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DEMONSTRATION OF FUEL RESISTANT TO PELLET-CLADDING INTERACTION PHASE 2 — FIRST SEMIANNUAL REPORT JANUARY — JUNE 1979

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DEMONSTRATION OF FUEL RESISTANT TO PELLET-CLADDING INTERACTION

PHASE 2

First Semiannual Report, January—June 1979

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ABSTRACT

This program has as its ultimate objective the demonstration of an advanced fuel design that is resistant to the failure mechanism known as fuel pellet-cladding interaction (PCI). Two fuel concepts are being developed for possible demonstration within this program: (a) Cu-barrier fuel and (b) Zr-liner fuel. These advanced fuels (known collectively as "barrier fuels") have special fuel cladding designed to protect the Zircaloy cladding tube from the harmful effects of localized stress and reactive fission products during reactor service. This is the first semiannual progress report for PHASE 2 of this program (January - June 1979). Progress in the irradiation testing of barrier fuel and of unfueled barrier cladding specimens is reported.

CONTRIBUTING AUTHORS

(All authors are from the General Electric Nuclear Fuel and Services Engineering Department.)

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HIGHLIGHTS

- Fuel rods which had been power ramp tested in the R-2 Test Reactor in 1978 were examined nondestructively during this report period and the earlier conclusions confirmed: (a) barrier fuel significantly outperformed reference (conventional) fuel; (b) PCI resistance was proven for Zr-liner fuel and for Cu-barrier fuel to burnups 16.6 and 12.4 MWd/kg-U, respectively.
- Tests on the behavior of barrier fuel during reactivity initiated accident (RIA) conditions showed no significant difference from that of fuel with conventional Zircaloy cladding.
- Four lead test assemblies (LTA's) of barrier fuel are under irradiation in the core of Quad Cities Unit 1 and have achieved a burnup of approximately 2.2 MWd/kg-U through June.

ACKNOWLEDGEMENTS

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J. S. Armijo of the GE Nuclear Fuel and Services Engineering Department for his continued encouragement and leadership.

C. Brown of the Quad Cities Nuclear Power Station staff, Commonwealth Edison Company, for his cooperation and invaluable assistance in collecting operating history data of the lead test assemblies (LTA's) as summarized in Section 3.

M. L. Ishikawa of the Japan Atomic Energy Research Institute for his continued interest in the RIA work and for planning and implementing the RIA tests which are summarized in Section 2.

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1. INTRODUCTION

An important mechanism for fuel failure in power reactors involves the direct interaction between the irradiated uranium fuel, including its inventory of fission products, and the Zircaloy fuel sheath. This fuel failure mechanism is known as "fuel/cladding interaction" or "pellet-cladding interaction" (PCI). This fuel failure mechanism is known to occur in all types of power reactors fueled with uranium oxide which is sheathed in Zircaloy. Building upon the General Electric Company's efforts from 1969 to 1977 to understand the PCI phenomenon and to develop potential remedies, this program was designed to exploit two remedies which General Electric (GE) had already idenfified as having good potential for success: (a) Cu-barrier fuel and (b) Zr-liner fuel. These fuel concepts are known collectively as barrier fuel, and they have been previously described[1,2].

This program leads ultimately to the large-scale demonstration of one of the two remedy concepts discussed here: Cu-barrier or Zr-liner. The overall program has been divided into three phases:

PHASE 1. Design and Supporting Tests
PHASE 2. Large-Scale Demonstration
PHASE 3. Demonstration Extending to High Burnup

PHASE 1 now has been completed and its final report issued.[3] PHASE 1 included:

1. A generic nuclear engineering study to show that the demonstration is feasible in a reactor of the BWR/3 type.

2. Laboratory and reactor tests to verify the PCI resistance of the Cu-barrier and the Zr-liner fuel types.

3. Laboratory tests of barrier cladding under simulated loss-of-coolant accident (LOCA) conditions.

4. Design, licensing documentation, fabrication and preirradiation characterization of four lead test assemblies (LTA's) for irradiation in the Quad Cities Nuclear Power Station, Unit 1, beginning in Cycle 5. Irradiation of the LTA's was begun in February 1979.

PHASE 2 will continue the work of PHASE 1, and includes:

1. Selection of the fuel design for the demonstration.

2. Nuclear design and core management of the demonstration, expanding from the generic feasibility study in PHASE 1 to a specific reactor and target cycle, including bundle nuclear designs. As presently contemplated, the reactor and target cycle are Quad Cities Unit 2, beginning with Cycle 6 (tentatively scheduled to begin in January 1982).

3. Design, licensing documentation, and manufacturing of the demonstration fuel.

4. The demonstration *per se; i.e.,* the irradiation (including specially designed power ramps to demonstrate PCI resistance). As presently perceived, PHASE 2 will include the irradiation through 1984 and is intended to include two full demonstration cycles (depending on reactor schedule).

5. Continued irradiation and evaluation of the four LTA's.

6. Continued testing of barrier fuel to assure PCI resistance at burnup levels relevant to the demonstration.

PHASE 3 is intended to extend the demonstration to high burnup. It is contingent on successful completion of PHASE 2, and details of the scope have yet to be defined.

This is the first semiannual progress report for PHASE 2; it includes work during January-June 1979.

The task structure of PHASE 2 is as follows:

Task I. Nuclear Design and Core Management

- Design of the core and the licensing of the demonstration including the mode of operation to power ramp the demonstration of fuel bundles in the test cells.
- Core management activities associated with the demonstration (but excluding that effort which would be normally required in the absence of the demonstration).

Task II. Support Tests

Subtask II.1. Laboratory Tests

- Characterization of barrier fuel and assessment of barrier stability.
- Simulated PCI tests.

Subtask II.2. Licensing Tests

- Tests to assess behavior of barrier fuel under accident conditions: (a) reactivity-initiated accident (RIA) and (b) loss-of-coolant accident (LOCA).

Subtask II.3. Fuel Irradiation Tests

- Power ramp tests in a test reactor to evaluate the PCI resistance of barrier fuel at various burnup levels.
- Stability of barrier fuel with a through-wall defect.

Task III. Demonstration

Subtask III.1. Design and Licensing of Demonstration Fuel

- Design and licensing activities associated with the barrier fuel itself.
- Integration of design and fabrication.

Subtask III.2. Fabrication of Demonstration Fuel

Subtask III.3. Irradiation and Evaluation

- Monitoring power history.
- Evaluation of performance of the demonstration fuel and analysis leading to engineering inferences.

Task IV. Lead Test Assemblies

- Evaluation of the in-reactor performance of the LTA's including correlations with power history.

During the current report period only a few of these tasks were active and only those active tasks will be discussed in this report.

2. TASK II. SUPPORT TESTS

2.1 SUBTASK II.1. LABORATORY TESTS

2.1.1 Expanding Mandrel Tests (S. B. Wisner, G. H. Henderson and R. B. Adamson)

The results of expanding mandrel tests conducted on specimens taken from the plenum sections of fuel rods irradiated in the Monticello and Millstone reactors in the GE Segmented Rod Program (SRP rods) and on unfueled specimens that were irradiated at 326°C in the Big Rock Point (BRP) commercial power reactor were reported previously.[3] Refined estimates of the fast neutron fluences (n/cm^2, E > 1 MeV) for the SRP specimens previously tested are given in Table 2.1-1, which is an updated version of Table 4.1-1 in Reference 3.

The fast fluence estimates were refined in two ways:

1. The burnup corresponding to a specific SRP plenum region was obtained from rod burnup profiles constructed from the SRP segment average burnups. The estimated burnups for the SRP plenum regions are given in Table 2.1-1.

2. Correlation of fast neutron fluence with burnup at specific rod locations (nodal position) were derived taking into account the neutron flux gradient existing near the core periphery (SRP bundle core locations). Using the burnup value for specific rod locations (e.g., plenum region) and the corresponding conversion factors, fast fluence estimates were derived for the expanding mandrel test specimens obtained from SRP plenum regions. The estimated fast fluences are 25% to 35% lower than the values previously reported.

The conclusions stated previously[3] are unaffected by these corrections.

2.1.2 Irradiations in the EBR-II (R. P. Tucker and R. B. Adamson)

Irradiation experiments in the Experimental Breeder Reactor (EBR-II) are being conducted to provide candidate barrier Zircaloy-2 cladding with high fluence exposures for postirradiation evaluations of the resistance of these materials to stress corrosion cracking under simulated PCI conditions, i.e., expanding mandrel tests.

The initial four nonfueled tubing specimens irradiated in EBR-II (and designated EBR-II-1) included reference (bright-etched) Zircaloy-2, copper on etched Zircaloy-2, copper on autoclave-oxidized Zircaloy-2 and 0.076 mm low-oxygen sponge zirconium-lined Zircaloy-2. These specimens have been irradiated and have been sectioned in accordance with a detailed cutting plan. The flux monitor rings obtained from two of these rods have been evaluated using the ^{58}Ni(n,p) ^{58}Co and ^{54}Fe(n,p) ^{54}Mn reaction rates. The neutron fluence values (E > 1 MeV) determined for the near-core-center tube (reference Zircaloy-2) at four axial locations are:

Location	Fluence (E > 1 MeV)
Core midplane	$2.1 \pm 0.5 \times 10^{21}$ n/cm^2
Near bottom of core	$1.9 \pm 0.5 \times 10^{21}$ n/cm^2
Mid-position of below core region	$1.9 \pm 0.5 \times 10^{20}$ n/cm^2
Bottom of rod/subassembly	$1.7 \pm 0.5 \times 10^{19}$ n/cm^2

The fluences determined for the rod located on the side of the subassembly away from the core center are approximately 20% lower at each corresponding position.

One mandrel specimen from the reference rod, trimmed to final test size, exhibited high radioactivity readings of $\sim$ 200 R/h on contact and $\sim$ 2-3 R/h at 1m. These activity levels exceed the limits permitted for the out-of-cell mandrel test facility. A plan is being developed to provide remote mandrel test capability in order to obtain test results in a timely manner.

