OVERVIEW OF THE PBF TEST RESULTS

MASTER

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EG&G Idaho, Inc. performs a variety of research activities in support of the U.S. Nuclear Regulatory Commission's (NRC) water reactor safety programs. Most of these research activities are conducted at the U.S. Department of Energy's Idaho National Engineering Laboratory (INEL). The current water reactor safety research conducted by EG&G Idaho is accomplished in the following programs: the Semiscale Program, the Loss-of-Fluid Test Experimental Program, the Thermal Fuels Behavior Program, the Code Development and Analysis Program, the Code Assessment and Applications Program, and Engineering Support Projects. These U.S. NRC-sponsored research programs are aimed at providing the NRC capability for independent assessment of the engineered safety features in nuclear plant designs to ensure that the consequences of postulated accidents can be mitigated.

The Thermal Fuels Behavior Program (TFBP) of EG&G Idaho conducts fuel behavior research in the Power Burst Facility (PBF) at INEL and at the Halden Reactor in Norway. The fuels behavior research in the PBF is directed toward providing a detailed understanding of the response of light water reactor (LWR) 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 that may be experienced in a pressurized water reactor or a boiling water reactor.

The TFBP-PBF fuels research is concerned with the following accident and transient conditions:

(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 such as FRAPCON[a] and FRAP-T[b].

Over the years significant changes in the focus of Thermal Fuels Behavior Program experiments have occurred. In 1975 the program was adjusted to a 40-test program. Since 1975, there have been 11 subsequent adjustments in the PBF test program. These adjustments represent changes in priorities and the impact of the experimental results obtained by TFBP and other fuels behavior research programs. For example, in 1978, several loss-of-coolant accident bundle tests were deleted to minimize duplication with LOCA testing planned elsewhere 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 priority assigned by the NRC to this area of fuels behavior research. The program now consists of 37 tests; 26 of which have been completed.

In addition to programmatic tests, the PBF is frequently utilized to conduct in-pile experiments in support of other NRC program needs. Two such nonprogrammatic test series recently performed are the Japanese Government-funded LOFT Lead Rod (LLR) Test Series consisting of four LOCA blowdowns and the Thermocouple Quench Test Series (Series TC-1).

The following sections provide an overview of each of the PBF test series, with emphasis on key results learned to date and what further results will be obtained from the remaining tests in the program.

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a. FRAPCON, MOD-1, Version 4. Idaho National Engineering Laboratory Configuration Control Number H00730IB. This code is available from the National Energy Software Center, Building 208 - Room C-230, 9700 South Cass Avenue, Argonne, Illinois 60439.

b. R. M. Oehlberg, M. V. Johnston, J. A. Dearien, "FRAP Fuel Behavior Computer Codes," Nuclear Safety, 19(5) (1978).

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 fuel rods can operate in film boiling and incur significant damage without failure.[1]

At temperatures below 902 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 retains sufficient ductility to accommodate the strains and preclude failure. The primary rod failure mechanism in both unirradiated and 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 zirconium oxide and oxygen-stablished alpha.[2,3] Zirconium hydriding has been noted as an embrittlement mechanism only in rods that have failed prior to or during film boiling. Molten fuel-cladding contact, a concern because of the increased potential for cladding melting, has been observed in a few PCM tests; however, cladding melting has not occurred. Fuel swelling has occurred in previously unirradiated rods due to thermal expansion, and to a larger extent in previously irradiated rods due to the retention of fission gases. However, fuel swelling has not resulted in rod failure or significantly affected the behavior of rods tested with burnups ranging up to 17 000 MWd/t. Fuel grain separation, powdering, has been observed in both fresh and previously irradiated test rods when these rods were quenched from temperature above 1900 K.[4]

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.[5] Therefore, use of the previously established DNB data base for individually shrouded fuel rods is made more credible for assessing the DNB response of an intrabundle fuel rod.

One more nine-rod cluster test, Test PCM-7, is planned to further substantiate the results obtained from the first cluster PCM test and to obtain additional DNB and rewet (return to nucleate boiling) data.

During most of the four-rod PCM tests, the rods were thermally and hydraulically isolated. 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 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 test, Test PR-1, is planned for the PCM program to investigate this rewet behavior. This four-rod test will consist of a large number of DNB and subsequent rewet cycles. Results from Test PR-1 are expected to 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

The major objective for the PBF LOCA test series is to experimentally evaluate the extent of fuel rod cladding deformation during severe loss-of-coolant conditions.

Three programmatic LOCA blowdown tests have been completed in the PBF LOCA series and two additional tests are planned. The mechanical behavior of the pressurized fuel rods during a LOCA test depends primarily on the peak cladding temperatures and internal pressures. The maximum cladding temperature achieved during the first PBF-LOCA test (LOC-11) was 1030 K. The resultant test rod deformation was small, but clearly defined the point of incipient deformation.

The second PBF-LOCA test (LOC-3) was conducted as planned with peak cladding temperatures of about 1200 K. All four of the fuel rods in this test ballooned and failed.

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. Both the previously irradiated and previously unirradiated 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, the third in the PBF LOCA series, was to achieve cladding deformation and ballooning at temperatures 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 cladding heatup rates for Test LOC-5 were greater than those for Test LOC-3 and a significant difference in time to failure was noted between the results of the two tests indicating that heatup rates as well as cladding temperature during blowdown may have significant influence on ballooning mechanisms of fuel rods during a LOCA.[6] Detailed evaluation of the data from these tests is underway.

The remaining two tests in the PBF-LOCA series will evaluate cladding ballooning in the alpha and beta phases and determine the influence on zircaloy deformation of rod internal pressure and prior irradiation.

On the basis of the differences in fuel rod behavior between in-pile and out-of-pile LOCA experiments, additional in-pile experiments appear to be warranted. Experiments with irradiated fuel rods with end-of-life internal pressure condition in a bundle configuration should be performed to ensure availability of in-pile data to address the safety issues that have been raised. Some of these areas, if not all, may be addressed by results from the PHEBUS Program in France, the ESSOR Program at ISPRA, and the NRC sponsored NRU Program.

