LIGHT WATER REACTOR FUEL RESPONSE DURING RIA EXPERIMENTS

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The rapid, inadvertent insertion of reactivity into a light water reactor (LWR) core, leading to high cladding temperatures has long been recognized as a potential mechanism for fuel rod failure. Reactivity initiated accidents (RIAs) are hypothesized to result from the mechanical separation of a control rod and control drive mechansim, followed by drop of the control rod from the core of a boiling water reactor or ejection of a control rod from a pressurized water reactor, with a resultant rapid increase in reactivity. The severity of the RIA depends on the energy deposited in the core, which increases with the rate of control rod removal and the worth of the control rod. On the basis of the analysis of previous RIA tests,[1,2] the U. S. Nuclear Regulatory Commission (NRC) has imposed a maximum fuel rod enthalpy limit of <280 cal/g UO_2 on commercial reactors to ensure minimal fuel rod damage and maintain the core in a coolable configuration in the event of an RIA. Complex analysis techniques are used to estimate the effects of postulated RIAs in LWRs. These techniques generally couple the transient neutronics behavior, fuel rod thermal and mechanical response, and the coolant hydrodynamic response. Assessment of these analytical models is incomplete due to limitations of existing fuel behavior data. Much of the applicable RIA experimental data were obtained several years ago in the Special Power Excursion Reactor Test (SPERT) and Transient Reactor Test Facility (TREAT) test programs, which investigated the behavior of single or small clusters of fuel rods under near room temperature and atmospheric (or near atomospheric) pressure conditions, no forced coolant flow, and zero initial power. Similar tests have been performed in the Japanese Nuclear Safety Research Reactor (NSRR).[2] Only a few irradiated fuel rods were tested in these programs.

An RIA behavior experimental program is now being performed by the Thermal Fuels Behavior Program of EG&G Idaho, Inc.,[3] for the NRC in the Power Burst Facility (PBF) reactor at the Idaho National Engineering Laboratory. The testing program is focused on the behavior of irradiated fuel rods tested under coolant conditions typical of hot-startup conditions in a commercial boiling water reactor (BWR).

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 285 and 185 cal/g UO_2, respectively. Results of the tests indicate that whereas the failure thresholds for unirradiated and irradiated fuel rods of 225 and 140 cal/g UO_2 (peak fuel enthalpy) are generally consistent with previous SPERT and NSRR results, the consequences of fuel rod failure at BWR hot-startup system conditions are more severe than observed in either SPERT or NSRR.

The mode of cladding failure for irradiated rods at a peak fuel enthalpy of 185 cal/g UO_2 appears to be pellet-cladding mechanical interaction (PCI). However, the irradiated rod that failed had the original fission product chemistry within the rod undisturbed, whereas three other rods subjected to the same energy insertion had been opened prior to testing and did not fail. The rod that failed had 22 longitudinal cracks starting at about 18 cm and extending to about 72 cm from the bottom of the 91-cm-long fuel stack. These results suggest that previously irradiated zircaloy cladding (which has experienced fast neutron damage) is unsuceptible to cracking due to PCI when the fission product inventory remains undisturbed.

The mode of cladding failure for unirradiated fuel rods tested at peak fuel enthalpies of 250 to 260 cal/g UO_2 was due to mechanical overstraining of the oxygen embrittled cladding during quench. Extensive cracking and crumbling of the embrittled cladding and fuel occurred at the peak power regions of the rods.

The mode of cladding failure for irradiated fuel rods tested at a peak fuel enthalpy of 285 cal/g UO_2 was by rupture caused from fuel-melting-induced and fuel-swelling-induced cladding strain during fuel heatup. The 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 those separately shrouded, irradiated fuel rods.

Metallographic examination of the RIA test fuel rods revealed extensive variation in wall thickness, involving considerable plastic flow. For example, a cross section of the test rod from RIA-ST-1 (245 cal/g UO_2 maximum fuel enthalpy) indicated wall thickening and thinning amounting to 170 and 60%, respectively, of the original wall thickness. The extensive cracking and crumbling of both irradiated and unirradiated fuel rods upon rewet was probably enhanced by the thinning of the cladding wall that occurred during the power burst.

REFERENCES

1. Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors, NRC Regulatory Guide 1.77 (May 1974).

2. Toshio Fujishiro et al, Light Water Reactor Fuel Response During Reactivity Initiated Accident Experiments, NUREG/CR-0269, TREE-1237 (August 1978).

3. P. E. MacDonald et al, "Light Water Reactor Fuel Response During Reactivity Initiated Accident Experiments," Proceedings of ANS Topical Meeting on Light Water Reactor Fuel Performance, Portland, Oregon, April 29-May 3, 1979.

Light Water Reactor Fuel Response During Reactivity Initiated Accident Experiments

Presented by
P.E. MacDonald

EG&G Idaho, Inc.

IDAHO NATIONAL ENGINEERING LABORATORY

Outline

- Fuel rod thermal response
- Overview of previous results
- Unirradiated test rods
- Irradiated test rods
- Conclusions

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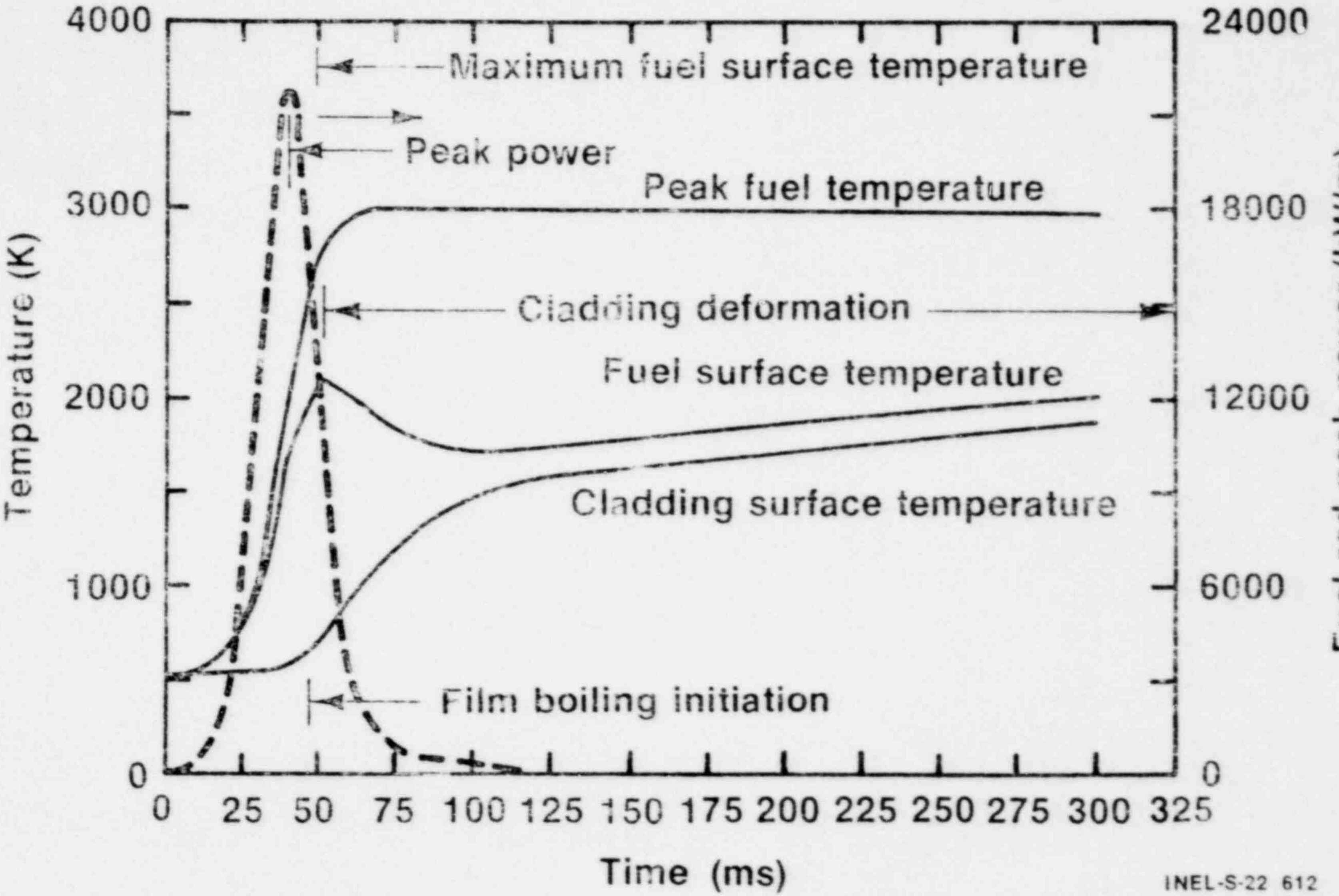
Fuel and Cladding Temperatures — Time Histories for RIA Peak Fuel Enthalpy of 225 cal/g UO_2



