GENERAL ELECTRIC

NUCLEAR POWER

SYSTEMS DIVISION

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125
MC 682, (408) 925-3344

MFN-156-80
REE-054-80

September 11, 1980

U.S. Nuclear Regulatory Commission
Division of Licensing
Washington, D.C. 20555

Attention: Robert L. Tedesco
Assistant Director for Licensing

Gentlemen:

SUBJECT: ADDITIONAL INFORMATION ON NEDO-24259, "GENERIC INFORMATION FOR BARRIER FUEL DEMONSTRATION BUNDLE LICENSING"

The additional information requested in your letter dated August 28, 1980 is provided for your review and approval in Attachment A and in its Attachments A1 thru A5.

Since you expressed a desire to have the background information as a part of the generic licensing document, the responses and attachments will be incorporated in the final document as appendices after approval has been received.

We are concerned that the target date of September 15, 1980 for approval cannot be met. We believe that your approval by September 30, 1980 is necessary so that hardware manufacture can begin in October 1980 for the timely implementation of this project.

We've noticed that most of the additional information requested deals with the nature of the program and its future plans. Very few of the requests dealt with safety issues. We are pleased with your apparent interest in the program, and we believe that there are no outstanding safety problems. For this reason, we feel that from a licensing standpoint, there should be no reason that the required approval date could not be met. We will be pleased to keep you fully informed of the future progress of the project.

B

8009160424

This project is considered to be of major importance to the General Electric Company and to the nuclear industry. We hope that you will make every effort possible to expedite the review and meet our necessary schedule.

Very truly yours,

R E Engel

R. E. Engel, Manager
Reload Fuel Licensing

REE:lm/1467, 1480

Attachment

cc: M. Tokar, NRC
R. O. Meyer, NRC
R. B. Bevan Jr. , NRC
L. S. Rubenstein, NRC

RESPONSE TO REQUESTS BY NRC ON NEDO-24259

REQUEST 231.1

Although the report is intended to provide a basis for the demonstration irradiation of PCI-resistant fuel in a commercial power reactor, hardly any background discussion is provided on the barrier fuels test program. Although references are cited in the text, it is desirable to have a brief summary discussion of the barrier fuel test program. The discussion should focus on (a) test results that have indicated that this particular barrier material and design is effective against PCI failure and (b) how the tests to date support the belief that the insertion of the demonstration bundle in Quad Cities Unit 2 will not result in reduced safety margin or increased fission product release. The response to this question should be incorporated into NEDO-24259, perhaps as an appendix, following completion of this review, so that the report constitutes a "stand-alone" document (which references other documents containing greater detailed information on specific points).

RESPONSE TO 231.1

The requested background information is provided in the attached text (Attachment A1), which has been excerpted from the report GEAP-23773-2 (Reference 6 of NEDO-24259). This information is a summary of Phase 1 of the program, "Demonstration of Fuel Resistant to Pellet-Cladding Interaction". Notice that within the Phase 1 work fuel test data were obtained that indicated that barrier fuel of the Zr-liner type had PCI-resistance to power ramps up to 18 KW/ft at a burnup of >16 MWd/kg U. More recent data are available which show similar PCI-resistance of the barrier fuel to a burnup of >23 MWd/kg U. These data are from ramp tests in a test reactor and are shown in Attachment A2. The portion of the figure on A2-2 within the dotted region shows the power-burnup range in which failures were observed in reference fuel (conventional Zircaloy-2 cladding). The Zr-liner fuel successfully withstood a ramp beyond the upper curve which marks the range in which 100% of the reference fuel rods have failed. The superior performance of the barrier fuel under severe PCI conditions is evident.

However, in Document NEDO-24259, licensing is sought for use of the specific barrier fuel bundles on a generic basis under normal operating conditions. Implicit is the inference that this barrier fuel will not result in reduced safety or in increased fission product release. The thermal-mechanical design work and the safety analyses for these bundles have shown all safety parameters to be within acceptable levels for a plant of the BWR/3 type.

In this program, tests have been performed for the effects of both the reactivity insertion accident (RIA) and the loss of coolant accident (LOCA), and results have indicated that Zr-liner barrier fuel behaves as well as conventional (non-barrier) fuel under these postulated accident situations. The results of the RIA tests appear here as Attachment A3 which is excerpted from the Phase 2 first semiannual report, GEAP-25163-1. The RIA tests conducted at the Nuclear Safety Research Reactor in Japan have been fully documented in the report by J. Hoshi, K. Iwata, J. Yoshimura, and M. Ishikawa, "Fuel Failure Behavior of PCI-Remedy Fuels Under the Reactivity Initiated Accident Conditions", JAERI-M 8836 (May 1980). Results of simulated LOCA tests of Zr-liner cladding appeared in the Phase 2 second semiannual report, GEAP-25163-2, which is hereby included as Attachment A4. See Section 3.3 for the LOCA data.

Release of fission products from barrier fuel rods which somehow become perforated has been addressed both in laboratory tests and in reactor tests of perforated fuel rods. No increased release of activity for the barrier fuel was observed or inferred from these tests. See Attachment A4, Sections 3.2.1 and 3.4.2.

Testing to date through the entire scope of the program has indicated that Zr-liner barrier fuel behaves at least as well as conventional fuel under all normal operating conditions. Further proof of this point is expected to be available in the Fall of 1980, when the two lead test assemblies of Zr-liner fuel will be examined after exposure to ~12 MWd/kg U in Quad Cities Unit 1 (substantially higher exposure than estimated earlier in the program). Those results will be made available to NRC in the normal reporting format for this program; see response to 231.4.

REQUEST 231.2

The barrier fuel demonstration in Quad Cities Unit 2 is intended to demonstrate the comparative resistance of the design to PCI failure, not only through normal reactor operation, but also when subjected to planned power ramps near the end of four fuel cycles. This is a large-scale demonstration (involving 132 bundles of which 64 will be ramped) and the ramp feature of the demonstration, in particular, is unique. Because the fuel design (Zr-liner cladding) is novel; because the demonstration involves a large number of bundles; because repeated power ramp testing of a new fuel design has not been carried out heretofore in a commercial power reactor (i.e., this sets a precedent); because some rods may fail (particularly during the ramps); and because it is desirable to obtain potentially useful information on the performance of the barrier fuel subjected to the demonstration (in preparation for possible future reload applications), there is a special need for adequate failure detection, fuel damage surveillance and analytical failure prediction capability. Please address, therefore, the following concerns:

a. On-Line Failure Detection

Describe the on-line failure detection capability at Quad Cities Unit 2. Discuss the utility's commitment to use the instrumentation; e.g., if gas samples are to be obtained at the steam jet air ejector, how often would such samples be taken? Provide any correlations that might be used to relate activity monitor readings or coolant radiochemistry to the number and/or location of leaking fuel rods. What is the minimum "delay time" (the time interval between the instant of cladding failure and the time at which the failure can be detected)? Discuss the potential benefits and costs of installing an on-line gamma spectrometer (GELI detector).

b. Post-Irradiation Surveillance

The report should contain a discussion of the PIE program to be undertaken on the discharged demonstration bundles as well as any interim examination to be performed on that fuel during refueling outages. The disposition of failed fuel should be addressed as well as any examination that might be directed toward detecting "partially damaged" fuel (i.e., incipient failures).

c. PCI Failure Prediction

Provide pre-ramp predictions of PCI failure probability using whatever model or code GE is currently using for that purpose. Briefly describe the analytical method.