Nonprogrammatic Tests Series

One of the primary objectives of the PBF LOFT Lead Rod Test Series was to evaluate the effects of collapsed cladding and pellet-cladding interaction on the mechanical response of fuel rods subjected to power increases, long-term preconditioning, and loss-of-coolant transients. The other objectives were more specifically related to LOFT needs.

During the third of the four LOCA transients in this series, Test LLR-4, the maximum measured cladding surface temperatures ranged from 1060 to 1170 K. The temperatures were probably higher at lower elevations on the test rods. Since test Rod 312-1, removed after Test LLR-4 for postirradiation examination, had waisted over a distance from 35 to 55 cm (from the bottom of the heated length), we feel confident that the three test rods that were subjected to an additional LOCA blowdown, Test LLR-4A, had also reached the waisting regime of mechanical deformation during Test LLR-4.

The important observation from the LOFT Lead Rod LOCA series is that even after the fuel rods have been exposed to temperatures of around 1200 K and have achieved the waisting regime of mechanical deformation, they retain adequate ductility to survive subsequent heatup, power cycling during preconditioning, and a LOCA transient without failure.

One of the key questions of concern to the LOCA-ECCS experimenters for some time was the effect of externally mounted cladding surface thermocouples on fuel cladding temperature and fuel rod behavior.

Test Series TC-1, consisting of four LOCA blowdowns, was performed in the PBF to specifically evaluate the influence of cladding surface thermocouples on the thermal-mechanical behavior of fuel rods during a LOCA. The PBF LOCA transient contained both the blowdown and the reflood phases and the PBF reactor was automatically controlled to achieve cladding temperatures in the range of 800 to 1000 K.

In addition, the blowdown valves were automatically cycled to force a two-phase liquid slug from the lower plenum past the fuel rods during the later phases of the blowdown to simulate the quench during the blowdown in the LOFT L-2 Test.

The data from the Test TC-1 experiment series indicates that the external thermocouples will delay CHF and have a favorable influence on fuel rod surface heat transfer. The surface thermocouples also enhance cladding quench during reflood.

A second LOCA series (Series TC-2) to investigate thermocouple effects during a blowdown quench with conditions that will more closely simulate the LOFT L-2 Test conditions is planned for later this year.

RIA Test Series

Detailed PBF RIA test results will be presented by Mr. R. McCardell; therefore, I will not dwell on these test results. The six tests completed in the PBF RIA Test Series, indicate that although the failure thresholds for unirradiated and irradiated fuel rods are consistent with the previous

Special Power Excursion Reactor Tests (SPERT) and Japanese Nuclear Safety Research (NSRR) results, the consequences of fuel rod failure due to RIA during BWR hot startup conditions are more severe.

Four experiments remain in the PBF RIA Test Series. The focus of these experiments will be to evaluate the severity of fuel rod damage as it relates to irradiated fuel rod bundle, coolant channel integrity, and bundle coolability with peak fuel enthalpy of 140 to 280 cal/g UO_2.

Two four-rod tests and the last nine-rod bundle test will utilize BWR/6 irradiated fuel rods recently provided by the General Electric Company. The results of these tests will be important because previous PBF RIA tests were performed using PWR type-fuel rods with relatively low burnup.

OPTRAN Test Series

A new area of research for the Thermal Fuels Behavior Program 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 late this year. Three of the tests will consist of four individually shrouded, previously irradiated test rods of BWR/5 or BWR/6 design. The fourth will be a 3 x 3 bundle test of the BWR/6 rod type. The first three tests will simulate the calculated rod power transient and fuel and cladding temperature histories during a turbine trip without bypass transient in a boiling water reactor. Since the frequency of occurrence of 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) simulate in a single power transient, the expected power history and dryout conditions predicted by the General Electric Company, 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

Planning for a small break LOCA fuel behavior test program in PBF has been started. This program will include bundle testing with slow system depressurizations and reductions of coolant flow similar to the conditions that occurred in 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 fragmented bundle heat transfer as a function of low rate and pressure.

The initial phase of this program will include tests with peak cladding temperatures up to approximately 2300 K. The second phase of the program will continue the evaluation of fuel rod behavior at 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.

REFERENCES

1. A. S. Mehner et al, "Damage and Failure of Unirradiated and Irradiated Fuel Rods Tested Under Film Boiling Conditions," Proceedings of ANS Topical Meeting on Light Water Reactor Fuel Performance, Portland, Oregon, April 29-May 3, 1979.

2. B. A. Cook, "Fuel Rod Material Behavior During Test PCM-1," NUREG/CR-0757, TREE 1333 (May 1979).

3. D. T. Sparks et al, "Film Boiling Behavior in a Nine Rod Cluster," Trans. Am. Nuc. Society, 30, (November 1978) p. 404.

4. A. W. Cronenberg and T. R. Yackle, "An Assessment of Intergranular Fracture Within Unrestructured UO_2 Fuel Due to Film Boiling Operation," NUREG/CR 0595, TREE 1330 (March 1979).

5. Fred S. Gunnerson and Daniel T. Sparks, "Behavior of a Nine-Rod Fuel Assembly During Power-Cooling-Mismatch Conditions - Results of Test PCM-5," NUREG/CR-1103, EGG-2002 (November 1979)

6. Quarterly Technical Progress Report on Water Reactor Safety Programs Sponsored by the Nuclear Regulatory Commission's Division of Reactor Safety Research, October - December 1979, NUREG/CR-1203, EGG-2012 (January 1980).

1.