Table 2.1-1
SUMMARY OF EXPANDING MANDREL TEST RESULTS FOR IRRADIATED COPPER PLATED ZIRCALOY-2, ZIRCONIUM-LINED ZIRCALOY-2 AND ZIRCALOY-2 TUBING
(CROSSHEAD SPEED — 0.025 mm min^{-1}, 335°C FOR I_2 AND LIQUID Cd, 300°C FOR SOLID Cd)

Source	Material	Estimated Burnup[d] (MWd/kg-U)	Estimated Fast Fluence n/cm^2(E>1 MeV)	Environment	Average Total Diametral Strain (%)	Plastic Strain (%)	Comments
BRP	Zircaloy-2		3.7×10^{20}	I_2 at 4 Pa	2.6	1.1	SCC Fracture
SRP3/14-1	Zircaloy + 5 μm Cu EP.UNB.	8.9	1.6×10^{21}	I_2 at 4 Pa	4.2	3.0	No Fracture
SRP3/17-2	Zircaloy + 0.076 mm Zr[c]	8.4	1.3×10^{21}	I_2 at 4 Pa	4.3	3.3	No Fracture
SRP3/18-2	Zircaloy + 0.076 mm Zr[c]	9.1	1.6×10^{21}	I_2 at 4 Pa	4.4	3.1	No Fracture
BRP	Zircaloy-2		3.7×10^{20}	I_2 at 40 Pa	1.8	0.5	SCC Fracture
SRP3/19-1	Zircaloy + 0.076 mm Zr[c]	7.0	1.5×10^{21}	I_2 at 40 Pa	2.0	1.1	SCC Fracture
BRP	Zircaloy + 10 μm Cu EP.UNB.[b]		5.4×10^{20}	Sol. Cd	3.7	2.5	No Fracture
BRP	Zircaloy + 0.076 mm Zr		4.4×10^{20}	Sol. Cd	4.6	2.9	No Fracture
SRP3/18-1	Zircaloy + 0.076 mm Zr[c]	9.1	1.6×10^{21}	Sol. Cd	4.1	2.8	No Fracture
SRP3/2-1	Zircaloy + 5 μm Cu EP.BND.[a]	4.2	7.7×10^{20}	Liq. Cd	1.4	0.5	SCC Fracture
SRP3/2-2	Zircaloy + 5 μm Cu EP.BND.	4.2	7.7×10^{20}	Liq. Cd	0.8	0.3	SCC Fracture
SRP3/13-1	Zircaloy + 5 μm Cu EP.UNB.[a]	8.4	1.3×10^{20}	Liq. Cd	4.2	3.6	No Fracture
SRP3/13-2	Zircaloy + 5 μm Cu EP.UNB.[a]	8.4	1.3×10^{21}	Liq. Cd	3.8	2.5	No Fracture
SRP3/17-1	Zircaloy + 0.076 mm Zr[c]	8.4	1.3×10^{21}	Liq. Cd	3.5	2.8	Fracture
SRP3/15-1	Zircaloy + 5 μm Cu EP.UNB.	7.3	1.5×10^{21}	Liq. Cd	4.2	2.9	No Fracture
SRP3/14-2	Zircaloy + 5 μm Cu EP.UNB.	8.9	1.6×10^{21}	Liq. Cd	4.4	3.0	No Fracture

SRP = Thick wall specimens — 0.028 inches (0.71 mm)
BRP = Thin wall specimens — 0.0165 inches (0.71 mm)
EP = Electroplated
BND = Diffusion bonded during fabrication
UNB = Not diffusion bonded
a = Nominal plating thickness is 5 μm; actual thickness varies from 5 to 17 μm around the circumference.
b = Ramp at 0.025 mm min^{-1} to 0.015 inches (0.38 mm) diametral expansion and hold for 4 hours.
c = Crystal bar zirconium.
d = At plenum location as interpolated from adjacent fuel above and below the plenum.

The coolant temperature below the EBR-II core and at the entrance to the core is nominally 371 °C (700 °F). Based on the subassembly power, coolant flow rate and heat transfer from surrounding subassemblies, the exit coolant temperature was calculated to be 462 °C (864 °F) for the test subassembly. A linear temperature rise from 371 °C is assumed over the 34.3 cm core length. Ex-reactor control tests with thermal histories corresponding to that of the in-reactor specimens are in progress. A second series (EBR-II-2) of four nonfueled tube specimens of the following types — reference (bright-etched) Zircaloy-2, copper on autoclave-oxidized Zircaloy-2, 0.076-mm crystal bar zirconium-lined Zircaloy-2, and 0.076-mm low-oxygen sponge zirconium-lined Zircaloy-2 — is currently under irradiation in Row 4 of EBR-II for Runs 100 through 103. The projected peak fluence (core midplane) for the rods at the conclusion of the EBR-II-2 irradiation is $\sim 8 \times 10^{21}$ n/cm² ($E > 1$ MeV). The projected fluence at the bottom of the core (lower coolant temperature) is $\sim 4.5 \times 10^{21}$ n/cm² ($E > 1$ MeV).

2.2 SUBTASK II.2. LICENSING TESTS (T. C. Rowland and L. D. Noble)

The objective of this task is to obtain experimental data on the performance of fuel with a copper or zirconium barrier as compared to that of standard fuel during RIA.

For a boiling water reactor (BWR), the design basis RIA is a hypothetical case in which the control rod (blade) becomes decoupled from the control drive while in the inserted position. It is then postulated that the control drive is withdrawn, but the control rod remains in the reactor, to drop out, suddenly, at some later time. Analysis indicates that the most severe transients occur during ambient or hot standby conditions. The barrier tubing will be tested under both of these conditions.

The tests at ambient temperature were conducted in 1978 and 1979 at the Nuclear Safety Research Reactor (NSRR) in Japan. These tests were arranged by the U.S. Nuclear Regulatory Commission (NRC) through their cooperative exchange agreement with the Japan Atomic Energy Research Institute (JAERI) for the exchange of safety research information. Tests at elevated temperatures are scheduled for late 1979. Hot standby tests will be performed in 1980 and 1981 at the Power Burst Facility (PBF) in Idaho, with irradiated fuel.

Currently, 31 tests with GE barrier tubing are planned for NSRR. Sixteen of the tests have been performed at room temperature and atmospheric pressure (0.1 MPa). Included were six reference pins with conventional tubing, and five pins, each of copper and zirconium barrier tubing. Thirteen tests will be performed at high temperature (286 °C) and pressure (7.2 MPa); eight with reference tubing, and five with either copper or zirconium barrier tubing. This test matrix may be modified as a result of test results.

The RIA tests deposit energy into the fuel very rapidly by pulsing the reactor (such as NSRR) with a large power burst of short duration. For NSRR, the power burst may typically have a half width of 4 to 5 ms, and the energy deposition may be as high as 500 cal/g (2092 J/g) of UO_2, depending on the fuel enrichment. At very high energy depositions, the fuel becomes fragmented, while at low magnitude depositions no visible change occurs. At intermediate energies, external cladding oxidation, cladding deformation, and small cracks may develop.

The tests planned for NSRR include energy depositions up to approximately 350 cal/g (1464 J/g) as indicated in the test matrix of Table 2.2-1. These span the expected range of test conditions from cladding oxidation through complete fragmentation. The energy depositions cited here include room temperature enthalpy and are averaged over the fuel column length.

The fuel pins (i.e. short fuel rods designed especially for testing in the NSRR) were manufactured at Battelle Northwest Laboratories. Tubing was provided by GE. The fuel and fabrication were provided by the NRC.

2.2.1 Nuclear Safety Research Reactor-RIA Test Results

The results of the RIA tests at NSRR will be published in a JAERI report by Hoshi, et al.[4] Here the results to date will be summarized briefly. The fuel pin characteristics are shown in Table 2.2-2, and the fuel pin is shown schematically in Figure 2.2-1. (See photograph of NSRR fuel pin assembly in Reference 2, p. 5-40.) There were no flux depressors at the ends of the fuel column so there was a fairly large amount of end flux peaking, especially at the lower end of the fuel column (Figure 2.2-2). Essentially all of the failures occurred at the flux peak at the

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Table 2.2-1
NUCLEAR SAFETY RESEARCH REACTOR TESTS

Fuel Type/Test Condition	Energy Deposition (cal/g-UO_2) 120	150	215 to 240	250	350	Total Number Planned	Number Completed
Reference/Ambient	1	1	4	1	1	8	6
Copper/Ambient	0	1	2	1	1	5	5
Zirconium/Ambient	0	1	2	1	1	5	5
Reference/High Temperature and Pressure	1	1	4	1	1	8	0
Copper or Zirconium/High Temperature and Pressure	0	1	2	1	1	5	0
						31	16

Table 2.2-2
CHARACTERISTICS OF NSRR RIA FUEL PINS

Type of Cladding Tested	
1. Reference type Zircaloy-2	
2. Zr-lined Zircaloy-2	
3. Cu-barrier Zircaloy-2	
Fuel Pellets	
Enrichment	10% U-235
Density	95% theoretical
Geometry	45° chamfered edge
Dimension	
Pellet o.d.	10.57 mm
Pellet length	10.7 mm
Cladding o.d.	12.52 mm
Cladding wall thickness	0.86 mm
Gap width	0.115 mm
Zr-liner thickness	~10% of wall thickness
Cu-barrier thickness	~0.01 mm
Fuel column length	135.15 mm