RESPONSE TO 231.2

This request refers to various aspects of the program work scope, but seems to express concern over fuel failures. Even with the power ramps at the ends of the cycles, all operation is planned to be within established limits and within the Technical Specifications. All test information indicates that fuel failure will not occur. Any additional restrictions on operation are unwarranted.

a. On-Line Failure Detection

Within the scope of this generic license request the utility has no commitment beyond normal operation and the normal failure detection instrumentation and procedures. Pending NRC review and approval of the generic license document, NEDO-24259, which will permit the utility to accept the barrier fuel bundles and to insert them in the core, a detailed operating plan for the demonstration will be developed jointly by GE and the utility. That plan will address the question in increased sampling before and after the ramp. The correlations of bundle failure with offgas and water analyses will be considered at that time.

It should be noted, however, that all operations will be within the plant's Technical Specifications, and the sampling methods and parameters described there are considered adequate, particularly since, with the barrier fuel, the probability of failures is decreased.

b. Post-Irradiation Surveillance

The demonstration program has been structured only through PHASE 2, which includes the first two power ramps in Quad Cities Unit 2 (through Cycle 7). Within the present scope of work the barrier fuel bundles will be sipped and visually examined at the discretion of the utility. The question of post-irradiation examination of discharged demonstration bundles has been deferred to PHASE 3, and the scope of work of this phase has yet to be determined.

c. PCI Failure Prediction

With the current state of the art, General Electric does not believe a reliable analytical model exists to accurately predict PCI fuel failure probabilities. The ramp test data (Attachment A2-2) indicate no failures in the Zr-liner barrier fuel in severe power ramp tests. Since the power ramps in the demonstration will not be nearly as severe, no failures of the Zr-liner barrier fuel are expected. With regard to the conventional fuel in the core, the demonstration will keep power levels within current BWR practice and the experience base clearly indicates a high probability that the demonstration will produce no fuel failures. Reliance on experience and experimental data is consistent with the General Electric philosophy of test-before-use and is deemed to be the best engineering practice.

REQUEST 231.3

Section 4 "Safety Analysis" of the report provides little specific information on the potential effects of change in cladding properties on the results of the safety analysis. While it may be necessary to wait for the establishment of a final fuel loading pattern before licensing calculations can be performed and reported, general trends can surely be determined now.

Therefore, for each of the 6 parameters and accidents listed on page 4-1 of the report, please list the associated major cladding property parameter change that will have the greatest effect on the analysis of the accident and discuss in general terms what that effect will consist of.

RESPONSE TO 231.3

Refer to NEDO-24259, p. 4-1/4-2. For items 1 through 3 and item 5, the cladding properties of interest are those involving the flow of heat from the fuel to the coolant. The flow of heat is influenced principally by the gap conductance and only secondarily by the thermal conductivity of the cladding. The barrier cladding is manufactured to the same specification for surface finish (i.e. roughness) as is conventional Zircaloy cladding, and no effect is anticipated regarding gap conductance. The thermal conductivity of zirconium so closely matches that of Zircaloy that no effect is anticipated. The flow of heat is assured by the metallurgical bond between the zirconium liner and the Zircaloy. No evidence of de-bonding has ever been observed, even after severe deformation tests (strain ~10%) of high exposure cladding samples. In manufacture the tubing is inspected for delamination over 100% of its length to assure soundness of bond and unimpeded flow of heat.

Item 4 refers to the loss of coolant accident (LOCA) situation, which focuses on the peak cladding temperature (PCT). This, too, is affected by the flow of heat radially through the cladding, and the response for Items 1-3 and 5 applies here as well. In addition, the strength of the cladding must be considered. Previous calculations were performed for the two Zr-liner lead test assemblies and were shown in the report, "Quad Cities Nuclear Power Station Unit 1 Reload 4 Supplemental Licensing Information for Barrier Lead Test Assemblies," NEDO-24147 (September 1978). In that analysis it was conservatively assumed that cladding strength was reduced 10% (i.e., the liner contributes zero strength). There was no effect on the calculational methods, however, the lower strength leads to an increase in the calculated number of perforations at a given PCT. Even with that unrealistically conservative assumption, the effect upon LOCA performance was deemed acceptable. In Attachment A4, pp. 3-18 through 3-24 are the results of simulated LOCA tests, which show that the behavior of Zr-liner cladding (unirradiated) is similar to that of conventional Zircaloy cladding.

Item 6 refers to the rod drop accident. Testing at the Nuclear Safety Research Reactor in Japan has demonstrated that the presence of Zr-liner barrier cladding has no effect under such conditions (Attachment A3 and the previously referenced report JAERI-M 8846).

REQUEST 231.4

The report contains no description of key milestones or schedules. Although some of that information was presented at the March 25, 1980 meeting between GE, DOE, Commonwealth Edison, and NRC staff, it would also be desirable to have that information in writing (to be incorporated into the report) along with a discussion of the intended plan for timely reporting of pertinent information to NRC regarding the results of the demonstration.

RESPONSE TO 231.4

Milestones and target dates are shown in Attachment A5. All data generated in this program are reported informally and in a preliminary manner each month, with formal progress reports at semiannual intervals. Several members of the NRC staff now receive the semiannual report.

Results of the demonstration will be reported within the normal program report format. Requests for additional information should be made through the utility, and the question of data transfer to NRC will be considered at the next meeting of the Program Steering Committee, which guides the program within the bounds set by the program contract.

CLH:cas:gmm/115-J7

1. INTRODUCTION

Experience in the nuclear industry with fuel rods of Zircaloy-clad urania has brought to light several causes of fuel rod failure. Most of these causes have been corrected by innovative design modifications and by improvements in manufacturing processes. However, there persists one class of fuel failures which has yet to be eliminated and which presently is controlled by reactor design and operational constraints. These failures are caused by the direct interaction between the irradiated urania fuel, including its inventory of fission products, and the Zircaloy fuel sheath, or cladding. This phenomenon is called "fuel/cladding interaction" or "pellet-cladding interaction" (PCI). The incidence of such failures is closely linked to the power history of the fuel rod and to the severity and duration of power changes. Pellet-cladding interaction fuel failures have occurred in all types of water-cooled reactors that are fueled with urania which is sheathed in Zircaloy: boiling water reactors (BWR), pressurized water reactors (PWR), Canadian deuterium-moderated reactors (CANDU), and the steam generating heavy water reactor (SGHWR).

Recently, national policy regarding light water reactor (LWR) technology has focused on the goal of improved uranium utilization in a fuel cycle that does not depend upon fuel reprocessing. Higher burnup can improve uranium utilization, but the design of fuel for high burnup service requires an acceptable resistance to PCI at those high burnup levels.

Of more immediate concern for utilities are the operational constraints which have been imposed on commercial power reactors to ameliorate the PCI phenomenon. While these operational procedures have been successful in reducing the incidence of fuel failure, the procedures constrain certain reactor operations and are costly in terms of capacity factor. There is a strong incentive to provide a remedy, *i.e.*, a fuel which is resistant to the PCI failure mechanism and which can be operated to high burnup with improved plant capacity factors.

Building upon the General Electric Company's extensive previous efforts (1969 — 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 identified as having good potential for success: (a) Cu-barrier fuel and (b) Zr-liner fuel. Copper-barrier fuel has a Zircaloy sheath with a very thin layer of copper plated on the inner surface. Zirconium-liner fuel has a Zircaloy fuel cladding with a metallurgically bonded layer of zirconium on the inner surface. Both the Cu-barrier fuel and the Zr-liner fuel are known collectively as "barrier fuel".

The ultimate objective of this program is to realize demonstration of the PCI-resistance of a fuel based on one of these potential PCI remedies. The demonstration will be in a commercial BWR and it is intended to test a sufficient quantity of fuel to form a reliable data base regarding the performance characteristics of the new fuel. While it is not yet known in which reactor the actual demonstration will occur, it probably will be in a reactor of the BWR/3 type with a steady-state core, where the barrier fuel will be introduced as part of a reload batch.