Presentation Outline

- Thermal Fuels Behavior Program
- TFBP Significant Test Results
- OPTRAN and Small Break Experiments

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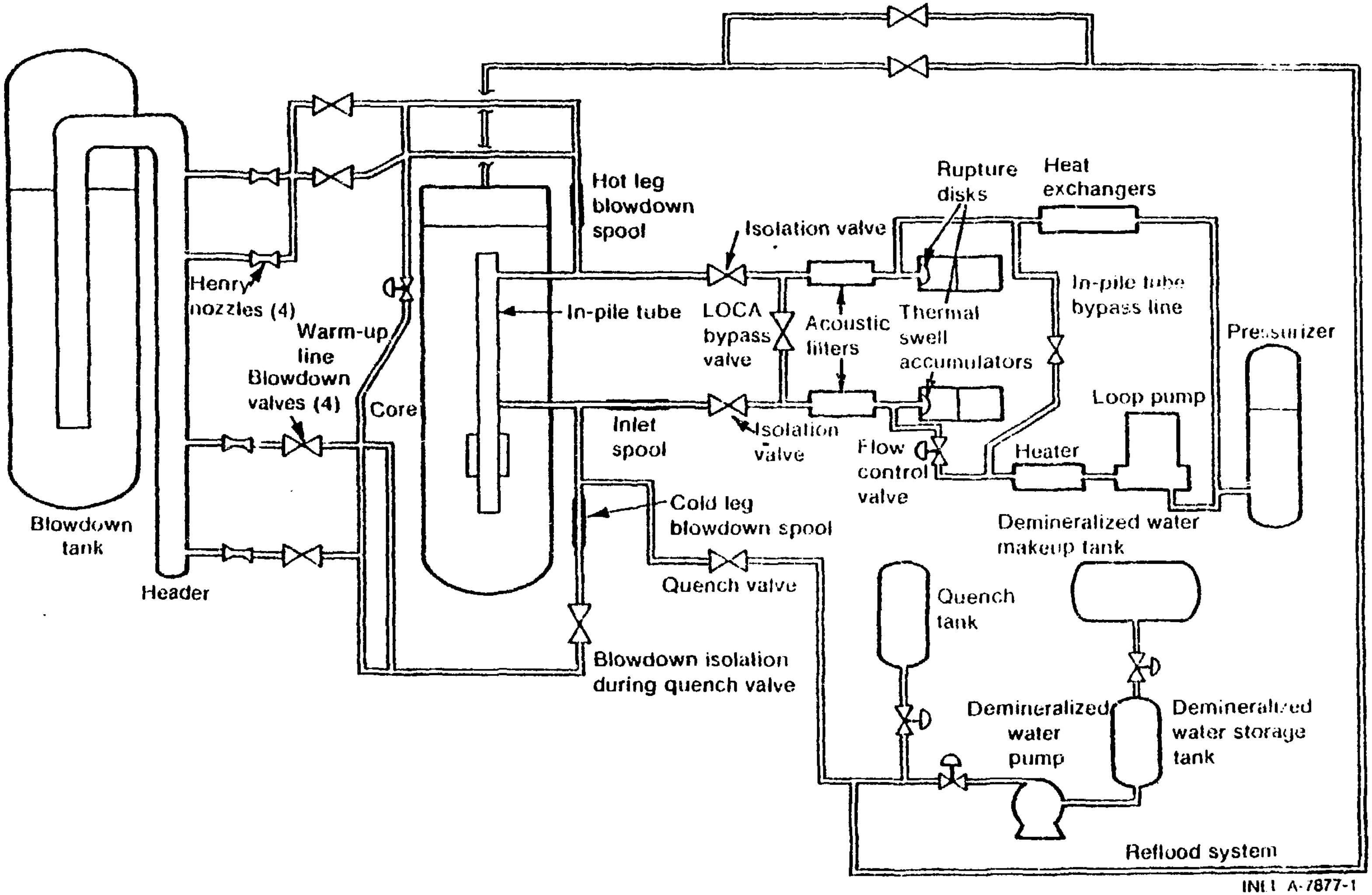
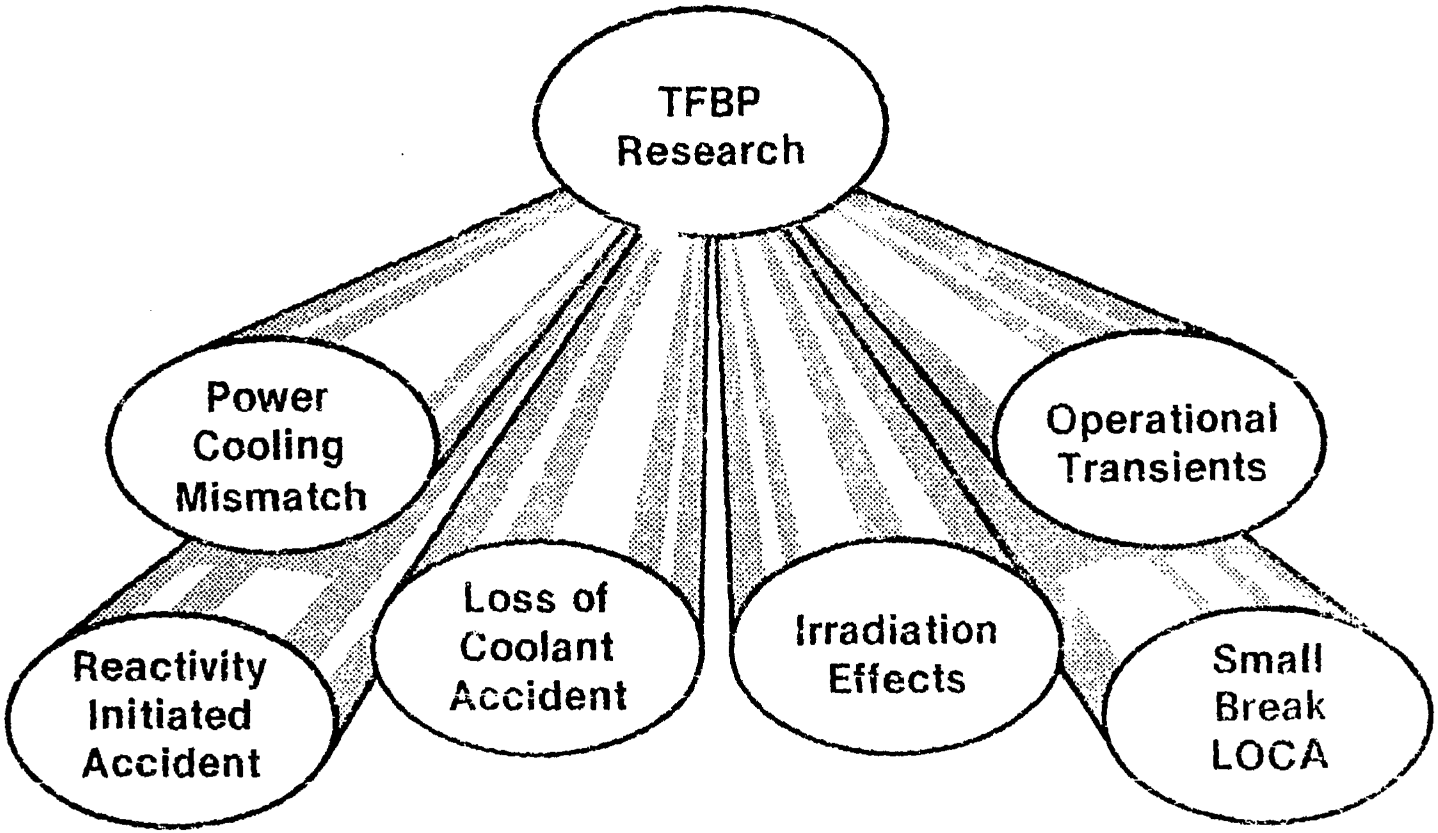