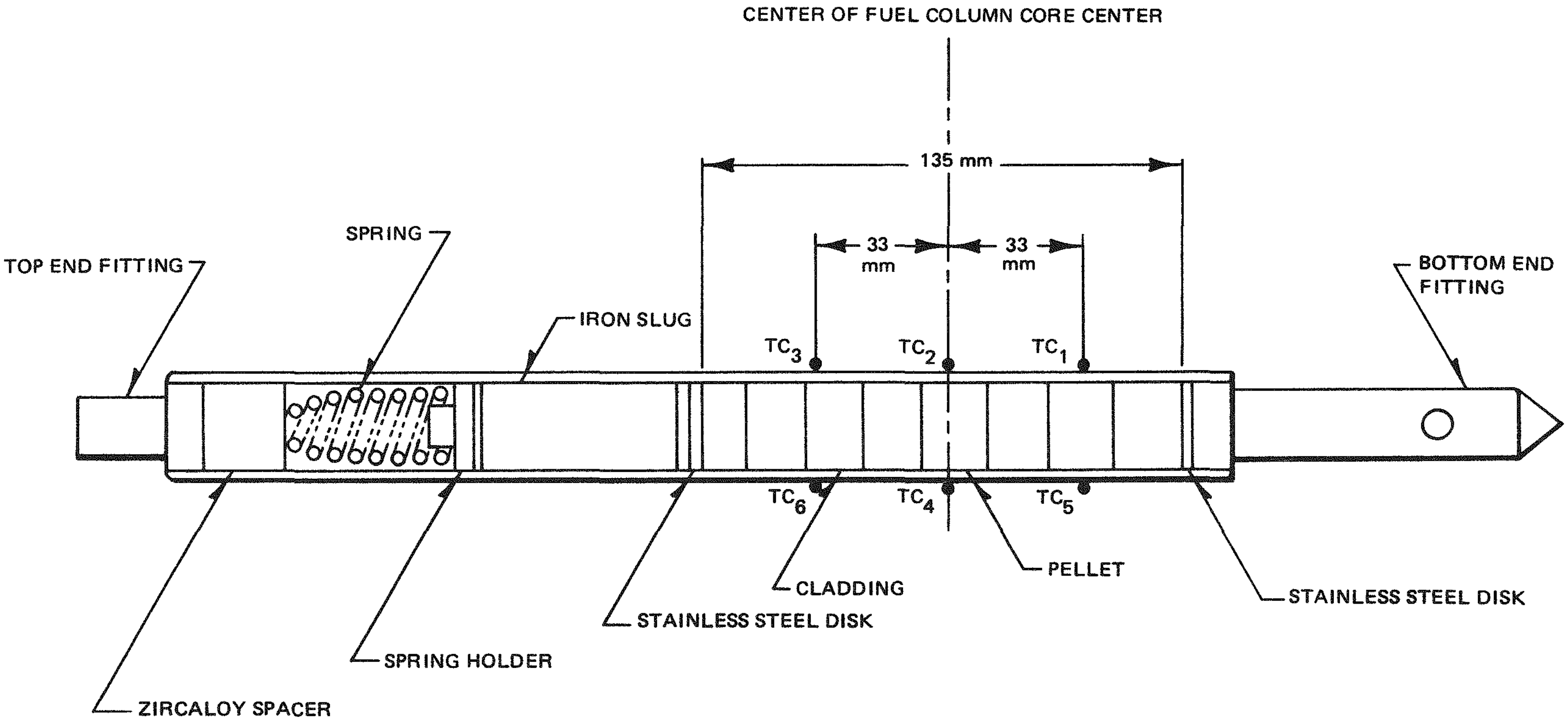

Figure 2.2-1. Nuclear Safety Research Reactor - RIA Fuel Pin

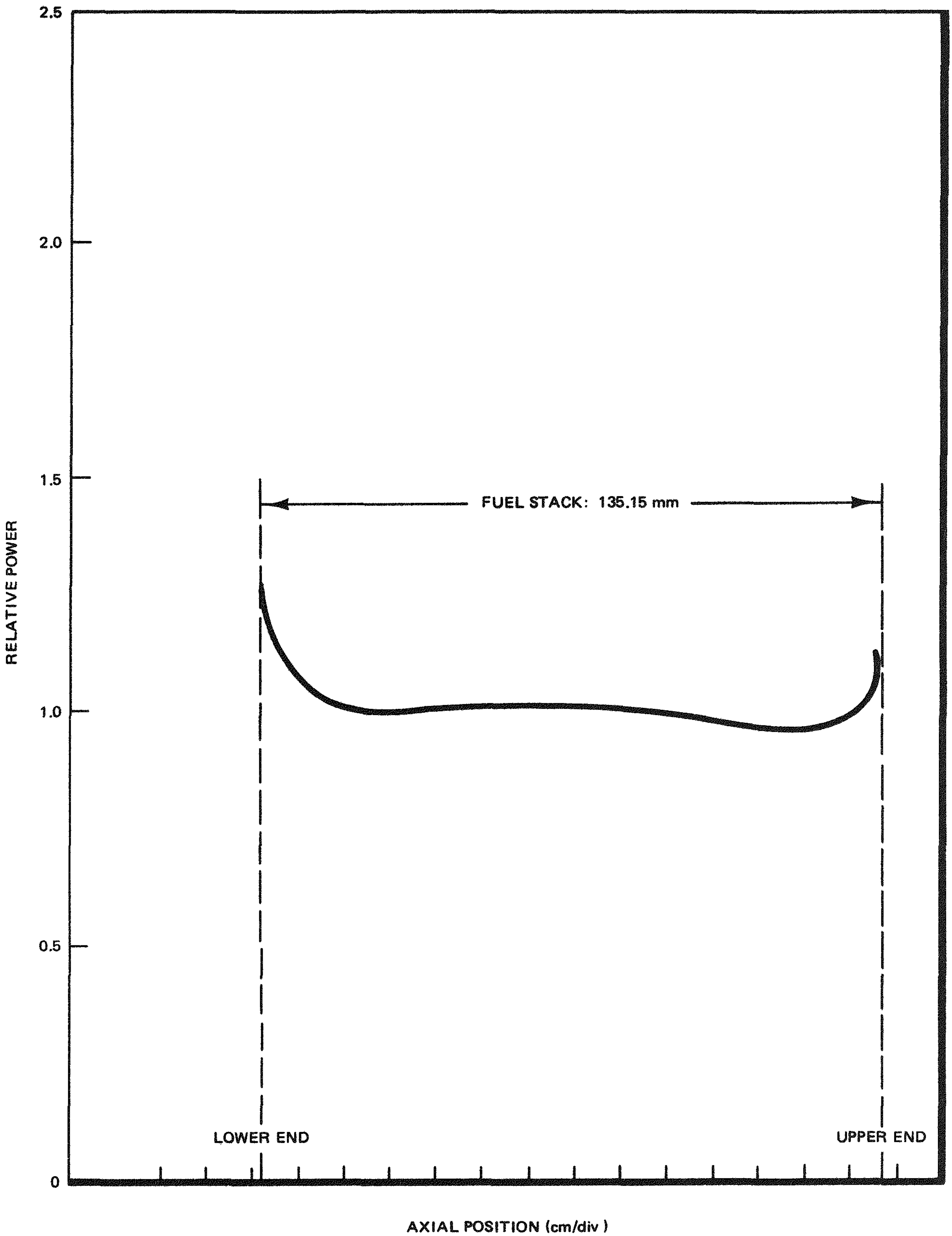

Figure 2.2-2. Typical Axial Power Distribution

bottom of the fuel column which would indicate that the energy deposition to failure should be 10 to 20% higher than the nominal. However, previous tests performed in the NSRR where the flux peaking was eliminated showed that the failure threshold was the same with flux peaking as it was without, so no correction was warranted.

Reference fuel pin results are shown in Table 2.2-3, Zr-liner fuel results in Table 2.2-4, and Cu-barrier fuel results in Table 2.2-5. The results are compared in Figure 2.2-3 and the maximum surface temperatures are compared in Figure 2.2-4.

The fuel failure threshold energy for the reference pins and the Cu-barrier fuel pins is 260 to 290 cal/g-UO_2 (1088 to 1213 J/g). This is about the same as for NSRR standard fuel rods and the SPERT-IV test results.[5] The Zr-liner fuel rod failure threshold appears to be slightly higher, approximately 300 cal/g-UO_2 (1255 J/g-UO_2). At higher energies, about 400 cal/g-UO_2 (1674 J/g), none of the fuel rods in the current tests fragmented while in previous NSRR tests and SPERT IV tests the cladding fragmented. The maximum cladding temperature of the Cu-barrier fuel rods was somewhat higher than the other types of fuel.

In summary, no significant differences were observed between the barrier fuel pins and the reference fuel pins in RIA conditions.

2.2.2 Planned PBF Tests

Segmented fuel rods which were irradiated in BWR's were shipped to EG&G, Idaho, in March 1979 for RIA testing sponsored by the NRC. The objective of the RIA testing is to determine if the failure threshold changes with exposure.

The fuel rods (length = approximately 1m) which have been shipped to EG&G for testing in the period June 1980 through February 1981 are listed in Table 2.2-6.

2.3 SUBTASK II.3. FUEL IRRADIATION TESTS

2.3.1 Segmented Rod Irradiation Tests (J. H. Davies, E. Rosicky, E. L. Esch, D. K. Dennison)

2.3.1.1 Bundle Status

The irradiation status of the three segmented test rod assemblies is updated in Table 2.3-1. Four segments were removed from the SRP-3 (Millstone) bundle during the end of Cycle 6 refueling outage in May. These segments are listed in Table 2.3-2.

2.3.2 Ramp Tests

2.3.2.1 Test Results — 1978

Ramp testing of twelve SRP segments in the R2 Reactor at Studsvik was described previously.[3] The preliminary results are reproduced in Table 2.3-3. These results were subsequently confirmed by visual examination (Table 2.3-4) and neutron radiography (Table 2.3-5).

Final test results, providing greater detail and more refined estimates of failure powers, are summarized in Table 2.3-6. Note the set of data under the heading, power "spike". This effect was briefly mentioned in the previous report.[3] During ramp testing in R2, rod power is monitored calorimetrically by measuring inlet and outlet temepratures in the rig plus coolant flow rate. Fission product activity in the loop is continuously monitored and a defect is indicated by a large increase in activity. Relative rod power and loop activity are recorded in parallel on a single chart. In five of the nine recorded defects the defect signal was preceded by a small spike or deflection on the strip chart output of the instrument monitoring relative power as a function of time. The time interval between the spike and a large activity release ranged from about 1 minute up to 98 minutes. An example where there was good separation of the two signals, is shown in Figure 2.3-1. These spikes