Prior to the actual demonstration there must be an adequate data base to enable design and licensing; fabrication and quality assurance problems must be addressed; and there must be extensive nuclear physics, fuel management, and power history analyses so that the experimental fuel is properly tested with minimum risk to other fuel in the core. Consequently, the program has been structured to consider each of these aspects.

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 includes all work from July 1, 1977, through February 28, 1979. 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 simulating loss-of-coolant accident (LOCA) conditions.

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

PHASE 2 will continue the work of PHASE 1, and it will also include:

1. Selection of the remedy fuel 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.

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 test PCI resistance) and the evaluations. As presently perceived, PHASE 2 will include the irradiation through September 30, 1984.

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

PHASE 3 is intended to extend the demonstration to high burnup. It is contingent on successful completion of PHASES 1 and 2, and details of the scope have yet to be defined.

This report is the Final Report for PHASE 1 of this program. It covers progress since the last progress report,[1] as well as a summary of the entire PHASE 1 effort (Section 2).[1,2]

2. SUMMARY FOR PHASE 1

2.1 DESIGN OF LARGE-SCALE DEMONSTRATION

Using Unit 1 of the Quad Cities Nuclear Power Station at the beginning of Cycle 6 as an example, the feasibility of the demonstration irradiation was determined. Quad Cities 1 is an operating commercial power reactor of the BWR/3 type with a steady-state core. The power density and other design features of a BWR/3 make such a reactor well-suited for the demonstration, but the results of the demonstration should be applicable generally to LWR's.

The demonstration, as presently perceived, will involve the insertion of the advanced, PCI-resistant fuel as part of a normal reload batch. The demonstration has been so designed that (a) the PCI resistance of the advanced fuel can be demonstrated by suitable power increases on certain nodes (*i.e.*, axial locations) of the advanced fuel bundles; (b) simultaneously, the power limits and power changes on the conventional, non-remedy fuel are held within limits specified to minimize the risk to the conventional fuel; (c) the demonstration does not inhibit unduly the reactor performances as a central station power plant; and (d) the operational procedures are compatible with the needs and capabilities of the utility which operates the plant. The demonstration plan which has been developed fulfills these goals.

The demonstration nuclear core design involves the use of test cells which contain the barrier fuel assemblies and which are symmetrically placed in the core. Each test cell contains four barrier fuel assemblies surrounding one cruciform control blade. The test cells are operated with their control blades inserted 60 to 100% of their length during most of the reactor cycle. Near the end of the cycle the control blades of the test cells are withdrawn in a stepwise fashion to impart a rapid increase in power in the barrier fuel to test its resistance to PCI. The test cells are surrounded by a buffer of high exposure (~20 MWd/kg-U) fuel bundles. This buffer zone serves to isolate the increase in power due to the withdrawal of the test cell control blades and thereby to protect the standard fuel in the core. The entire process has been analyzed to compare the suggested operation using the demonstration fuel and the demonstration mode of operation to what would normally be expected for Cycles 6 through 13. The demonstration mode produces very nearly the same energy as the normal mode of operation, but the extra neutron absorption of the copper barrier (~0.25% reactivity penalty) results in ~0.14% energy penalty for the demonstration. Use of Zr-liner fuel results in no energy penalty. Power ramp simulation analyses using a 3-dimensional coupled nuclear-thermal hydraulic simulation have shown that the demonstration is feasible, providing both adequate design margin and a good demonstration of PCI resistance of the barrier fuel, while protecting the conventional fuel bundles in the core from power changes likely to produce fuel cladding penetrations by PCI. Also, margins for transient and accident situations were shown to be adequate in the demonstration mode.

2.2 SUPPORT TESTS FOR LARGE-SCALE DEMONSTRATION

2.2.1 Laboratory Tests

2.2.1.1 Expanding Mandrel Tests

Expanding mandrel tests provide a controlled, localized, noncompliant stress system on cladding specimens along with exposure to temperature and corrosive environments that produce stress corrosion effects in Zircaloy. Thus, the expanding mandrel tests constitute a laboratory simulation of PCI. Such tests were done with both irradiated and with unirradiated barrier cladding and with suitable control specimens. The expanding mandrel tests were used to compare the stress corrosion resistance of irradiated Cu-barrier, Zr-lined and conventional cladding. Tests were done also with unirradiated materials to explore the effects of fabrication parameters.

Unirradiated specimens were tested in several environments: (a) flowing iodine (I_2) in a carrier gas of argon, (b) pure cadmium (either above or below the melting temperature), and (c) liquid cesium saturated with cadmium (Cs/Cd). Cold-worked Zircaloy reference specimens were consistently embrittled in such tests. While some tests were done with unirradiated Cu-barrier tubing, the emphasis was on Zr-lined tubing, where the effects of liner thickness, purity (oxygen content), and grain size were explored. It was found that in the unirradiated condition the resistance of

Zr-lined Zircaloy tubing to simulated PCI is insensitive to either the thickness or the purity of the liner. However, if the grain size of the zirconium liner was allowed to grow to >35 μm, the resistance to PCI was degraded. In tests with unirradiated Cu-barrier Zircaloy tubing it was shown that the PCI-resistance of the Cu-barrier tubing could be degraded if the Cu-barrier tubing is given an anneal sufficient to significantly interdiffuse the copper and the Zircaloy.

Irradiated specimens were obtained mainly from the unfueled plenum regions of test fuel rods (removable segments from segmented fuel rods) which had been irradiated in commercial BWR's and subsequently power ramp tested for PCI in a test reactor. Expanding mandrel tests were done at reactor temperatures (300 to 335°C) in atmospheres containing either I_2 vapor or pure Cd. These tests at burnup levels up to ~10 MWd/kg-U (fluences up to 2.4×10^{21} n/cm^2, E>1 MeV) showed that when the copper barrier had been diffusion-bonded, the Cu-barrier tubing lost much of its PCI resistance. Irradiated Zircaloy tubing having either an unbonded copper barrier or a zirconium liner showed good resistance to simulated PCI at these fluence levels.

2.2.1.2 Barrier Characterization and Stability

Although the early experiments were done by General Electric on electroplated Cu-barrier tubing, the Lead Test Assemblies (LTA's) (see Part 3) were prepared by an electroless copper plating method. The electroless technique was thought to have advantages both in ease of fabrication and in product uniformity. Much of the effort in this subtask was devoted to chemical and metallurgical characterization of tne electroless copper barrier. The Zr-lined Zircaloy tubing was characterized with respect to crystallographic texture.

The electroless copper barrier was found to have very fine grain size (mean intercept distance <0.5 μm). Chemical analyses by means of ion microprobe showed the major impurities of the electroless product to be hydrogen and carbon. The crystallographic texture of the zirconium liner and the Zircaloy tube in which it was bonded was not markedly different from that of standard Zircaloy tubing.

The Cu-barrier tubing received further attention with respect to its stability, especially its behavior in the presence of steam-hydrogen mixtures as might occur in a fuel rod with a cladding penetration. It was found in laboratory experiments that the presence of the copper barrier promoted the absorption of hydrogen by the Zircaloy in an environment of steam and hydrogen. The use of copper plated on oxidized Zircaloy (a thin layer of zirconia separating the copper and the Zircaloy) tended to retard the absorption of hydrogen, but did not prevent it entirely.

