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**DEMONSTRATION OF FUEL RESISTANT
TO PELLET-CLADDING INTERACTION:
PHASE 2, — ^{3.11}THIRD SEMIANNUAL REPORT,
JANUARY-JUNE 1980**

COMMONWEALTH RESEARCH CORPORATION
SUBCONTRACT 3-20-46
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UC-78
September 1980
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**DEMONSTRATION OF FUEL RESISTANT TO
PELLET-CLADDING INTERACTION**

PHASE 2

Third Semiannual Report, January-June 1980

Complied by
H. S. Rosenbaum
Program Manager

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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 are being developed for possible demonstration within this program: (a) Cu-barrier fuel and (b) Zr-liner fuel. These advanced fuels (known collectively as "barrier fuels") have special fuel cladding designed to avoid the harmful effects of localized stress and reactive fission products during reactor service. Within the work scope of this program one of these concepts is to be selected for demonstration in a commercial power reactor.

In the current report period the decision was reached to demonstrate Zr-liner fuel and to include bundles which have liners of either crystal-bar zirconium or of low-oxygen sponge zirconium in approximately equal numbers. The demonstration is intended to include a total of 132 barrier bundles in the reload for Quad Cities Unit 2, Cycle 6.

The nuclear design calls for 64 barrier bundles to be power ramped in special ramp cells whose control blades will remain inserted >62% until near the end of the cycle. The design of these bundles was completed in this report period, as was the design of the 68 barrier bundles which are not to be power ramped in the ramp cells. Work is in progress to match the nuclear design of the demonstration to the energy utilization plan for the Quad Cities Unit 2 in Cycles 6 through 9. The licensing of the barrier bundles was initiated.

Irradiation testing of barrier fuel was continued, and the superior PCI resistance of Zr-liner fuel was further substantiated in the current report period. Furthermore, an irradiation experiment in which Zr-liner fuel, having a deliberately fabricated cladding perforation, was operated at a linear heat generation rate of 35 kW/m to a burnup of ~ 3 MWd/kg U showed no unusual signs of degradation compared with a similarly defected reference fuel rod.

Four lead test assemblies of barrier fuel (two of Zr-liner and two of Cu-barrier) are presently under irradiation in Quad Cities Unit 1, and they have achieved a burnup of approximately 11 MWd/kg U thus far.

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)
 J. A. Baumgartner, Nuclear Fuel and Services Engineering Department, (NFSED)
 R. E. Blood, Nuclear Fuel and Services Engineering Department, (NFSED)
 B. Cheng, Nuclear Fuel and Services Engineering Department, (NFSED)
 J. H. Davies, Nuclear Fuel and Services Engineering Department, (NFSED)
 R. E. Donaghy, Wilmington Manufacturing Department, (WMD)
 E. L. Esch, Nuclear Fuel and Services Engineering Department, (NFSED)
 C. C. Guitton, Nuclear Fuel and Services Engineering Department, (NFSED)
 J. O. Hegland, Nuclear Fuel and Services Engineering Department, (NFSED)
 G. H. Henderson, Nuclear Fuel and Services Engineering Department, (NFSED)
 R. J. Mack, Wilmington Manufacturing Department, (WMD)
 L. D. Noble, Nuclear Fuel and Services Engineering Department (NFSED)
 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. L. 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

- The decision was reached to choose Zr-liner fuel for the large-scale demonstration and to include within the demonstration bundles made with liners of low-oxygen sponge zirconium as well as those with liners of the high purity crystal bar zirconium. The demonstration is intended to involve a total of 132 barrier fuel bundles in the Quad Cities Unit 2 starting with Cycle 6.
- Ramp test data continue to support the superior PCI resistance of Zr-liner fuel.
- Bend tests both with unirradiated Zr-liner tubing material and with irradiated fuel cladding (burnup $\cong 19.0$ MWd/kg U) have shown no tendency for the delamination of the liner from the Zircaloy when bending strains $\geq 9\%$ at the zirconium-Zircaloy interface.
- The irradiation test of deliberately perforated Zr-liner fuel was completed with burnup of approximately 3 MWd/kg U. The release of fission products indicated no unusual degradation of the barrier fuel compared with similarly perforated reference fuel.
- A scheme has been devised for power ramp testing of fuel rods at very high ramp rates $> 328 \text{ kW} \cdot \text{m}^{-1} \cdot \text{min}^{-1}$ to simulate the rates expected when a control blade is withdrawn. The new test scheme not only provides tests under more realistic ramp rate conditions, but also permits two ramp tests to be obtained on a single test fuel rod. Hitherto, only one test could be done on each test fuel rod.
- Licensing of the demonstration fuel bundles was initiated.
- Four lead test assemblies (LTA's) of barrier fuel are under irradiation in the core of Quad Cities Unit 1 since February 1979. These assemblies include two of Cu-barrier fuel and two of Zr-liner fuel. At the end of this report period the average burnup achieved is approximately 11 MWd/kg U.

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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.¹⁻⁵ The current effort was started in 1977 with the support and encouragement of the Commonwealth Research Corporation and the U.S. Department of Energy.

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

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

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

1. A generic nuclear engineering study to show that the demonstration is feasible in a reactor of the BWR/3 type.
2. Laboratory and reactor tests to verify the PCI resistance of the Cu-barrier and the Zr-liner fuel types.
3. Laboratory tests of barrier cladding under simulated loss-of-coolant accident (LOCA) conditions.
4. Design, licensing documentation, fabrication and preirradiation characterization of four lead test assemblies (LTA's) for irradiation in the Quad Cities Nuclear Power Station Unit 1, beginning in Cycle 5. Irradiation of the LTA's was begun in February 1979.

PHASE 2 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 third semiannual progress report for PHASE 2; it includes work during January-June 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 132 demonstration bundles of barrier fuel.

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.