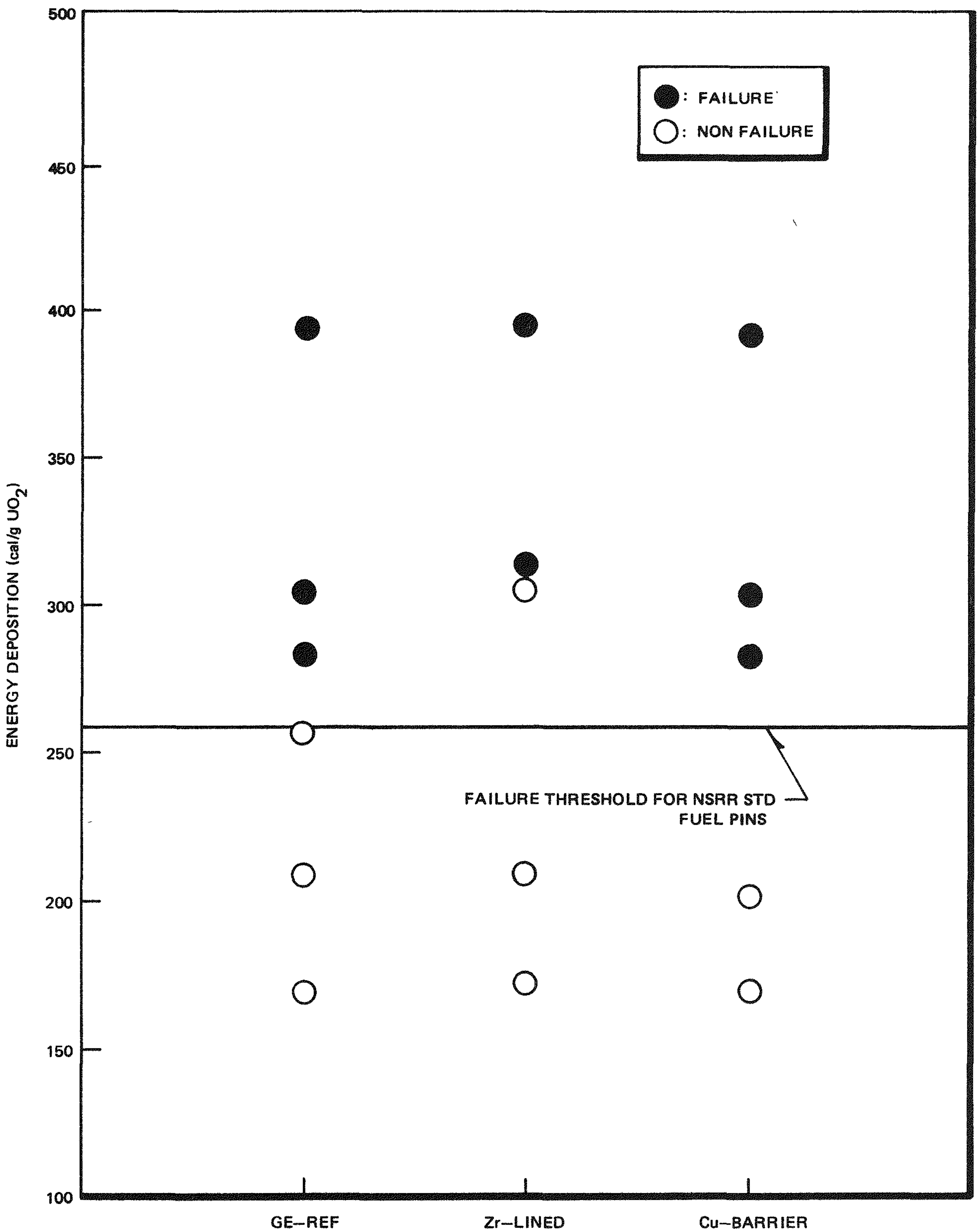

Figure 2.2-3. Comparison of Failure Threshold for Reference and Barrier Clad Fuel Rods

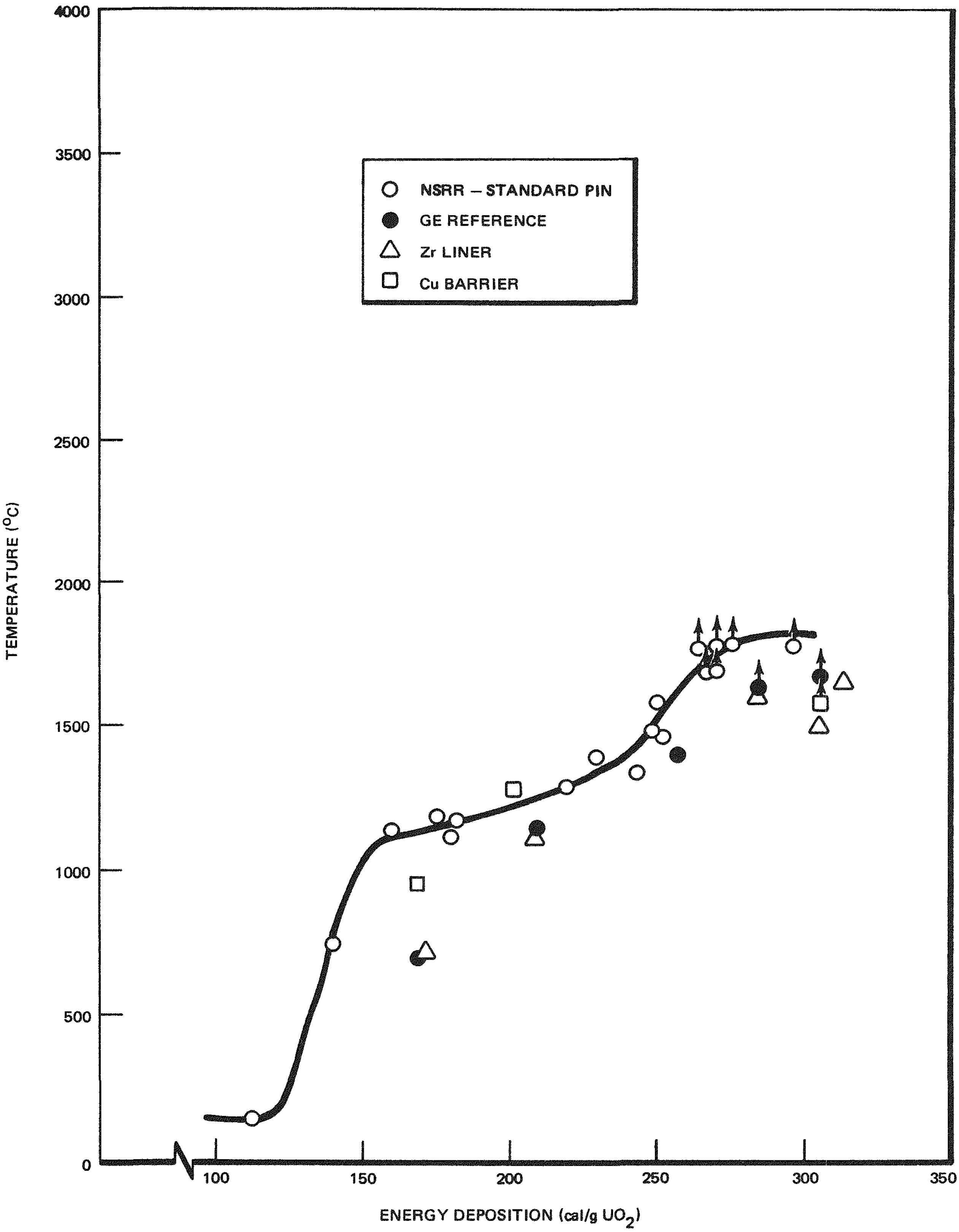

Figure 2.2-4. Maximum Cladding Surface Temperature at Axial Midpoint (Data with arrows are lower-bound estimates)

Table 2.2-3

SUMMARY OF TEST RESULTS ON REFERENCE FUEL PINS (GE CLADDING)

Energy Deposition (cal/g UO_2)	Test No.	Maximum Cladding Surface Temperature[a] No. 1 (°C)	No. 2 (°C)	No. 3 (°C)	No. 4 (°C)	No. 5 (°C)	No. 6 (°C)	Maximum Capsule Pressure bar (MPa)	Maximum Water Column Velocity (m/s)	Post-Test Observation
169	501-2	720	700	380	540	600	450	—	—	Cladding surface was slightly discolored in the fuel region. ~1s film boiling.
209	501-1	1120	1150	1120	1180	1120	1130	—	—	Cladding surface was discolored black over the entire fuel region. Fuel pin did not fail, 7-10s film boiling.
257	501-3	1420	1410	1350	1400	1420	1330	—	—	Cladding surface was discolored over the entire fuel region and oxidized Zircaloy film flaked away. Fuel pin did not fail, 8-13s film boiling.
284	501-4	1690	>1520	1530	1630	1520	1530	—	—	Melted cladding/or fuel was pushed out from the cladding and relatively large ballooning of cladding was observed in the lower portion of the pin. Several holes were at the thermocouple locations.
305	501-7	1620	>1670	1600	1640	1640	1620	—	—	Fuel rod fractured into two pieces during disassembly. A large void was observed in the fuel in the lower fractured portion.

Table 2.2-4
SUMMARY OF TEST RESULTS ON Zr-LINED FUEL PIN

Energy Deposition (cal/g UO_2)	Test No.	Maximum Cladding Surface Temperature[a] No. 1 (°C)	No. 2 (°C)	No. 3 (°C)	No. 4 (°C)	No. 5 (°C)	No. 6 (°C)	Maximum Capsule Pressure bar (mPa)	Maximum Water Column Velocity (m/s)	Post-Test Observation
393	501-8	>1150	>1030	>1120	>1120	—	—	2.5	2.2	Cladding was melted extensively at the lower portion of the rod. Fuel was expelled from the rod and fragmented. Oxidation of the cladding was observed in a portion of the plenum.
171	502-2	640	700	480	710	810	800	—	—	Cladding surface was slightly discolored over the fuel region, $\sim$1s film boiling.
208	502-1	1110	1130	1130	1110	1250	1140	—	—	Cladding surface was discolored black over the entire fuel region. Fuel pin did not fail, 6-9s film boiling.
304	502-4	1510	1500	1480	1500	1410	1510	—	—	Cladding surface was discolored and dulled over the entire fuel region. Fuel pin did not fail, 7-11s film boiling.
313	502-3	1690	1530	1510	1650	1600	1650	—	—	Fuel pin fractured into two pieces during disassembling. A void was observed in the fuel and the cladding was melted.
394	502-5	1130	1100	1480	1170	—	—	3.4 (0.34)	4.8	Cladding was melted extensively. Fuel was expelled from the pin and fragmented. Oxidation of cladding was observed in a portion of the fuel plenum.

Table 2.2-5
SUMMARY OF TEST RESULTS ON Cu-BARRIER FUEL PINS

Energy Deposition (cal/g UO_2)	Test No.	Maximum Cladding Surface Temperature No. 1 (°C)	No. 2 (°C)	No. 3 (°C)	No. 4 (°C)	No. 5 (°C)	No. 6 (°C)	Maximum Capsule Pressure bar (mPa)	Maximum Water Column Velocity (m/s)	Post-Test Observation
169	503-2	710	730	970	950	830	890	—	—	Cladding surface was discolored over the entire fuel region, 1-6s film boiling.
201	503-1	1370	1280	1200	failed	1350	1350	—	—	Cladding surface was discolored black over the entire fuel region. Fuel pin did not fail, 12-18s film boiling.
283	503-3	1090	1610	1580	1410	1640	failed	—	—	Fuel pin fractured into two pieces during disassembling. A void was observed in the melted cladding in the broken portion.
304	503-4	1500	1550	1430	1590	1580	1620	—	—	Fuel pin fractured into two pieces. A large void was observed in the fuel and the cladding was melted.
392	503-5	1670	1100	1520	1540	—	—	1.5 (0.15)	3.4	Cladding was melted extensively. Fuel was expelled from the pin and fragmented. Oxidation of cladding was observed in a portion of the fuel plenum.