2.2.1.3 Effects of Irradiation on Zirconium of Various Purities

Irradiated specimens of zirconium (flat stock) of different purity levels were tested for resistance to stress corrosion cracking (or embrittlement) in environments of pure cadmium or liquid cesium saturated with cadmium (Cs/Cd). The intention was to determine the influence of purity on the resistance of irradiated zirconium to stress corrosion and thereby to PCI. With sheet specimens the strain rate can be directly controlled and varied by orders of magnitude. Results showed that both crystal bar zirconium and sponge zirconium retain a high degree of resistance to embrittlement in these environments at strain rates ~0.01 min^{-1}, but at much higher strain rates (~0.1 min^{-1}) some embrittlement was seen.

2.2.2 Licensing Tests

2.2.2.1 Simulated Loss-of-Coolant Accident (LOCA)

Laboratory experiments were done to compare the behavior of barrier fuel with that of reference fuel under simulated LOCA conditions. These experiments involved the use of cladding with urania dummy fuel pellets; the cladding was subjected to a pressure/temperature transient expected in a postulated LOCA situation. In terms of overall cladding deformation and tendency for perforations to occur, the behavior of barrier cladding (both Cu-barrier and Zr-lined) was not markedly different from that of reference Zircaloy cladding. Under the extreme temperatures involved with a LOCA the copper barrier and the Zircaloy interacted in a eutectic reaction as had been expected.

2.2.2 Reactivity Initiated Accident

Reactivity Initiated Accident (RIA) comparisons between barrier fuel and reference fuel are to be done at the Nuclear Safety Research Reactor (NSRR) in Tokai, Japan. Barrier and reference cladding samples were supplied to Battelle Pacific Northwest Laboratory where special fuel pins were fabricated and shipped to Japan. Tests are scheduled to occur in 1979 and will be reported in PHASE 2.

2.2.3 Fuel Irradiation Tests

This task addresses directly the question of PCI resistance. It involves the irradiation of experimental fuel rods and subjects them to power ramp tests in a test reactor especially equipped for such tests. In PHASE 1 such tests were done at burnup levels ranging to ~16 MWd/kg-U under power history and power ramp conditions that consistently produce cladding perforations by PCI in conventional fuel (i.e., fuel with nonbarrier reference Zircaloy cladding) of equivalent burnup.

At burnup levels up to 10 MWd/kg-U both the Cu-barrier fuel and the Zr-liner fuel demonstrated superior resistance to PCI. Not only did these barrier fuel rods remain sound after a ramp to high powers, but careful post-test nondestructive and destructive examinations revealed no incipient cracks of the barrier cladding. Tests at higher burnup showed that while all of the barrier configurations appeared to be superior to conventional fuel, cladding failures did occur in fuel with certain copper barrier configurations. Copper barrier fuel with 5 μm thick copper failed as did the diffusion-bonded Cu-barrier fuel. Copper barrier cladding with 10 μm thick copper which had not been diffusion bonded during fabrication continued to resist failure by PCI at a burnup of 12.5 MWd/kg-U (rod average). In tests thus far the Zr-liner fuel has resisted failure by PCI to burnups up to 16.6 MWd/kg-U (rod average) and linear powers as high as 59.1 kW/m (18 kW/ft).

Fuel ramp testing is scheduled to continue in PHASE 2 to support the large-scale demonstration in a commercial power reactor.

2.3 LEAD TEST ASSEMBLIES

The actual demonstration of the Cu-barrier and the Zr-liner fuel concepts starts with the fabrication, irradiation and evaluation of lead test assemblies (LTA's). The LTA's have full sized fuel rods in an 8x8 array and are intended to provide both fabrication experience and lead irradiation experience for the barrier fuels. Thus, the LTA's will start their irradiation before the large-scale demonstration fuel is inserted into a reactor, and the LTA's will lead the large-scale demonstration in burnup. No special power maneuvers are anticipated for the LTA's; their function is to be irradiated under service conditions like that of conventional BWR fuel and to be evaluated for licensing purposes with respect to dimensional stability and ability to perform at relatively high power ratings. In PHASE 1 the LTA's were designed and fabricated. Irradiation was initiated in Cycle 5 of the Quad Cities-1 power reactor beginning in February, 1979. Their continued irradiation and evaluations are scheduled for PHASE 2 and PHASE 3.

Four LTA's have been fabricated, two with Cu-barrier fuel and two with Zr-liner fuel as follows:

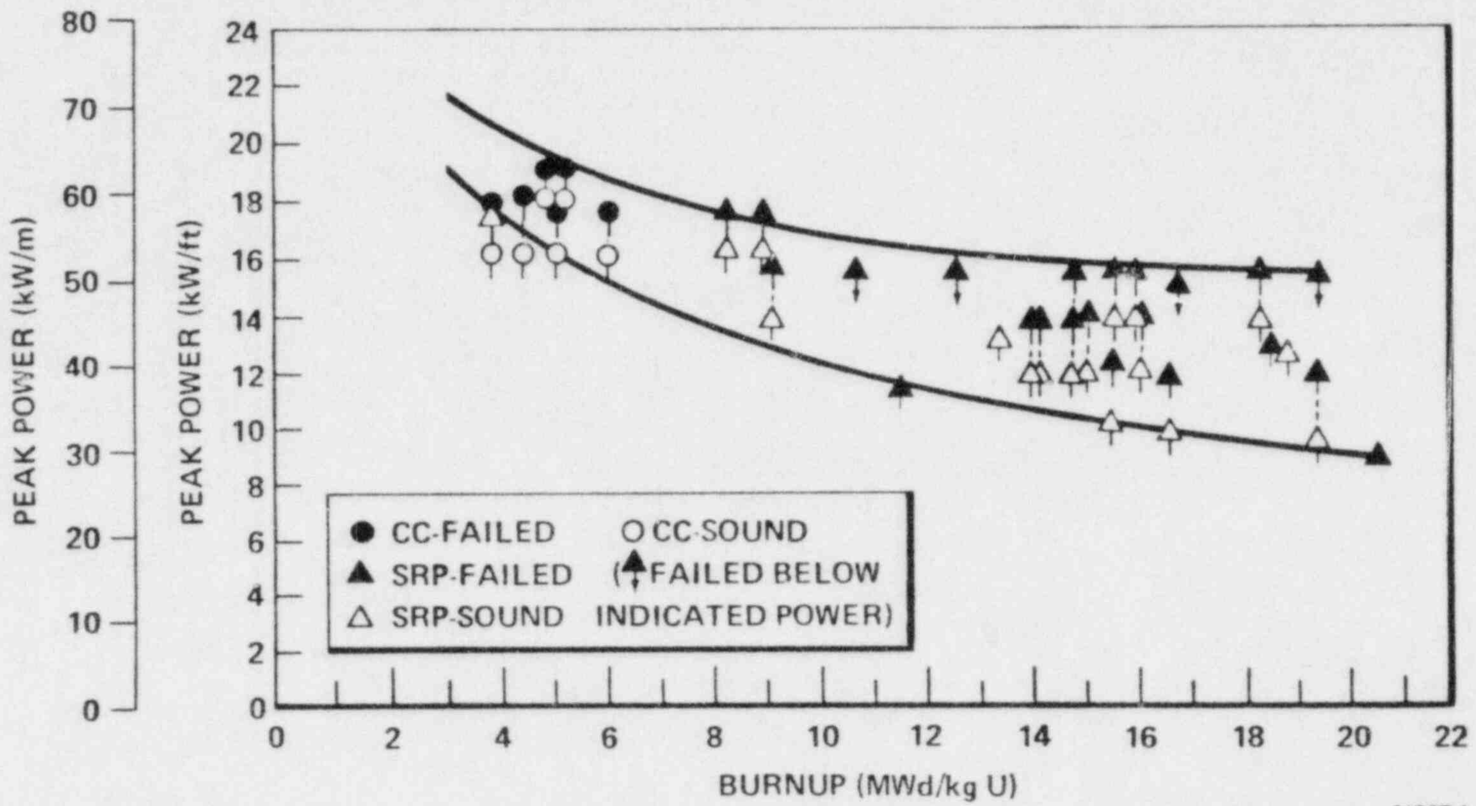
—Copper barrier plated on etched Zircaloy
—Copper barrier plated on autoclave-oxidized Zircaloy
—Zirconium liner (crystal bar zirconium) coreduced with Zircaloy
—Zirconium liner (low oxygen sponge zirconium) coreduced with Zircaloy

Each LTA contains 60 full-length fuel rods, two water rods, and two rods which are segmented. The segmented rods were included as a contingency; they each are composed of four short (~1m long) fuel rod segments which can be readily disassembled for detailed examinations or for fuel power ramp tests in a test reactor should such tests be deemed desirable.