An important development in the current report period was the selection of the fuel design for the demonstration. It was decided to choose Zr-liner fuel as the barrier fuel to be demonstrated and to include cladding whose liner is of two purity levels: low oxygen sponge zirconium and crystal bar zirconium. The selection is discussed as part of Task III.

2. TASK I. NUCLEAR DESIGN AND CORE MANAGEMENT (H. H. Paustian, NFSED)

2.1 OBJECTIVES

1. Design the demonstration irradiation for a specific reactor and target cycle (tentatively 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

Work during the current report period has involved several technical areas. These include completion of the bundle design for the special barrier bundles to be ramped, iterations on batch sizes and loading patterns as the Energy Utilization Plan (EUP) evolved to its current version, reexamination of several safety and transient analyses that were done in the generic studies of PHASE 1 but which were updated to match the conditions used in PHASE 2 studies, trial control rod patterns for Cycle 6, and an evaluation to determine whether nuclear design requirements of the demonstration, as well as requirements applicable to all reload cores, could be met. Ongoing efforts continue in optimizing the ramp cell and buffer region loading patterns to increase peak linear heat generation rates (LHGRs) during the power rampings in the special barrier bundles.*

2.3 SPECIAL BARRIER BUNDLE DESIGN

The purpose of a unique set of bundle nuclear designs for the special barrier bundles that are to be ramped at the end of each of Cycles 6 through 9 is to enhance the peaking in the wide-wide corner fuel pin immediately adjacent to the control blade. This enhanced peaking results in a higher peak linear heat generation rate (LHGR) in that location in each of the special barrier bundles during each pull of the control blade in that ramp cell. In all cases the wide-wide corner pin will have the largest LHGR during the power ramping sequence. This is due to the control blade history (CBH) effect. Stated briefly, the CBH effect results from operation of a fuel bundle with the adjacent control blade inserted. The control blade depresses the neutron flux and hardens the neutron energy spectrum in fuel pins near the blade. This results in a reduced burnup rate and a larger U to Pu conversion ratio in these fuel pins compared to fuel pins located farther away from the control rod. When the

*Definitions of demonstration nomenclature:

- *Ramp cells* are comprised of the four fuel assemblies surrounding a single cruciform control blade which is planned for programmed withdrawal near the end of a fuel 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).

control blade is subsequently withdrawn, because of the lower burnup and higher Pu content, the wide-wide corner pins will operate at higher relative powers than they would if they had been uncontrolled for the entire period. The magnitude of this increased corner pin peaking is dependent upon the amount of burnup that a fuel segment accumulates while controlled. The planned operational strategy for the demonstration uses the CBH effect to advantage by operating with the fuel segments that are to be power ramped at the end-of-cycle in a controlled state through the entire cycle until the power ramp is begun. Thus the CBH effect is used to increase the peak LHGR above what it would be if the fuel segment were power ramped at the same burnup, but without any accumulated control blade history.

Current standard reload lattices were designed with a low-enrichment W-W corner pin to mitigate the CBH effect. For the bundle design for the special barrier bundles to be ramped, however, these low-enrichment corner fuel pins were replaced with higher enrichment pins. Figure 2.3-1 shows the enrichments for the four fuel pins closest to the wide-wide corner for both the standard reload designs and for the special barrier bundles to be ramped. The remaining fuel pins in each bundle design for the special barrier bundles are not changed from the standard reload bundle design. It should be noted that the limiting constraints on the enrichment that can be used for the wide-wide corner pin in the special barrier bundles are 1) the LHGR in the bundle while it is being incubated prior to the power ramp must not exceed the threshold for preconditioning and 2) in the cycle after a bundle has been ramped, wide-wide corner pin peaking must be low enough to enable the bundle to be used normally in the core.

STANDARD BUNDLE

SPECIAL BARRIER BUNDLE

FOR CYCLE 6 – 2.65 AVG BUNDLE ENRICHMENT – LOW Gd_2O_3 LOADING

STANDARD BUNDLE

W-W

1.3	1.7
1.7	2.0

SPECIAL BARRIER BUNDLE

W-W

1.7	1.7
1.7	2.0

FOR CYCLE 7 – 2.65 AVG BUNDLE ENRICHMENT – MEDIUM Gd_2O_3 LOADING

STANDARD BUNDLE

W-W

1.3	1.7
1.7	2.0

SPECIAL BARRIER BUNDLE

W-W

1.7	1.7
1.7	2.0

FOR CYCLES 8 AND 9 – 2.98 AVG BUNDLE ENRICHMENT – HIGH Gd_2O_3 LOADING

STANDARD BUNDLE

W-W

1.5	2.0
2.0	2.4 + Gd

SPECIAL BARRIER BUNDLE

W-W

2.0	1.7
1.7	1.7 + Gd

Figure 2.3-1. Special Barrier Bundle Design

2.4 ENERGY UTILIZATION PLAN (EUP)

Table 2.4-1 shows the current version of the EUP that is being used for PHASE 2 studies (cf. Table 2.3-1 of the PHASE 2—Second Semiannual Report). Table 2.4-2 lists current preliminary reload bundle requirements for the demonstration cycles. Note that these values will probably be revised as optimization studies continue. Table 2.4-2 also shows the cycle exposure capabilities of the loadings listed.

Table 2.4-1
ENERGY UTILIZATION PLAN

Cycle	Cycle Energy		(GWd)	Includes Coastdown
	(MWd/ST)	(MWd/kg U)		
5	6781	7.48	990	Yes
6	5772	6.36	830	No
7	7314	8.06	1040	No
8	6941	7.65	980	No
9*	6800	7.50	958	No

* This cycle energy was not updated in the most recent EUP, so the preliminary value of 6800 Mw/ST has been retained.