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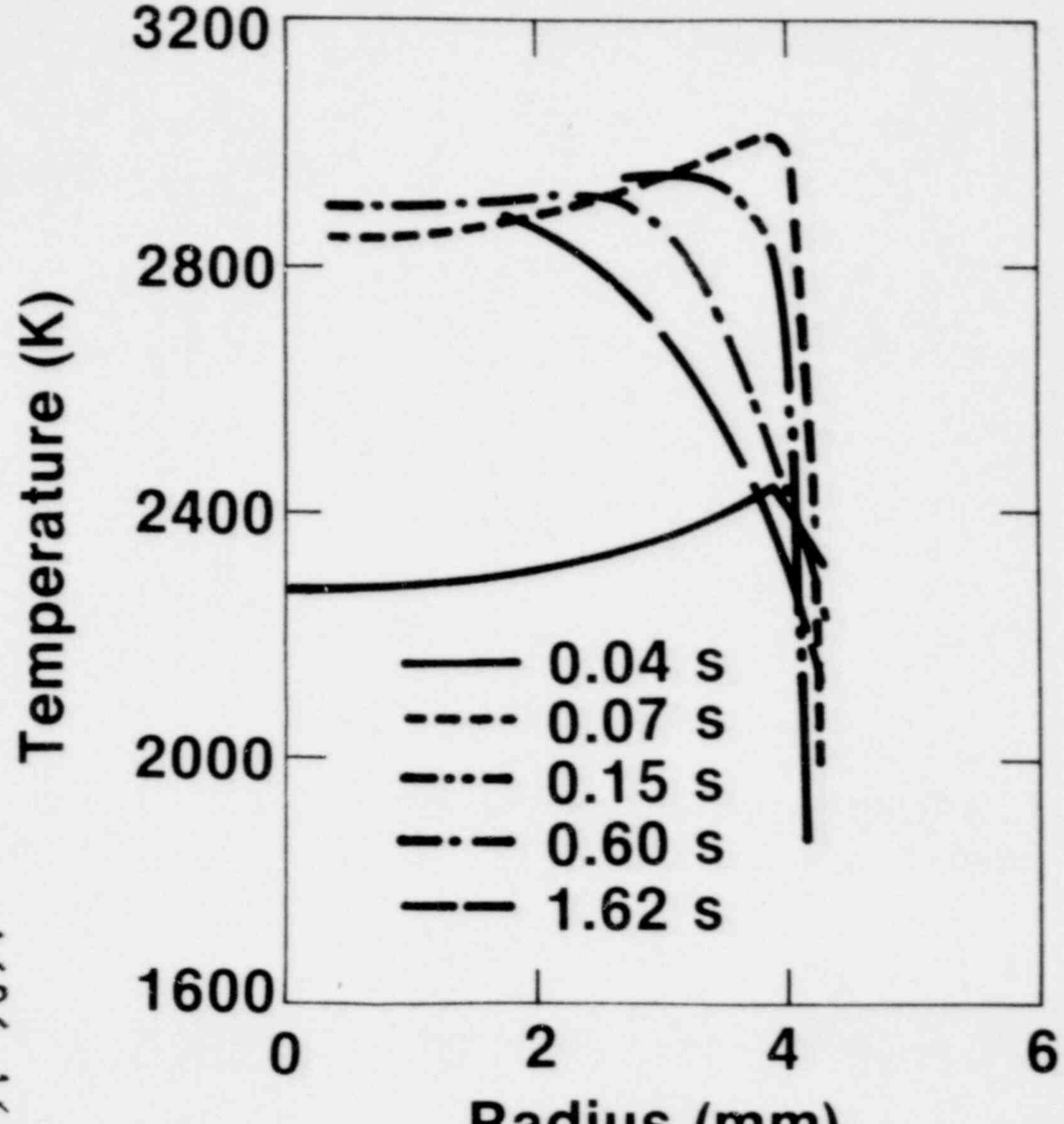
Fuel Radial Temperature Distributions During a 230 cal/g UO_2 RIA



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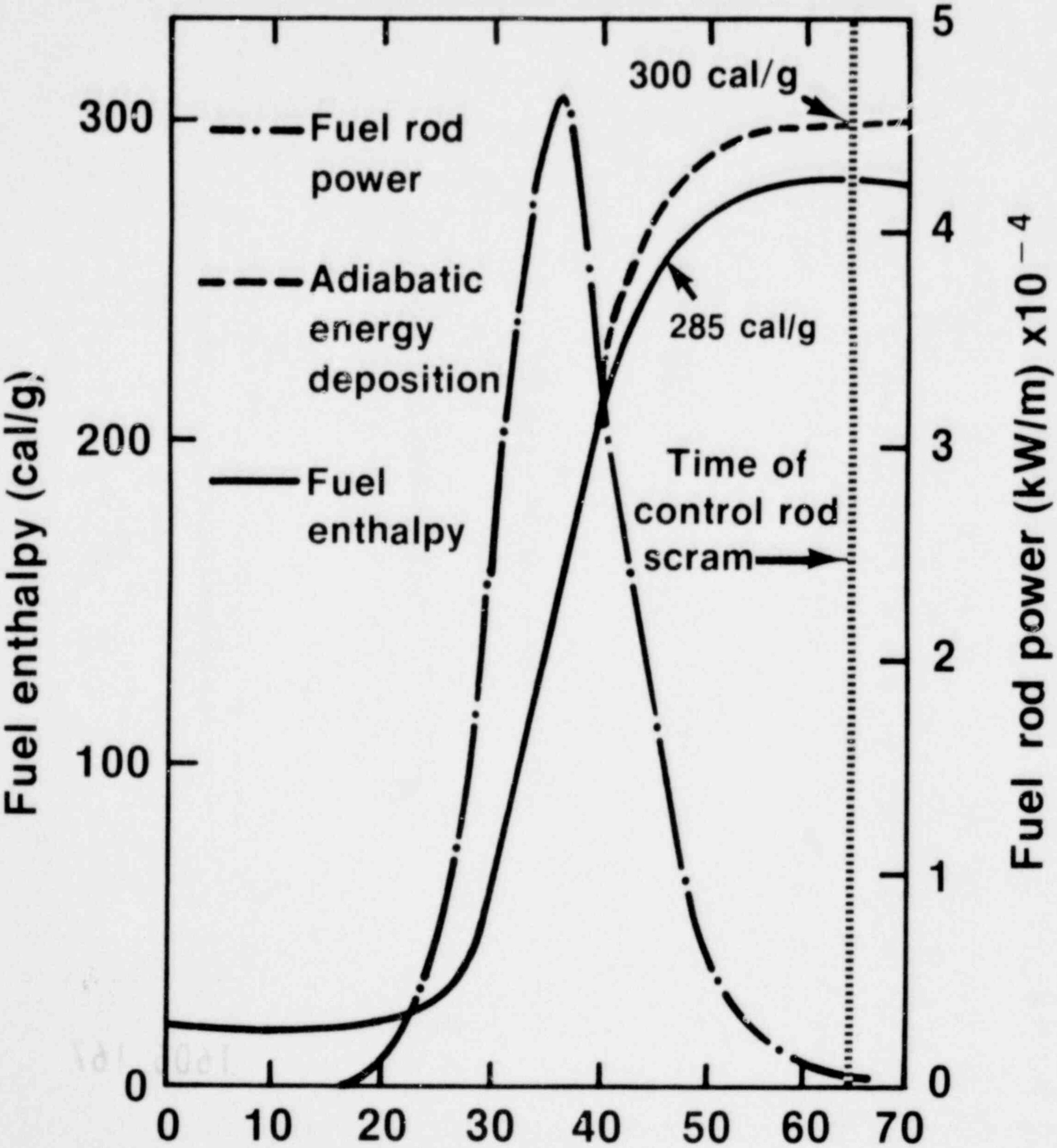
Test RIA 1-1
Fuel Enthalpy vs Time



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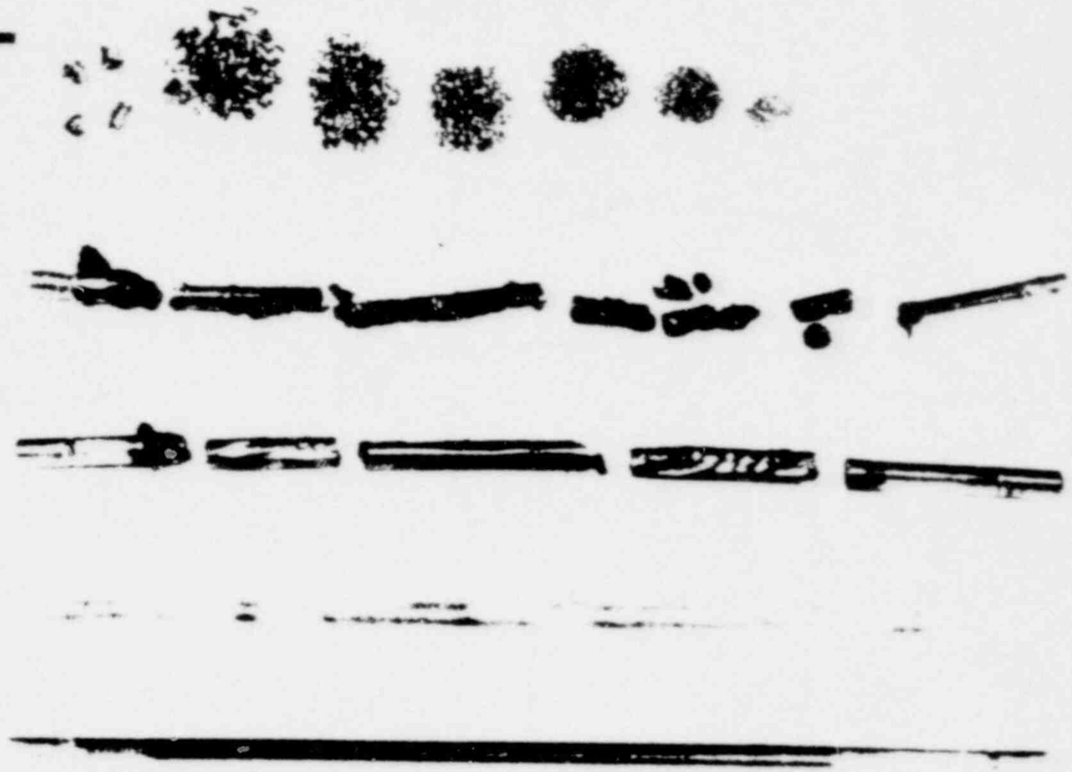
Posttest Photographs of SPXM Fuel Rods