The LTA's were designed for inclusion in symmetric locations in the core of Quad Cities Unit 1, at the start of Cycle 5. Fabrication was completed and the LTA's were delivered to the Quad Cities Nuclear Power Station in December 1978 in time for the scheduled refueling outage (starting January 1979) prior to the start of Cycle 5 (February 27, 1979).

During the fabrication of the LTA's they were thoroughly characterized to facilitate evaluations at subsequent refueling outages of the Quad Cities 1 power plant. Also, preliminary work was done at the Quad Cities Nuclear Power Station to assure the availability of a thorough and complete power history record to aid in the performance evaluations of the LTA's.

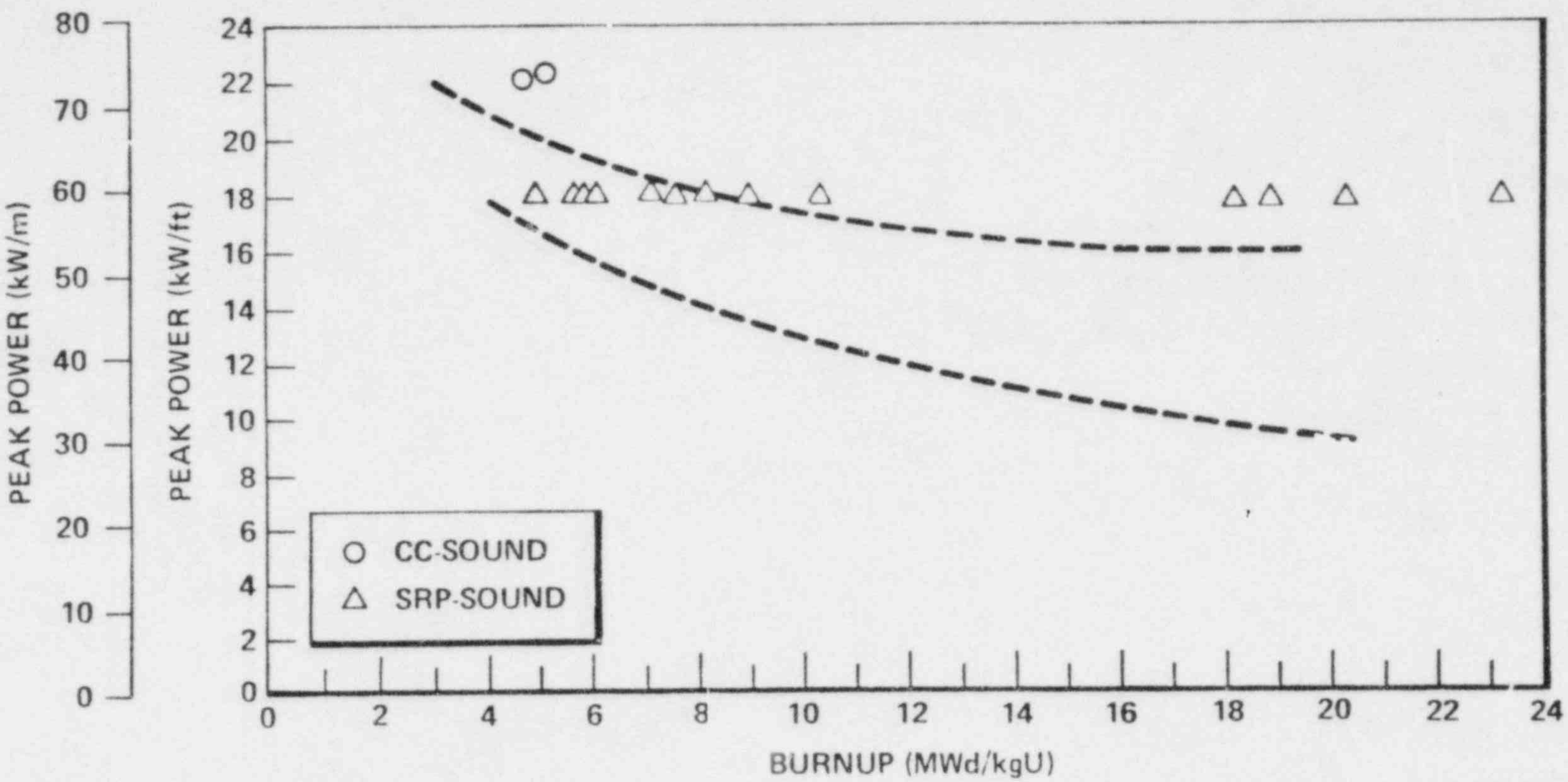
RAMP TEST RESULTS - REFERENCES



01807-1

Attachment A2
A2-7

RAMP TEST RESULTS - Zr LINER



01807-3

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

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

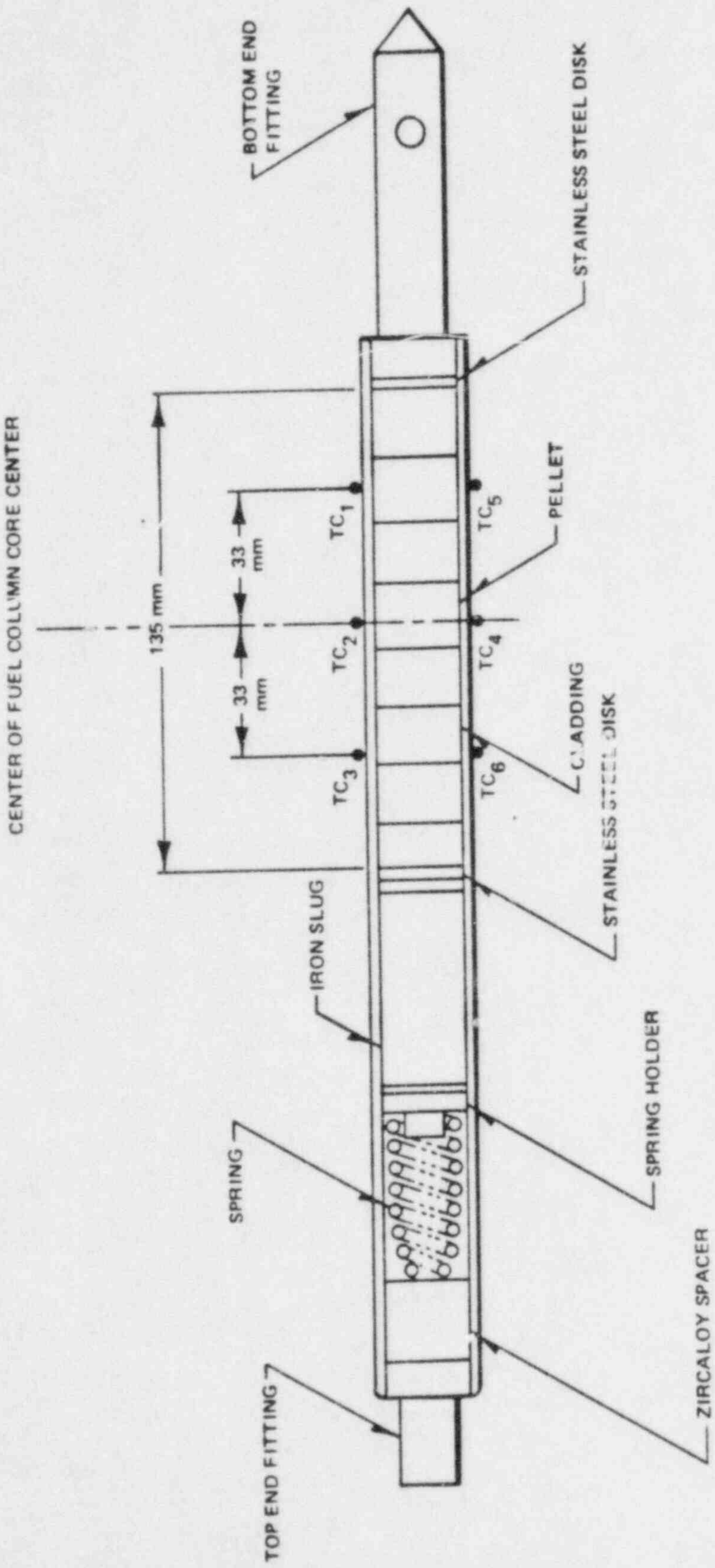


Figure 2.2-1. Nuclear Safety Research Reactor - RIA Fuel Pin

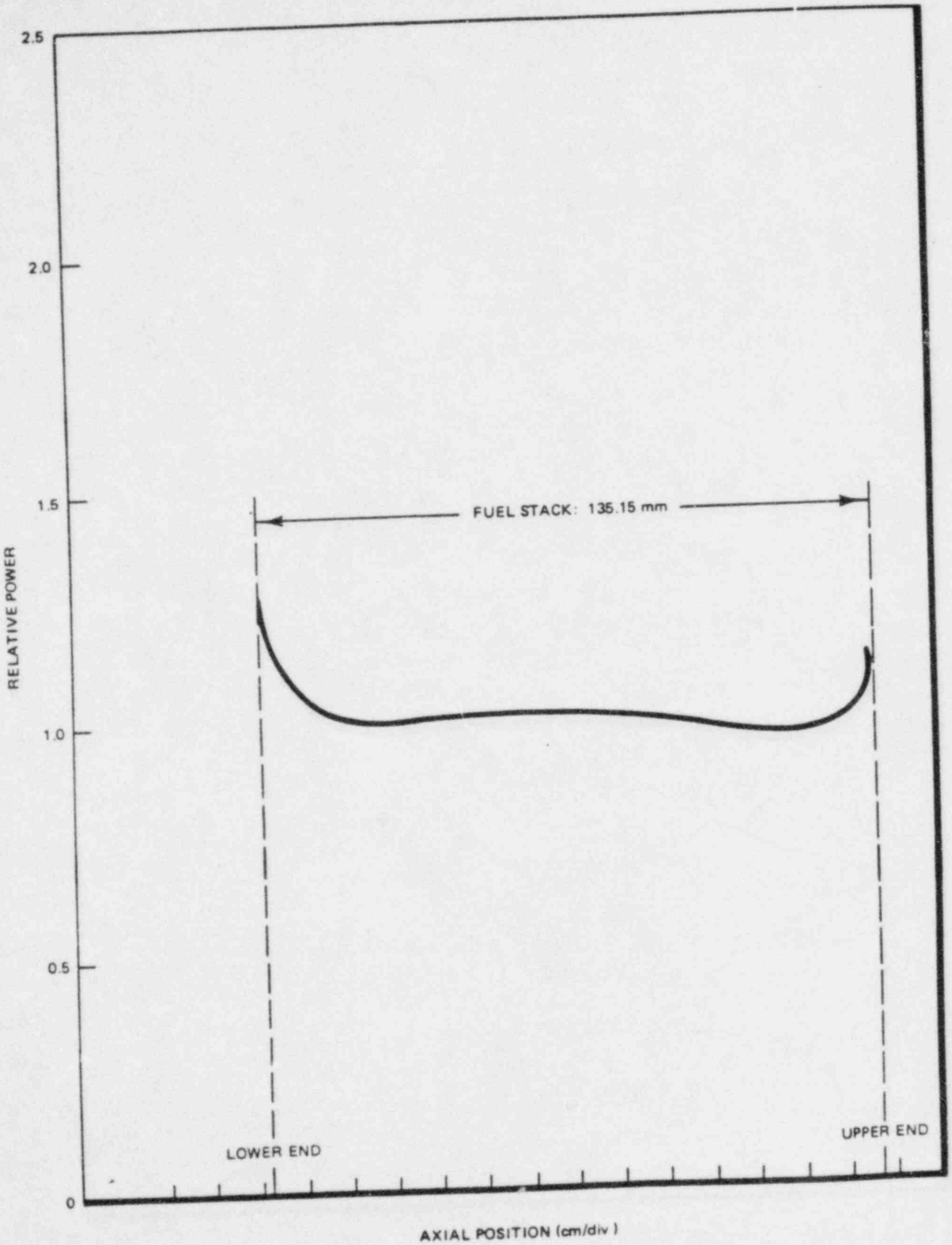


Figure 2.2-2. Typical Axial Power Distribution

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 correctio 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

