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**DEMONSTRATION OF FUEL RESISTANT
TO PELLET-CLADDING INTERACTION
PHASE 2 — FOURTH SEMIANNUAL REPORT
JULY - DECEMBER 1980**

COMMONWEALTH RESEARCH CORPORATION
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DEMONSTRATION OF FUEL RESISTANT TO PELLET-CLADDING INTERACTION

PHASE 2

Fourth Semiannual Report, July-December 1980

Compiled by
H. S. Rosenbaum
Program Manager

Approved:

J. S. Armijo, Manager
Core Materials Technology

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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 have been developed for possible demonstration: (a) Cu-barrier fuel and (b) Zr-liner fuel. These advanced fuels (known collectively as "barrier fuels") have special fuel cladding designed to avoid the harmful effects of localized stress and reactive fission products during reactor service. Within the scope of this program one of these concepts had to be selected for a large-scale demonstration in a commercial power reactor. The selection was made to demonstrate Zr-liner fuel and to include bundles which have liners prepared from either low oxygen sponge zirconium or of crystal bar zirconium. The demonstration is intended to include a total of 132 barrier bundles in the reload for Quad Cities Unit 2, Cycle 6.

In the current report period changes in the nuclear design were made to respond to changes in the Energy Utilization Plan for Quad Cities Unit 2. Bundle designs were completed, and were licensed for use in a BWR/3. The core specific licensing will be done as part of the reload license for Quad Cities Unit 2, Cycle 6.

Irradiation testing of barrier fuel was continued, and the superior PCI resistance of Zr-liner fuel was further substantiated in the current report period. The fuel tests included some power ramp tests in which the power was increased very rapidly ($>328 \text{ kW} \cdot \text{m}^{-1} \cdot \text{min}^{-1}$), which is in the range of rates expected when a BWR control blade is moved.

Examination of fuel rods which had been operated at $\sim 35 \text{ kW/m}$ to a burnup of $\sim 3 \text{ MWd/kg U}$ with a deliberately fabricated cladding perforation showed no signs of excessive degradation, although the liner material was oxidized in the vicinity of the perforation.

The four lead test assemblies of barrier fuel (two of Zr-liner and two of Cu-barrier) were examined at the end of one full cycle of operation at a bundle average burnup of 11.7 MWd/kg U . The performance and appearance of these bundles were normal in all respects.

CONTRIBUTING AUTHORS

R. B. Adamson, Nuclear Fuel and Services Engineering Department, (NFSED)
 L. D. Albert, Nuclear Fuel and Services Engineering Department (NFSED)
 D. E. Allen, Wilmington Manufacturing Department, (WMD)
 R. E. Blood, Nuclear Fuel and Services Engineering Department, (NFSED)
 J. H. Davies, Nuclear Fuel and Services Engineering Department, (NFSED)
 M. J. DeVoe, Nuclear Fuel and Services Engineering Department, (NFSED)
 E. L. Esch, Nuclear Fuel and Services Engineering Department, (NFSED)
 C. C. Guitton, Nuclear Fuel and Services Engineering Department, (NFSED)
 L. V. Hampton, Nuclear Fuel and Services Engineering Department, (NFSED)
 G. H. Henderson, Nuclear Fuel and Services Engineering Department, (NFSED)
 E. Hoshi, Nuclear Fuel and Services Engineering Department, (NFSED)
 R. N. Ikemoto, Nuclear Fuel and Services Engineering Department, (NFSED)
 G. R. Lundeen, Nuclear Fuel and Services Engineering Department, (NFSED)
 R. J. Mack, Wilmington Manufacturing Department, (WMD)
 H. M. Nguyen, Nuclear Fuel and Services Engineering Department, (NFSED)
 H. H. Paustian, Nuclear Fuel and Services Engineering Department, (NFSED)
 M. B. Reynolds, Nuclear Fuel and Services Engineering Department, (NFSED)
 H. S. Rosenbaum, Nuclear Fuel and Services Engineering Department, (NFSED)
 E. Rosicky, Nuclear Fuel and Services Engineering Department, (NFSED)
 T. C. Rowland, Nuclear Fuel and Services Engineering Department, (NFSED)
 S. R. Schutz, Nuclear Fuel and Services Engineering Department, (NFSED)
 J. R. Thompson, Nuclear Fuel and Services Engineering Department, (NFSED)
 R. P. Tucker, Nuclear Fuel and Services Engineering Department, (NFSED)
 S. B. Wisner, Nuclear Fuel and Services Engineering Department, (NFSED)

HIGHLIGHTS

- A favorable Safety Evaluation Report was issued by the U.S. Nuclear Regulatory Commission for the demonstration barrier fuel bundles. This is a major step in licensing the large-scale demonstration at Quad Cities Unit 2.
- The four barrier fuel lead test assemblies, which have been under irradiation in Quad Cities Unit 1 since February 1979, were examined after the completion of one full cycle of service at a bundle average burnup of 11.7 MWd/kg U. The two Cu-barrier assemblies were examined visually and by sipping. The two Zr-liner assemblies, one with liners prepared from crystal bar zirconium and one with liners prepared from low oxygen sponge zirconium, were disassembled and selected fuel rods examined individually in addition to the sipping and visual examination of the fuel bundles. Results of profilometry, eddy current, and ultrasonic inspections of these selected Zr-liner fuel rods indicated normal behavior in all respects.
- In severe power ramp tests in a test reactor the barrier fuel continues to prove resistant to PCI. Under power history and ramp conditions which consistently causes failure in reference fuel rods, the Zr-liner fuel successfully sustains the ramp. During this report period a very fast ramp ($>328 \text{ kW} \cdot \text{m}^{-1} \cdot \text{min}^{-1}$) was introduced and successfully implemented as a second ramp on a previously tested rod. This fast ramp rate simulates that in a BWR when a control blade is moved.
- Barrier fuel rods (Zr-liner type) which were exposed for 3 MWd/kg U with a manufactured cladding perforation showed no undue signs of degradation in the post-irradiation examination.
- The nuclear design of the predemonstration has been changed to match the latest energy utilization plan for Quad Cities Unit 2 Cycles 5-8. The changes indicate somewhat higher burnup for the demonstration fuel scheduled to be ramped at the end of Cycles 7 and 8.

1. INTRODUCTION

An important fuel failure mechanism in water-cooled power reactors involves the direct interaction between the irradiated uranium dioxide fuel, including its inventory of fission products, and the Zircaloy fuel sheath. This fuel mechanism is known as "fuel/cladding interaction" or "pellet-cladding interaction" (PCI). 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 identified 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.¹⁻⁶ The present program was started in 1977 with the support and encouragement of the Commonwealth Research Corporation and the U.S. Department of Energy.

This program involves 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.⁴ 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 continues 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 (scheduled for delivery in August 1981).
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 and end-of-cycle ramps (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; details of the scope of PHASE 3 have yet to be defined.

This is the fourth semiannual progress report for PHASE 2; it includes work during July-December 1980.

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

- Fabrication of demonstration bundles of barrier fuel (the target number is 132.).

Subtask III.3 Irradiation and Evaluation

- Monitoring power history.
- Evaluation of performance of the demonstration fuel.

Task IV. Lead Test Assemblies

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

In the previous report period¹ the selection of the fuel design for the demonstration was made. It was decided to demonstrate the Zr-liner fuel and to include both the crystal bar zirconium liner and the low oxygen sponge zirconium liner. That is, the Zr-liner fuel will have liners prepared either from crystal bar zirconium or from low oxygen sponge zirconium. The demonstration is intended to include 132 bundles of barrier (Zr-liner) fuel; but the two grades (purity levels) of zirconium liner, the crystal bar and the low-oxygen sponge, are to be segregated into separate fuel bundles.

The demonstration is scheduled for Quad Cities Unit 2 beginning with Cycle 6.

2. TASK I. NUCLEAR DESIGN AND CORE MANAGEMENT (M. J. DeVoe and H. H. Paustian, NFSED)

2.1 OBJECTIVES

1. Design the demonstration irradiation for a specific reactor and target cycle (Quad Cities-2, Cycle 6) including technical evaluation of the fuel loading pattern and the reactor operational mode to effect both the demonstration and the efficient production of electricity as dictated by the needs of the Commonwealth Edison Company (CECo).
2. Develop and implement a core management plan for the demonstration irradiation.
3. Design the demonstration fuel bundles.
4. Provide the nuclear licensing support required for the demonstration irradiation and the associated reactor operating plan.
5. Work with CECo
 - (a) to define the operational constraints imposed by the reactor owner-operator, and
 - (b) to obtain mutually satisfactory arrangements required to conduct the demonstration irradiation.

2.2 INTRODUCTION

During the past six months, efforts have been concentrated on details of Cycles 6 and 7 at Quad Cities Unit 2, the first and second cycles of the large-scale demonstration of the PCI-resistant barrier fuel concept. Cycle 6 is scheduled for start-up in the fall of 1981 and plans remain to load a target quantity of 132 barrier fuel bundles, including 64 special barrier bundles. Of these, 16 are scheduled to be power-ramped at end-of-cycle 6 (EOC 6) and 48 will be available for power-ramping in Cycles 7, 8, and 9. Also included in the target 132 bundles of barrier fuel are 68 standard barrier bundles which will basically be operated as normal fuel.*

A number of changes have been made in the plans for the fresh fuel bundles to be loaded in Cycle 6 since the last semi-annual report¹ was compiled. These include both changes in the standard non-barrier fuel as well as changes in the special barrier bundles that will be power-ramped in cycles subsequent to Cycle 6. The Energy Utilization Plan (EUP) has also been updated as more accurate estimates of energy to be produced in Cycle 5 and energy required in Cycle 6 have been made. These changes are discussed in more detail in the following sections.

2.3 ENERGY UTILIZATION PLAN (EUP)

Table 2.3-1 shows the current version of the EUP. This differs from that previously reported¹ by increased energies in Cycles 5 and 6. These values are viewed as best estimates and are those on which the final design is based.