Total Energy Deposition (cal/g)	Peak Fuel Enthalpy (cal/g)
378	~305
338	~275
287	~240
240	~205
168	~145

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SPERT Unirradiated Test Rod Results

	Total Energy Deposition (cal/g)	Peak Fuel Enthalpy (cal/g)
Failure threshold	240-265	205-225
Loss-of-coolant geometry	~300	~245
Prompt fuel dispersion	~370	~300

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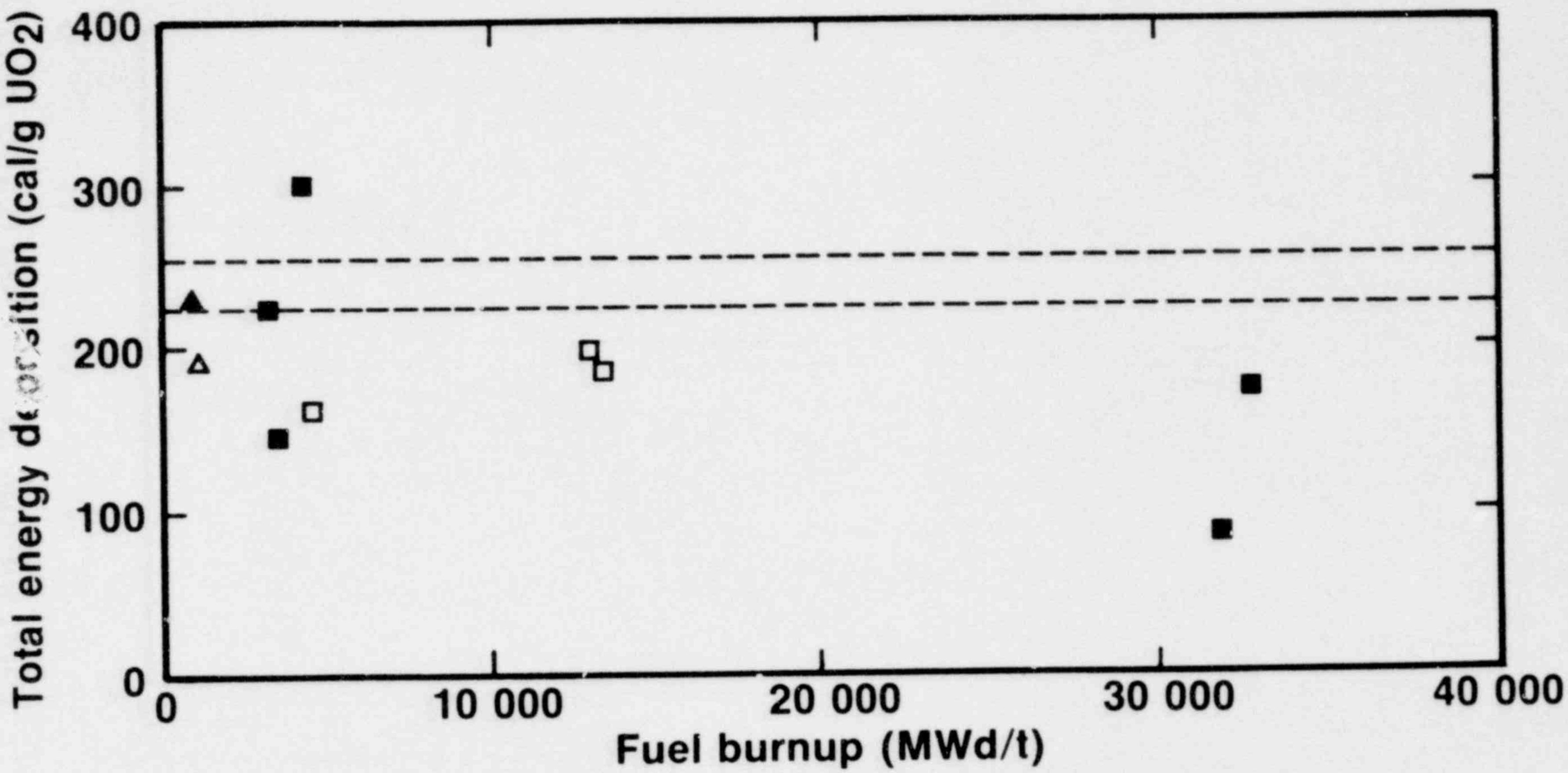
Failure Energy for Irradiated Rods



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RIA Scoping Test Objectives

- Evaluate calorimetry techniques for determining test rod energy deposition.
- Define the peak fuel enthalpy failure threshold for unirradiated test rods operated at BWR hot-startup conditions.

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Initial Conditions

Coolant temperature	538 K
Coolant pressure	6.45 MPa
Shroud flow	85 cm^3/s
Rod power	0

INEL-S-14 131

RIA ST Energy Data

Test	Total Energy Deposition (cal/g)	Peak Fuel Enthalpy (cal/g)	Failure
RIA ST-1			
PB-1	255	185	No
PB-2	335	250	Yes
RIA ST-2	350	260	Yes
RIA ST-3	300	225	No
RIA ST-4	~700		Yes

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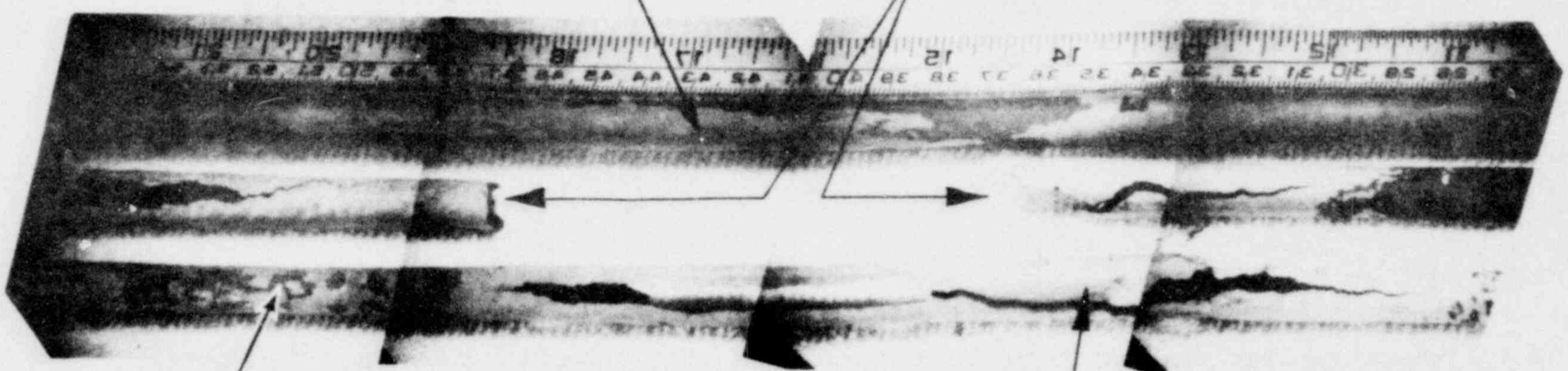
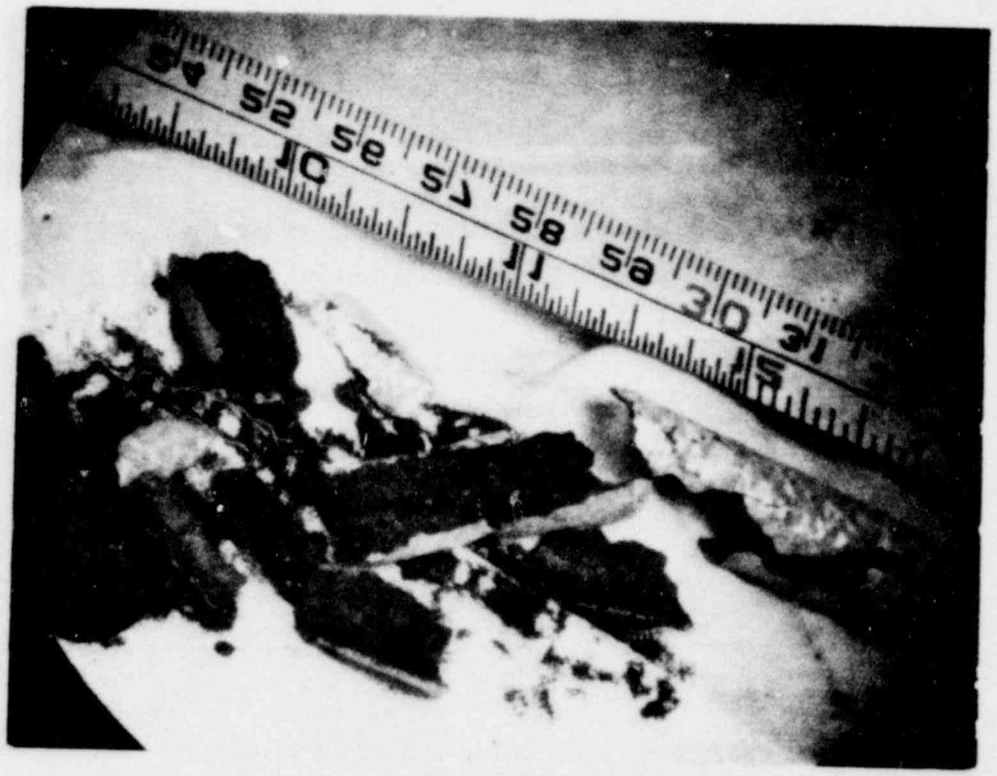
RIA Scoping Test 1, 2, and 3