Blowdown tank
Header
Henry nozzles (4)
Warm-up line
Blowdown valves (4)
Core
Hot leg blowdown spool
In-pile tube
Isolation valve
LOCA bypass valve
Acoustic filters
Rupture disks
Thermal swell accumulators
Heat exchangers
In-pile tube bypass line
Pressurizer
Loop pump
Heater
Inlet spool
Isolation valve
Flow control valve
Cold leg blowdown spool
Demineralized water makeup tank
Quench valve
Quench tank
Blowdown isolation during quench valve
Demineralized water pump
Demineralized water storage tank
Reflood system
INEL A-7877-1


TFBP Research
Power Cooling Mismatch
Operational Transients
Reactivity Initiated Accident
Loss of Coolant Accident
Irradiation Effects
Small Break LOCA

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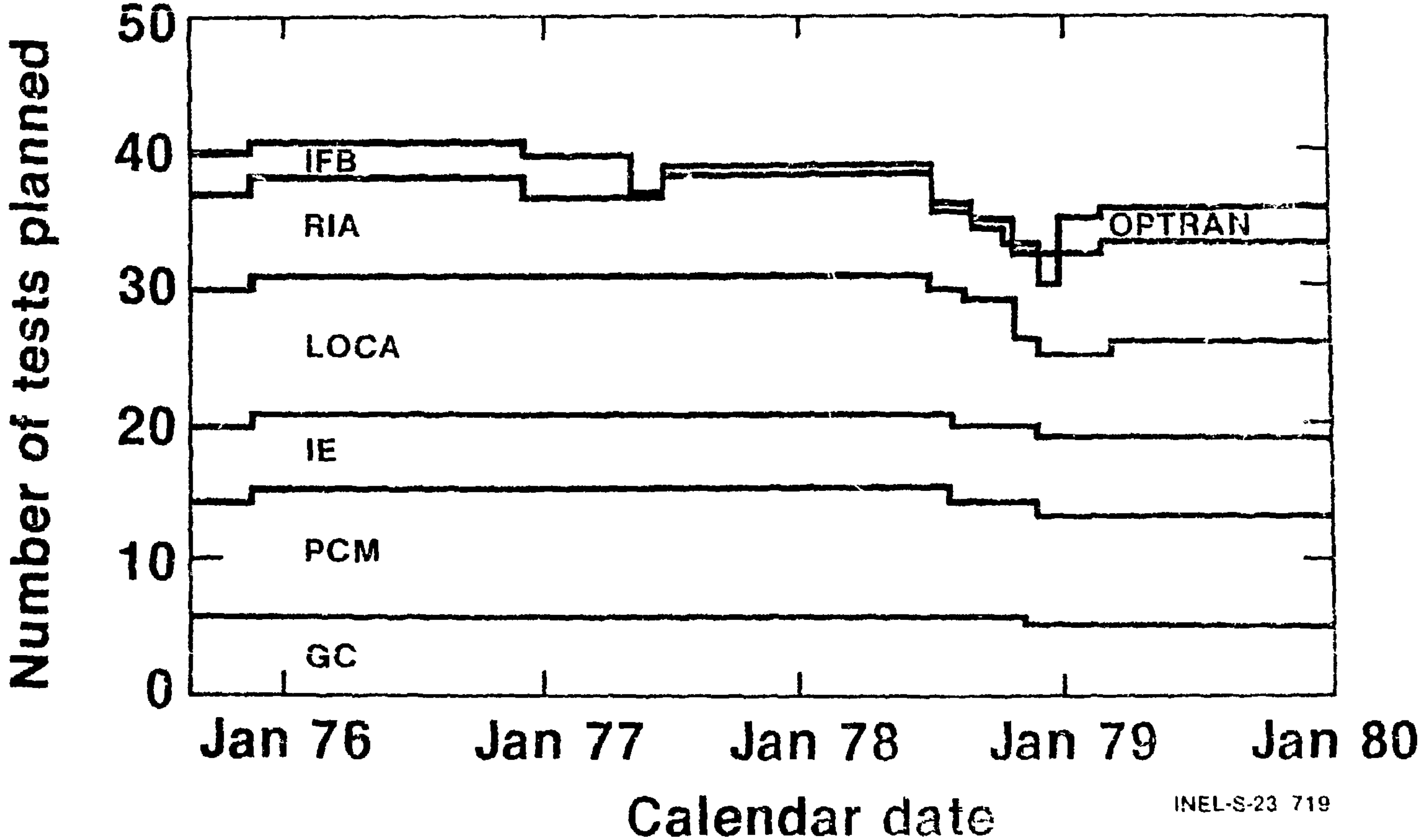
EG&G Thermal Fuels Behavior Test Program



Single Rod PCM Test Results

- Fuel rods operate in film boiling and incure damage without failure.
- Cladding deformation occurs $>$ 920 K.
- Zircaloy embrittled by oxygen due to Zr/steam and Zr/UO_2 reactions.
- At-power oxidation failure after complete reaction of cladding.