Table 2.2-6
GENERAL ELECTRIC-FUEL ROD SEGMENTS[a]
PBF TESTS

Test[b]	SRP Rod No.	Pellet Cladding Gap mm/in.	Approximate Exposure MWd/kg-U	Fuel Type
RIA 1-7	W5-2	0.229/0.009	14.4	Reference
RIA 1-7	0D07-2	0.299/0.009	15.3	Reference
RIA 1-7	STR-137	0.229/0.009	9.6	Reference
RIA 1-7	0C08-4	0.229/0.009	9.1	Cu-barrier[c]
RIA 1-3	9C07-1	0.229/0.009	13.1	Cu-barrier[c]
RIA 1-3	STR-134	0.229/0.009	13.3	Reference
RIA 1-7	8D15-3	0.178/0.007	14.0	Reference
RIA 1-3	0A06-1	0.178/0.007	13.1	Reference
RIA 1-7	5D05-5	0.229/0.009	15.3	Reference
RIA 1-7	DTB-2406	0.229/0.009	5.3	Zr-liner 0.076 mm

[a]Rod o.d. = 12.52 mm (0.493 inches), cladding wall thickness = 0.864 mm (0.034 inches)

[b]Test — RIA reactivity insertion accident test
RIA 1-3 Test Date Sep 1980
RIA 1-7 Test Date Feb 1981
OPT-ATWS tests OPT 1-1 scheduled for June 1980.
All filled with 1 atmosphere (0.1 MPa) He except DTB — 2406 Fuel — 3 atm. (0.3 MPa) Fuel Rod Length 955 mm (37.6 in.).

[c]0.01 mm (0.0004 in.) Cu — Diffusion bonded

Table 2.3-1
SRP IRRADIATION STATUS

STR Bundle	Segment Tier	Average Exposure (MWd/kg-U)	Highest SRP Segment Average Exposure (MWd/kg-U)	Date
SRP-1	Top	10.7	12.7	April 1979
(Quad Cities-1)	Middle Top	15.6	18.6	
	Middle Bottom	15.6	19.4	
	Bottom	13.2	16.7	
	Bundle Average	13.8		
SRP-2	Top	12.0	16.0	May 1979
(Monticello)	Middle Top	18.4	24.4	
	Middle Bottom	20.8	27.4	
	Bottom	18.5	24.3	
	Bundle Average	17.4		
SRP-3	Top	11.8	16.1	April 1979
(Millstone)	Middle Top	17.1	22.0	
	Middle Bottom	19.2	24.6	
	Bottom	17.6	23.6	
	Bundle Average	16.4		

Table 2.3-2
SEGMENTS RETRIEVED DURING FOURTH RECONSTITUTION OF BUNDLE SRP-3 (MILLSTONE)

Segment Serial No.	Design Feature[a]	Cladding Wall Thickness (mm)	He Pressure (MPa)	Estimated Average Burnup[b] (MWd/kg-U)
STR 046	Zr-Liner (crystal bar)	0.71	1.7	13.4[c]
STR 049	Zr-Liner (crystal bar)	0.71	1.7	20.0
DTC 2303	5 μm Cu-Barrier (on oxide)	0.86	0.3	5.6
DTC 2305	5 μm Cu-Barrier (on oxide)	0.86	0.3	6.7

[a] Cladding heat treatment, recrystallization anneal; fuel density 95.5%; diametral gap 0.23 mm.

[b] Estimated segment burnups subject to revision following evaluation of ^{137}Cs gamma scan data.

[c] Top segment; peak/average burnup ratio approximately 1.5.

Table 2.3-3
1978 R2 RAMP TESTS — RESULTS SUMMARY

Segment Identification	Design Feature	Average Burnup (MWd/kg-U)	Maximum Peak Power[a] (kW/m)	(kW/ft)	Time at Maximum Power[b]	Test Results
SRP-2/10	Reference	16.4	39.4	12	16 min	Failed
SRP-2/11	Reference	15.4	45.9	14	0[c]	Failed
SRP-2/13	Reference	15.4	52.5	16	30 s	Failed
SRP-2/14	Reference	14.6	45.9	14	30 s	Failed
SRP-2/19	Reference	14.6	52.5	16	29 min	Failed
SRP-2/20	Reference	13.8	45.9	14[d]	59 min	Failed
SRP-2/1	10 μm Cu-Barrier (nonbonded)	12.4	59.1	18	12 h	Sound
SRP-2/4	10 μm Cu-Barrier (bonded)	14.4	59.1	18	100 min	Failed
SRP-3/33	5 μm Cu-Barrier (nonbonded)	16.7	52.5	16	39 min	Failed
SRP-3/34	5 μm Cu-Barrier (nonbonded)	16.1	59.1	18	0[c]	Failed
SRP-3/35	Zr-Liner (crystal bar)	16.6	59.1	18	12 h	Sound
SRP-3/36	Zr-Liner (crystal bar)	15.3	59.1	18	12 h	Sound

[a]Nominal Value after last ramp step.
[b]Failure times based on loop activity level.
[c]Failed during ramp step to peak power shown.
[d]Segment ramped to 54.5 kW/m (16 kW/ft) before failure indication recorded (2½ minute delay in loop activity monitor).

Table 2.3-4
1978 R2 RAMP TESTS
HOTCELL VISUAL EXAMINATION RESULTS

Segment Identification	Feature	Result	Observation	Axial Location[a] (mm)
SRP-2/10	Reference	Failed	Crack[b]	261.8
			Transverse Crack	294.5
SRP-2/11	Reference	Failed	Crack	303.2
SRP-2/13	Reference	Failed	"Comet Mark"[c]	302.3
SRP-2/14	Reference	Failed	"Comet Mark" and Heavy Oxide	278.6
			"X-Mark"	279.0
			2 "X-Marks"	300.8
			Water Seep	722.1
SRP-2/19	Reference	Failed	Crack	394.7
SRP-2/20	Reference	Failed	Crack with Deposit	316.8
SRP-2/1	10 μm Cu, nonbonded	Sound	No Defect	
SRP-2/4	10 μm Cu, bonded	Failed	Crack with Deposit	387.9
SRP-3/33	5 μm Cu, nonbonded	Failed	Heavy Oxide[d]	470.7
			Heavy Oxide	487.8
			Heavy Oxide	508.0
SRP-3/34	5 μm Cu, nonbonded	Failed	Crack with Deposit	286.4
			Crack with Deposit	310.1
SRP-3/35	Zr-Liner	Sound	No Defect	
SRP-3/36	Zr-Liner	Sound	No Defect	

[a]Axial location measured from Bottom End Plug Weld.
[b]Several cracks had water seeping out.
[c]Usually due to deposits on cladding downstream from a defect.
[d]At a Millstone spacer contact point.

Table 2.3-5
1978 R2 RAMP TESTS NEUTROGRAPHY RESULTS

Segment Identification	Hydrogen in Getter	Moisture at End Plug Welds
SRP-2/1	0	0
SRP-2/4	Medium Concentration	+
SRP-2/10	Large Concentration	+
SRP-2/11	Medium Concentration	+
SRP-2/13	Medium Concentration	+
SRP-2/14	Small Concentration	+
SRP-2/19	Barely Visible	+
SRP-2/20	Very Small Concentration	+
SRP-3/33	Small Concentration	+
SRP-3/34	Small Concentration	+
SRP-3/35	0	0
SRP-3/36	0	0

Table 2.3-6
1978 R2 RAMP TESTS — FINAL RESULTS SUMMARY

				Ramp Testing					
	Standardization		First Power	Power "Spike"		Activity Release		Max Power Level	
Segment Identification	Power (kW/m)	Time (h)	Level (kW/m)	Power (kW/m)	Delay Time (min)	Power (kW/m)	Delay Time (min)	Power (kW/m)	Time (min)
SRP-2/13	27.1	304	27.0	~47	0	51.8	1	51.8	4.5
SRP-2/14	24.4	304	24.0	~41	0	46.0	1	46.0	4.5
SRP-2/11	22.7	304	22.7	39.1	10.5	~41	0	46.2	2.5
SRP-2/20	28.1	304	27.9	—		45.8	59	52.0	1
SRP-2/10	26.3	301	27.0	39.2	4	39.2	15.5	39.2	20
SRP-3/36	24.2	301	24.7	—		—		58.6	720
SRP-3/34	24.4	301	24.6	—		~57	0	59.0	4
SRP-2/19	24.5	572	24.6	—		52.2	29.5	52.2	35
SRP-3/35	26.3	572	26.0	—		—		58.6	720
SRP-3/33	26.3	572	26.5	—		52.1	36	52.1	41
SRP-2/1	24.1	572	24.1	—		—		58.2	720
SRP-2/4	26.3	301	25.8	~55	0	58.5	98	58.5	102

were not observed on chart output from sound rods, nor were they observed when the defected rod was prepressurized (four rods). The precise cause of the spikes has not yet been established; but there is no reason to believe that they are spurious signals, and it has been hypothesized that they indicate the time of the actual defect. This hypothesis is consistent with an observation in the INTER-RAMP program,[6] which indicated that activity release to the coolant may lag considerably behind actual defection. A ramped fuel rod, which was taken off test prior to a defect signal, was shown nevertheless to be defected.

The results are presented graphically in Figure 2.3-3. As expected, all of the reference rods failed and were outperformed by the barrier rods. The presently proposed barrier designs, crystal bar Zr-liner and 10 μm nonbonded Cu-barrier, both performed well in the power ramp test, but three other Cu-barrier rods defected.