INEL-S-22 348

Peak Flux Region of RIA ST-2
(260 cal/g)



INEL-S-14 097

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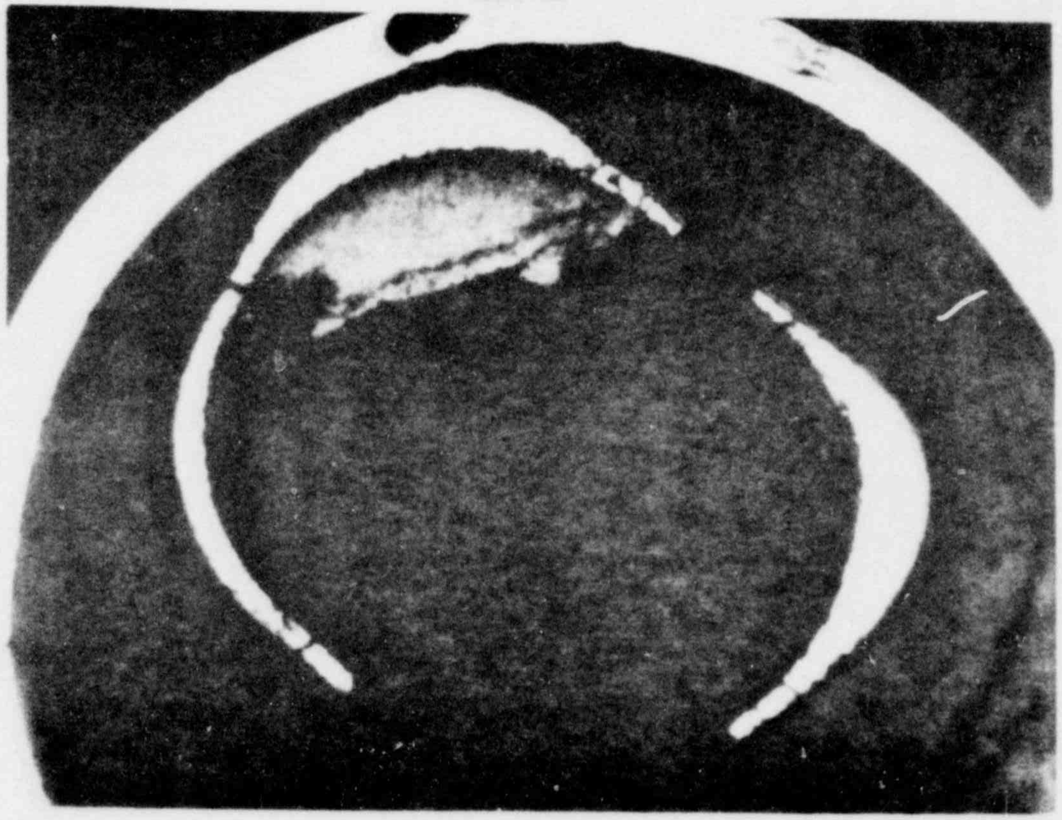
RIA ST-1 Cladding at 0.35-m Elevation



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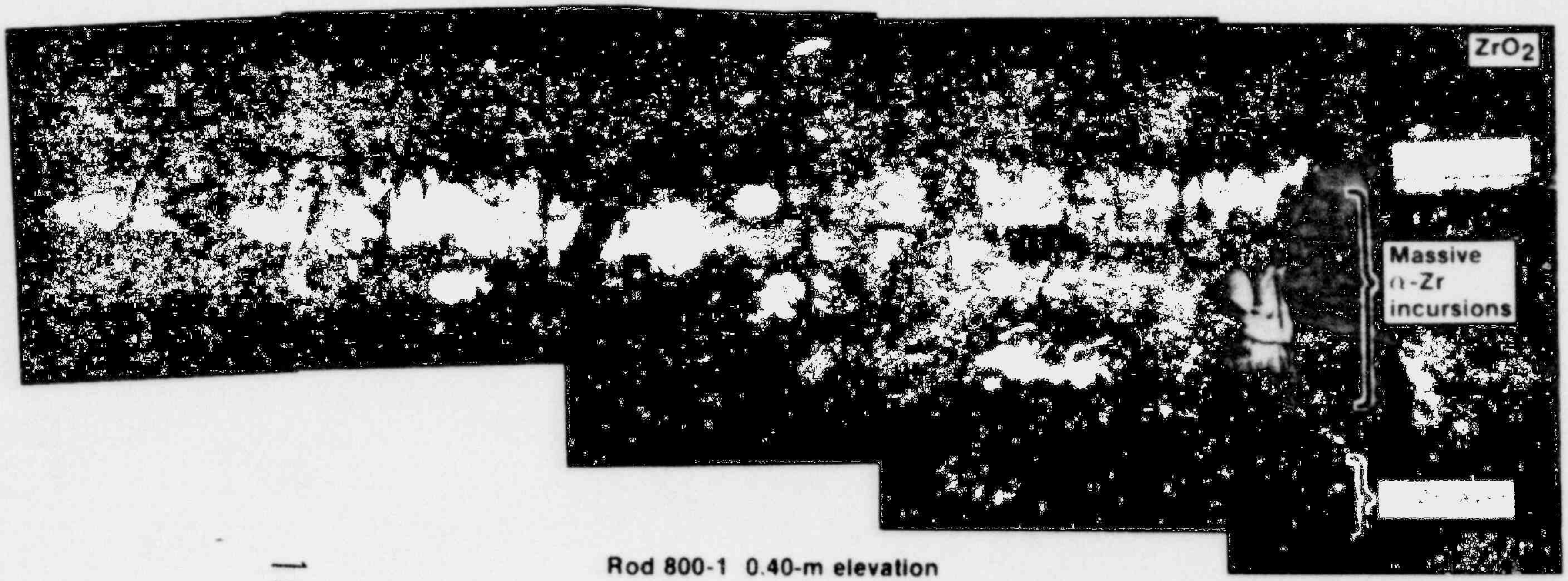
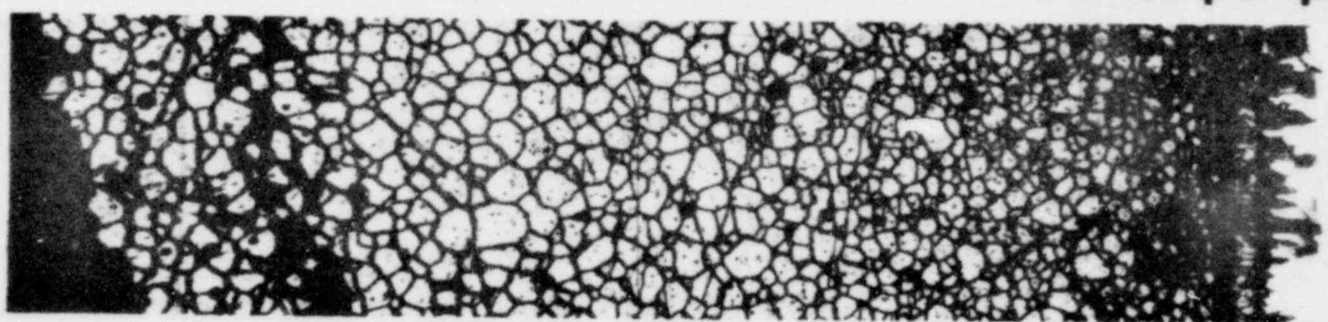
RIA ST-1 (250 cal/g) Cladding Microstructures



Rod 800-1 0.40-m elevation

RIA ST-1 (250 cal/g)
Fuel Shattering at 0.35-m Elevation, 240° Orientation

POOR ORIGINAL



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Tests RIA 1-1 and RIA 1-2 Objectives

- Characterize the response of previously irradiated fuel rods during a RIA event at BWR hot-startup conditions
- Evaluate the effect of internal rod pressure on preirradiated fuel rod response
- Provide data on failure threshold enthalpy for previously irradiated rods