* Definitions of demonstration nomenclature:

- *Ramp cells* are comprised of four fuel assemblies surrounding a single cruciform control blade which is planned for programmed withdrawal near the end of a reactor cycle to effect a power ramp on those fuel assemblies.
- *Demonstration fuel bundles or barrier fuel bundles* are any of the 132 bundles of barrier fuel which are planned for use in this demonstration.
- *Special demonstration bundles or special barrier bundles* are any of the 64 barrier bundles which have a special nuclear design to peak the power in the wide-wide corner and which are scheduled to be power ramped in the ramp cells.
- *Standard demonstration bundles or standard barrier bundles* are any of the 68 demonstration bundles having a standard nuclear design (i.e., are not designed to peak the power in the wide-wide corner).

**Table 2.3-1
ENERGY UTILIZATION PLAN**

Cycle	Cycle Energy		(GWd)	Includes Coastdown?
	(MWd/ST)	(MWd/kg U)		
5	6854	7.56	1000.7	yes
6	6166	6.80	886.0	no
7	7314	8.06	1040.0	no
8	6941	7.65	980.0	no
9	6800	7.50	958.0	no

2.4 CHANGES IN FUEL LOADING PLANS FOR CYCLE 6

Changes in the fuel loading plans for Cycle 6 are of two basic types. The first involved changing all the non-barrier bundles of the type* 2.83 $\bar{e}$ -medium Gd to 2.65 $\bar{e}$ -low Gd non-barrier fuel. This change was made to lower the separative work units (SWUs) required and the U_3O_8 ore required for the Cycle 6 reload.

The second major change was in the bundle design anticipated for the EOC 7 power ramp. Prior plans (see Table 2.4-2 of Reference 1) indicated special barrier bundles of the type 2.65 $\bar{e}$ -medium Gd for the EOC 7 power ramp. However, in the most recent EUP (Table 2.3-1) the estimated exposure for Cycles 6 and 7 have been increased. With the new EUP it is now reasonable to consider ramping higher enrichment bundles in Cycle 7. Thus, the possible ramping of 2.97 $\bar{e}$ -high Gd special barrier bundles has been analyzed.

Figure 2.4-1 shows the variation of k_{∞} with exposure for the 2.65 $\bar{e}$ -medium Gd design and the 2.97 $\bar{e}$ -high Gd design in a generic manner. For most nodes of interest in the EOC 7 ramp, current exposures are such that the 2.97 $\bar{e}$ -high Gd design has a higher k_{∞} than would the 2.65 $\bar{e}$ -medium Gd (exposures greater than E_2 in Figure 2.4-1) and would therefore have a higher linear heat generation rate (LHGR) in more fuel rods for a better demonstration.

Several possible ramp cell loadings or configurations are possible that utilize the 2.97 $\bar{e}$ -high Gd bundles in Cycle 7. One possibility is to load four 2.97 $\bar{e}$ -high Gd bundles in each of the four Cycle 7 ramp cells. A second possibility is to load two 2.97 $\bar{e}$ bundles and two 2.65 $\bar{e}$ bundles in each ramp cell. These two ramp cell configurations are the most likely to be used in Cycle 7. Bundles are loaded and incubated in Cycle 6 so as to keep both of these options open. Figure 2.4-2 compares global powers in these two ramp cell configurations during the ramp with the original four 2.65 $\bar{e}$ -medium Gd configuration. Note that because the 2.97 $\bar{e}$ -high Gd bundle design does appear to be a better choice for the EOC 7 ramp, it is possible to eliminate the 2.65 $\bar{e}$ -medium Gd special barrier bundle design and replace it with additional bundles of the 2.65 $\bar{e}$ -low Gd design. This reduces the number of fresh fuel types to be loaded and tracked in Cycle 6 from five to four. The most limiting constraint in loading the Cycle 7 ramp cells is the meeting of a cold shutdown margin of 1% at 1/3 to 1/2 of the way through Cycle 7. This determines how "hot" the buffer region around each ramp cell may be loaded. Also it should be noted that current tentative loading plans for the buffer region around each Cycle 7 ramp cell include loading two of the 2.65 $\bar{e}$ -low Gd special barrier bundles already ramped at EOC 6 in the "back rows" of each Cycle 7 ramp cell as shown in Figure 2.4-3. This has three advantages: (1) It places barrier fuel in the buffer region, thereby reducing the risk to standard fuel; (2) It enables making the buffer region "hotter" to produce higher LHGRs in the EOC 7 ramps; (3) It gives those bundles ramped at EOC 6 a second ramp, although less severe than their first ramp.

Note that because of these same EUP changes which made the 2.97 $\bar{e}$ -high Gd best for EOC 7 ramping, the EOC 9 ramp is becoming less of interest due to the difficulty of obtaining significant LHGRs. Current plans for the EOC 9 ramp maintain the option of "re-ramping" the 2.97 $\bar{e}$ bundles previously ramped at EOC 7.

* $\bar{e}$ indicates the bundle average enrichment in weight % ^{235}U relative to total heavy element content (total uranium). The gadolinium content is indicated "high, medium, or low" Gd, although the gadolinium is added to the fuel as the oxide Gd_2O_3 .

Figure 2.4-1. Neutron Multiplication Factor (k_{∞}) as a Function of Nodal Exposure

CYCLE 7 RAMP CELL
2-2.65 $\bar{e}$ – LOW Gd (L)
2-2.97 $\bar{e}$ – HIGH Gd (H)

(H)	(L)
1.92	1.84
<hr/>	
(L)	(H)
1.85	1.99

cc[‡]

CYCLE 7 RAMP CELL
4-2.97 $\bar{e}$ – HIGH Gd (H)

(H)	(H)
1.99	1.98
<hr/>	
(H)	(H)
1.99	2.04

cc[‡]

CYCLE 7 BASE CASE
4-2.65 $\bar{e}$ – MEDIUM Gd

1.78	1.79
<hr/>	
1.79	1.82

cc[‡]

‡“cc” SHOWS CORNER
OF FOUR-BUNDLE CELL
CLOSEST TO CORE CENTER

*THESE VALUES ARE OBTAINED FROM A HALING ANALYSIS RATHER THAN A DETAILED RAMPING SIMULATION AND ARE THE RATIO OF THE POWER IN THE PEAK-POWER AXIAL NODE OF A GIVEN BUNDLE TO THE AVERAGE NODEL POWER THROUGH THE CORE. THUS, THESE NUMBERS DO NOT INCLUDE LOCAL PEAKING EFFECTS.

Figure 2.4-2. Global Powers in Candidate Cycle 7 Ramp Cell Configurations

Figure 2.4-3. Cycle 7 Ramp Cell and Buffer Region Configuration

Some changes have also been made in the incubation locations of the 48 special barrier bundles not ramped in Cycle 6. Formerly all 48 special barrier bundles were incubated in the second row from the core perimeter. Now, only 32 bundles remain in those locations. The remaining 16 special barrier bundles have been moved to the third row from the core perimeter. This change results in a significant increase in Cycle 6 energy. Half of these 16 bundles are close to the buffer region (see Figure 2.4-4). The remainder of these bundles are 2.65 $\bar{e}$ -low Gd special barrier bundles and are located near the quarter core axes of symmetry.

Figures 2.4-5 and 2.4-6 show hot excess reactivity and cold shutdown margin trends for a loading pattern utilizing the bundle loading plans for Cycle 6 shown in Table 2.4-1, with the exception that 2.65 $\bar{e}$ -medium Gd bundles were not yet replaced by 2.65 $\bar{e}$ -low Gd bundles. Design of a final loading pattern in Cycle 6 utilizing the bundles in Table 2.4-1 is now underway. That loading pattern currently meets the EUP-requested Cycle 6 energy capability and may exceed it somewhat as optimization efforts continue.

Table 2.4-1
BUNDLE LOADING STRATEGY FOR CYCLE 6

Number of Bundles Loaded	Bundle Average Enrichment and Gd Content	
32	2.65 $\bar{e}$ — low Gd*	} barrier
32	2.97 $\bar{e}$ — high Gd*	
68	2.83 $\bar{e}$ — medium Gd‡	
92	2.65 $\bar{e}$ — low Gd	
<u>224</u>		

* Special barrier bundles to be ramped

‡ Remainder of barrier bundles of standard nuclear design

Figure 2.4-4. Location of Special Barrier Bundles in Cycle 6 Loading

Figure 2.4-5. Cycle 6 Hot Excess Reactivity

Figure 2.4-6. Cycle 6 Cold Shutdown Margin

2.5 ACKNOWLEDGMENTS

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C. S. Heller is with the Fuel Projects Department. All of the others named here are with the Nuclear Fuel and Services Engineering Department.

3. TASK II. SUPPORT TESTS

3.1 OBJECTIVES

- 1 Assess the long-term stability of the advanced fuel and develop supporting information needed for licensing
- 2 Develop materials performance data needed to support the demonstration in a commercial power reactor
- 3 Assess Cu-barrier and Zr-liner fuel performance under Loss of Coolant Accident (LOCA) and Reactivity Initiated Accident (RIA) conditions
- 4 Determine the capability of the advanced fuel to resist PCI
- 5 Assess the in-reactor behavior of barrier fuel with cladding perforation
- 6 Develop data needed to justify recommendation of a primary PCI-resistant fuel design for the demonstration and to maintain a viable backup design

3.2 SUBTASK II.1 LABORATORY TESTS

3.2.1 Subtask II.1.1 Characterization and Stability

3.2.1.1 Yield Strength of Crystal Bar Zr-Lined Tubing — (R. E. Blood, M. B. Reynolds and R. B. Adamson, NFSED)

3.2.1.1.1 Introduction

In support of the mechanical design of Zr-liner barrier fuel two tests were conducted in a closed-end pressurized tube configuration to measure the yield strength (0.2% flow stress) of crystal bar zirconium lined tubing. These tests are in addition to and companions to similar tests on reference and sponge zirconium lined tubing described previously.¹

3.2.1.1.2 Results

The average 0.2% yield hoop stress, measured at 367°C for the crystal bar barrier tubing was 31,125 psi (214.7 MPa). The crystal bar liner thickness was approximately 9.4% of the wall thickness. The observed yield strength for crystal bar liner cladding was lower than that measured in companion tests on reference Zircaloy-2 tubing by approximately 4.7% and lower than sponge Zr-lined tubing by approximately 2.3%.