Table 2.4-2
DEMONSTRATION LOADING STRATEGY

Cycle	No. of Bundles Loaded	Bundle Average Enrichment and Gd Content	Cycle Energy (GWd)	EUP Requested Cycle Energy (GWd)
6	16	2.66 low Gd ₂ O ₃ *	840.7	830
	16	2.65 medium Gd ₂ O ₃ *		
	32	2.97 high Gd ₂ O ₃ *		
	68	2.83 medium Gd ₂ O ₃ ‡		
	84	2.83 medium Gd ₂ O ₃		
	216			
7	192	2.65 medium Gd ₂ O ₃	1059.0	1040
8	16	2.65 low Gd ₂ O ₃	978.3	980
	(drivers)			
	176	2.65 medium Gd ₂ O ₃		
	192			
9	16	2.65 low Gd ₂ O ₃	967.9	958
	(drivers)			
	184	2.65 medium Gd ₂ O ₃		
	200			

* Special barrier bundles to be ramped.

‡ Remainder of 132 barrier bundles.

2.5 FUEL CYCLE COMPARISON

Tables 2.5-1 and 2.5-2 show the results of a fuel cycle comparison that was done for a previous intermediate version of the EUP. Based on this comparison between loading patterns for Cycles 6-13, one case including the demonstration in Cycles 6-9 and the second, nominal case, not including the demonstration; these two cases appear equivalent within the uncertainties of the comparison. It appears that any apparent fuel cycle penalty associated with the demonstration can be recovered in subsequent cycles. This same conclusion was reached in the generic studies in PHASE 1.

Table 2.5-1
FUEL CYCLE COMPARISON

Cycle	Demonstration and Follow-On			Nominal		
	No. of Bundles Loaded	Bundle* Average Enrich.	Cycle Energy (GWd)	No. of Bundles Loaded	Bundle* Average Enrich.	Cycle Energy (GWd)
6	16	2.66 ^(a)	821.5	232	2.83 ^(b)	841.5
	16	2.65 ^(b)				
	32	2.97 ^(c)				
	168	2.83 ^(b)				
7	168	2.65 ^(b)	971.0	52	2.82 ^(b)	961.7
				120	2.65 ^(a)	
8	184	2.65 ^(b)	977.2	44	2.82 ^(b)	957.4
				128	2.65 ^(a)	
9	16	2.65 ^(a)	948.4	72	2.82 ^(b)	963.6
	184	2.65 ^(b)		120	2.65 ^(a)	
10	72	2.82 ^(b)	954.3	80	2.28 ^(b)	963.3
	120	2.65 ^(a)		120	2.65 ^(a)	
11	64	2.82 ^(b)	967.5	64	2.82 ^(b)	968.0
	120	2.65 ^(a)		120	2.65 ^(b)	
12	68	2.82 ^(b)	955.6	68	2.82 ^(b)	951.3
	120	2.65 ^(a)		120	2.65 ^(a)	
13	68	2.82 ^(b)	969.8	72	2.82 ^(b)	963.1
	120	2.65 ^(a)		120	2.65 ^(a)	
Total	1536		7565.3	1532		7569.9
		$\bar{e} = 2.706$			$\bar{e} = 2.726$	

* Superscripts on bundle enrichments have the following meanings: (1) "a" indicates a "low" Gd₂O₃ loading
(2) "b" indicates a "medium" range Gd₂O₃ loading and (3) "c" indicates a "high" Gd₂O₃ loading.

Table 2.5-2
DISCHARGE DATA
DISCHARGE EXPOSURE AND ENERGY PRODUCED BY DISCHARGE BATCH

Cycle	Demonstration and Follow-On			Nominal		
	Number Discharged	Batch Exposure (MWd/kg U)	Batch Energy (GWd)	Number Discharged	Batch Exposure (MWd/kg U)	Batch Energy (GWd)
6	232	26.716	1157.0	232	26.660	1151.9
7	168	26.741	831.7	172	26.963	853.9
8	184	29.457	988.4	172	29.366	923.6
9	200	28.620	1017.5	192	29.606	1004.5
10	192	28.171	955.3	200	28.953	1022.6
11	184	29.152	947.0	184	28.344	921.4
12	188	28.753	954.5	188	29.674	964.6
13	188	28.319	940.4	192	28.793	976.4
Total	1536		7791.8	1532		7818.9
	$\bar{e} = 2.628\%$			$\bar{e} = 2.647\%$		
	Aver. dis. exposure = 28.214 MWd/kg U			Aver. dis. exposure = 28.506 MWd/kg U		

2.6 POWER RAMP SIMULATION

Table 2.6-1 shows results from power ramp simulations for the loading patterns used to generate the data in Table 2.4-2. Peak LHGRs achieved during the ramping are shown. These values include the effect of CBH and of modifying the wide-wide corner pin enrichments for the special barrier bundles to be ramped. These results are preliminary, and it is anticipated that re-optimization of loading patterns derived for the most recent EUP can increase the values shown in Table 2.6-1, particularly for Cycle 7.

Table 2.6-1
ESTIMATED PEAK LHGRs DURING RAMPS

Cycle	Peak Bundle Location	Peak Axial Fuel Segment	Nodal Exposure (MWd/kg U)	W-W Corner Rod LHGR and ΔP After Blade Movement kW/m (kW/ft)	
				LHGR	ΔP
6	(8, 8)	8	6.0	41.3 (12.6)	34.1 (10.4)
7	(8, 8)	5	8.9*	33.1 (10.1)	27.9 (8.5)
8	(9, 6)	11	20.3	33.5 (10.2)	27.4 (8.4)
9	(10, 5)	11	26.1	27.9 (8.5)	23.0 (7.0)

* Nodes with exposures of ~11-12 MWd/kg U also show W-W corner peak LHGRs during the ramp of approximately 3.05 kW/m (10.0 kW/ft).