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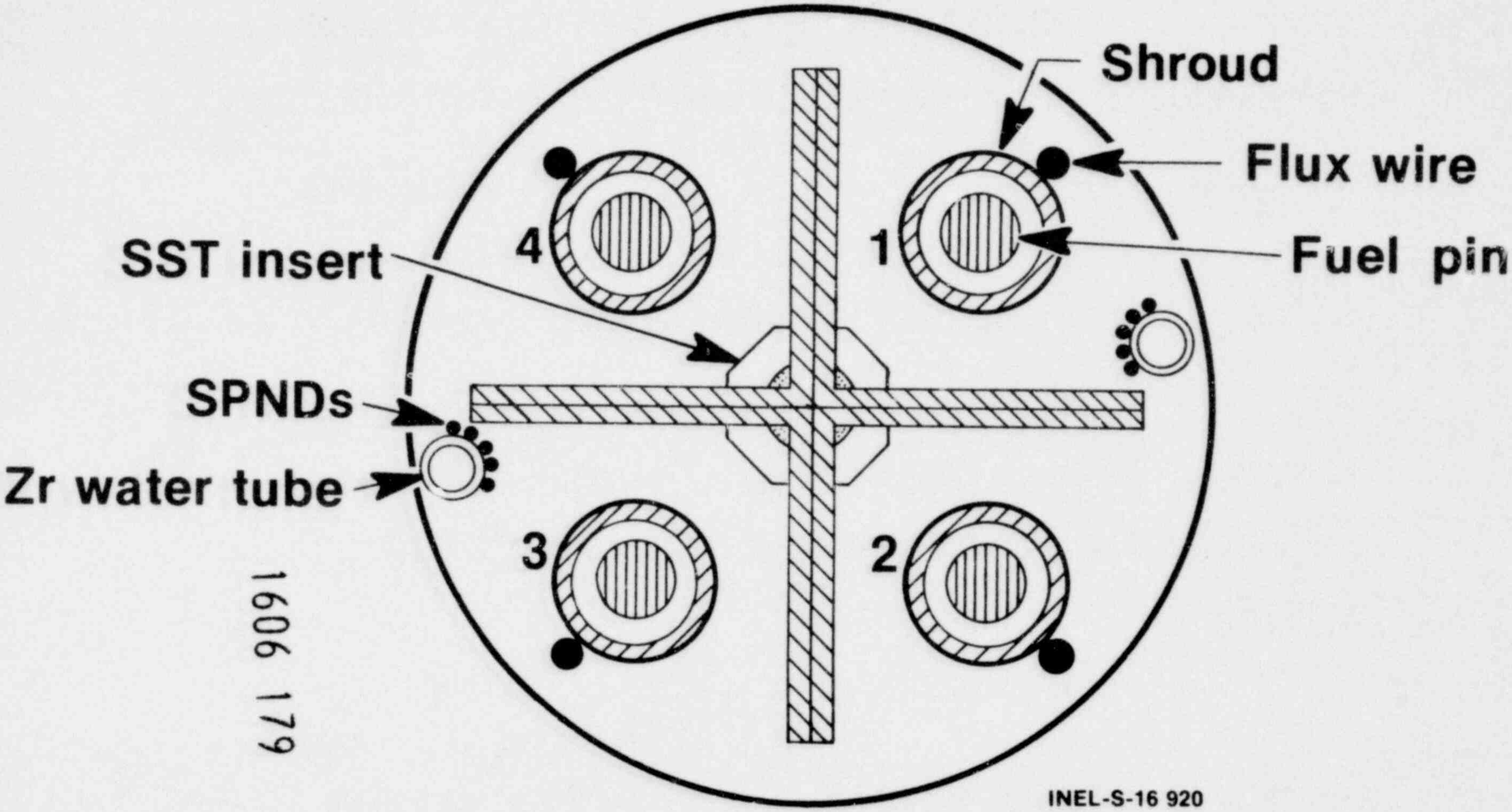
Four Rod Test Configuration Schematic



INEL-S-16 920

RIA Energy Data

Test	Total Energy Deposition (cal/g)	Peak Fuel Enthalpy (cal/g)	Failure
RIA ST-1			
PB-1	255	185	No
PB-2	335	250	Yes
RIA ST-2	350	260	Yes
RIA ST-3	300	225	No
RIA 1-1	365	285	Yes
RIA 1-2	245	185	3 No/ 1 Yes

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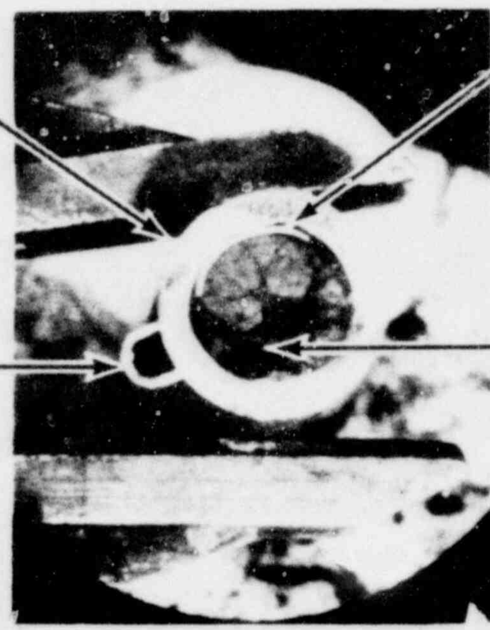
Test RIA 1-1 Rod 801-1
(285 cal/g)



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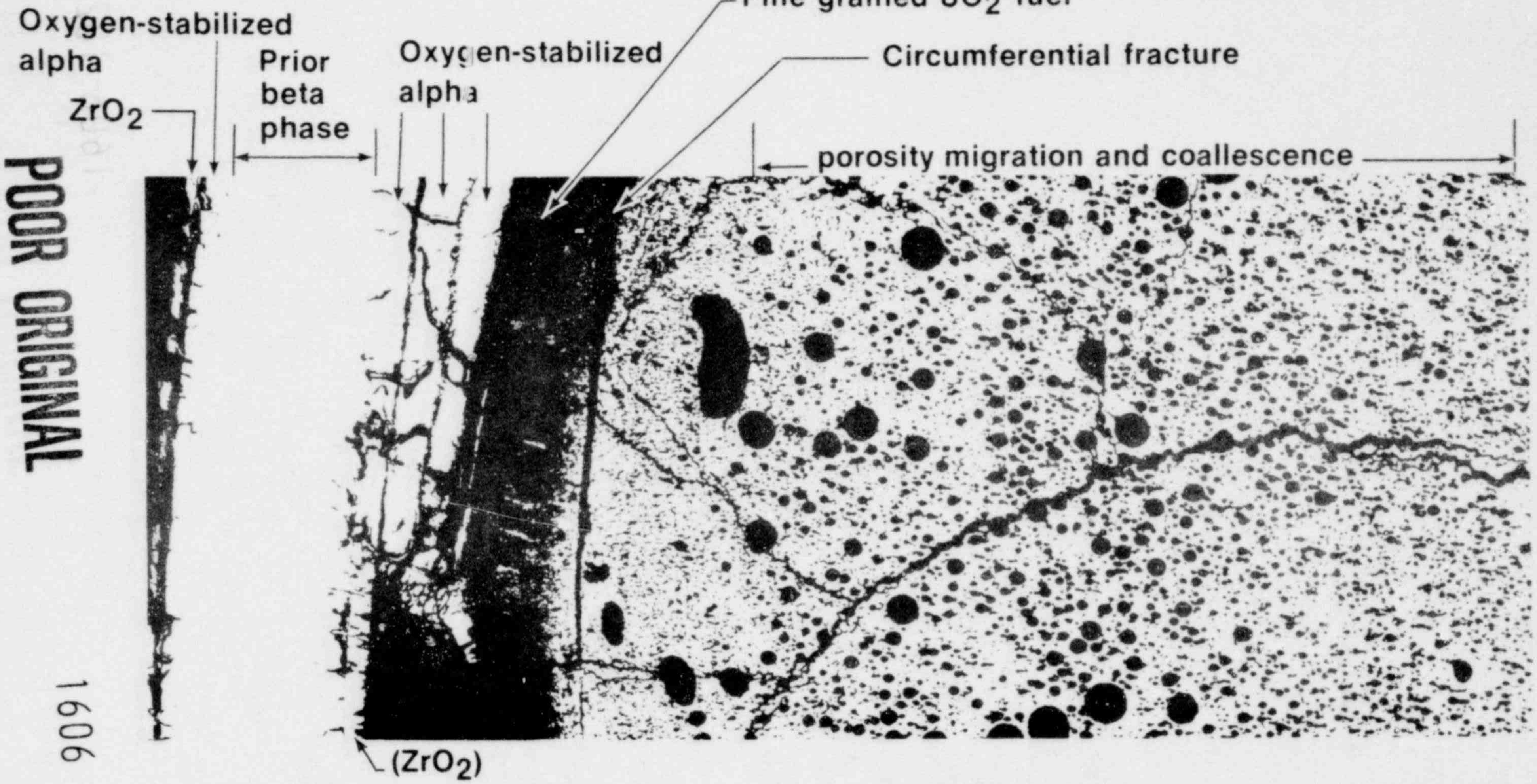
Test RIA 1-1 Rod 801-1