3.2.1.2 Zr-liner Metallurgical Bond (R. P. Tucker, G. R. Lundeen and R. B. Adamson NFSED)

Optical metallography was done to examine the metallurgical bond in Zr-lined Zircaloy-2 tubing. The nature of the bond can be seen in Figure 3-2-1. This is an example of low-oxygen sponge Zr-lined tubing, typical of that in the lead test assembly, which is now under irradiation in Quad Cities Unit 1. (See Section 5.)

(a)

500X

(b)

1500X

Figure 3 2-1 Metallurgical Bond at Zirconium Zircaloy interface in low-oxygen sponge Zr-lined Zircaloy-2

3.2.2 Subtask II.1.2 PCI Simulation Tests

3.2.2.1 Irradiations in the EBR-II — (R. P. Tucker and R. B. Adamson, NFSED)

3.2.2.1.1 Introduction

The series of irradiation experiments of barrier cladding continue in the Experimental Breeder Reactor (EBR-II) to provide high fluence specimens for evaluation in expanding mandrel tests of the resistance to stress corrosion cracking under simulated PCI conditions.

3.2.2.1.2 Results

Irradiation of the third series of rods (EBR-II-3) has been extended to a total of seven runs and currently is scheduled for completion in January 1981. The projected fluences ($E > 1$ MeV) are $\sim 7.3 \times 10^{21}$ n/cm² at core midplane and $\sim 6.6 \times 10^{21}$ n/cm² near the bottom of the core.

Selection of cladding has been made, and capsule rod fabrication has been initiated for the fourth EBR-II experiment. The materials in EBR-II-4 are: (1) reference (bright-etched) Zircaloy-2, (2) 0.076-mm low-oxygen sponge Zr-lined Zircaloy-2, (3) thin (~ 0.05 mm) low-oxygen sponge Zr-lined Zircaloy-2 and (4) 0.076-mm reactor grade sponge Zr-lined Zircaloy-2 (~ 1000 ppm oxygen in liner). The material vendor for items 1-3 is Cezus. A companion series of rods to items 1-3 was in EBR-II-3 with Teledyne Wah Chang Albany as the material vendor. Irradiation of EBR-II-4 is scheduled to begin in May 1981. The projected fluences ($E > 1$ MeV) for four reactor runs are $\sim 4.2 \times 10^{21}$ n/cm² at core midplane and $\sim 3.8 \times 10^{21}$ n/cm² near the bottom of the core.

3.3 SUBTASK II.2 LICENSING TESTS

3.3.1 Loss-of-Coolant-Accident Simulation

3.3.1.1 Metallographic Evaluation of Zr-Lined and Reference Cladding Tested Under Simulated Loss-Of-Coolant-Accident Conditions (H. M. Nguyen and R. B. Adamson, NFSED)

3.3.1.1.1 Introduction

This section presents the results of metallographic analyses of reference and Zr-lined Zircaloy-2 specimens tested under simulated Loss-Of-Coolant-Accident (LOCA) conditions. The metallographic samples were obtained from specimens from the previously reported series of LOCA tests.¹ In those experiments, a segment of cladding tubing containing depleted UO₂ pellets was surrounded by saturated steam at atmospheric pressure and heated electrically at rates postulated for a LOCA event. Fission gas pressure loading on the tube test segment was simulated by pressurizing with argon. Tests were carried out at fixed argon pressures corresponding to hoop stresses of 5.17 and 6.90 MPa. The flows of electrical current for heating and argon gas for pressurization were terminated immediately upon perforation of the cladding specimen.

3.3.1.1.2 Experimental Details

The following materials were tested under simulated LOCA conditions and representative specimens from each type of tubing were examined metallographically:

- 1) Bright reference Zircaloy-2 tubing (unlined).
- 2) Low oxygen sponge Zr-lined tubing.
- 3) Crystal bar, Zr-lined tubing.

Each of these materials was in the fully recrystallized (annealed) condition.

The LOCA test specimens examined are identified in Table 3.3-1.

The tubing specimens became bulged during the LOCA testing. Transverse sections were obtained from each test specimen at two locations. One section was cut to include the maximum of the bulge, and the other was cut in the region of uniform deformation (not in the bulge), see Figure 3.3-1. Photomicrographs were taken to document the thickness of the surface oxide layers, the microstructure, hydrides, and precipitates.

3.3.1.1.3 Results and Discussions

The series of specimens listed in Table 3.3-1 were metallographically and visually examined. Both bulged and uniformly deformed regions were investigated.

A — Microstructures

Zirconium oxide (ZrO_2), oxygen-stabilized α -zirconium and β -quenched Zircaloy structures were found, in sequence, from the outside surface to the inside of the reference and Zr-lined tubes, as can be seen in Figures 3.3-2, 3.3-3 and 3.3-4. The β -transformed structure extended to the inside surface, and the Zr-Zircaloy boundary can no longer be clearly defined in the Zr-lined tubes, see Figures 3.3-5 and 3.3-6.

B — Microcracks and Flaws

Microcracks broke through the outside surface oxide layers and penetrated into the oxygen-stabilized α -zirconium structure. Some open microcracks at the outside surface of the bulged regions are larger than those microcracks in the uniformly deformed regions. There are no microcracks, but only shallow flaws, observed at the inside surface of the specimens, see Figures 3.3-2, 3.3-3, and 3.3-4.

C — Oxide

The tube outside surfaces were heavily oxidized. The oxide thicknesses were measured and are listed in Table 3.3-1. The oxide formed at the tube inside surfaces was so thin that its thickness could not be measured. The outside surface oxide layer of the bulged region is much thicker than that of the uniformly deformed region, on the same tube.

D — Hydrides

Hydrogen content in the β -quenched structure was estimated to be 50 ppm — 150 ppm. The hydrogen content in the bulged region is slightly greater than that in the uniformly deformed region, on the same tube. The hydrogen content in the sponge Zr-lined specimens (50 ppm — 70 ppm) is somewhat lower than that in the reference and crystal bar zirconium lined specimens (120 ppm — 150 ppm).

E — Precipitate Particles

Precipitates particles cannot be seen in the oxygen-stabilized α -zirconium but occur in large numbers in the structure which had been transformed from the β -phase. The precipitate particles tend to be aligned in the α -phase within prior β -grains. (See Figures 3.3-7 and 3.3-8.) These precipitates are expected to be the intermetallic phases common to the Zircaloy system.

3.3.2 Conclusions

New microstructures formed in both reference and Zr-lined Zircaloy-2 during the LOCA tests, namely, from the outside to the inside of the tubes, in order: Zr oxide, oxygen-stabilized α -zirconium and β -transformed Zircaloy. The hydride content has remained at relatively low levels (estimated ≤ 150 ppm hydrogen) in these tests. The oxide layers on the tube outside surfaces were $< 9\mu\text{m}$ thick. Very little evidence of oxidation was observed on the inside surfaces of either lined or reference types of tubing. These LOCA tests indicate comparable performance of Zr-lined and reference Zircaloy-2 under these test conditions.

**Table 3.3-1
LOCA TEST SUMMARY**

Specimens	LOCA Test Temperature (°C)		Hoop Stress MPa (psi)	Failure Time (min)	CFE ^a , %	CUE ^b , %	Outer Surface Oxide Thickness (μm)	
	Center	Surface					Bulged Region	Uniform Region
Zircaloy	922	1008	5.17 (750)	4.75	55	8.3	7.11	6.35
Zircaloy	907	981	6.90 (1000)	4.30	130	5.1	6.35	3.81
Low-Oxygen Sponge	919	967	6.90 (1000)	4.30	124	4.0	5.58	4.06
Low-Oxygen Sponge	984	1006	5.17 (750)	5.30	98	8.0	8.63	6.35
Crystal Bar	971	1049	5.17 (750)	4.60	67	6.5	8.63	6.35
Crystal Bar	909	964	6.90 (1000)	4.00	127	4.7	5.58	4.31

^a Circumferential fracture elongation

^b Circumferential uniform elongation

Figure Deleted

Figure 3.3-1. (Figure Deleted)

Figure 3 3-2 Unlined Zircaloy ZrO_2 (A), Oxygen-Stabilized α -Zirconium (B) and β -Quenched Zircaloy (C), in Sequence from the Outer Surface (O D) to the Inner Surface

(a)

1000X

(b)

1000X

Figure 3 3-3 Sponge Zr-Lined Zircaloy ZrO_2 (A), Oxygen-Stabilized α -Zirconium (B) and β -Transformed Zircaloy (C), in Order, from the Outer Surface (O D.) to the Inside

Figure 3 3-4 Crystal Bar Zr-Lined Zircaloy, ZrO_2 (A), Oxygen-Stabilized α -Zirconium (B) and β -Transformed Zircaloy (C), in Order, from the Outer Surface (O D) to the Inside

(a) LOW OXYGEN SPONGE-LINED ZIRCALLOY-2

(b) CRYSTAL BAR Zr-LINED ZIRCALLOY-2

Figure 3.3-5. Uniformly Deformed Regions. Typical β -Transformed Structure Extended to the Inner Surface (I.D.), and the Zirconium-Zircaloy Interface Is Not Clearly Definable

(a) CRYSTAL BAR Zr-LINED ZIRCALLOY-2 250X

(b) SPONGE Zr-LINED ZIRCALLOY-2 250X

(c) UNLINED ZIRCALLOY-2 (REFERENCE) 250X

Figure 3.3-6 Bulged Regions, Typical β -Transformed Structure Extended to the Inner Surface (I.D.). The Zr-Zircaloy Interface Is Not Clearly Definable