The failure of SRP-2/4, bonded Cu-barrier, was predictable on the basis of expanding mandrel tests results,[2] which showed that this design was embrittled by liquid cadmium, whereas the nonbonded barrier was not. Defection of SRP-3/33 and -3/34, 5 μm nonbonded Cu barrier, was not expected, however because similar designs had survived earlier ramp tests in both the CC and SRP tests series.[1,2] An evaluation was performed of possible "wear out" mechanisms for this loss of PCI-protectiveness. Based on earlier results[2] which showed small but significant Cu-Zr interdiffusion at cladding temperatures, it was concluded that the formation of a brittle intermetallic layer at the barrier-cladding interface was most likely responsible. If this turns out to be the case, and planned 1979 ramp tests should provide some insight (see below), the Cu-on-oxide barrier concept should be more resistant to defection, as the oxide layer forms a barrier to interdiffusion.

2.3.2.2 Test Plans — 1979

Thirty-six (36) irradiated fuel rod segments have been selected for ramp testing in 1979. Again the program will be performed in the R2 Test Reactor at Studsvik. Following detailed precharacterization (visual examination, profilometry, gamma scanning, neutron radiography, pulsed eddy-current) the segments were shipped to Studsvik in July. The segments, which are listed in Table 2.3-7, included 14 reference rods, 6 Zr-liner rods and 16 Cu-barrier rods. The test program has four main objectives:

1. Investigation of the need for an extended low power standarization period prior to ramp testing.

   The 300-hour standarization irradiation, which was used in the 1978 R2 test series, was a somewhat arbitrary requirement, included to preserve test parity with earlier tests at the GE Test Reactor (GETR). This phase of the test is time-consuming and expensive and should be discarded if unnecessary.

   Seven reference rods (see Table 2.3-7) have been selected to evaluate a 6-hour standarization. The ramping sequence (Figure 2.3-4) will be similar to that used previously[2] and results will be compared to previous results. Depending on the outcome of this comparison, all subsequent 29 tests will employ either the 300-hour or the new 6-hour standarization period.

2. Testing the PCI resistance of Cu-barrier fuel at lead exposures.

   Three Cu-barrier rods (10 μm, nonbonded) at burnups in the range 12.9 to 15.2 MWd/kg-U will be tested following the standard "staircase" ramp sequence to confirm the previous results (SRP-2/1 in Table 2.3-3) at somewhat higher burnups. One electroless Cu-barrier rod will be tested at ~5.9 MWd/kg U.

3. Testing the ability of zirconium and Cu-barrier fuel to survive a large power step at low to medium exposures.

Table 2.3-7
1979 FUEL RAMP TEST
PROGRAM PLAN

GE Rod Designation	Rod Characteristics	Rod Average Exposure (MWd/kgU)	Test Objective
SRP-2/7	Reference	15.9	Investigation of requirement for an extended low-power operating period prior to ramp testing.
SRP-2/8	Reference	14.9	
SRP-2/9	Reference	13.8	
SRP-2/16	Reference	15.8	
SRP-2/25	Reference	18.1	
SRP-2/26	Reference	20.4	
SRP-2/27	Reference	19.2	
SRP-2/28	10 μm Cu-barrier (nonbonded)	13.1	Confirmation of PCI resistance of Cu-barrier fuel at lead exposures.
SRP-2/29	10 μm Cu-barrier (nonbonded)	15.2	
SRP-2/30	10 μm Cu-barrier (nonbonded)	13.0	
SRP-3/40	10 μm Cu-barrier (electroless)	6.3	
SRP-2/12	Reference	12.4	Demonstration of Cu- and Zr-barrier fuel ability to survive large power increase steps at low-to-medium exposure
SRP-2/15	Reference	16.4	
SRP-2/21	Reference	10.4	
SRP-2/22	Reference	15.9	
SRP-2/24	Reference	19.4	
SRP-3/7	Reference	3.6	
SRP-3/29	Reference	9.0	
SRP-2/2	10 μm Cu-barrier (non-bonded)	10.5	
SRP-2/31	10 μm Cu-barrier (non-bonded)	8.1	
SRP-2/32	10 μm Cu-barrier (on oxide)	4.8	
SRP-3/16	5 μm Cu-barrier (nonbonded)	5.6	
SRP-3/41	10 μm Cu-barrier (electroless)	6.2	
SRP-3/42	5 μm Cu-barrier (electroless)	5.7	
SRP-2/34	Zr-barrier (sponge)	4.8	
SRP-3/20	Zr-barrier (crystal bar)	5.6	
SRP-3/44	Zr-barrier (crystal bar)	5.9	
SRP-3/45	Zr-barrier (crystal bar)	5.7	
SRP-3/46	Zr-barrier (crystal bar)	7.4	
SRP-3/47	Zr-barrier (crystal bar)	7.0	
SRP-3/24	10 μm Cu-barrier (bonded)	9.6	Exploration of Cu-barrier performance limits
SRP-3/25	10 μm Cu-barrier (bonded)	9.3	
SRP-3/31	5 μm Cu-barrier (bonded)	10.9	
SRP-3/32	5 μm Cu-barrier (bonded)	10.5	
SRP-3/38	10 μm Cu-barrier (bonded)	15.5	
SRP-3/39	10 μm Cu-barrier (bonded)	15.0	

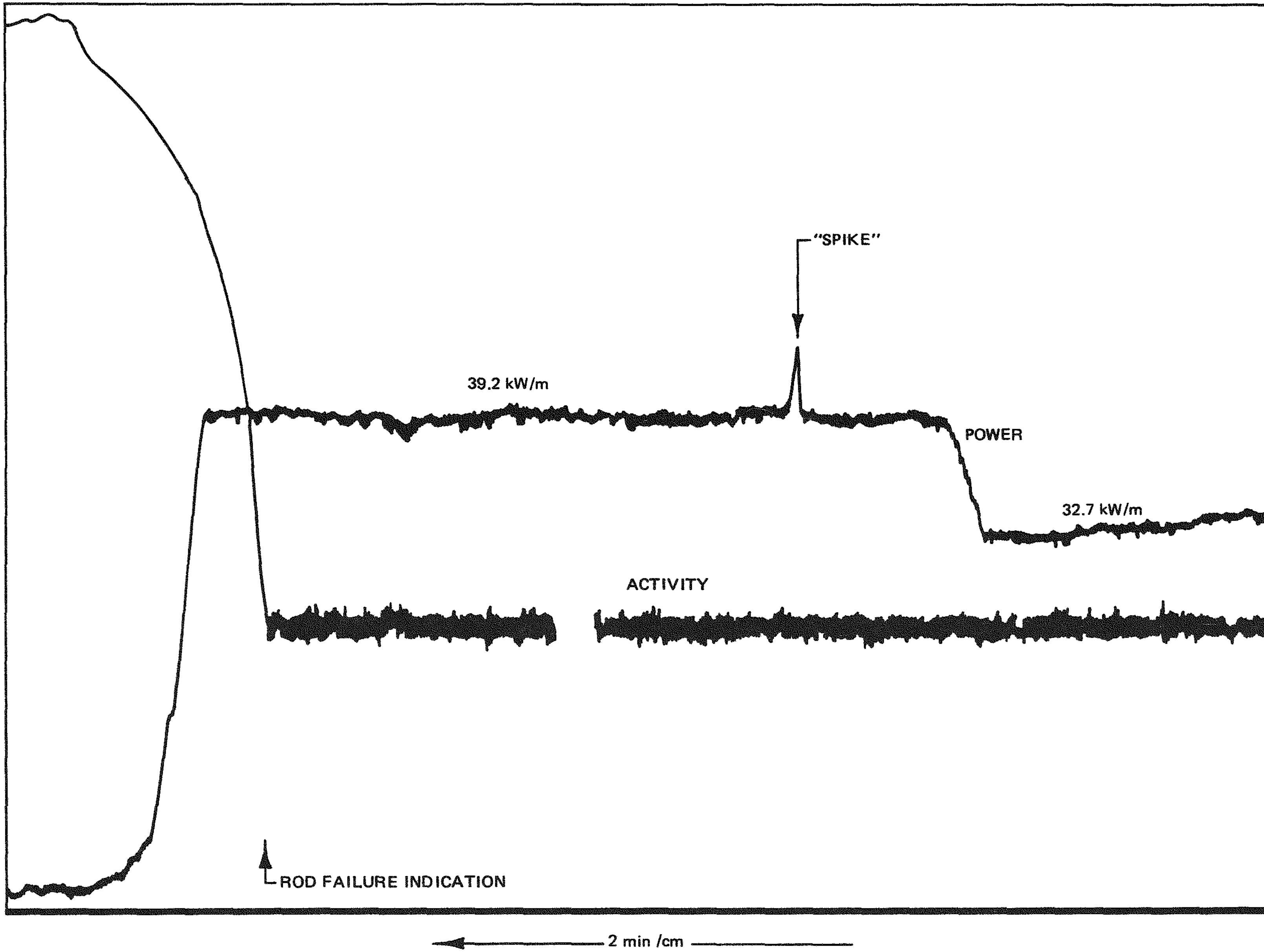

Figure 2.3-1. Recorder Chart Output of Relative Rod Power (Flow x Temperature Increment) and Loop Activity Level Indicating Failure of SRP - 2/10