INEL-S-18 207

Test RIA 1-1 Rod 801-1



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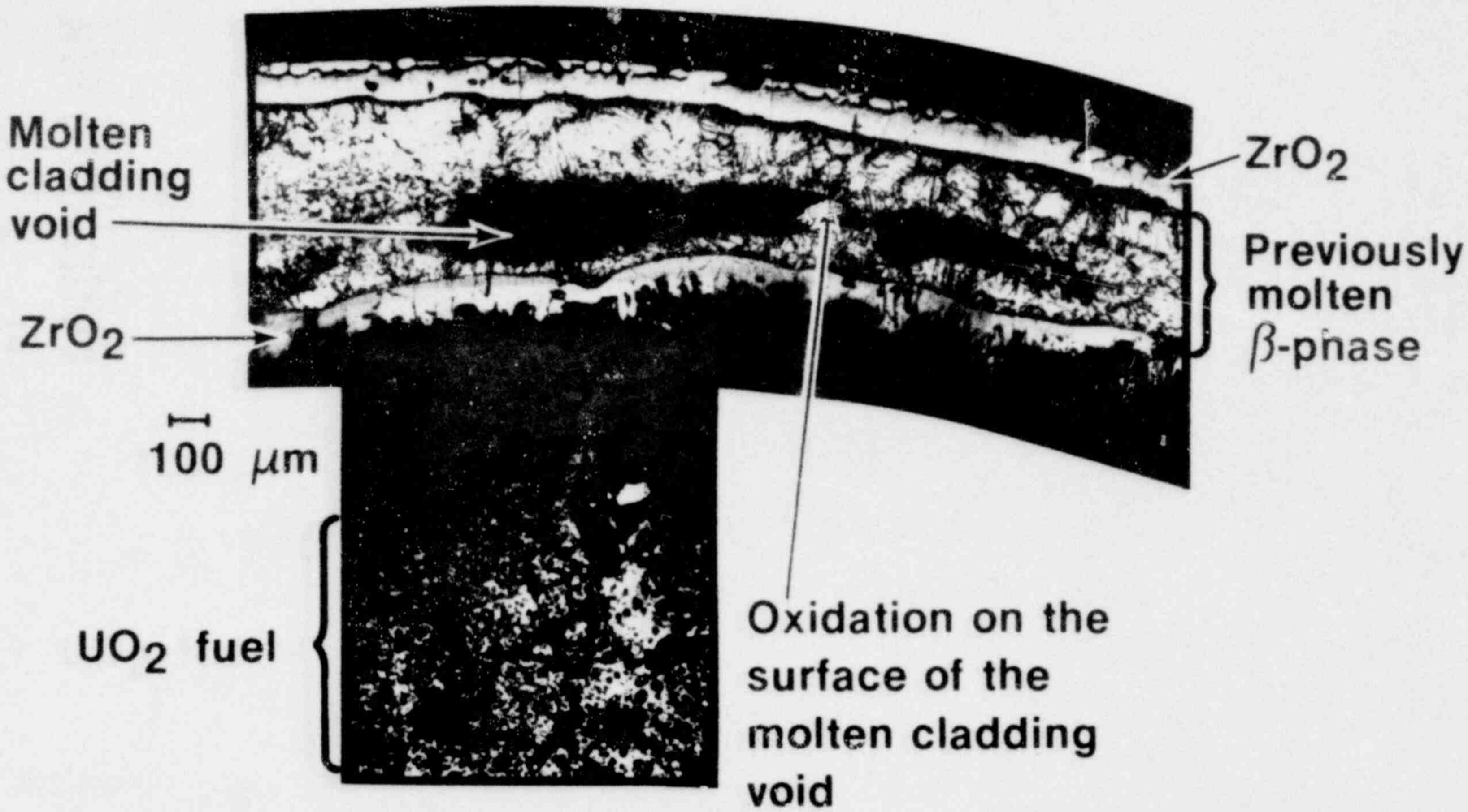
Molten Fuel and Cladding from Test RIA 1-1, Rod 801-2



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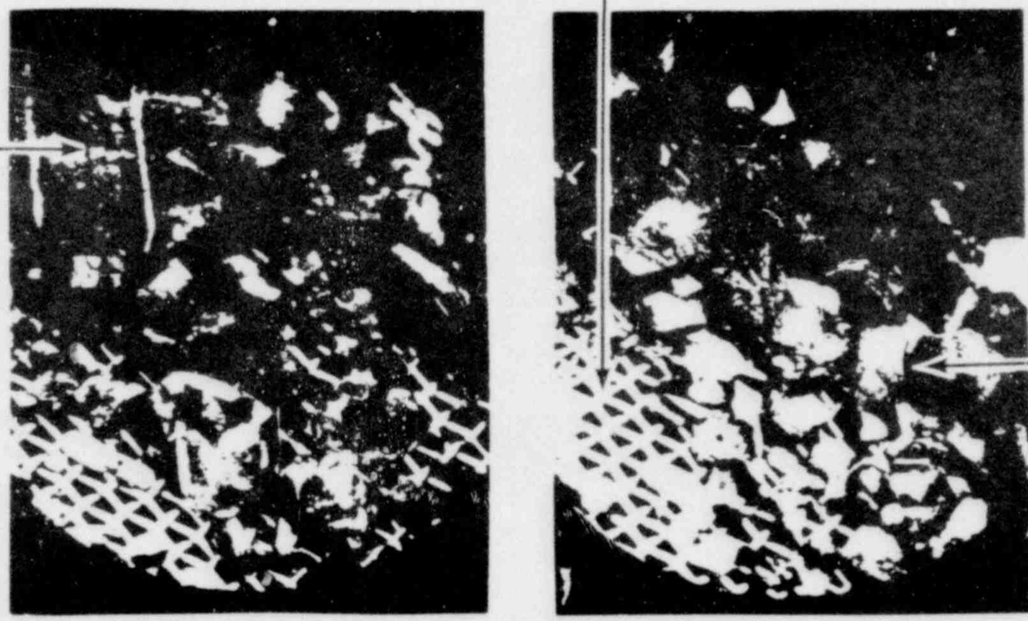
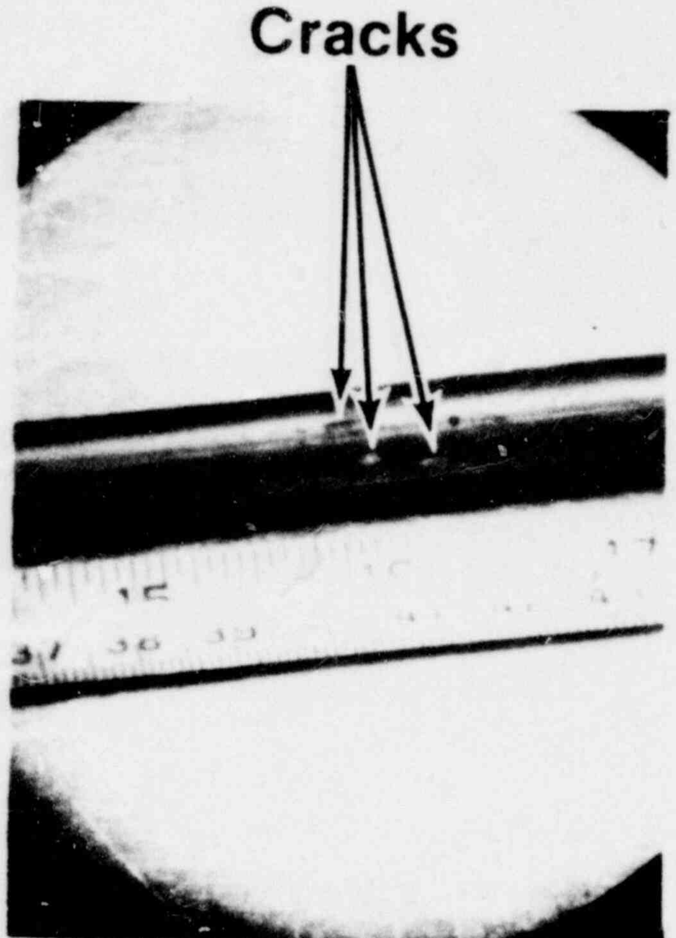
Test RIA 1-1 Rod 801-1



INEL-S-16 922

Fuel and cladding fragments collected from the shroud flow blockage
(size >> 2 mm)

Test RIA 1-2, Rod 2-3
Cracks in Cladding (185 cal/g)