(a) LOW OXYGEN SPONGE Zr-LINED ZIRCALOY 2 MATRIX 1000X

(b) CRYSTAL BAR Zr LINED ZIRCALOY 2 MATRIX 1000X

(c) UNLINED ZIRCALOY 2 (REFERENCE) 1000X

Figure 3 3-7 Uniformly Deformed Regions Precipitates Exist in the β -Transformed Structure

(a) CRYSTAL BAR Zr LINED ZIRCALOY 2 1000X

(b) SPONGE Zr LINED ZIRCALOY 2 1000X

(c) UNLINED ZIRCALOY 2 (REFERENCE) 1000X

Figure 3 3-8 Bulged Regions Precipitates Exist in the β -Transformed Structure

3.3.3 Acknowledgments

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3.4 SUBTASK II.3 FUEL IRRADIATION TESTS

3.4.1 Subtask II.3.1 Segmented Rod Irradiation Tests (J. H. Davies, E. Rosicky L. D. Albert, E. L. Esch, E. Hoshi, S. R. Schutz and T. C. Rowland NFSED)

3.4.1.1 Bundle Status

The irradiation status of the these segmented test rod (STR) assemblies in the segmented rod program (SRP) is updated in Table 3.4-1. During the current reporting period the Quad Cities Unit 1 bundle (SRP-1) and the Millstone bundle (SRP-3) were reconstituted. Twenty segments, including 12 Zr-liner segments, were retrieved from SRP-3 for ramp testing in 1981.

**Table 3.4-1
SRP IRRADIATION STATUS**

Bundle	Segment	Average Exposure (MWd/kg U)	Average Exposure of the Highest Exposure Segment (MWd/kg U)	Date
SRP-1 (Quad Cities-1)	Top	12.6	15.8	December 1, 1980
	Mid-Top	17.9	22.6	
	Mid-Bottom	18.3	23.9	
	Bottom	14.8	19.6	
	Bundle Avg.	15.9		
SRP-2 (Monticello)	Top	14.1	19.0	October 31, 1980
	Mid-Top	21.4	28.8	
	Mid-Bottom	23.6	31.7	
	Bottom	20.0	27.2	
	Bundle Avg.	19.8		
SRP-3 (Millstone)	Top	13.6	18.7	December 1, 1980
	Mid-Top	20.4	29.1	
	Mid-Bottom	22.7	32.2	
	Bottom	19.9	27.3	
	Bundle Avg.	21.8		

3.4.1.2 Power Ramp Tests

3.4.1.2.1 Introduction

Twenty-seven fuel rods were selected for ramp testing in the Studsvik (Sweden) R2 reactor in 1980. The rods are described in Table 3.4-2. As before, ¹⁻⁶ reference rods were defined broadly as those rods with unmodified Zircaloy cladding, although there were differences among the reference rods in conventional design features (cladding wall thickness, diametral gap, helium pressure). The rods were tested in two batches. The first eight tests (see Table 3.4-2) were performed in August and the remainder were performed in November.

Table 3.4-2
DESCRIPTION OF FUEL RODS TESTED IN 1980

Segment Identification	Axial Location In Bundle	Design Feature ^a	Cladding Wall (mm)	Diam Gap (mm)	He Pressure (MPa)	Ave BU ^{bc} (MWd/kg U)
SRP-3/49	Top	Reference	0.71	0.23	1.7	9.7
SRP-3/48	Bottom	Reference	0.71	0.23	1.7	12.2
SRP-2/35	Bottom	Reference	0.71	0.18	2.2	17.1
SRP-2/36	Bottom	Reference	0.71	0.18	0.1	16.4
SRP-3/53	Top	76 μ m Zr-liner (crystal bar)	0.71	0.23	1.7	14.9
SRP-3/50	Bottom	76 μ m Zr-liner (crystal bar)	0.71	0.23	1.7	21.4
SRP-3/51	Middle	5 μ m Cu-barrier (on oxide)	0.86	0.23	0.3	7.8
SRP-3/52	Middle	5 μ m Cu-barrier (on oxide)	0.86	0.23	0.3	5.9
SRP-2/51	Top	Reference	0.86	0.23	0.3	3.4
SRP-2/53	Top	Reference	0.86	0.23	0.1	18.4
SRP-2/55	Top	Reference	0.86	0.18	0.1	15.6
SRP-2/48 ^d	Middle	Reference	0.86	0.23	0.1	25.7
SRP-2/47 ^d	Middle	Reference	0.86	0.23	0.1	26.8
SRP-2/54	Bottom	Reference	0.86	0.18	0.1	20.3
SRP-2/52	Bottom	Reference	0.86	0.23	0.1	24.6
SRP-2/45	Middle	76 μ m Zr-liner (crystal bar)	0.86	0.23	0.3	11.2
SRP-2/43	Bottom	76 μ m Zr-liner (crystal bar)	0.86	0.23	0.3	10.7
SRP-2/49	Middle	76 μ m Zr-liner (crystal bar)	0.86	0.23	0.3	6.3
SRP-2/44	Middle	76 μ m Zr-liner (sponge)	0.86	0.23	0.3	12.7
SRP-2/46	Top	76 μ m Zr-liner (sponge)	0.86	0.23	0.3	6.6
SRP-2/50 ^e	Middle	76 μ m Zr-liner (sponge)	0.86	0.23	0.3	5.8
SRP-2/39	Middle	10 μ m Cu-barrier (on oxide)	0.86	0.23	0.3	12.5
SRP-2/38	Middle	10 μ m Cu-barrier (on oxide)	0.86	0.23	0.3	14.5
SRP-2/41	Bottom	10 μ m Cu-barrier (on oxide)	0.86	0.23	0.3	9.6
SRP-2/42	Top	10 μ m Cu-barrier (on oxide)	0.86	0.23	0.3	6.4
SRP-2/40	Top	10 μ m Cu-barrier (nonbonded)	0.86	0.23	0.1	13.3
SRP-2/37	Bottom	10 μ m Cu-barrier (nonbonded)	0.86	0.23	0.1	22.1

NOTES^a All cladding is Zircaloy 2 (recrystallized) fuel density is 95.5%^b Burnup calculated from ¹³⁷Cs gamma scans using comparison to known standards^c See Reference 1 for approximate axial burnup distributions and peak to average (P/A) ratios for Top, Middle, and Bottom segments^d Shot-peened cladding^e Sponge Zr liner where the sponge had not been selected for low oxygen concentration

The specific objectives of these tests were

1. Define Reference failure threshold for single step (B' Sequence) ramps (Five out of five Reference rods with burnups >10 MWd/kg U, tested in this mode in 1979 failed B' Sequence ramps to 16 kW/ft)
2. Qualify a new, very fast ramp sequence (There is a need to demonstrate that the superior performance of Zr-liner fuel extends to very fast ramp rates, similar to those experienced in a BWR by withdrawing a control blade)

3. Extend definition of Reference failure curve to higher burnups.
4. Test barrier fuel at lead burnups.
5. Test the capability of barrier fuel to withstand the new, very fast ramp sequence.
6. Compare performance of sponge and crystal bar Zr-liners at low to moderate burnups.

3.4.1.2.2 Test Description — C Ramps

As described in the previous progress report,¹ the 1980 Test Specification was considerably revised to meet program objectives. The principal new feature was the addition of a Ramp Sequence C, which is a single step ramp at an average ramp rate $>100 \text{ kW} \cdot \text{ft}^{-1} \cdot \text{min}^{-1}$ ($328 \text{ kW} \cdot \text{m}^{-1} \cdot \text{min}^{-1}$). Ramp testing in R2 is generally performed using a ^3He neutron attenuation device and previous experience had shown that ramp rates were limited by the rate at which ^3He could be purged from the system (i.e., $\sim 6 \text{ kW} \cdot \text{ft}^{-1} \cdot \text{min}^{-1}$ or $19.7 \text{ kW} \cdot \text{m}^{-1} \cdot \text{min}^{-1}$). To meet the Sequence C ramp rate requirement the test is performed with the peak power near the lower end of the fuel column simply by raising the fuel rod a predetermined distance in the ^3He coil (following a successful, i.e., no failure, A or B' ramp sequence). Because the axial flux profile is cosine-shaped, this axial relocation of the fuel column subjects its lower end to a substantial power shock at a ramp rate which reasonably simulates control blade withdrawal ramp rates. (The ramp sequences, A, B' and C are depicted in the previous progress report¹.)

By definition, Sequence C Ramps are second ramps on previously tested rods and can only be performed on rods that have survived a prior A or B' sequence ramp.

To minimize overlapping of the power profile, from the first (A or B') ramp and the second (C) ramp, a new ramp test rig was used. This had more hafnium shielding about the bottom of the fuel column to depress the local neutron flux at Node 9 prior to raising the rod (i.e., performing the second ramp). The general shape of the axial flux profile in the new rig was mapped with flux wires and is compared with that used in prior tests in Figure 3.4-1.

The relative axial location of the active fuel column before and after execution of the C ramp sequence is also shown in Figure 3.4-1. The ramps were executed by raising the rods a standard distance of 200 mms. The power change, ΔP , experienced at the principal test node (Node 9) can be estimated from the 1980 curve, and the average ramp rate is given by ΔP divided by the time taken to effect the axial shift.

The required movement of the rod in executing Ramp Sequence C was performed (see Figure 3.4-2) by rotation of a wheel, located on a platform above the test position, which was linked to the actuating rod of the test assembly by means of a stainless steel wire. On initial insertion of the rod for the primary test, the test assembly was secured in the correct position by a locking mechanism that incorporated parts in the hydraulic drive tube and on the test assembly actuating rod. The locking mechanism also secured the test train in the proper position after the upward movement of the rod was completed. The system was designed to make it impossible to exceed the required span of the test rod's upward movement, as the initial and final positions of the test assembly were fixed by the locking mechanism.