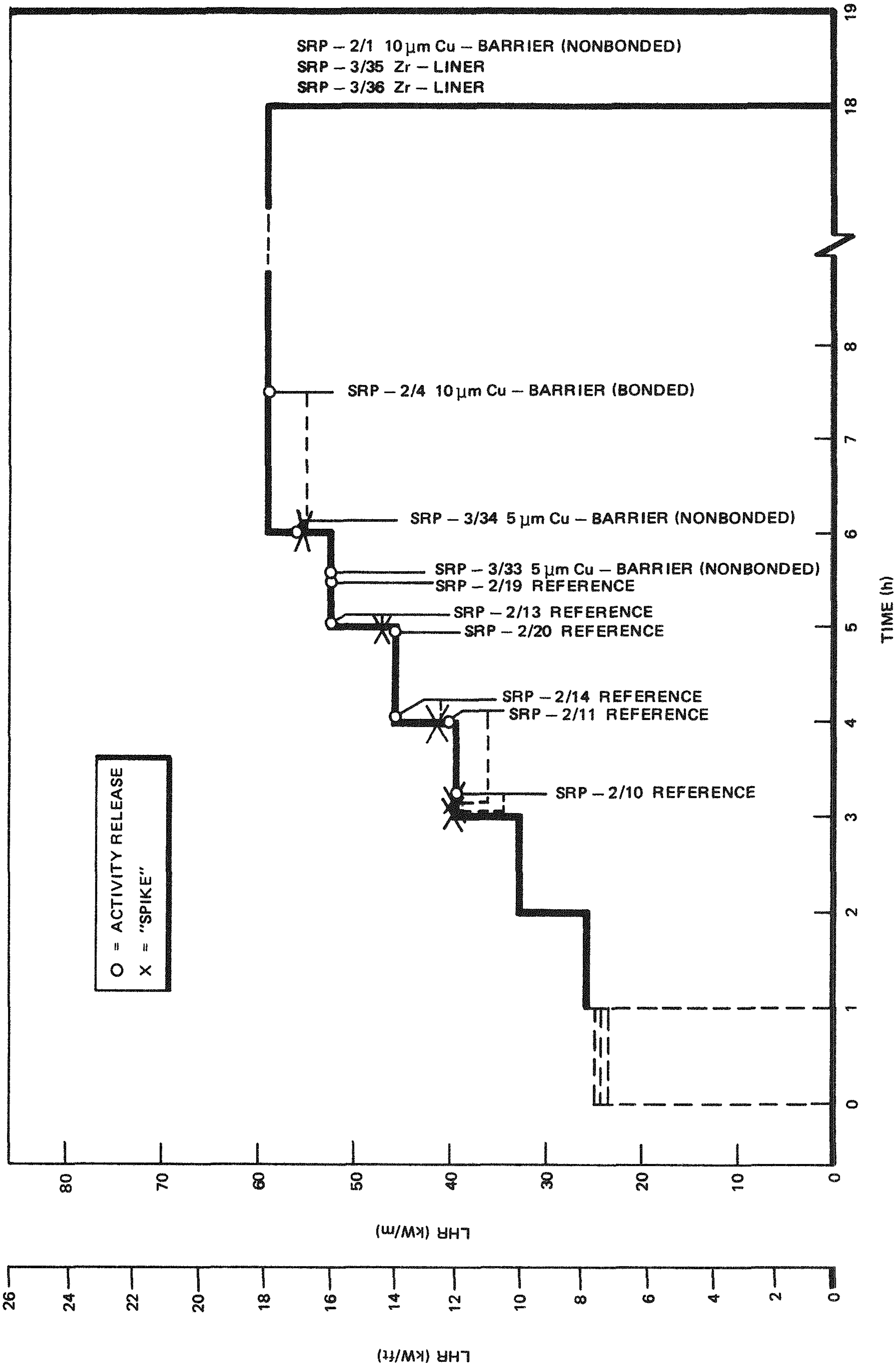

Figure 2.3-2. Ramp Test Results Summary

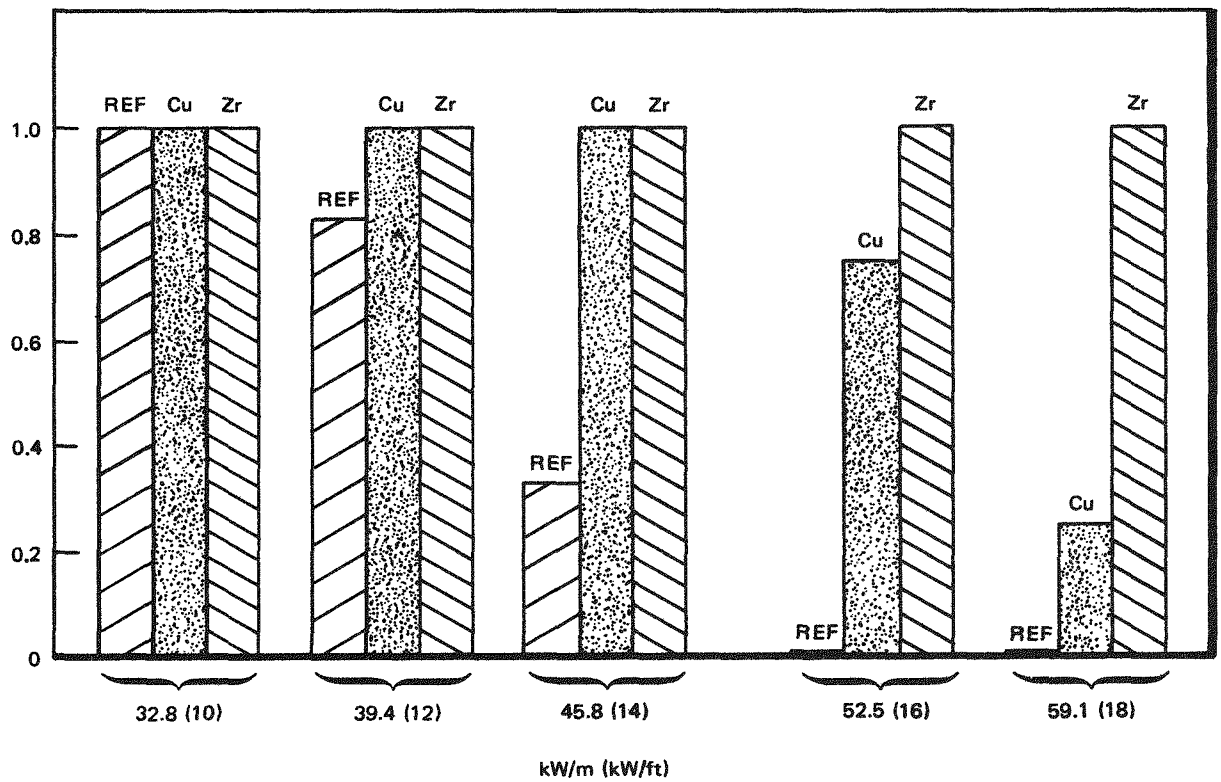

Figure 2.3-3. Fraction of SRP Rods Surviving Ramp Test in R2 to Peak Power Shown

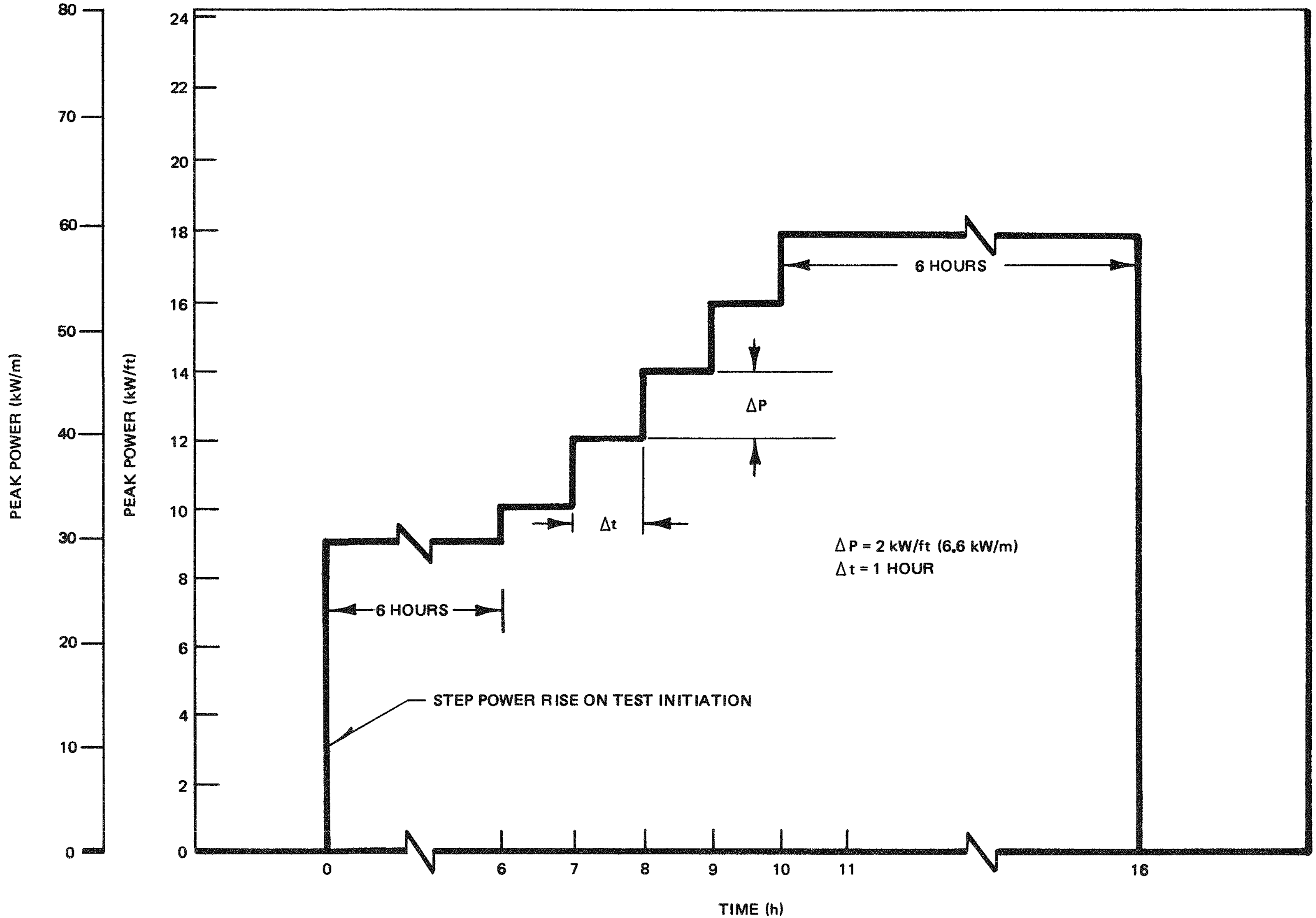

Figure 2.3-4. Ramp Test Sequence to Investigate Effect of Short (6 Hours) Standardization Phase

This test is a closer simulation of a control blade withdrawal than the standard "staircase" ramp test. Following standardization (300 hours or 6 hours), the fuel will be subjected to a severe step ramp of ~26.2 kW/m at a ramp rate of ~19.7 kW/m-min. If the fuel rod survives 2 hours, additional information will be provided by subjecting the rod to an additional small step (see Figure 2.3-5) if possible.

Twenty (20) rods will be tested in this manner, including 6 Cu-barriers, 6 Zr-liners and 7 references.

4. Investigation of performance limits of Cu-barrier fuel.

Failure of two thin (5 μm) Cu-barrier rods in the 1978 test series, after universally good performance at lower exposures, has indicated a possible exposure-dependent loss of protectiveness of the Cu-barrier. This "wear-out" phenomenon is believed to be due to the formation of a brittle Cu-Zr intermetallic layer by interdiffusion. A small test matrix has been set up in this task to test this hypothesis and to explore the effects of residual copper layer thickness and exposure.