90°

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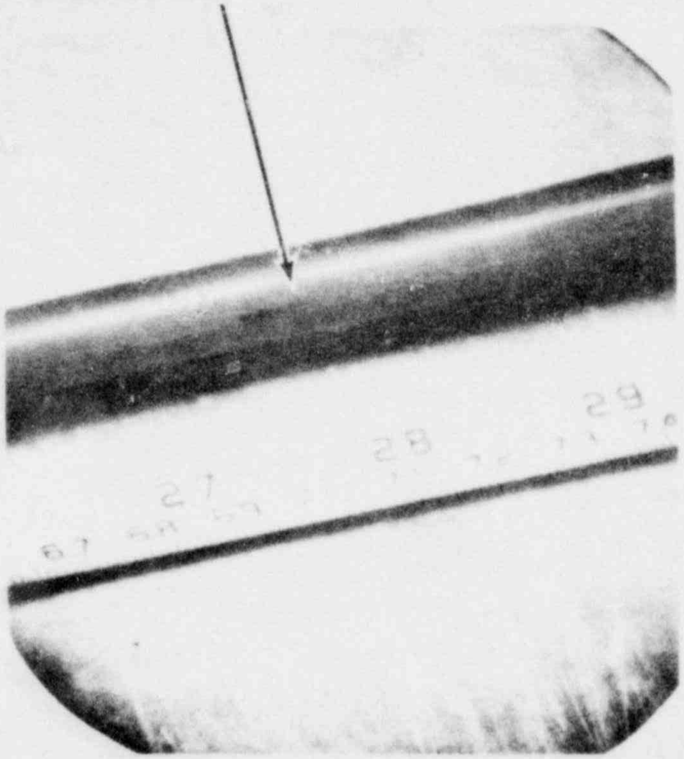
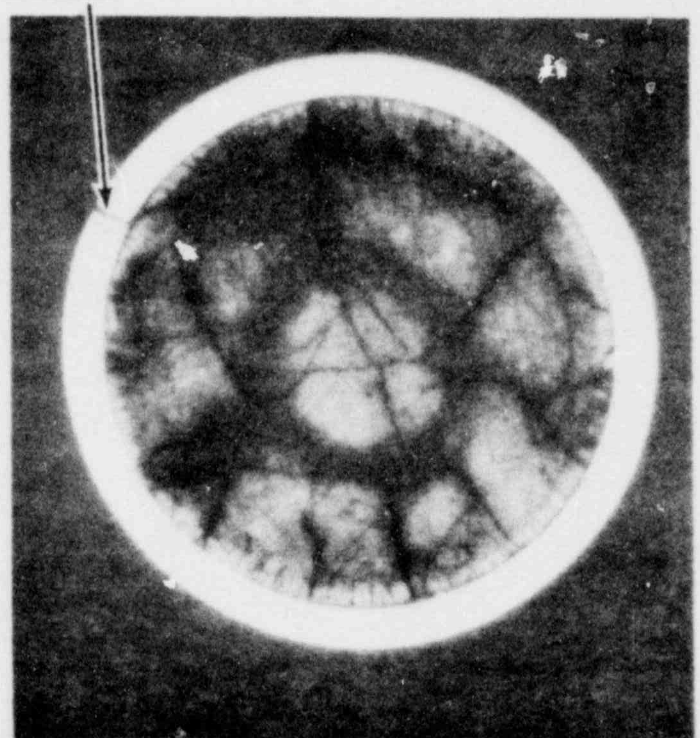
RIA 1-2 Rod 802-3
0.688-m Elevation

Transverse mount location



Crack through clad wall



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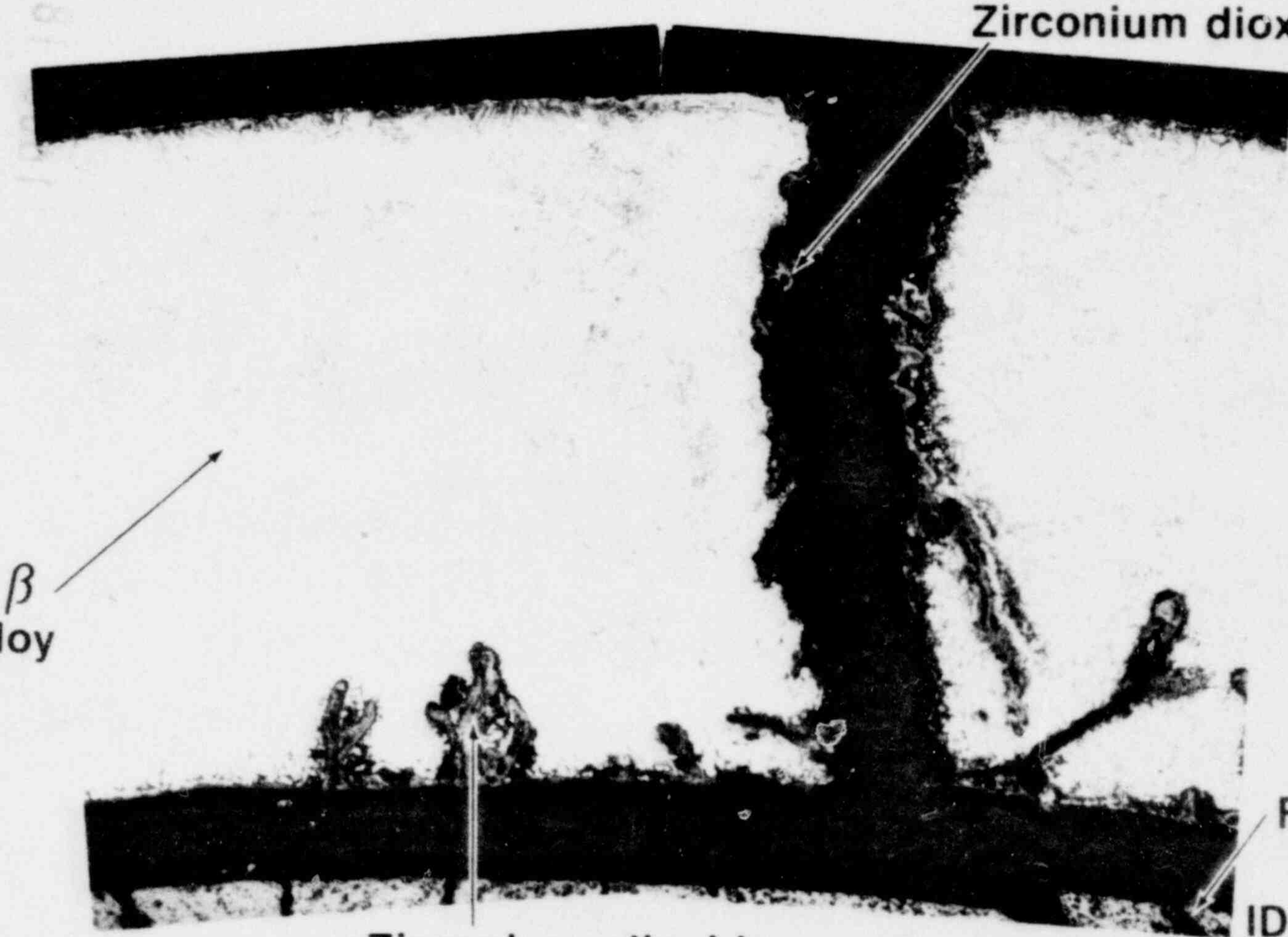
RIA 1-2 Rod 802-3 0.688-m Elevation



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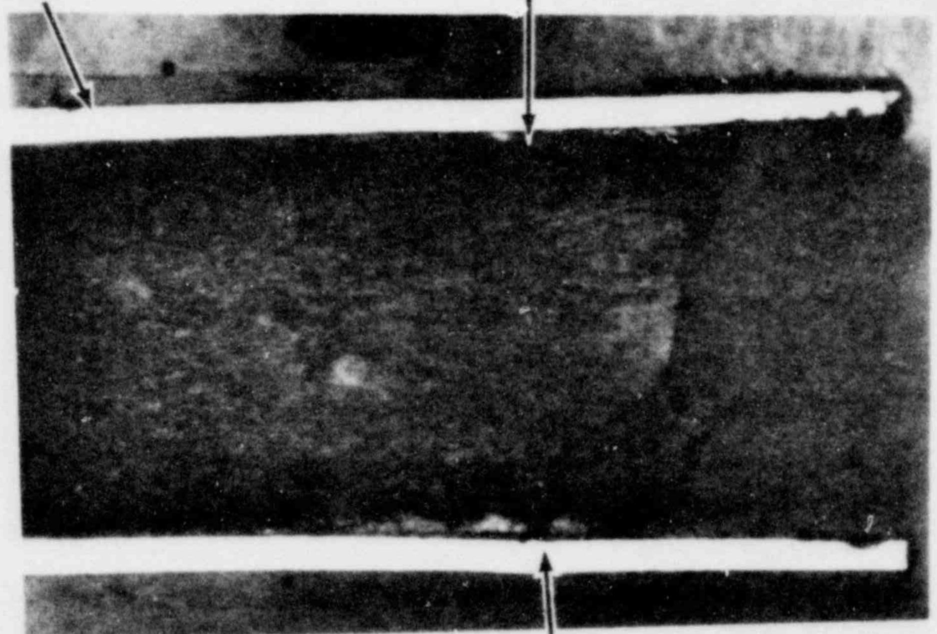
1606 188