2.7 CYCLE 6 THERMAL LIMITS CALCULATIONS

As part of the PHASE 2 work, a trial set of control rod patterns was developed for Cycle 6. The primary purpose for developing these rod patterns was to reaffirm that the loading patterns and special bundle designs required for the demonstration did not adversely impact normal operation of the core. Trial rod patterns were also developed in PHASE 1 and indicated no adverse impact, but it was necessary to redevelop them in PHASE 2 because of substantial changes in cycle energy capability, in detailed loading patterns, and in the bundle design of the barrier bundles.

Figures 2.7-1 through 2.7-3 show calculated values of maximum linear heat generation rate (MLHGR), minimum critical power ratio (MCPR), and maximum ratioed average planar linear heat generation rate (MRAPLHGR) through Cycle 6. Note that probably the actual operating rod patterns in Cycle 6 will differ somewhat from those used here. However, these results are expected to be representative. These rod patterns included deep rod-shallow rod exchanges at cycle exposures of 3.3 MWd/kg and 5.2 MWd/kg. The purpose of these exchanges is to keep effects of control blade history in the control cells within easily manageable bounds such that control rods can be fully withdrawn at the end of the cycle.

Figure 2.7-1. Maximum Linear Heat Generation Rate (MLHGR) Cycle 6 Exposure

Figure 2.7-2. Minimum Critical Power Ratio (MCPR) Cycle 6 Exposure

Figure 2.7-3. Maximum Ratioed Average Planar Heat Generation Rate (MRAPLHGR) Cycle 6 Exposure

2.8 SAFETY AND TRANSIENT ANALYSES

Also as part of PHASE 2, a number of safety and transient analyses done in PHASE 1 were reexamined under PHASE 2 conditions to determine that results were still applicable. Emphasis was placed on the analyses for Cycle 6, because it is both the earliest cycle as well as the cycle that is most constrained in bundle types committed to being loaded. The basic conclusions drawn from this reexamination were the following: (1) Void coefficients and scram coefficients for Cycles 6 and 7 were similar to those of Cycle 5; thus the transient response and thermal-hydraulic stability of the plant would be expected to be similar to that for the current cycle. (2) Although the special barrier bundles have a somewhat different pin-by-pin enrichment distribution than do the standard bundle designs, the impact on LOCA analyses is expected to be negligible. (3) Preliminary rod withdrawal error (RWE) and control rod drop accident (CRDA) analyses for Cycle 6 showed results comparable to any reload core without demonstration barrier bundles or ramp cells.

2.9 SUMMARY

Work on the nuclear design continues with no apparent major impediments to carrying out the demonstration. Optimization studies to improve the loading patterns to obtain a higher peak LHGR during the end-of-cycle power ramping are in progress. Bundle designs for the barrier bundles have been completed and are fixed.

2.10 ACKNOWLEDGMENTS

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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 Oxidation of Zirconium-Lined Zircaloy (B. Cheng, J. O. Hegland, and R. B. Adamson, NFSED)

3.2.1.1.1 Introduction

The corrosion performance of zirconium liners under conditions which simulate those inside a perforated fuel rod has been evaluated in a laboratory simulator. Early test results obtained in 1979, as reported previously⁽¹⁾, indicated that the reaction rates both of crystal bar and of low oxygen sponge liners are linear at both 350 and 400°C, and the rates are ~ (80–100) times higher than for Zircaloy-2 cladding at 350°C. The rapid oxidation of the liner continues until the whole liner is consumed, and the rate of oxidation then diminishes markedly.

During this reporting period, tests were carried out to provide additional data.

3.2.1.1.2 Results and Discussions

Tests were carried out on both crystal bar and low oxygen sponge at 350° and 400°C for 4 and 17 days using the same simulator and procedures as reported previously.⁽¹⁾ These data are shown in Table 3.2-1 and in Figure 3.2-1 along with the earlier results. The new data fit the linear reaction curves for both crystal bar and low oxygen sponge at both 350 and 400°C.

3.2.1.2 Yield Strength of Zirconium Barrier and Reference Tubing — (M. B. Reynolds, R. E. Blood and R. B. Adamson, NFSED)

3.2.1.2.1 Introduction

In support of the mechanical design of Zr-liner barrier fuel (cf. Section 4.3 of this report) it is desired to measure the yield strength (0.2% flow stress) of the tubing. Therefore, tests were performed to measure the flow stress at relatively low strain levels (circumferential plastic strain 0–1%) in a closed-end pressurized tube configuration.

Table 3.2-1
WEIGHT GAIN DATA OF SAMPLES TESTED IN THE SIMULATOR AT 350 AND 400 °C

Temperature (°C)	Duration (days)	Zircaloy-2	Crystal Bar Only*	Low Oxygen Sponge Only*
Previous Results (mg/dm²)				
352 ± 3	3	7.9	276 ± 20	192 ± 56
356 ± 6	12	14.7	963 ± 48	750 ± 89
354 ± 4	36	19.3	1451 ± 2	1486 ± 54
407 ± 7	3	—	894 ± 57	877 ± 3
403 ± 2	12	26.1	1443 ± 93	1556 ± 0
Current Results (mg/dm²)				
353 ± 3	17	15.4	1460 ± 18	1098 ± 207
402 ± 2	4	12.1	1283 ± 76	1104 ± 120

* Average of two samples.

3.2.1.2.2 Experimental Procedure and Results

Circumferential strain was measured as a function of internal pressure in a number of specimens of sponge zirconium lined cladding tubing. The tubing was in the annealed condition and initially autoclave-oxidized on its outer surface. The tubing was pressurized in the closed end mode at 367°C using silicone oil as the pressurizing medium. Circumferential strain was measured with a ribbon type circumferential extensometer developed specifically for these tests. The walls of the "gage sections" of the tubing specimens were reduced slightly by chemically etching the outside surface. The strain rate was 0.037 min⁻¹ in the circumferential direction.