In the first series of tests the actuating wheel (Figure 3.4-2) was operated manually. Later the system was improved by the addition of a motor drive. In each case the rod displacement versus time during the performance of each rapid ramp sequence was recorded on a strip chart recorder located in the R2 control room. The displacement axis input to the recorder was supplied by a potentiometer attached to the test assembly actuating wheel.

An example of a chart recording of a C Ramp Sequence execution is shown in Figure 3.4-3. To meet the specified ramp rate, completion of the 200 mm upward movement was required in a period of 3 to 5 seconds. A review of the forces exerted on the rod, as a result of the upward movement, has shown that these are negligible. The maximum acceleration to which the rod was subjected is estimated to be less than 0.1 times that of gravity. Because the movement was conducted at low displacement rates with the rod in a water medium, there were no significant movement initiation or ending impact forces generated.

Figure 3.4-1. Flux Profiles In Ramp Test Rigs at the R2 Reactor

Figure 3.4-2. Rapid Ramp System

Figure 3.4-3. Displacement versus Time During Execution of C Ramp — Chart Recording

Typical power profiles experienced by fuel rods subjected to A or B' ramps followed by a C ramp are shown in Figures 3.4-4, 3.4-5 and 3.4-6. The burnup distribution along the rods are also plotted. Local power and burnup information can be interpolated from these curves for data analysis and manipulation. Because of the burnup gradients along Top and Bottom segments, it was felt to be more appropriate to correlate the results in terms of local burnup rather than average burnup.

3.4.1.2.3 Results and Discussion

The apparent test results are presented in Table 3.4-3 (pending receipt of the Studsvik final irradiation report). The results are largely consistent with expectation. All of the Zr-liner rods and Cu-on-oxide barrier rods survived, but the two electroplated Cu-barrier rods (SRP-2/37 and SRP-2/40) failed, confirming earlier experience that this PCI remedy loses its effectiveness at higher exposures.

Table 3.4-3
1980 RAMP TEST RESULTS SUMMARY

Rod Identification	Rod Design ^a	First Ramp						Remarks ^d	Second Ramp						Remarks ^d
		Ramp Sequence	Exposure ^b (MWd/kg)	Peak Power (kW/ft) (kW/m)		Time to Failure ^c (h)	Result		Ramp Sequence	Exposure ^b (MWd/kg)	Peak Power (kW/ft) (kW/m)		Time to Failure ^c (h)	Result	
SRP-2/43	Zr-liner (crystal bar)	A	11.3	17.9	58.6	—	Sound		C	8.0	18.6	61.0	—	Sound	
SRP-2/44	Zr-liner (sponge)	A	13.6	18.2	59.6	—	Sound		C	12.9	17.1	56.0	—	Sound	
SRP-2/45	Zr-liner (crystal bar)	A	11.4	18.1	59.2	—	Sound		C	12.0	17.8	58.3	—	Sound	
SRP-2/46	Zr-liner (sponge)	A	7.6	18.0	59.0	—	Sound		C	8.3	17.2	56.3	—	Sound	
SRP-2/49	Zr-liner (crystal bar)	A	6.5	18.0	59.0	—	Sound		C	6.1	15.4	50.6	—	Sound	
SRP-2/50	Zr-liner (sponge)	A	6.3	18.0	59.0	—	Sound		C	6.1	17.5	57.5	—	Sound	
SRP-3/50	Zr-liner (crystal bar)	A	23.3	17.4	57.1	—	Sound		C	16.3	13.9	45.5	—	Sound	
SRP-3/53	Zr-liner (crystal bar)	A	17.2	17.9	58.8	—	Sound		C	18.5	13.2	43.2	—	Sound	
SRP-2/37	Cu-barrier (nonbonded)	B'	23.8	16.0	52.6	3.0	Failed	Power Spike (9 min.)	—	—	—	—	—	—	
SRP-2/38	Cu-barrier (on oxide)	A	15.3	17.9	58.8	—	Sound		C	15.3	17.0	55.8	—	Sound	
SRP-2/39	Cu-barrier (on oxide)	A	13.1	18.0	59.0	—	Sound		C	13.4	17.7	58.0	—	Sound	
SRP-2/40	Cu-barrier (nonbonded)	A	15.6	18.0	59.0	—	Sound		C	16.6	17.0	55.8	2.8	Failed	Power Spike (5 min.)
SRP-2/41	Cu-barrier (on oxide)	A	10.8	18.0	59.1	—	Sound		—	—	—	—	—	—	
SRP-2/42	Cu-barrier (on oxide)	A	7.6	18.2	59.8	—	Sound		C	8.2	18.2	59.6	—	Sound	
SRP-3/51	Cu-barrier (on oxide)	B'	8.1	16.0	52.5	—	Sound		C	8.1	15.5	50.8	—	Sound	
SRP-3/52	Cu-barrier (on oxide)	A	6.1	18.2	59.7	—	Sound		C	6.3	15.8	51.8	—	Sound	
SRP-2/35	Reference	B'	19.0	13.6	44.5	—	Sound		C	12.5	13.6	44.6	—	Sound	
SRP-2/36	Reference	B'	18.5	13.7	44.9	0.6	Failed		—	—	—	—	—	—	
SRP-2/47	Reference	B'	28.4	10.0	32.8	—	Sound		C	27.2	9.8	32.0	—	Sound	
SRP-2/48	Reference	B'	27.1	12.1	39.8	1.5	Failed		—	—	—	—	—	—	
SRP-2/51	Reference	B'	4.0	15.9	52.2	—	Sound		C	4.3	16.3	53.6	—	Sound	
SRP-2/52	Reference	A	26.6	16.1	52.8	0.3	Failed		—	—	—	—	—	—	
SRP-2/53	Reference	B'	21.2	12.0	39.4	3.0	Failed		—	—	—	—	—	—	
SRP-2/54	Reference	A ^e	22.3	12.1	39.6	2.0	Failed		—	—	—	—	—	—	
SRP-2/55	Reference	B'	18.4	13.8	45.2	1.5	Failed	Power Spike (51 min.)	—	—	—	—	—	—	
SRP-3/48	Reference	B'	13.5	14.0	45.9	—	Sound		C	9.2	13.0	42.6	—	Sound	
SRP-3/49	Reference	B'	11.4	16.0	52.5	0.3	Failed		—	—	—	—	—	—	

NOTE:
^a See Table 3.4-2 for Design details.
^b Exposure is the local exposure at the peak power node.
^c Failure times based on loop activity level. Time to failure from end of last ramp step.
^d Power spikes discussed previously.^{2,3} Times in parenthesis is the time from end of the last ramp step to the spike.
^e Truncated A ramp sequence. Maximum peak power held at 12.1 kW/ft (39.6 kW/m). See text.

NODAL POWER/BURNUP PROFILE - SRP 3/50 (GE STR-049, STUDSVIK 1656)

Figure 3.4-4. Maximum Power and Burnup Profiles during Ramp Tests of SRP-3/50 (Bottom Segment)

NODAL POWER/BURNUP PROFILE - SRP 2/45 (GE DTB2408, STUDSVIK 1673)

Figure 3.4-5. Maximum Power and Burnup Profiles during Ramp Tests of SRP-2/45 (Middle Segment)

NODAL POWER/BURNUP PROFILE - SRP 2/40 (GE DTB0136, STUDSVIK 1676)

Figure 3.4-6. Maximum Power and Burnup Profiles during Ramp Tests of SRP-2/40 (Top Segment)

The Cu-on-oxide barrier was conceived to address the reduction in Cu-barrier effectiveness, assuming that it was caused by the formation of a Cu-Zr intermetallic layer; a ZrO_2 film on the cladding prevents Cu-Zr interdiffusion. Results to date have been encouraging, but the Cu-on-oxide barrier fuel has not yet been extensively tested in the exposure range where Cu-barrier defects have occurred.

Most of the Reference rods were used to define the failure threshold for single step (B' Sequence) ramps and four of the eleven rods survived. It was noteworthy that each of these four rods also survived the subsequent C ramp, implying that these very fast ramps ($>100 \text{ kW} \cdot \text{ft}^{-1} \cdot \text{min}^{-1}$ or $>328 \text{ kW} \cdot \text{m}^{-1} \cdot \text{min}^{-1}$) are no more severe than normal B' ramps ($\sim 6 \text{ kW} \cdot \text{ft}^{-1} \cdot \text{min}^{-1}$ or $19.7 \text{ kW} \cdot \text{m}^{-1} \cdot \text{min}^{-1}$). In an attempt to generate a C ramp failure in a Reference rod (SRP-2/54), an A ramp sequence was interrupted at 12.1 kW/ft (39.6 kW/m). Since this rod was a bottom segment during incubation, the fissile isotope distribution was such that following execution of the C ramp operation (raising the rod) without adjusting the pressure in the ^3He coil, the peak power would have been $\sim 14 \text{ kW/ft}$ (45.9 kW/m). However, the rod failed two hours into the four hour hold period prior to the C ramp. This result showed that A ramp sequence tests (with one hour hold periods following each step) may overshoot failure powers. This is taken into account in the data analysis.

The Cu-barrier rod, SRP-2/40, was the only rod in the test series to fail a C ramp. It was examined nondestructively in the hot cell, but it was not possible to positively locate the cladding defect (this must await eddy current testing at VNC). Because of this, the C ramp test is yet to be fully qualified. However, the performance of the Zr-liner rods in surviving these very fast ramp rates is encouraging.