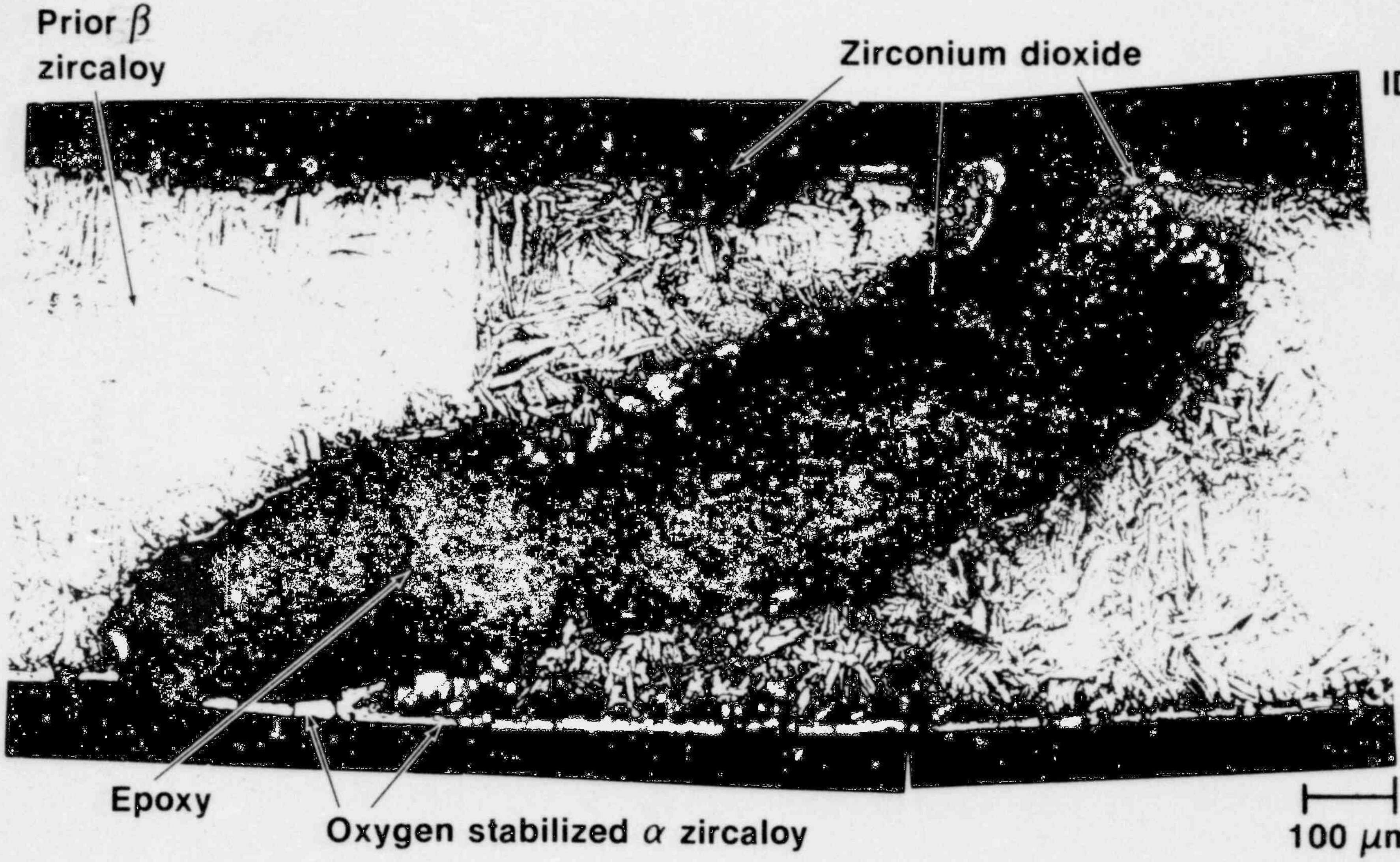
RIA 1-2 Rod 802-3
0.49 to 0.51-m Elevation



INEL-S-22 289

1606 189

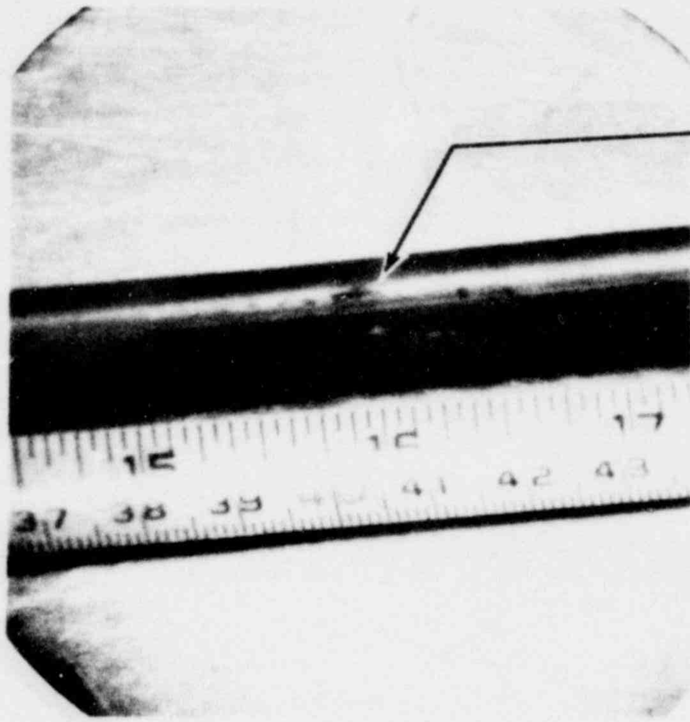
RIA 1-2 Rod 802-3
0.49 to 0.51-m Elevation



INEL-S-22 281

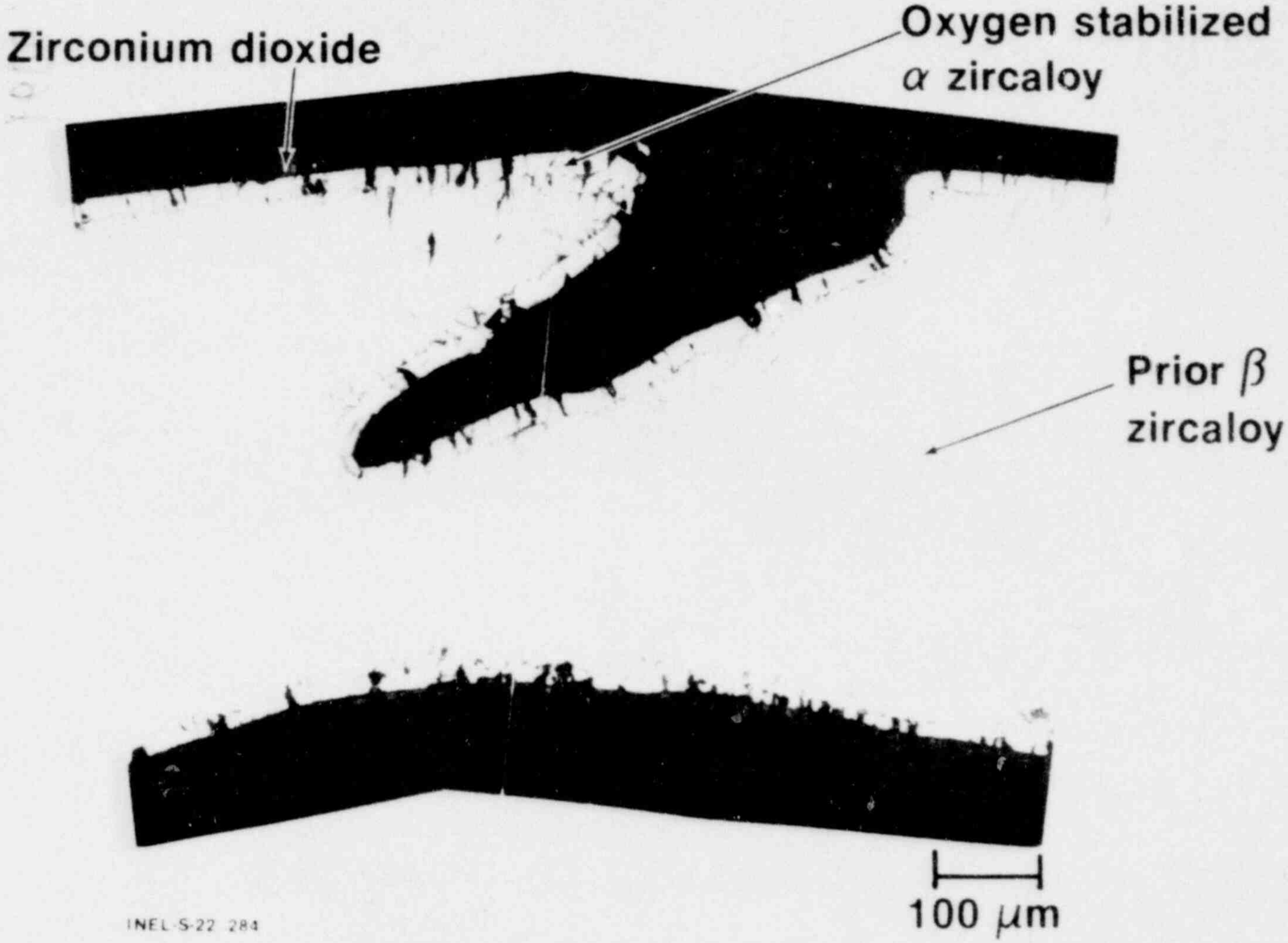
1606 190

RIA 1-2 Rod 802-3
0.38-m Elevation



INEL-S-22 290

RIA 1-2 Rod 802-3
0.3800-m Elevation



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RIA 1-2 Rod 802-3
0.3795-m Elevation

INEL-S-22 283

Consequences of Fresh Rod Failure at 260 cal/g

- Large variations in cladding thickness.
- Oxidation of the cladding on both the inside and outside surfaces.
- Fracture of the cladding at "thin" locations.
- Crumbling of the rod.
- Fuel shattering along grain boundaries.

1606 194

INEL-S-18 597

Conclusions from Unirradiated Rod Tests

- The failure threshold of fuel rods tested under BWR hot startup conditions is slightly higher than observed in SPERT (205-225 cal/g)

- Failure at 260 cal/g and BWR hot startup conditions is as severe as previously observed in SPERT (loss of coolable geometry)

INEL-S-22 293

Consequences of Irradiated Rod Failure at 285 cal/g

- Large variations in cladding thickness
- Fuel swelling and foaming; cladding failure
- Oxidation of the cladding on both the inside and outside surfaces
- Fracture of the cladding at "thin" locations fuel shattering upon quench
- Coolant flow blockage

INEL-S-22 292

Conclusions from Preirradiated Rod Tests

- The failure threshold was less than for unirradiated fuel and was about 140 cal/g.

- Failure at 285 cal/g and BWR hot startup conditions is more severe than previously observed in SPERT and NSRR.

- Upon failure at 285 cal/g the rods sweiled and blocked the flow channels.

INEL-S-22 291