Pressure and circumferential extension were recorded with a conventional X-Y analog plotter. Pressures at 0.2% circumferential strain were read from the plotter record.

After reducing the specimen wall thickness to an average value of 0.076 cm, the sponge zirconium barrier layer represented approximately 8.3% of the wall thickness.

Average internal pressure at 0.2% yield was 4177 psig (28.8 MPa in excess of atmospheric pressure) for the barrier tubing compared to 4365 psig (30.1 MPa) for non-barrier type annealed and autoclaved tubing. Corresponding hoop stresses for the barrier and non-barrier tubing were 31,837 psi (218.5 MPa) and 32,584 psi (224.7 MPa), respectively. Considering the thickness of the barrier layer, these pressure values are consistent with a barrier flow stress between two-thirds and three-fourths that of the Zircaloy matrix.

3.2.1.3 Zirconium Liner Cladding Bend Tests — (H. M. Nguyen, R. B. Adamson, NFSED)

3.2.1.3.1 Introduction

This experiment is intended to determine if severe bending deformation of Zr-liner fuel cladding can lead to delamination at the zirconium-Zircaloy interface. Tests were done on both unirradiated tubing and on irradiated fuel cladding.

Figure 3.2-1. Corrosion Weight Gains for Zr-liner Cladding and Reference Zircaloy-2 as a Function of Time at 350°C and 400°C in 7.0 MPa (1000 psig) Steam in the Simulator

3.2.1.3.2 Test Procedure

Longitudinally quartered sections of tubing 7.6 cm long were subjected to 3-point bending over a 6.35 mm diameter mandrel. The zirconium liner was on the convex (tension) side of the bend curvature. Specimens freely moved in a slot of the bending device in the desired direction. During bending, the mandrel advanced at the rate of 3.8 cm/min to bend a specimen into a U-shape.

The most severely deformed regions of bent specimens were metallographically examined to look for separation of the zirconium liner from the Zircaloy tubing, and to identify any significant changes in the microstructure.

3.2.1.3.3 Materials

The unirradiated specimens were from low-oxygen sponge Zr-liner tubing.

The irradiated specimen was sectioned from the irradiated test fuel rod (crystal bar Zr-liner) SRP 3/36 (cf. Section 3.4) 5.1–15.2 cm from the bottom of the rod, from portions of the segment not plastically deformed by ramp-testing in the R-2 Test Reactor. The burnup in the region of this specimen is estimated to be approximately 19.0 MWd/kg. This fuel cladding had a total wall thickness of 0.7 mm of which ~0.076 mm was the zirconium liner portion.

3.2.1.3.4 Results

Unirradiated Specimens

The bent specimen is shown in Figure 3.2-2. Micrographs were taken at the most severely deformed regions of the unirradiated bent specimens.

The micrographs did not show any microcracks or delamination, see Figures 3.2-2 and 3.2-3. This result indicated that the unirradiated zirconium-lined Zircaloy tubing is very ductile and was well bonded.

Some hydrides, about 20 ppm, existed in the zirconium liner and in the Zircaloy cladding as well.

Twinning associated with grain deformation was found in both zirconium liner and Zircaloy matrix, see Figures 3.2-3 and 3.2-4.

The thicknesses at the undeformed and severely deformed regions were approximately 0.81 mm and 0.84 mm respectively (see Table 3.2-2).

Irradiated Specimens

A bent irradiated specimen is shown in Figure 3.2-5 with micrographs in Figures 3.2-6 through 3.2-9. Figure 3.2-6 shows some oxide layers formed on the outside surface of the cladding. Oxide thicknesses ranged from 10–20 μm . This oxide was formed during irradiation.

Several microcracks extending 5–10 μm into the zirconium (Figure 3.2-6) were found in the undeformed regions and were not caused by the bend test. However, the cracks have been somewhat enlarged during the bend test (Figure 3.2-7).

The zirconium-Zircaloy interface was carefully examined, but no evidence of separation was found. It was very well bonded.

Figure 3.2-2. Unirradiated Specimen as Bent and Sectioned Metallographically. Plane of Section is Parallel to the Original Tube Axis and Normal Tube Bend Axis. a Indicates the Zirconium Liner and b the Zircaloy Portion of the Bent Tubing.

Figure 3.2-8 showed that twins were scattered in the zirconium liner. A much greater density of twinning was observed in the region of the Zircaloy, which had compressive strain near the mandrel (Figure 3.2-8b).

Thicknesses of the irradiated specimens were measured and recorded in Table 3.2-3. Metal thickening apparently took place at the deformed regions during bending.

The maximum tensile strain at points 0 in Figures 3.2-2 and 3.2-5 are estimated to be 11.6% and 10.7% respectively. At the zirconium-Zircaloy interface the strains are 10.6% for the unirradiated specimen in Figure 3.2-2 and 9.0% for the unirradiated specimen in Figure 3.2-5. Despite these very severe bending strains no delamination of the Zr-liner cladding occurred, even in the irradiated fuel cladding.

Figure 3.2-3. Photomicrograph at Region of Maximum Deformation (at O in Figure 3.2-2). No Separation Was Found at the Interface of Zirconium-Zircaloy. A Twin is Indicated by the Arrow, (a) Bright Field, (b) Polarized Light, Same Area as (a).

Figure 3.2-4. Microstructure at the Mid-Wall of the Zircaloy. Hydride Particles are indicated by Arrowheads. Twinning is Shown by Arrows. (a) Bright Field, (b) Polarized Light, Same Area as (a).