3.4.1.3 Data Analysis

A total of 84 fuel rods have been tested in R2 and GETR and the results have been analysed using a general purpose statistical program with options for censored data analysis. Since many of the tests were stopped prior to failure, they were treated as right censored. But even the failed fuel rods were considered as censored data, because all the tests employed step ramps (albeit small, 2 kW/ft , steps for A ramps) and the failure power may have been anywhere in the interval between the penultimate step and the final step of the ramp sequence. The computer program analyzes interval censored data by placing the failure in about the middle of the interval. This approach is reasonably consistent with actual experience — see, for example, the discussion on Rod No. SRP-2/54 (Section 3.4.1.2.3) and the discussion of power spikes in previous progress reports.^{2,4} Maximum log-likelihood methods are used to fit statistical models to the censored data. The correlating variables, based on our general understanding of the PCI phenomenon were exposure and peak power during the ramp.

The Reference (non-barrier) fuel rods included several variations in design and fabrication parameters: cladding wall thickness; diametral gap; He fill gas pressure; cladding inner surface treatment — autoclaved oxide, grit-blasted, shot-peened. The statistical analysis confirmed an earlier impression that, within the range tested, there was no significant effect of any of these parameters on failure probability. In addition, there was no effect of pre-ramp standardization time (between 6 hours and 300 hours) nor ramp sequence (A or B') on failure probability. However, a strong exposure effect was confirmed.

All the Reference rod data were combined to construct cumulative failure fraction curves, which are compared to literature⁷ correlations in Figure 3.4-7. The agreement between the CANDU and BWR failure correlations is further testimony to the contention that PCI susceptibility is not sensitive to changes in conventional design variables.

Barrier fuels are immune or significantly less susceptible to PCI. Actual barrier fuel experience is superimposed on the Reference rod failure correlations in Figures 3.4-8 and 3.4-9. Since no failures of Zr-liner rods have yet been observed, the design is highly resistant to PCI. Reference rod C ramp results are superimposed on the cumulative failure fraction curves in Figure 3.4-10. Clearly, unless the very fast ramp rates were more severe than A or B' ramps, none of these tests had an even chance of failing. As pointed out in the discussion section (Section 3.4.1.2.3), those rods that would have had a high probability of failing the C ramp test failed the prior A or B' ramp. Barrier fuel C ramp test results are compared with the 50 percent cumulative failure probability curve in Figure 3.4-11, further emphasizing the superior performance of the barriers, particularly Zr.

Figure 3.4-7. Cumulative Failure Fractions of Reference Rods Ramp Tested in GETR and R2 Compared to Literature⁷ Values

Figure 3.4-8. Ramp Test Results — Zr-Liner Rods

Figure 3.4-9. Ramp Test Results — Cu Barrier Rods

Figure 3.4-10. "C" Sequence Ramp Test Results — Reference Rods

Figure 3.4-11. "C" Sequence Ramp Test Results — Barrier Fuel

3.4.1.4 Post Irradiation Examination

Fourteen fuel rods were selected for post irradiation examination (PIE) from among 36 rods tested in the R2 reactor in 1979. Detailed descriptions of all 36 rods, the ramp test results, and the results of neutron radiography and visual examinations performed at Studsvik were reported previously.^{2,1} The general objectives of the PIE were:

1. Increase the data base on Zr-liner fuel. In particular:
 - resolve the cladding strain discrepancy between rods tested in 1977 in the GE Test Reactor (GETR) and those tested² in R2 in 1978.
 - characterize the condition of the liner in the first Zr sponge liner rod tested (SRP-2/34)
 - determine a relationship of between exposure and Zr liner microcracks (see Ref ¹)
2. Characterize failure mechanism of lead burnup Cu-barrier fuel rod (SRP-2/29). Determine extent of Cu-Zr intermetallic layer formation in high exposure, non-failed Cu-barrier rod (SRP-2/28).
3. Characterize failure mode of reference rod that failed at 29.8 kW/m (9.1 kW/ft) during standardization phase of ramp test (SRP-2/26).
4. Establish a PIE data base for Bottom and Top segments (sound and failed).
5. Compare fission gas releases from A sequence and B sequence ramps.

The non-destructive phase of the examination has been completed at Vallecitos. Results are summarized in Table 3.4.4. With respect to the examination objectives, it appears that cladding strain is a strong function of exposure in the power reactor prior to ramp testing. For example, the diameter profile of SRP-3/45 in Figure 3.4-12 may be compared with that of SRP-3/35, reported previously² and reproduced in Figure 3.4-13. Both these Zr-liner rods are of identical design (0.71 mm cladding wall; 0.23 mm gap; 1.7 MPa He pressure) and differ mainly in exposure (Table 3.4-5).

A plot of permanent cladding strain versus burnup for the Zr-liner rods, which have been examined to date, is presented in Figure 3.4-14. All of the rods were ramped to maximum peak powers of ~ 18 kW/ft (~ 59 kW/m). The high strains at higher burnups are believed to reflect exposure-dependent fuel relocation and cladding creepdown effects.

Pulsed eddy current (PEC) indications of rod condition (failed or sound) were consistent with the irradiation test results and neutron radiography. PEC defect signals were recorded on all failed rods and none were recorded on the sound rods. Experience has shown that the PEC method is very reliable and through-wall cladding defects can be located with good precision. In contrast, visual evidence of PCI defects locations is unreliable. True defects are often difficult to distinguish from sundry artifacts and blemishes on the cladding surface. Most PEC defect signals were in a band extending from 216 mm below to 74 mm above the ramp peak power locations. The effect of segment type (Top, Middle, Bottom) on this distribution will be examined.

Two PEC defect signals were recorded near the peak power location on SRP-2/26, the reference rod that failed at 29.8 kW/m (9.1 kW/ft). There was no visual evidence of an unusual failure mechanism (e.g. accelerated corrosion) and the rod profile was essentially identical to the pre-test trace. Metallographic sections have been selected at the defect locations.

Metallography and other destructive examination results will be reported in the next progress report.

Table 3.4.4
1979 RAMP TESTS — NONDESTRUCTIVE PIE RESULTS SUMMARY

Fuel Rod Identification	Axial Location ^a in Bundle	Design Feature	Average Burnup (MWd/kg U)	Maximum Peak Power (kW/m)	Rod Condition	Maximum Cladding Strain % ^b	No. of Cladding Ridges	No. of PEC Through-Wall Indications
SRP-2/12	BOT	Reference	12.4	53.2	Failed	0.16	27	6 ^c
SRP-2/21	TOP	Reference	10.4	53.2	Failed	0.20	11	7 ^c
SRP-2/26	LM	Reference	20.4	29.8	Failed	None	None	2
SRP-2/27	UM	Reference	19.2	39.4	Failed	None	None	1
SRP-2/28	BOT	Cu-Barrier	13.1	58.8	Sound	0.16	21	None
SRP-2/29	LM	Cu-Barrier	15.2	58.0	Failed	0.14	21	3 ^c
SRP-2/32	UM	Cu-Barrier (on oxide)	4.8	59.0	Sound	0.06	None	None
SRP-2/34	UM	Zr-Liner (sponge)	4.8	59.0	Sound	0.06	None	None
SRP-3/20	TOP	Zr-Liner (crystal bar)	5.6	59.4	Sound	0.16	4	None
SRP-3/40	LM	Cu-Barrier	6.3	58.7	Sound	0.04	None	None
SRP-3/44	LM	Zr-Liner (crystal bar)	5.9	59.3	Sound	0.04	None	None
SRP-3/45	UM	Zr-Liner (crystal bar)	5.7	58.5	Sound	0.10	2	None
SRP-3/46	LM	Zr-Liner (crystal bar)	7.4	58.8	Sound	0.08	6	None
SRP-3/47	UM	Zr-Liner (crystal bar)	7.0	59.1	Sound	0.14	3	None

^a Bottom, Lower Middle, Upper Middle, Top

^b Does not include ridging strain

^c Includes possible through-wall signals

Table 3.4-5
IRRADIATION HISTORIES (SRP-3/45 and SRP-3/35)

Rod Identification	Peak Exposure (MWd/kgU)		Standardization (R2)		Ramp Testing		
	Power Reactor	R2	Max. Peak	Hold	Sequence	Max. Peak	Hold
			Power (kW/m)	Time(h)		Power (kW/m)	Time (h)
SRP-3/45	6.2	—	26.2	6	B	58.5	6
SRP-3/35	19.3	0.9	26.3	572	A	58.6	12

Figure 3.4-12. Nondestructive Test Results — Zr-liner Rod SRP-3/45

Figure 3.4-13. Nondestructive Test Results — Zr-liner Rod SRP-3/35

Figure 3.4-14. Effect of Burnup on Cladding Strains Following Ramp Tests of Zr-liner Fuel Rods

3.4.2 Subtask II.3.2 Fuel with Cladding Perforation — (J. H. Davies, L. V. Hampton, E. L. Esch, and S. B. Wisner, NFSED)

3.4.2.1 Introduction

The design and fabrication of an experiment to determine the effect of operating Zr-liner fuel in a defected condition have been described previously.^{2,3} Three Zr-liner fuel rods plus one reference standard rod, each containing one small artificial defect, were operated at ~35 kW/m (10.7 kW/ft) in a loop of the R2 Test Reactor. Irradiation details and initial experience have also been described.² At the previous reporting period, the results of the preliminary post-irradiation examination (PIE) of the fuel rod (non-destructive examination conducted at the R2 reactor site) were highlighted along with activity release rate data.¹ During the current period, destructive examination of the defected rods was started. The results of investigations made thus far are herein presented and discussed.

3.4.2.2 Defected Rod Destructive Examination

Samples have been cut from the four defected segments at the locations indicated in Table 3.4-6. Preliminary evaluation of metallography shows that the drilled defect holes remained open during the test period (~80 days); no oxidation or reactor crud blockage interfered with the coolant ingress. Figures 3.4-15 and 3.4-16 show the post-test condition of the drilled holes for the segments DTC-2105 (Reference cladding) and DTB-2407 (crystal bar Zr-liner).