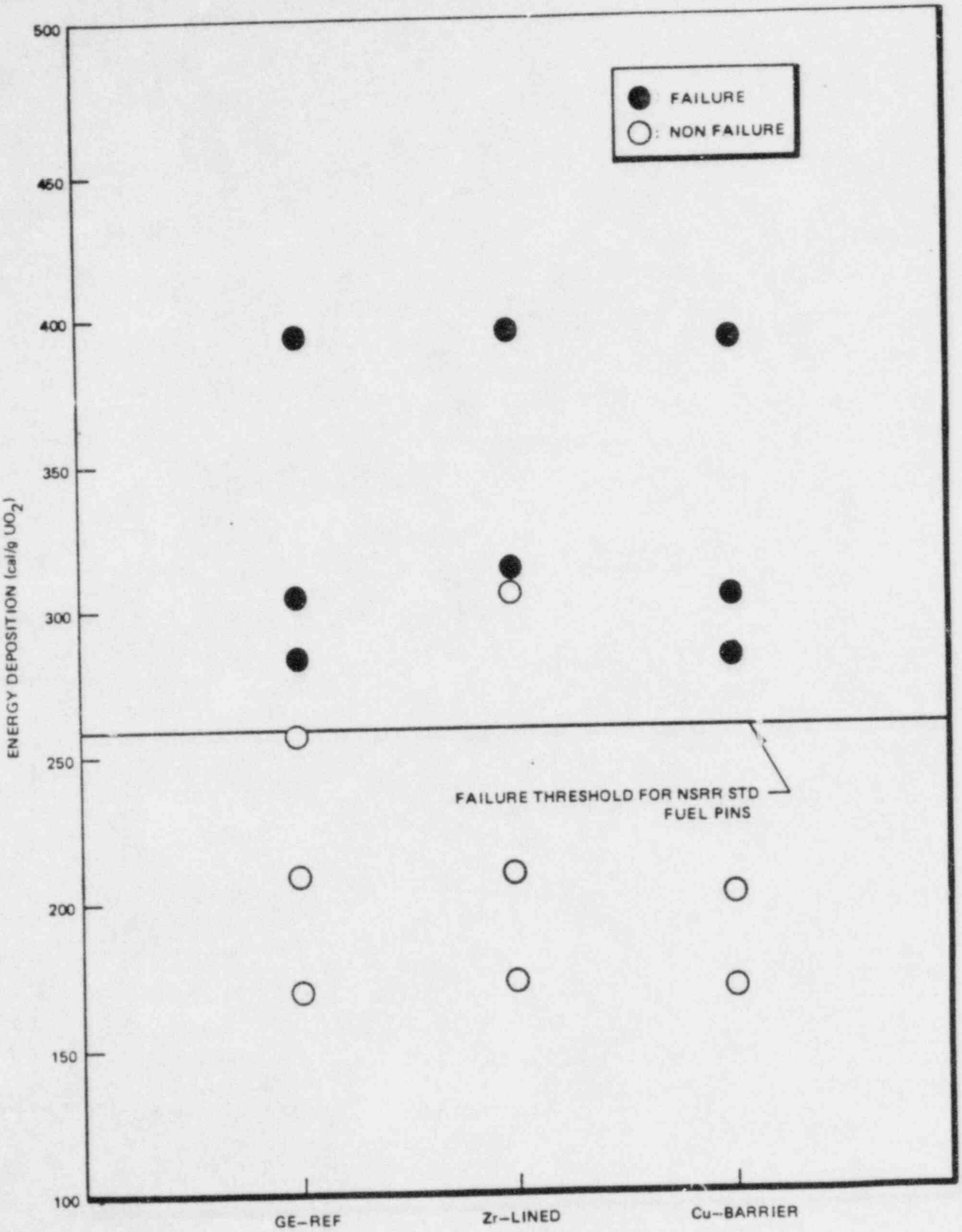


Figure 2.2-3. Comparison of Failure Threshold for Reference and Barr. · Clad Fuel Rods

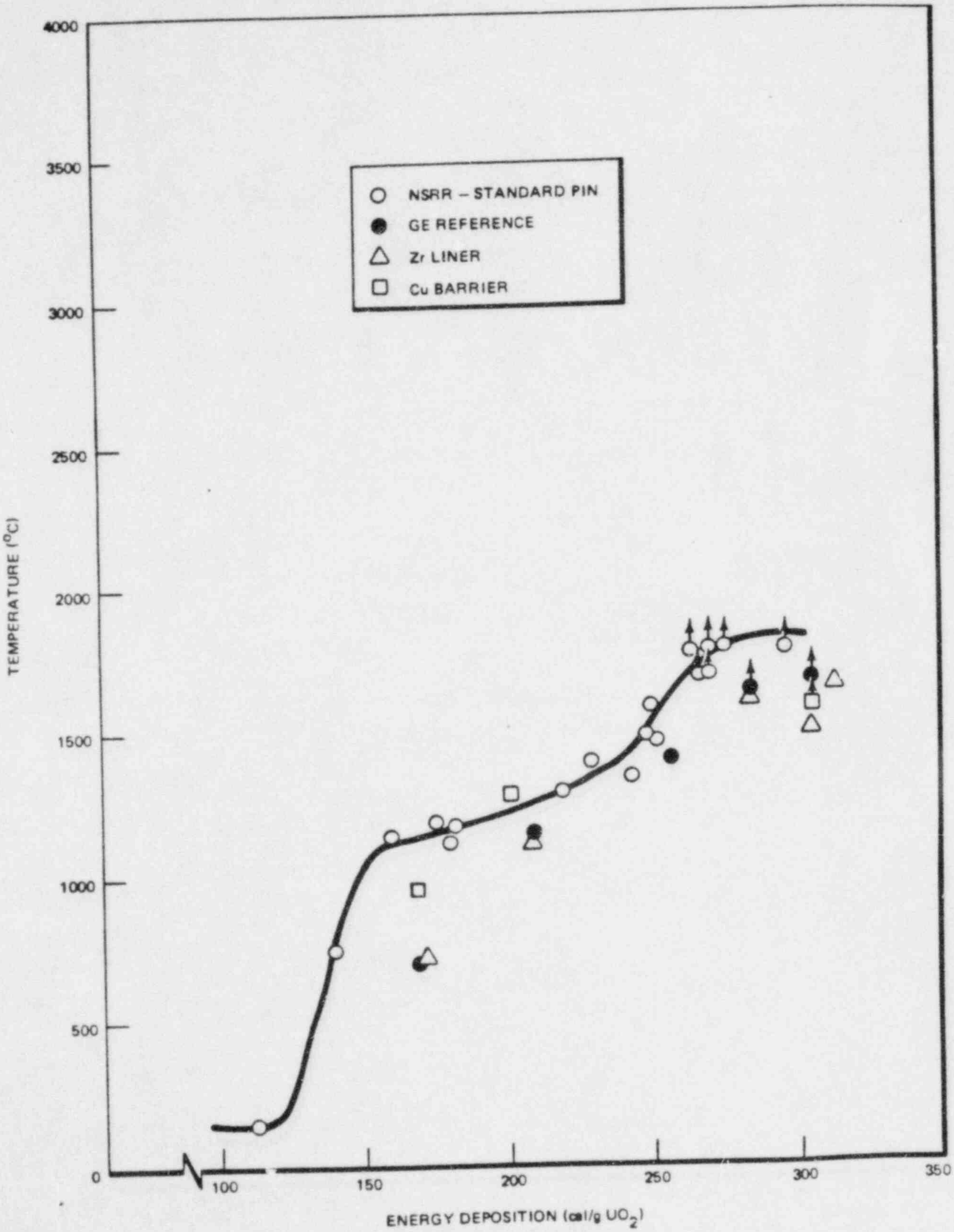


Figure 2.2-4. Maximum Cladding Surface Temperature at Axial Midpoint (Data with arrows are lower-bound estimates)

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, ~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 (Quad Cities-1)	Top	10.7	12.7	April 1979
	Middle Top	15.6	18.6	
	Middle Bottom	15.6	19.4	
	Bottom	13.2	16.7	
	Bundle Average	13.8		
SRP-2 (Monticello)	Top	12.0	16.0	May 1979
	Middle Top	18.4	24.4	
	Middle Bottom	20.8	27.4	
	Bottom	18.5	24.3	
	Bundle Average	17.4		
SRP-3 (Millstone)	Top	11.8	16.1	April 1979
	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.