Table 3.2-2
AVERAGE THICKNESSES OF THE UNIRRADIATED SPECIMENS
MEASURED AT THE UNDEFORMED AND DEFORMED REGIONS

Specimens	Undeformed	Deformed	Metal Thickening
As received (unbent) specimen	0.032 inch 0.81 mm		
Bent specimen #1	0.032 inch 0.81 mm	0.033 inch 0.84 mm	0.001 inch 0.025 mm
Bent specimen #2	0.032 inch 0.81 mm	0.033 inch 0.84 mm	0.001 inch 0.025 mm

Figure 3.2-5. Irradiated Specimen as Bent: a Shows Zirconium Liner, b Shows Zircaloy Magnification 4X

Figure 3.2-6. Photomicrograph of Irradiated Specimen at the Undeformed Region. (a) Microcracks in the Zirconium Liner, Indicated by Arrowheads. (b) Oxide Layer Formed at Outer Surface of Tubing, Indicated by Arrowheads.

Figure 3.2-7. Irradiated Bent Specimen at the Most Severely Deformed Region (Marked O in Figure 3.2-5). (a) Region a is Zirconium Liner. Region b is Zircaloy. (b) Microcracks in the Zirconium Liner are Indicated by Arrowheads.

Figure 3-2-8 Irradiated Specimen at the Most Severely Deformed Region as Viewed in Polarized Light Arrows Indicate Twins (a) At Tubing Inner Surface, Zirconium Liner Indicated by A and Zircaloy by B (b) At Mid-Wall Location Showing Heavy Twinning in the Region of the Zircaloy Deformed by Compression During the Bend Test

Table 3.2-3
AVERAGE THICKNESSES OF THE IRRADIATED SPECIMENS
MEASURED AT THE UNDEFORMED AND DEFORMED REGIONS

Specimens	Undeformed	Deformed	Metal Thickening
A2B	0.028 inch 0.71 mm	0.032 inch 0.81 mm	0.004 inch 0.10 mm
A2C	0.028 inch 0.71 mm	0.030 inch 0.76 mm	0.002 inch 0.05 mm
A2D	0.028 inch 0.71 mm	0.029 inch 0.73 mm	0.001 inch 0.025 mm

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)

Irradiation experiments continue to be conducted in the Experimental Breeder Reactor (EBR-II) to provide candidate barrier cladding with high fluence exposures. Post-irradiation expanding mandrel tests of materials in the first and second EBR-II experiments have evaluated the resistance of barrier cladding to stress corrosion cracking under simulated PCI conditions. The results of these tests are reported in this section.

Experimentally determined neutron fluences are now available for the rods in the second EBR-II experiment, namely: (1) reference (bright-etched) Zircaloy-2, (2) copper on autoclave-oxidized Zircaloy-2, (3) 0.076-mm low-oxygen sponge zirconium-lined Zircaloy-2 and (4) 0.076-mm crystal bar zirconium-lined Zircaloy-2. The neutron fluences ($E > 1$ MeV) for tubes (1) and (2) were 6.3×10^{21} n/cm² at the core midplane ($T \approx 415^\circ\text{C}$) and 4.9×10^{21} n/cm² near the bottom of the core ($T \approx 373^\circ\text{C}$) and for tubes (3) and (4) the corresponding fluences were 5.8×10^{21} n/cm² and 4.4×10^{21} n/cm², respectively.

Irradiation of the third series of rods (EBR-II-3) began in December 1979 and is currently scheduled for completion in October 1980. The projected fluences ($E > 1$ MeV) are $\sim 5.2 \times 10^{21}$ n/cm² at core midplane and $\sim 4.7 \times 10^{21}$ n/cm² near the bottom of the core. The materials in EBR-II-3 are: (1) reference (bright-etched) Zircaloy-2, (2) 0.076-mm low-oxygen sponge zirconium-lined Zircaloy-2, (3) 0.076-mm low-oxygen sponge zirconium-lined Zircaloy-2 heat treated to recrystallize the liner and to stress relieve the cladding and (4) thin (~ 0.03 -mm) low-oxygen sponge zirconium-lined Zircaloy-2. The thin-liner rod will provide irradiated material for evaluation of the effect of liner thickness on the resistance to failure under simulated PCI conditions.

3.2.2.2 Expanding Mandrel Tests — (S. B. Wisner, R. P. Tucker, G. H. Henderson and R. B. Adamson, NFSED)

3.2.2.2.1 Introduction

To evaluate the performance of high fluence Zircaloy-2 barrier cladding under simulated PCI conditions two types of expanding mandrel tests in iodine have been conducted remotely in the Test Handling Facility at the GE Vallecitos Nuclear Center. In constant extension rate tests (CERTS), cladding specimens were strained to failure by continuous application of an increasing load. These tests were conducted at 337°C in 40 Pa (0.3 torr) flowing iodine at a strain rate of 5×10^{-4} min⁻¹. In ramp and hold (RH) tests, cladding specimens were rapidly ramped (loaded) to a predetermined strain (generally 2.0% nominal) at which time the position of the

loading member was fixed and the specimen held in test for 20 hours (maximum) or until through wall cracking occurred. The RH tests were conducted at 337°C in 40 Pa flowing iodine at a nominal ramp strain rate of 0.031 min⁻¹. During the holding period the load relaxed, in part, due to a cladding creep mechanism, simulating the behavior in fuel cladding after a power ramp (i.e., control blade withdrawal). Additional experimental details for the expanding mandrel test are in previous reports in this series.²⁻⁵

Most of the mandrel specimens were sectioned from non-fueled cladding tubes irradiated as part of two experiments in the EBR-II reactor, namely: (1) reference and low-oxygen sponge zirconium lined Zircaloy at fluences of $1.4\text{--}2.0 \times 10^{21}$ n/cm² ($E > 1$ MeV) from EBR-II-1 and (2) reference, low-oxygen sponge zirconium lined and crystal bar zirconium lined Zircaloy at fluences of $3.5\text{--}6.2 \times 10^{21}$ n/cm² from EBR-II-2. The irradiation temperatures for the specimens tested were estimated to be 373–408°C.