Table 3.4-6
INITIAL DESTRUCTIVE EXAMINATION SAMPLES — DEFECTED RODS

Segment	Sample Location (cm) ^a	Selection Basis	Sample Type
DTC 2105 (reference, 0.15 mm hole)	10.85-12.62	Defect Hole Condition	Transverse Met.
	21.59-22.86	Peak Power Region	Burnup and Retained Fission Gas
	22.86-24.64	Peak Power Region	Transverse Met.
	24.64-25.27	Peak Power Region	Cladding Hydrogen
	35.56-37.34	Low Power Region	Transverse Met.
DTC 2602 (crystal bar, 0.15 mm hole)	21.59-22.86	Peak Power Region	Burnup and Retained Fission Gas
	22.86-24.64	Peak Power Region	Transverse Met.
	24.64-25.27	Peak Power Region	Cladding Hydrogen
	29.46-31.50	Compare-Cladding Hydride at Mid-Pellet and Pellet Interface Locations	Longitudinal Met.
DTB 2407 (crystal bar, 0.25 mm hole)	10.85-12.62	Defect Hole Condition	Transverse Met.
	35.56-37.34	Low Power Region	Transverse Met.
DTB 2513 (sponge, 0.15 mm hole)	22.86-24.64	Peak Power Region	Transverse Met.
	29.46-31.50	Compare Cladding Hydride at Mid-Pellet and Pellet Interface Locations	Longitudinal Met.

^a Referenced from bottom end plug shoulder

C756-08

AS-POLISHED

50X

Figure 3 4-15 Drilled (0 006-Inch Diameter) Hole-Reference Segment DTC-2105

C761-02

AS POLISHED

50X

Figure 3 4-16 Drilled (0 010-Inch Diameter) Hole-Crystal Bar Liner Segment DTB-2407

A comparison of examination results at the peak power locations of DTC-2105 (Reference), DTC-2602 (crystal bar Zr-liner) and DTB-2513 (sponge Zr-liner) furnishes a representative description of the relative response of the three configurations.

1. **DTC-2105 (Reference):** A uniform 0.010 mm layer of oxide is observed on the inner cladding surface as shown in Figure 3.4-17. The cladding hydride content is shown in Figure 3.4-18. Distribution is relatively uniform across the section and the hydrogen content can be visually estimated to be 25-30 ppm.

Figure 3.4-17. Cladding Oxide Film-Reference Segment DTC-2105 at Peak Power Location

Figure 3.4-18. Cladding Hydride-Reference Segment DTC-2105 at Peak Power Location

2. **DTC-2602 (crystal bar Zr-liner):** The Zr-liner has been partially or totally converted to oxide on about 75 percent of the cladding circumference. Average oxide layer thickness in regions where total conversion has occurred is approximately 0.10 mm. Figure 3.4-19 illustrates oxidation characteristics in a region where varying degrees of oxidation have occurred. There is no appreciable oxidation of the underlying Zircaloy cladding at any point on the sample circumference. The distribution of Zr hydride in the Zircaloy cladding is shown in Figure 3.4-20. The cladding radial thermal gradient has caused most of the hydrogen generated by liner oxidation to migrate to the cladding outer edge where the local hydrogen concentration is estimated to be ~1000 ppm. This outer "rim" of heavily hydrided zircaloy is ~0.08 mm wide (<10% of the original 0.86 mm cladding/liner wall thickness).
3. **DTB-2513 (sponge Zr-liner):** Approximately 50 percent of the liner circumference has been converted to oxide "nodules", as illustrated in Figure 3.4-21. Average oxide thickness at locations of complete liner oxidation is approximately 0.08 mm, as compared to the average thickness of 0.10 mm observed on the crystal bar Zr sample. The oxide resulting from oxidation of the low-oxygen sponge Zr appears to be denser than that of the crystal bar Zr. No oxidation of the Zircaloy cladding can be observed on this sample. Cladding hydride distribution is shown in Figure 3.4-22. Some radial migration effects are observed but there is no "rim" of heavy hydriding like that seen on the crystal bar liner sample. The hydrogen concentration was estimated to be about 100 ppm.

C759-02

AS-POLISHED

50X

Figure 3.4-19. Liner Oxidation-Crystal Bar Liner Segment DTC-2602 at Peak Power Location

C759-16

ETCHED

100X

Figure 3.4-20. Cladding Hydride Distribution-Crystal Bar Liner Segment DTC-2602 at Peak Power Location

Figure 3 4-21 Liner Oxidation-Sponge Liner Segment DTB-2513 at Peak Power Location

Figure 3 4-22 Cladding Hydride Distribution-Sponge Liner Segment DTB-2513 at Peak Power Location

It appears from this study that under these test conditions the low oxygen sponge Zr-liner is more resistant to oxidation than is the crystal bar Zr. However, both crystal bar liner rods were exposed to conditions which may be deemed more severe than those of the sponge liner rod. That is, one rod was exposed at a somewhat higher power (DTC-2602) and the other had a larger defect (DTB-2407).

Knoop microhardness measurements were made at the cladding inner radius, mid-radius and outer radius of all three (peak power) samples. Results are presented in Table 3.4-7. Both the reference and crystal bar samples appear to be harder near the outer surface than near the inner surface, although the disparity is much greater on the crystal bar sample. The sponge liner sample appears to have a higher hardness near the inner surface. The relationship of these results to the estimates of the local hydride concentrations is good for the crystal bar sample, but not as good for the reference and sponge liner samples. It is possible that these hardness variations involve variables other than local hydride concentrations.

Table 3.4-7
MICROHARDNESS MEASUREMENTS — PEAK POWER SAMPLES

Cladding Remedy	Segment Sample Number	Radial Location	Estimated Hydrogen (ppm)	Knoop Number (100 g)
Reference	DTC-2105-E	Inner Surface	25-30	244 ^{+1/-3}
		Middle		256 ^{+2/-0}
		Outer Surface		254 ^{+2/-2}
Crystal Bar Zr-Liner	DTC-2601-C	Inner Surface	<15	244 ^{+1/-1}
		Middle	100	244 ^{+1/-1}
		3/4	150	245 ^{+2/-3}
		Outer Surface	1000	269 ^{+5/-2}
Sponge Zr-Liner	DTB-2513-B	Inner Surface	<15	242 ^{+1/-2}
		Middle	80	243 ^{+4/-2}
		Outer Surface	100	240 ^{+3/-5}

3.4.2.3 Summary

Metallographic results show that whereas the reference segment formed a uniform 0.01mm oxide film on the Zircaloy cladding inner surface, the Zr-liner of the barrier segments have oxide layers in the range 0.08-0.10 mm. In no case did the liner oxidation continue into the underlying Zircaloy cladding.

Initial results also indicate that the sponge Zr-liner may be more corrosion resistant than the crystal bar liner; the circumferential fraction of sponge liner oxidation is less and the sponge liner oxide appears more dense.

Zircaloy hydriding due to inner surface corrosion was very low for the reference sample (~25-30 ppm), somewhat higher (~100 ppm) for the sponge liner sample and substantially higher for the crystal bar liner sample, where there appeared to be a heavy concentration of hydride precipitates at the cladding outer surface.

4. TASK III. DEMONSTRATION

4.1 OBJECTIVES

1. Design and manufacture of PCI-resistant fuel for demonstration of PCI resistance under steady state and under power ramp conditions.
2. Irradiate, test, and evaluate the performance of the demonstration fuel in a commercial BWR power station.

4.2 SUBTASK III.1. DESIGN AND LICENSING OF DEMONSTRATION FUEL (C. C. Guitton, NFSED)

As reported earlier¹ the generic license document for the demonstration fuel bundles was submitted to the U.S. Nuclear Regulatory Commission in May 1980. That document contains the license information for the specific bundle types with Zr-liner fuel and is generic in that approval will allow use of those particular barrier fuel bundles in a reactor of the type BWR/3. In November 1980 a favorable Safety Evaluation Report was issued by the Nuclear Regulatory Commission; thus, the licensing in a generic sense has been completed, with all bundle-specific licensing issues resolved. There remains the non-specific license documentation, which will be included by the utility, Commonwealth Edison Company, as part of its normal reload licensing for Quad Cities Unit 2, Reload 5 (Cycle 6).

4.3 SUBTASK III.2. FABRICATION OF DEMONSTRATION FUEL (D. E. Allen and R. J. Mack, WMD)

4.3.1 Introduction

The fabrication portion of this program is divided into three major parts:

1. Manufacturing Development Run
 - Fabrication parameters are varied and evaluated using small batches of tubeshells.
2. Preproduction Run
 - Having decided on a set of fabrication parameters, a sustained run tried and the resulting tubing evaluated. The preproduction run is intended to confirm the process.
 - Tubing produced in the preproduction is expected to be usable for the demonstration.
3. Production Run
 - Actual fabrication of the demonstration fuel cladding.

Subsequent fabrication of fuel rods and bundles.

In addition to the main work classifications, the work included processing and evaluation of four developmental tubeshells which had been supplied by Compagnie European Zirconium Cezus. The evaluation of those four tubeshells was described previously.²

4.3.2 Manufacturing Development Run

Sixty coextruded zirconium lined tubeshells were obtained for the manufacturing development run (MDR). The suppliers of these tubeshells were Teledyne Wah Chang Albany (TWCA) and Cezus as follows:

- 20 with liners of crystal bar zirconium — from TWCA
- 20 with liners of low oxygen sponge zirconium — from TWCA
- 20 with liners of low oxygen sponge zirconium — from Cezus

The MDR was divided into three stages: developmental studies were done in Stages I and II as described previously.¹ In Stage III approximately 800 tubes were produced from 25 tubeshells to confirm the process parameters defined by the work in Stages I and II. The tube reduction was by a 3-pass pilger process with intermediate anneals at 620°C and a final anneal at 577°C, similar to the process which had been used to produce the tubing for the Zr-liner fuel lead test assemblies.⁵

4.3.3 Preproduction Run

Sixty-five coextruded zirconium lined tubeshells comprised the material for the preproduction run (PPR):

- 25 with liners of crystal bar zirconium — from TWCA
- 20 with liners of low oxygen sponge zirconium — from TWCA
- 20 with liners of low oxygen sponge zirconium — from Cezus

These have been tube reduced using the same fabrication parameters of the MDR Stage III. A total of approximately 2400 tubes were prepared during the PPR; these are now awaiting final inspection.