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Single Rod PCM Test Results (Continued)

- Posttest fracture due to embrittlement predictable by 95% O_2 saturation and 0.7 wt% criteria.
- Hydriding embrittlement only in failed rods.
- Clad failure due to MF unlikely unless MF superheated.

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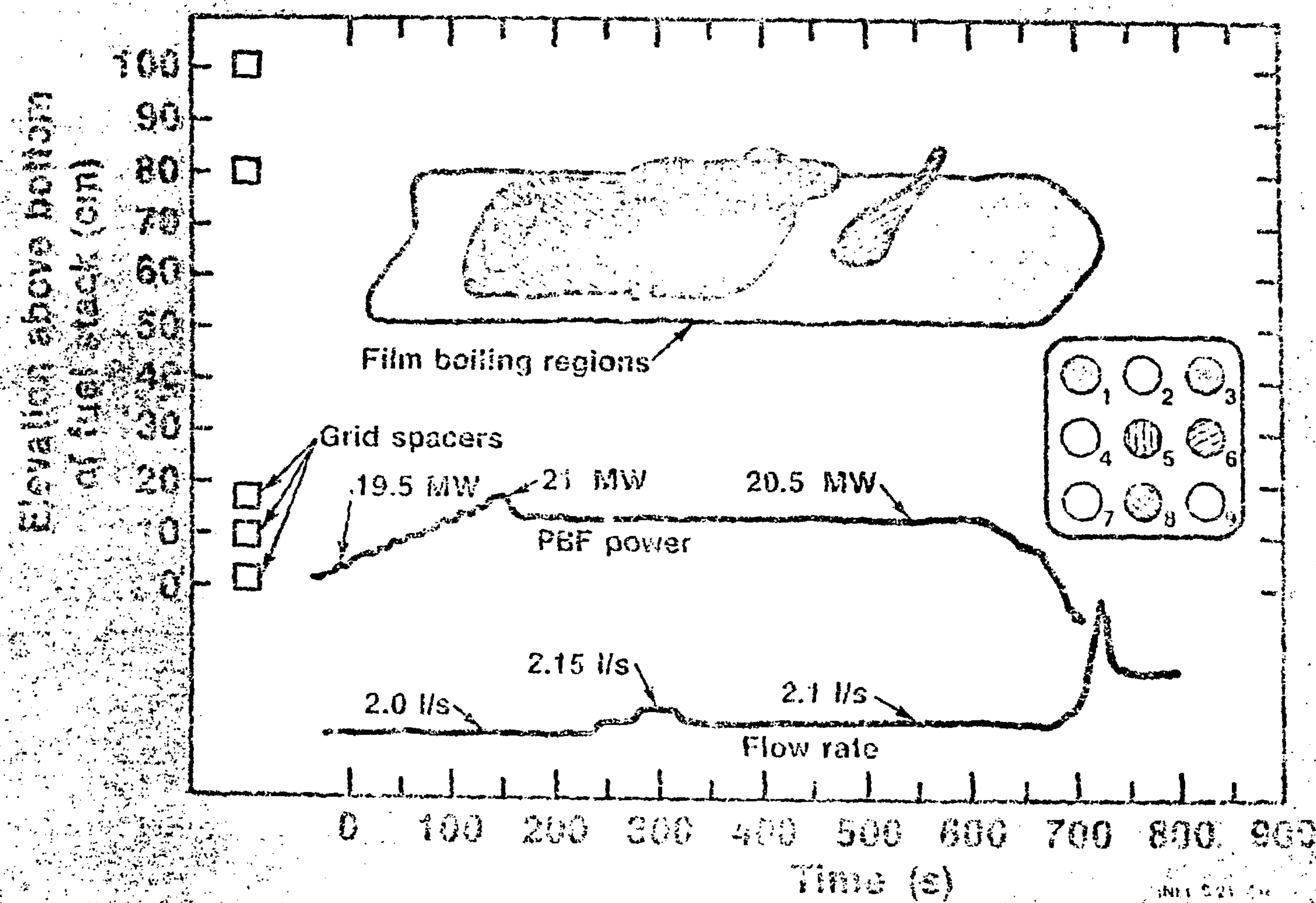
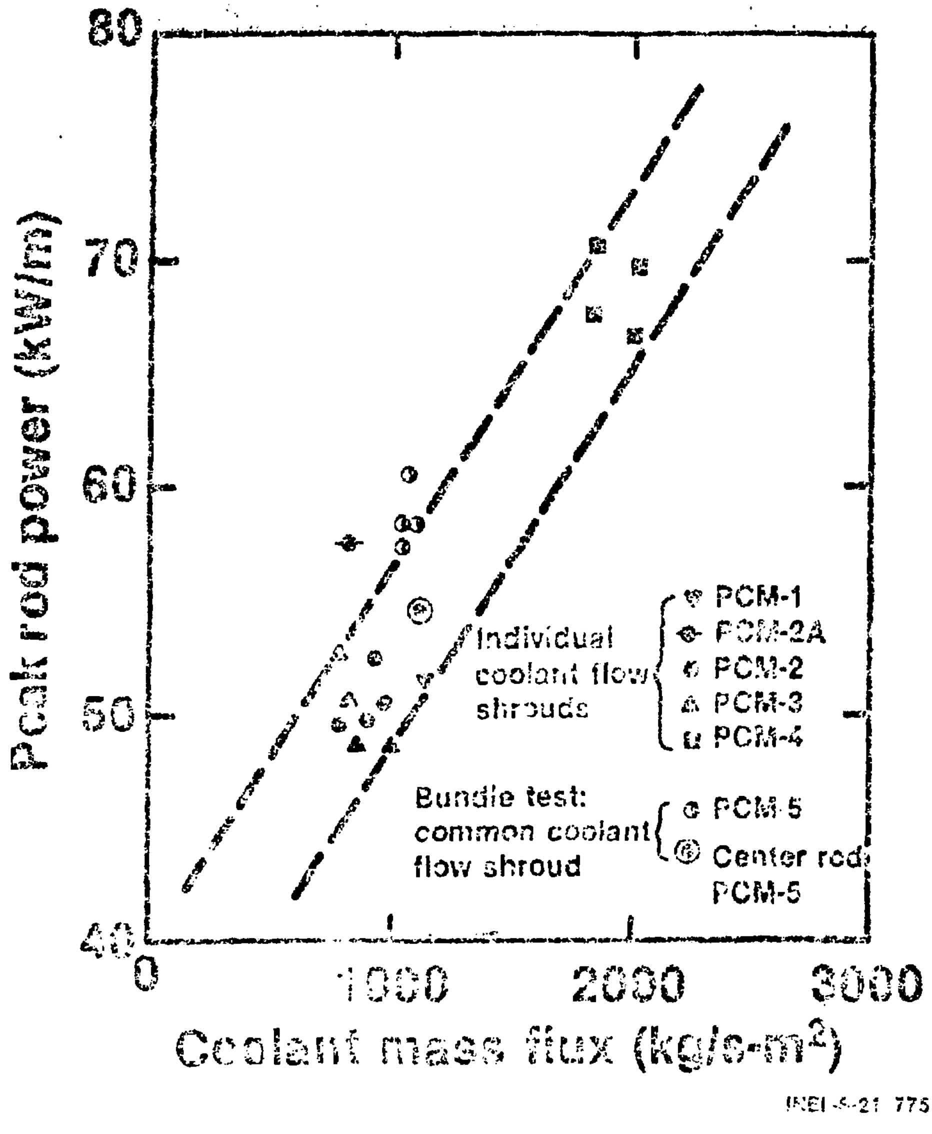
Single Rod PCM Test Results
(Continued)