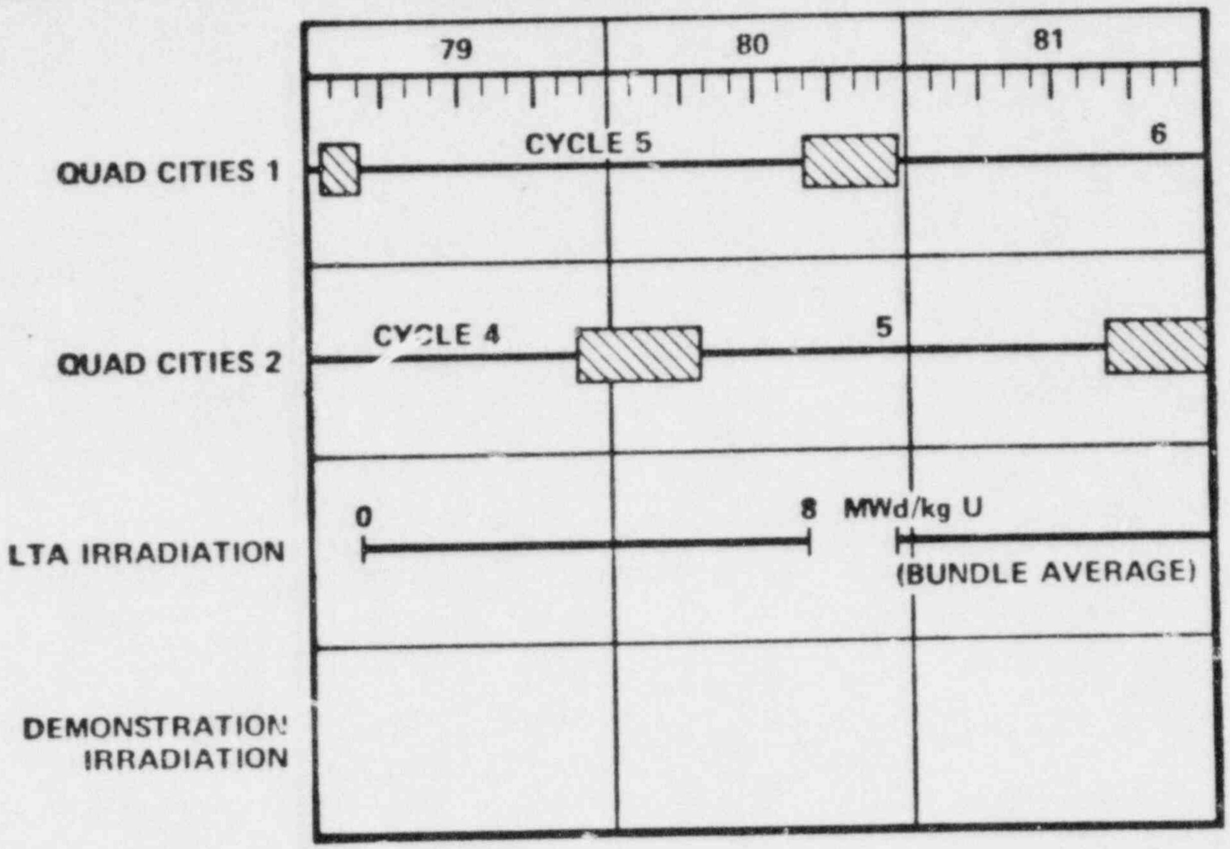
[b]Estimated segment burnups subject to revision following evaluation of ^{137}Cs gamma scan data.

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

PHASE 2 - KEY MILESTONES/DECISIONS

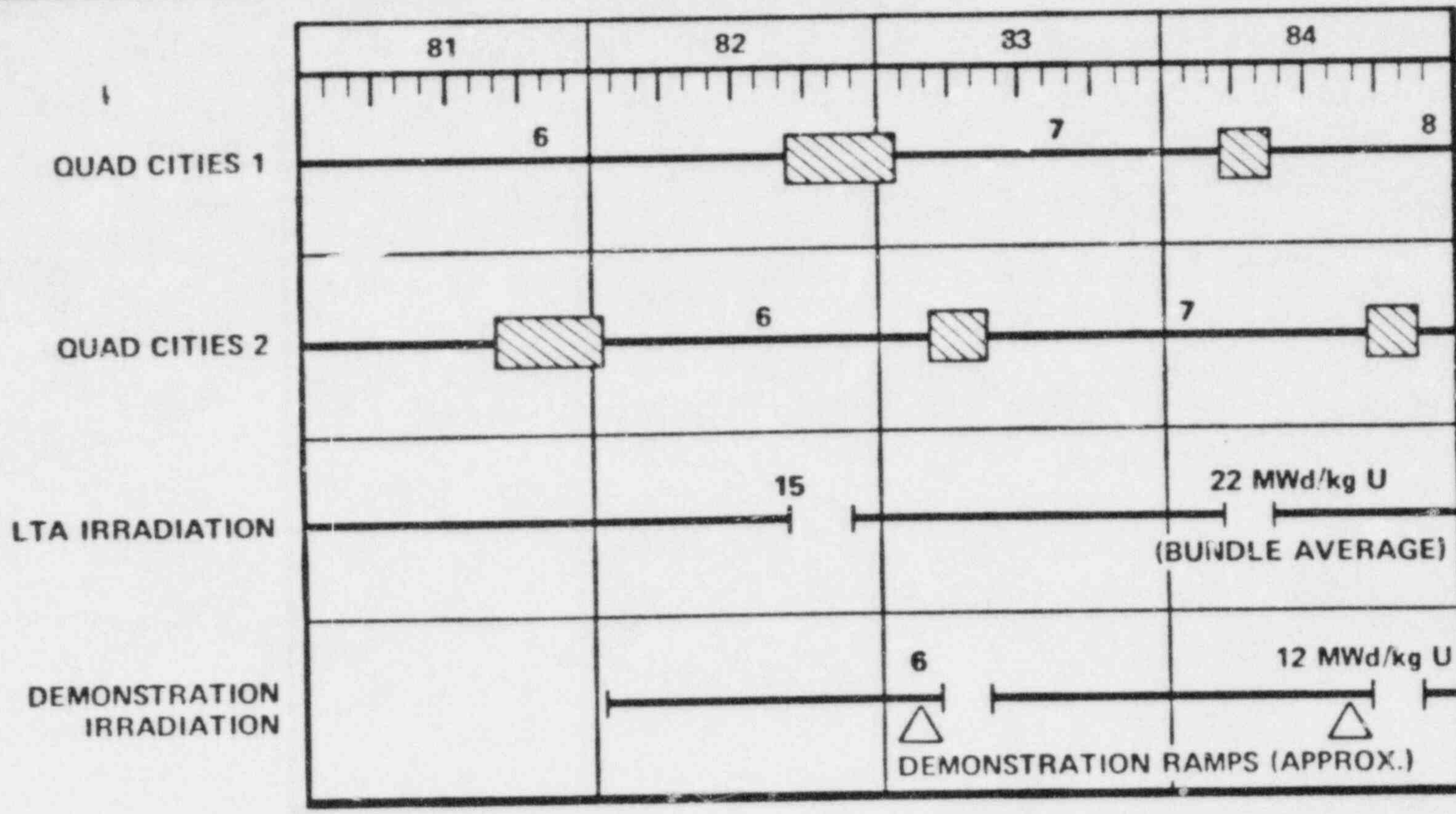
MILESTONE	TARGET DATE	INFORMATION TO SUPPORT DECISION
START IRRADIATION OF LTA'S	FEB 79	- SRP RAMP TESTS TO ~9 GWD/MT
SELECT REMEDY FOR DEMONSTRATION	JAN 80	- SRP RAMP TESTS TO ~15 GWD/MT - ALL PROGRAM DATA (LOCA, ...)
START FABRICATION OF CLADDING	OCT 80	- LICENSING OF BARRIER FUEL BY NRC - LABORATORY DATA ON DESIGN PARAMETERS - DEFECTED FUEL TESTS
START FUEL ROD AND BUNDLE FABRICATION	APR 81	- COMPLETION OF NUCLEAR DESIGN - CORE LICENSE SUBMITTAL - FIRST INTERIM EXAM OF LTA's
START IRRADIATION OF DEMO FUEL	DEC 81	- RAMP TEST DATA AT ~20 GWD/MT
FIRST DEMONSTRATION RAMP	JAN 83	- SECOND INTERIM EXAM OF LTA'S AT ~15 GWD/MT - RAMP TEST DATA AT ~26 GWD/MT
SECOND DEMONSTRATION RAMP	SEP 84	- RAMP TESTS ON SPONGE ZR-LINER AT ~15 GWD/MT - THIRD INTERIM EXAM OF LTA's AT ~22 GWD/MT

SCHEDULE FOR BARRIER FUEL DEMONSTRATION
PHASE 2



00755.5A

SCHEDULE FOR BARRIER FUEL DEMONSTRATION
PHASE 2



00756-58

SCHEDULE FOR GENERIC APPROVAL

NRC SUBMITTAL	MAY 1980
APPROVAL REQUIRED	OCTOBER 1980

PLANT/CYCLE UNIQUE SCHEDULE
BEGINS NOVEMBER 1980

JC:JLK/1340
3/25/80