4.3.4 Microfissures

Evaluation of the MDR Stage III tubing revealed the presence of small longitudinally oriented tears in the zirconium liner at the inner surface of the tubing. Typically, these tears were 10-30 μm deep and several millimeters in length. It was determined that these microfissures could be removed by heavy etching, but dimensional constraints on the finished tubing limited the amount of material removal which could be tolerated.

Prior to the start of the production run, the fabrication process was altered to permit etch removal of microfissures. In a production test run of 12 production tubeshells it was determined that the revised process practically eliminated microfissures in the final tubing, with any microfissures remaining having depth $<12 \mu\text{m}$ and a rounded shape which was judged to be innocuous. Preparations are in progress for tube reduction of the production run.

WIDE-WIDE CORNER GAP

	A	B	C	D	E	F	G	H
1	* 6	* 5	4	4	4	4	4	5
2	5	4	2	Gd	2	2	2	4
3	4	2	2	2	* 1	1	Gd	2
4	4	* Gd	2		H ₂ O	1	1	2
5	4	2	1	H ₂ O	1	1	Gd	2
6	4	2	1	1	1	1	1	2
7	4	2	Gd	1	Gd	1		3
8	5	4	* 2	2	2	2	3	4

ROD TYPES:

- NUMERALS 1 THROUGH 6 INDICATE UO₂ RODS IN ORDER OF DESCENDING ENRICHMENT
- Gd INDICATES RODS WITH UO₂ + Gd₂ O₃
- H₂O INDICATES WATER RODS
- T INDICATES TIE RODS

— SEGMENTED ROD POSITIONS

— FULLY CHARACTERIZED ROD

— SELECTED FOR INTERIM EXAMINATION

Figure 5.1-1. Cross Section of Load Test Assembly Bundle with 2.82% Average Enrichment and 6 Rods with $\bar{e}$ Gd

5.2 INSPECTION RESULTS

All four LTA's were sipped and found to be sound. Three bundles were visually inspected without dissassembly. All four sides of these bundles were examined. Particular attention was given to upper and lower tie plates, finger springs and spacers. These parts were examined for crud, corrosion, and other anomalies. The tie rod nuts were checked to insure they were down on the upper tie plate. The lock tabs were examined to insure they were properly formed and up against the nuts. The expansion springs were checked for proper compression. All the peripheral rods were examined for the proper seating in the lower tie plate, as well as for crud, corrosion, rod-to-rod spacing or any unexpected conditions. The bundles showed no unusual characteristics and the crud deposition was normal. A shiny autoclaved surface was visible beneath a very light coat of crud. The rod spacing was proper and the expansion springs had adequate free space. The upper tie plates were level within acceptable limits. In addition to the visual mechanical inspection, all of the LTA's were gamma scanned at 24 elevations. The results are currently being evaluated to help provide a better definition of power histories and additional data for nuclear design verifications. The bundle which had not been inspected visually (LJB 587) was subsequently dissassembled and the mechanical condition evaluated.

Both LJB 586 and LJB 587 were dissassembled for individual rod examinations. Five rods (A1, B1, B4, E3 and C8) from each bundle were individually visually inspected using the periscope. All the rods showed a light layer of crud with a shiny surface just visible underneath. Some of the interior rods showed a slightly heavier layer of crud.

The inspection included non-destructive testing (NDT) of the individual fuel rods from Fuel Assemblies LJB 586 and LJB 587. The NDT consisted of an ultrasonic (UT) and an eddy current (EC) testing system. The amplitude of the sinusoidal wave of the EC Traces were very small and uniform indicating clean and uniform cladding. The monitoring of the UT output showed no indication of rod failure.

Rod length measurements were also performed on the five fuel rods from each of the two designated fuel assemblies. The results of this measurement were compared to the pre-irradiation length measurement. These data are presented in Table 5.2-1.

Full length helical fuel rod profilometry was also performed on the five characterized rods from bundles LJB 586 and LJB 587. The calibrated outputs were combined electronically and summed. The summed output was amplified and recorded on a strip chart recorder which plotted the helical trace of the fuel rod diameters. Maximum ovalities were tabulated by identifying the largest ovalities on each trace, measuring its maximum amplitude, using the proper scaling factor to determine the magnitude of the ovality and recording the magnitude together with the location on the rod where the ovality occurred. Table 5.2-2 shows the maximum ovality and its axial location for each of the fuel rods.

After completion of the individual rod examinations, all the rods were reinserted into their respective locations within the bundles. All the rods were then checked to assure proper seating into the lower tie plate. All final adjustments were made in accordance with Quality Assurance Procedures. The bundles were then returned to the core.

The results of the interim examinations of the LTA's after the first cycle of operation were very favorable. The LTA bundles were all characterized by typical results for fuel after initial irradiation and were normal in every respect.

The refueling/maintenance outage was completed and startup of Quad Cities 1 Unit began December 20, 1980. The average exposure at the end of December 1980 was 11.8 GWd/kg U.

**Table 5.2-1
FUEL ROD TIP-TO-TIP ROD LENGTH MEASUREMENTS**

Rod Location	Rod Serial No.	Rod Type	Interim	Pre-Irradiation		Length Change	
			Irradiated Measurement (inch)	Measurement (inch)	(mm)	(inch)	(mm)
Bundle Assembly LJB 586 (Crystal Bar Zr-Liner)							
A-1	VA00102	Standard	159.1385	158.949	4,037.305	0.1895	4.8133
B-1	VE00102	Standard	159.1792	158.958	4,037.533	0.2212	5.6185
E-3	ZA00109	Standard	159.1695	158.942	4,037.127	0.2275	5.7785
C-8	ZK00101	Tie	159.6361	159.334	4,047.084	0.3021	7.6733
B-4	ZP00101	Gadolinia	160.0493	159.815	4,059.301	0.2343	5.9512
Bundle Assembly LJB 587 (Low Oxygen Sponge Zr-Liner)							
A-1	VB00101	Standard	159.1267	158.950	4,037.330	0.1767	4.4882
B-1	VG00103	Standard	159.1449	158.951	4,037.355	0.1939	4.9251
E-3	ZB00113	Standard	159.1514	158.951	4,037.355	0.2004	5.0902
C-8	ZL00102	Tie	159.5928	159.330	4,046.982	0.2628	6.6751
B-4	ZS00101	Gadolinia	160.0336	159.808	4,059.123	0.2256	5.7302
					Average	Δ = 0.223	5.664
						±0.075(2σ)	1.905

**Table 5.2-2
FUEL ROD MAXIMUM OVALITY**

Rod Location	Rod Serial No.	Rod Type	Axial Location		Interim Measurement After Irradiation	
			(inch)	(mm)	(mils)	(mm)
Bundle Assembly LJB 586 (Crystal Bar Zr-Liner)						
A-1	VA00102	Standard	78	1,981	0.5	0.0127
B-1	VE00102	Standard	35	889	1.2	0.0305
E-3	ZA00109	Standard	94	2,388	2.0	0.0508
C-8	ZK00101	Tie	Several		1.8	0.0457
B-4	ZP00101	Gadolinia	94	2,388	1.8	0.0457
Bundle Assembly LJB 587 (Low Oxygen Sponge Zr-Liner)						
A-1	VB00101	Standard	135	3,429	1.0	0.0254
B-1	VG00103	Standard	Several		0.4	0.0102
E-3	ZB00113	Standard	29	737	0.7	0.0178
C-8	ZL00102	Tie	7	178	1.6	0.0406
B-4	ZS00101	Gadolinia	7/Several	178/several	2.2/1.6	0.0559/0.0406

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5. TASK IV. LEAD TEST ASSEMBLIES (J. R. Thompson and R. N. Ikemoto, NFSED)

5.1 INTRODUCTION

Four barrier lead test assemblies (LTA's) continued irradiation in Quad Cities Unit 1. The purpose of these assemblies is to precede the large scale demonstration fuel assemblies with significant exposures. Quad Cities Unit 1 was shut down on August 31, 1980 for a planned refueling and maintenance outage. The average exposure of the four bundles at shutdown was 11.7 MWd/kg U after the first full cycle of exposure (EOC 5).

During this outage the four lead test assemblies were scheduled for peripheral visual inspection and gamma scanning. In addition, the two Zr-liner LTA's were to be disassembled to permit detailed examination of five fuel rods from each of these bundles. The bundle serial numbers and barrier types are shown in Table 5.1-1. Five fuel rods from each of the Fuel Assemblies LJB 586 and LJB 587 were to be characterized by profilometry, length measurement, non-destructive testing (NDT) and individual rod visuals. The rod selection basis is shown in Table 5.1-2, and the fuel rod locations are shown in Figure 5.1-1.

**Table 5.1-1
LEAD TEST ASSEMBLY BARRIER TYPES**

Bundle Serial No.	Barrier Type
LJB 586	Zr-liner — Crystal bar Zr
LJB 587	Zr-liner — Low oxygen sponge
LJB 588	Copper barrier — Electroless copper plating on "as etched" Zircaloy-2 tubing
LJB 589	Copper barrier — Electroless copper plating on autoclaved (oxidized) Zircaloy-2 tubing

Table 5.1-2
ROD SELECTION BASIS FOR INTERIM EXAMINATION

Rod Location	Description
A-1	Largest Power Swing (Uncontrolled to Controlled)
B-1	Highest Combined Power Plus Power Swing (Uncontrolled to Controlled)
B-4	Highest Local Power Gadolinia Rod (All Exposures)
E-3	Highest Power Rod (at 25 MWd/kg U)
C-8	Highest Power Tie Rod (Beginning of Life — BOL)