- Energetic molten fuel/coolant interaction did not occur as a result of failure at high test rod peak powers (78 kW/m).
- Fission gas induces fuel swelling.
- Fuel powdering may occur when quenched from temperatures > 1900 K.

PCM-5 Film Boiling Scenario



Comparison of CHF Conditions Bundle Rod vs Single Rod



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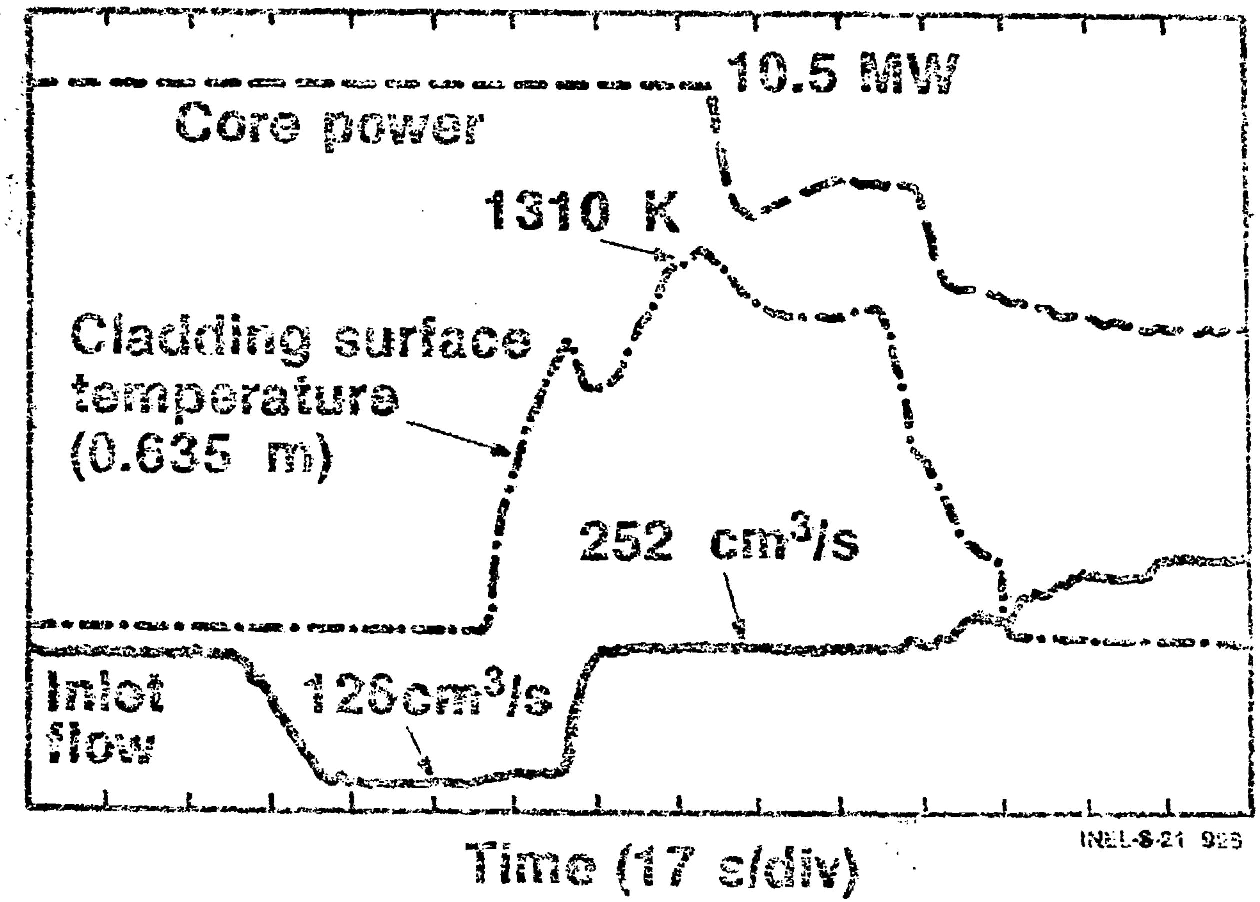
Thermal Fuels Behavior
Program Results of PCM-2 Test