In addition, one mandrel specimen was cut from the non-fueled plenum region of each of two sound crystal bar zirconium lined Zircaloy-2 SRP (3/35 and 3/36) rods (cf. Section 3.4). These fuel rods were irradiated in the Millstone reactor and then ramp tested in the R-2 reactor at Studsvik, Sweden. The estimated peak burnup obtained from neodymium isotopic gamma scans was 20.2 MWd/kg-U for 3/35 and 19.0 MWd/kg-U for 3/36. The estimated fast fluences ($E < 1$ MeV) based on the burnups were 3.7×10^{21} n/cm² and 3.5×10^{21} n/cm² for 3/35 and 3/36, respectively.

3.2.2.2 Results

Tables 3.2-4 and 3.2-5 summarize the results of the CERT and RH mandrel tests, respectively. The final plastic strains reported in the tables were obtained from post-test diametral measurements taken for each specimen as illustrated in Figure 3.2-9. Note the necking of the cladding at mandrel segment interfaces, segment and alumina sleeve fragmentation, extrusion of the zirconium slug between the segments and the non-uniform segment separation. This post-test appearance is typical for the specimens in CERT and RH expanding mandrel tests.

CERT Tests

For the CERT tests in Table 3.2-4, comparison of the strains at maximum load indicates improved performance by both types of zirconium-lined cladding relative to unlined reference cladding. The reference specimens were embrittled and cracked in gaseous iodine at considerably smaller strains than the sponge and crystal bar lined specimens. However, both types of zirconium liners experienced liner penetrating cracks when strained to relatively high levels in the range from 2.8 to 3.0%. At lower strains small ($< 10 \mu\text{m}$) non-penetrating cracks were observed in one sponge zirconium liner tested to 2.6% (Figure 3.2-10), whereas a companion higher fluence specimen (LX7X-S15) tested to the same strain exhibited only very small microcracks ($< 3 \mu\text{m}$). On the other hand, a crystal bar liner tested to a strain of 2.4% exhibited non-penetrating cracks up to about 50 μm in length (Figure 3.2-11), which is a depth greater than observed for comparable sponge lined specimens.

Figures 3.2-12 through 3.2-14 are scanning electron micrographs (SEM) of the fracture surfaces of reference, sponge, and crystal bar zirconium lined specimens. The fracture surface for the reference specimen is shown in Figure 3.2-12. A fracture morphology typical of iodine embrittled Zircaloy is clearly seen. The fracture appears to be predominantly transgranular with cleavage facets and some fluting features seen. Figure 3.2-13 shows the fracture surface for a sponge zirconium lined specimen. Some intergranular features can be seen along with large cleavage facets (Figure 3.2-13b). At this fracture location, the crack had penetrated approximately half-way into the liner. The remainder of the surface has a ductile appearance and was formed by fracture in liquid nitrogen prior to SEM examination. In Figure 3.2-14, the fracture surface for a crystal bar zirconium lined specimen tested in iodine is shown. Similar to the sponge liner, large cleavage facets are evident as well as some fluting features. However, at this fracture location, the crack had penetrated through the liner into the Zircaloy-2 base metal.

Figure 3.2-9. *Transverse View of the Post-Test Appearance of an Expanding Mandrel Specimen Showing Strain Measurement Points and Local Necking.*

Figure 3.2-15 presents the relative strain performance in CERT tests of reference, sponge lined, and crystal bar lined Zircaloy-2 tested in iodine. Although there is a strain beyond which the sponge and crystal bar liners are not immune to iodine stress corrosion cracking (SCC), improved performance was demonstrated over the reference clad in terms of strain at maximum load. Strains in excess of about 2.6% for sponge and about 2.4% for crystal bar liners appear to be required to cause penetrating cracks of the liner.

RH Tests

For the RH tests, examination of the results in Table 3.2-5 indicates improved performance for the zirconium lined cladding compared to reference Zircaloy cladding. Typical load-time/strain-time curves for the RH tests on reference and zirconium lined specimens are presented in Figure 3.2-16. From the curves it can be seen that during the 20 hour holding period the load decreased, in part, due to a cladding creep mechanism. All the reference specimens were embrittled and exhibited through wall cracks. The zirconium lined specimens had small non-penetrating micro-cracks ($<30 \mu\text{m}$) in the liner. (For the SRP crystal bar lined specimens, stereoscopic examination of clam shell sections revealed no evidence of incipient cracks. Transverse metallography on these specimens is in progress.) Figure 3.2-16 shows the typical microcracks that were observed in both zirconium liner types.

Table 3.2-4
SUMMARY OF EXPANDING MANDREL CONSTANT EXTENSION RATE TESTS FOR EBR-II-1 AND 2 IRRADIATED REFERENCE
AND BARRIER MODIFIED CLAD TESTED IN FLOWING IODINE AT $\dot{\epsilon} = 5 \times 10^{-4} \text{ min}^{-1}$ AND 337°C