The six (6) rods selected for this test matrix are identified in Table 2.3-7. The copper barriers in all these rods were diffusion bonded to the cladding during fabrication. Thus, the test is an accelerated test of the deleterious effect of intermetallic layer formation. The rods will be ramped according to the standard "staircase" sequence and the results compared among themselves and with the results of the nonbonded Cu-barrier tests above.

2.3.3 Test of Fuel with Cladding Perforation

An in-reactor test is being designed to investigate the performance of barrier fuel relative to that of conventional fuel should a defect (cladding perforation) occur. Experimental details have not yet been established, but the intent is to irradiate intentionally perforated barrier and reference fuel rods for a meaningful period at typical power levels.

Based on literature data, an empirical relationship has been defined by Locke[7] which distinguishes between successful operation of defected fuel (i.e., secondary damage is minor), and failure (i.e., severe secondary damage or hydriding). This relationship is shown in Figure 2.3-6 in terms of heat flux and days at power. The boundary between "successful defect operation" and "failure" is distinguished by the severity of the cladding damage and activity release to the coolant. The coordinates of the proposed barrier fuel test in the context of Locke's correlation are also shown in Figure 2.3-6. Unless the barriers have a deleterious effect on defected fuel operation, the rods should operate in the safe zone.

It is intended to initiate the test before the end of 1979. Current work is aimed at preparing the test specification, defining the test matrix and developing techniques to fabricate the artificial defects.

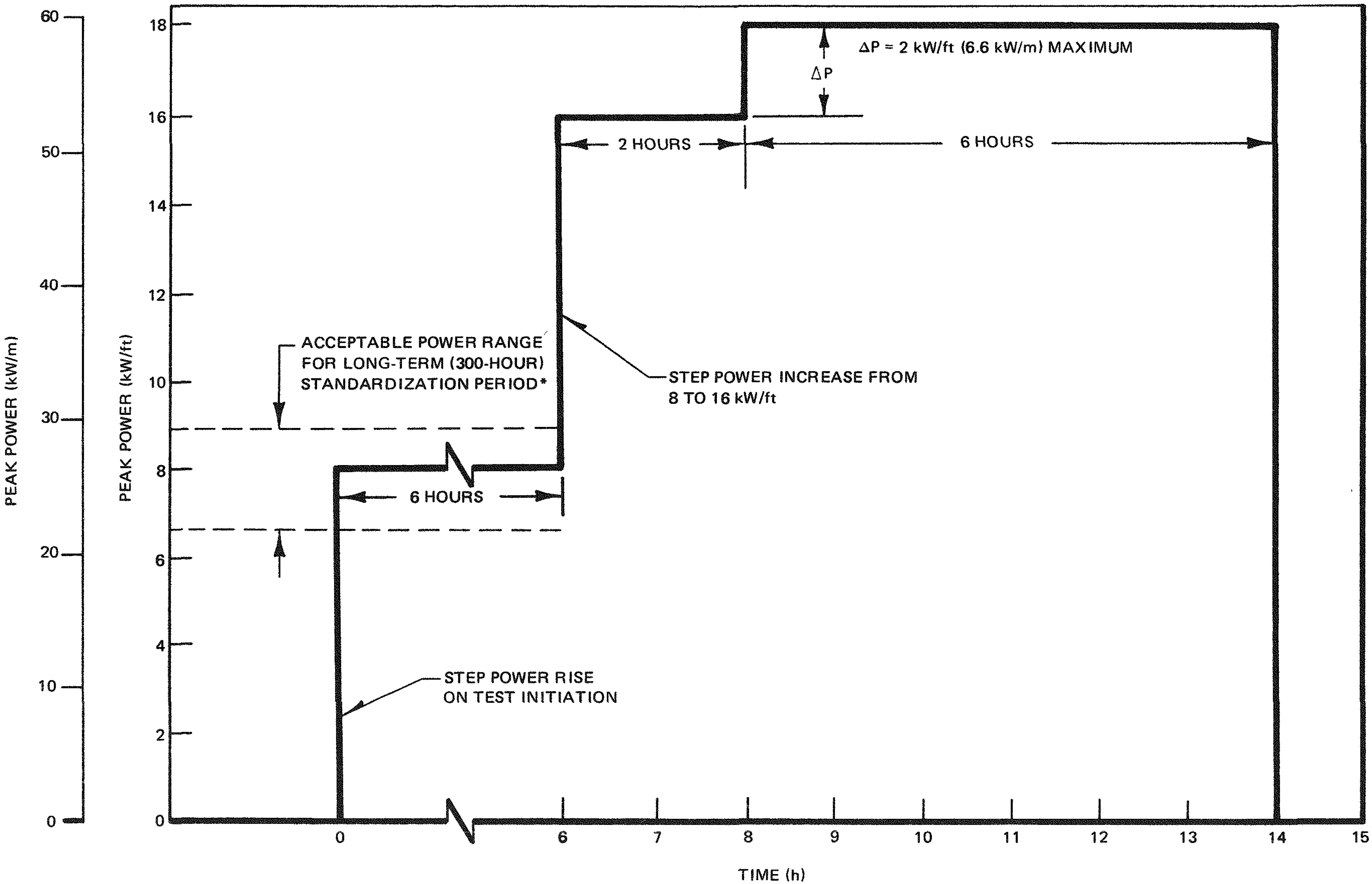

Figure 2.3-5. Ramp Test Sequence to Simulate Control Blade Withdrawal

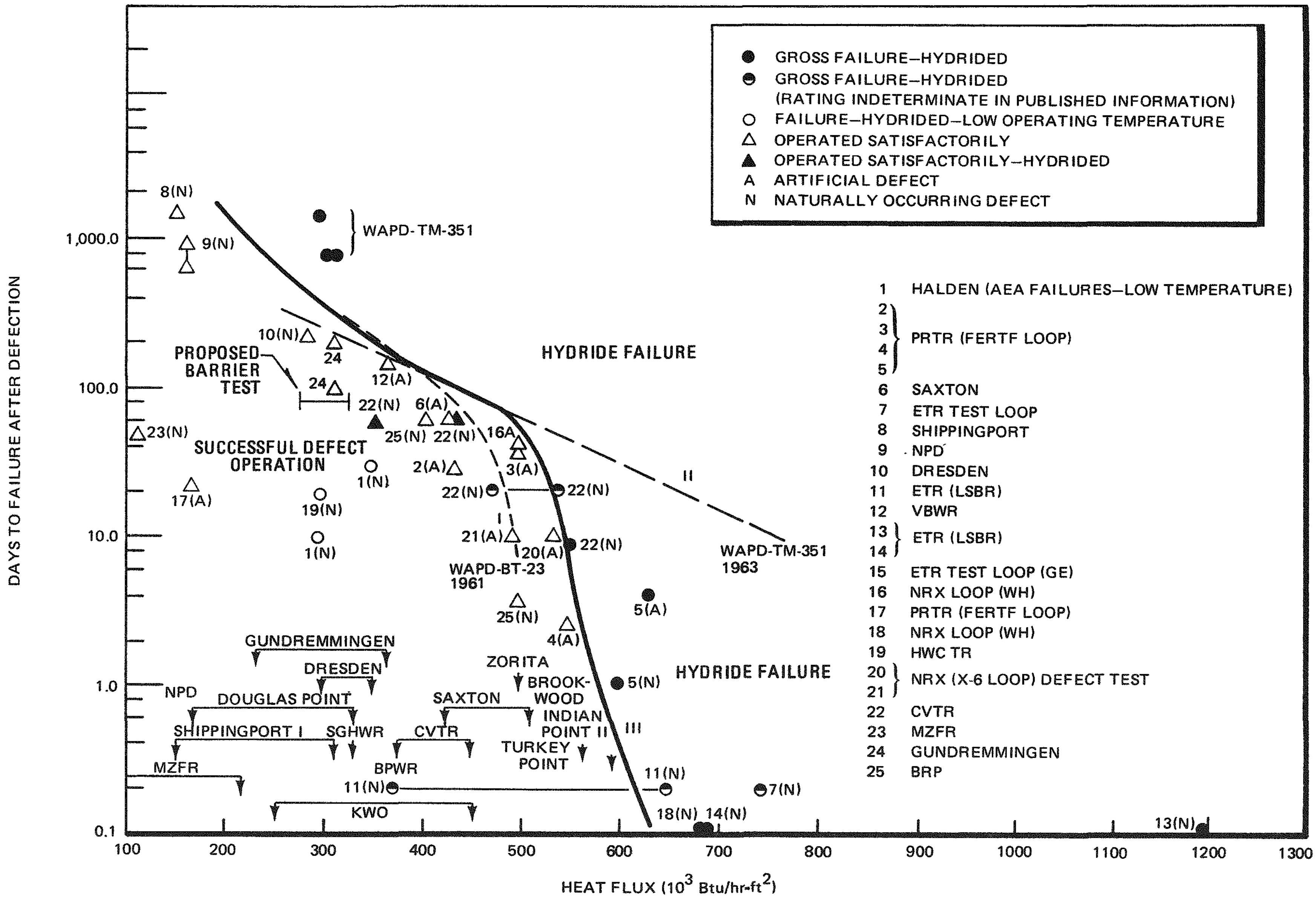

Figure 2.3-6. Assessment of the Failure of Defective Zircaloy Clad Fuel (after Locke[7])

3. TASKS III AND IV DEMONSTRATION AND LEAD TEST ASSEMBLIES
(S. Sen, and B. B. Sarma)

The barrier LTA's are being irradiated at Quad Cities Unit 1. The exposure of the LTA's through June 1979 is approximately 2.2 MWd/kg-U. These assemblies are intended to lead in burnup those of the large scale demonstration fuel. The demonstration fuel is scheduled to be irradiated in Quad Cities Unit 2 from the beginning of Cycle 6 (January 1982).

To acquire power history data for the barrier LTA's the Fuel Performance Analysis and Data Acquisition System (FPADAS) has been installed in Quad Cities Unit 1, and data are being recorded continuously through that system. Since Quad Cities is a dual unit plant controlled by the same process computer, it is planned to develop a new software program so that core performance and detailed power history data can be collected from both the units and recorded onto the same magnetic tape once the demonstration fuel bundles are loaded into Quad Cities Unit 2.

In order to analyze the composite site tape (containing data from both plants) a software program will also be developed which will separate the data from the two plants.

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