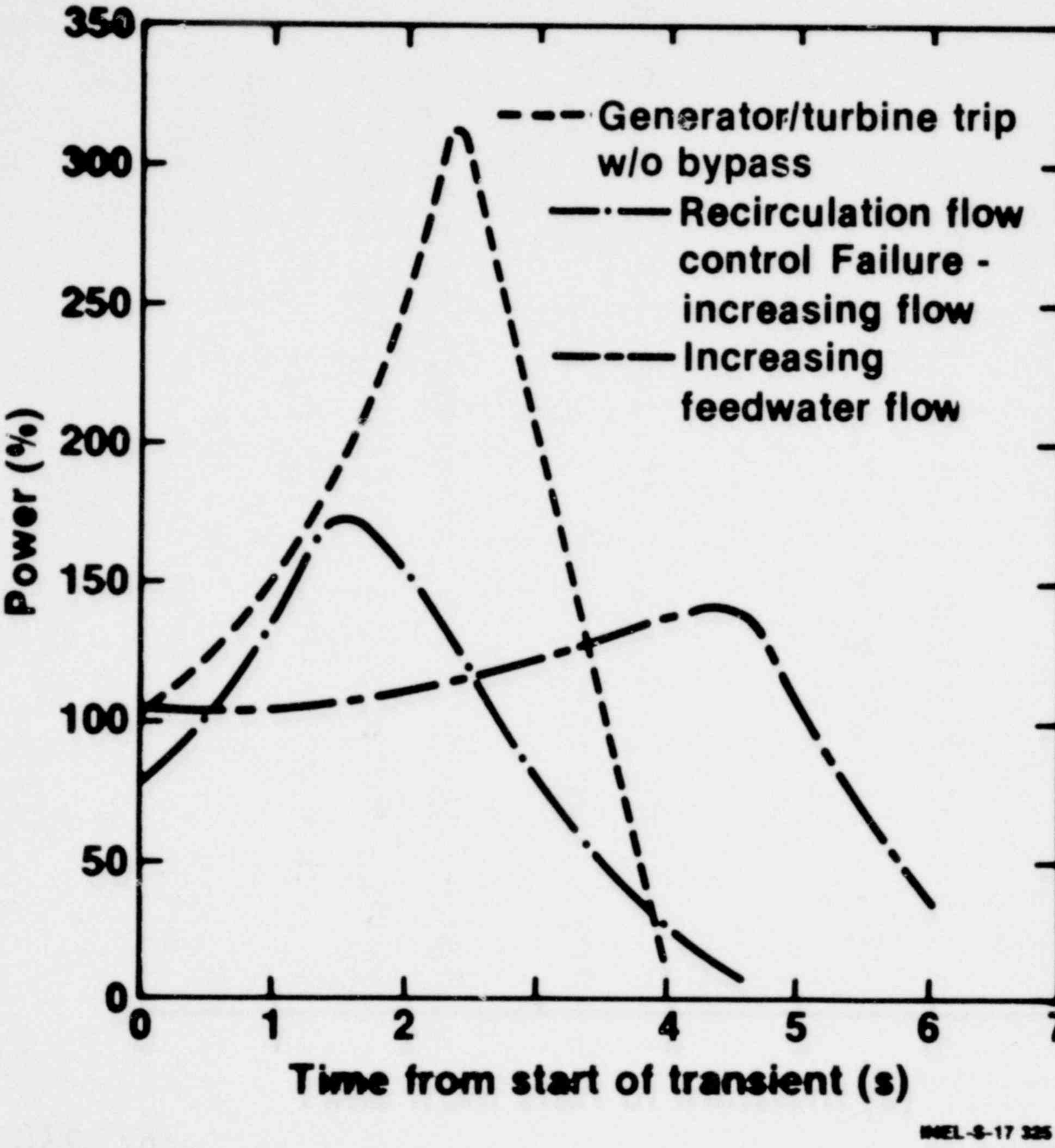
BWR Anticipated Transients



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The fuel damage mechanisms of particular interest:

- Cladding collapse where film boiling or dryout is indicated
- Pellet-cladding interaction combined with stress corrosion cracking for the overpower transients

INEL-S-20 058

Summary of OPTRAN Tests

OPTRAN Test Number	Number of Transients	Number of Rods	Type of Accident
1-1	17	4	BWR/6 turbine trip w/o bypass
1-2	15	4	BWR/5 turbine trip w/o bypass
1-3	17	9	BWR/6 turbine trip w/o bypass
1-4	1	4	MSIV closure w/o scram

INEL-S-21 935

Small Break LOCA Fuel Behavior Test Program

- Evaluate fuel rod behavior during non-design basis LWR accidents

- System response characteristics

    - Slow depressurization

    - Coolant flow starvation at bundle inlet

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INEL-S-21 933

Small Break LOCA Test Objectives

- Characterize core damage
  - Oxidation
  - Hydriding
  - Zr/UO_2 eutectic formation
  - Rod fragmentation
- Evaluate the effects of heatup rate and prior oxidation on eutectic melting and rod fragmentation
- Evaluate the effects of rod internal pressure and subsequent ballooning on cladding oxidation, eutectic melting
- Monitor fission product release and transport
- Determine fragmented bundle heat transfer as a function of flow rate and pressure

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Test Program

- Phase One
  - 6-10 tests
  - Peak temperatures of ~2300 K
  - Slow system depressurization with flow reduction
- Phase Two
  - Peak temperatures of ~3300 K
  - System response to be determined