Information Expected from Test PR-1

- 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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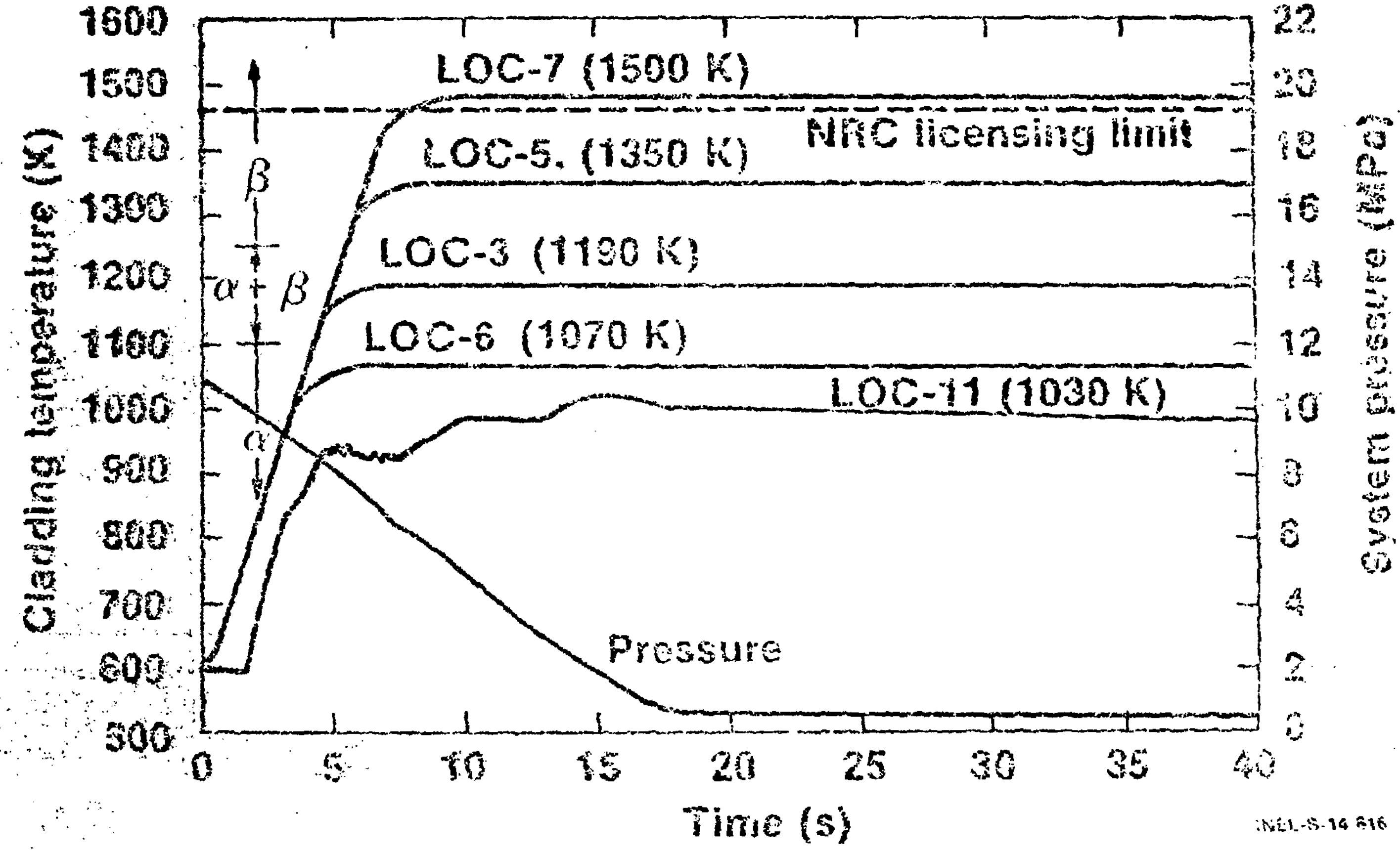
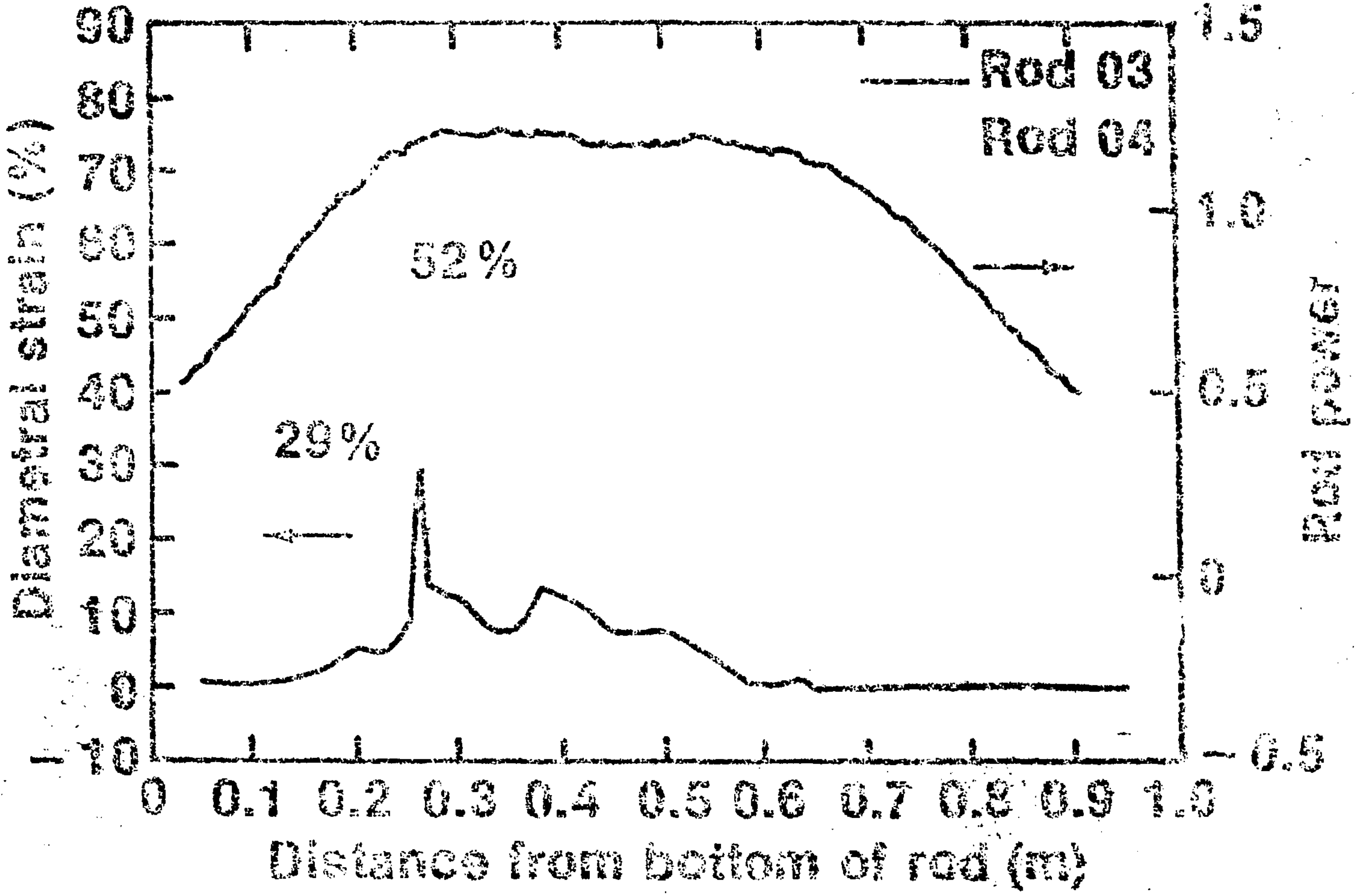
PBF LOCA Test Clad Temp Histories



LOC-[illegible] High Pressure
Rods 03 and 04



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PBF/LLR Test Objectives

- Experimentally evaluate the anticipated behavior of the LOFT core during the L2 Power Ascension Test Series
  - Extent of cladding collapse and waisting during the LOCA transients
  - Effects of PCI during preconditioning cycles
- Benchmark the fuel rod analysis package (FRAP) used to requalify the LOFT core after each test.

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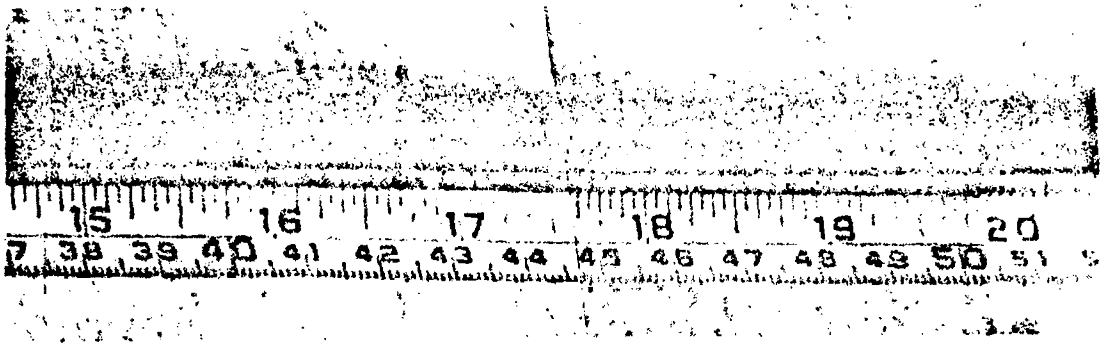
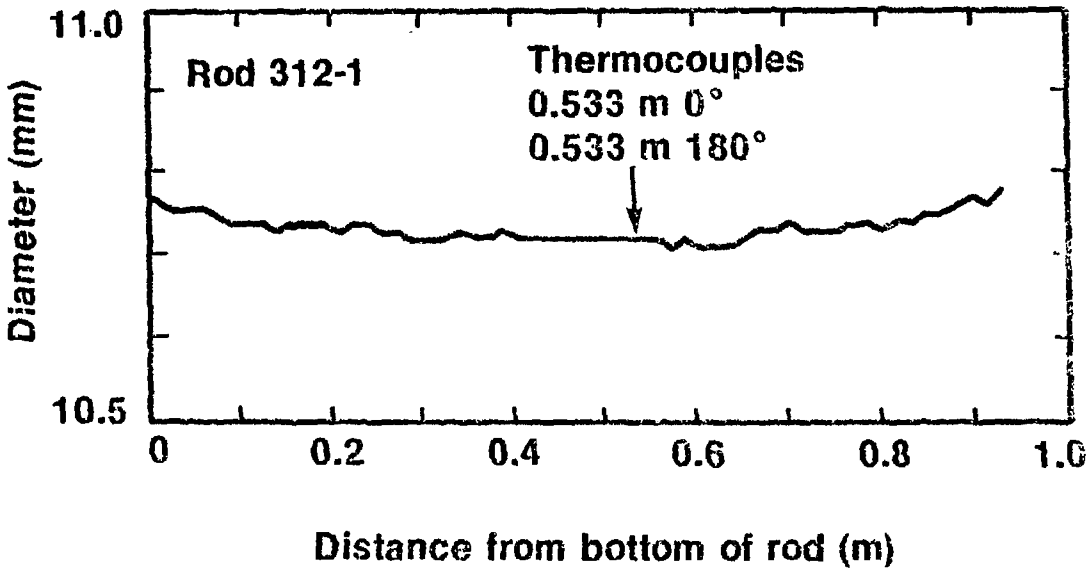
Post Irradiation Examination Results of Rod 312-1





TC-1 Test Objectives

- Do cladding surface thermocouples influence fuel rod thermal behavior during a LOCA?
- Do cladding thermocouples accurately measure cladding temperatures?

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Conclusions

- Surface thermocouples accurately measure local rod behavior
- All rods responded to the two-phase slug
- Surface thermocouples did influence the rod thermal behavior by:
  - Delaying CHF
  - Inducing early rewet during reflood

(18)

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

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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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