Specimen Number	Material	Fluence n/cm ² E > 1 Mev	T _{irr} , °C	Maximum Load Attained lbs	Strain at Max. Load, % ε _{max}	Final Plastic Strain Meas. at Seg. Midpoints, %		Avg. Plastic Strain at Seg. Midpoints, % ε _m	Final Plastic Strain Meas. at Seg. Interfaces, %		Avg. Plastic Strain at Interfaces, % ε _i	Comments
						ε _{m1}	ε _{m2}		ε _{i1}	ε _{i2}		
EBR-II-1												
LZ1-S15 ⁽¹⁾	Reference	~2.0 × 10 ²¹	~397	2260	0.8 ⁽²⁾	—	—	1.2	—	—	?	Cracks (through wall)
LZ5-S15	Sponge Zr Liner	~1.6 × 10 ²¹	~397	2610	2.9 ⁺	3.0	2.9	2.9	1.0 ⁽³⁾	1.7	1.4	Cracks (through wall)
LZ5-S13	Sponge Zr Liner	~1.4 × 10 ²¹	~373	2650	2.6 ⁺	2.6	2.6	2.6	1.6	1.4	1.5	Small (<10μm) cracks (non-penetrating)
EBR-II-2												
LX1X-S13	Reference	~3.8 × 10 ²¹	~373	2600	1.4 ⁽²⁾	1.9	1.7	1.8	1.3	0.7 ⁽³⁾	1.0	Cracks (through wall)
LX1X-S15	Reference	~5.9 × 10 ²¹	~397	2640	1.8 ⁽²⁾	2.2	2.2	2.2	0.6 ⁽³⁾	0.9	0.8	Cracks (through wall)
LX7X-S13	Sponge Zr Liner	~3.5 × 10 ²¹	~373	2710	3.0 ⁽²⁾	3.4	3.2	3.3	1.9	1.6 ⁽³⁾	1.8	Cracks (through wall)
LX7X-S15	Sponge Zr Liner	~5.4 × 10 ²¹	~397	2570	2.6 ⁺	2.6	2.6	2.6	1.3	1.1	1.2	Very small cracks (2–3 μm)
LZ7-S13	Xtal Bar Zr Liner	~3.5 × 10 ²¹	~373	2670	2.8 ⁺	2.9	2.8	2.8	1.8	0.8 ⁽³⁾	1.3	Cracks (through wall)
LZ7-S15	Xtal Bar Zr Liner	~5.4 × 10 ²¹	~397	2710	2.4 ⁺	2.4	2.4	2.4	1.1	1.0	1.1	Medium (<50μm) cracks (non- penetrating)

(1) Individual segment midpoint strains not measured.

(2) Value based on ratio of $\epsilon_{\text{max}}/\epsilon_i$ taken from load-strain curves times ϵ_m

(3) Crack location

Table 3.2-5
SUMMARY OF EXPANDING MANDREL RAMP AND HOLD TESTS FOR EBR-II AND MILLSTONE IRRADIATED
REFERENCE AND BARRIER MODIFIED CLAD TESTED IN FLOWING IODINE AT 337°C
(RAMP RATE = 0.031 min⁻¹, HOLD PERIOD = 20 HOURS)

Specimen Number	Material	Fluence n/cm ² , E > 1 Mev	T _{irr} , °C	Maximum Load lbs	Total Extension (Elastic + Plastic) at Max. Load, % ε _{max}	Final Plastic Strain Meas. at Seg. Midpoints, %		Avg. Plastic Strain at Seg. Midpoints, % ε _m	Final Plastic Strain Meas. at Seg. Interfaces, %		Avg. Plastic Strain at Interfaces ε _i	Comments
						ε _{m1}	ε _{m2}		ε _{i1}	ε _{i2}		
EBR-II-1												
LZ1-S14 ⁽¹⁾	Reference	~2.0 × 10 ²¹	~387	3040	2.5	2.8	2.7	2.8	1.4	0.9 ⁽³⁾	1.2	Cracks, <20 hrs
LZ5-S14	Sponge Zr Liner	~2.0 × 10 ²¹	~387	3070	2.0	3.0	2.7	2.9	1.7	2.0	1.9	Small cracks*
EBR-II-2												
LX1X-S14	Reference	~5.5 × 10 ²¹	~387	3000	2.0	2.6	2.4	2.5	0.8 ⁽³⁾	1.9	1.4	Cracks, <5 hrs
LX1X-S16	Reference	~6.2 × 10 ²¹	~408	3100	2.1	3.0	2.8	2.9	1.0 ⁽³⁾	1.9	1.5	Cracks, <5 hrs
LX7X-S14	Sponge Zr Liner	~4.9 × 10 ²¹	~387	2980	2.1	2.5	2.3	2.4	1.1	1.4	1.3	Small cracks*
LX7X-S16 ⁽²⁾	Sponge Zr Liner	~5.8 × 10 ²¹	~408	2920	2.6	3.3	3.1	3.2	1.7	1.8	1.8	Small cracks*
LZ7-S14	Xtal Bar Zr Liner	~4.9 × 10 ²¹	~387	3250	2.1	2.9 3.4	2.6 3.2	2.8 3.3	1.1 2.3	1.7 1.4	1.4 1.9	Small cracks*
LZ7-S16 ⁽⁴⁾	Xtal Bar Zr Liner	~5.8 × 10 ²¹	~408	3025	3.1	3.4	3.2	3.3	2.3	1.4	1.9	No cracks
SRP												
3/35	Xtal Bar Zr Liner	~3.7 × 10 ²¹	288	2800	2.1	3.1	3.0	3.0	1.0	1.4	1.2	No cracks
3/36	Xtal Bar Zr Liner	~3.5 × 10 ²¹	288	2890	2.1	2.3	2.2	2.2	1.3	0.6	1.0	No cracks

(1) Ramp target total extension of 2.0%.

(2) Specimen ramped to total extension of 2.5% prior to hold.

(3) Crack location.

(4) Metallographic examination not complete.

* Cracks not penetrating through the liner.

Figure 3.2-10. Photomicrographs of Small ($<10\mu\text{m}$) Incipient Cracks Found at One Segment Interface Location in the Irradiated Sponge Liner of Specimen Tested to 2.6% Strain.

Figure 3.2-11. *Photomicrograph of Small ($<50\mu\text{m}$) Incipient Cracks Found at One Segment Interface Location in the Irradiated Crystal Bar Liner of Specimen LZ7-S15 Tested to 2.4% Strain.*

(a)

300X

(b)

1000X

Figure 3.2-12. Scanning Electron Micrograph Showing the Fracture Surface of LX1X-S15 (Reference Zircaloy-2) Tested in 337° C, 40 Pa Iodine.

Figure 3.2-13. Scanning Electron Micrographs Showing the Fracture Surface of LX7X-S13 (0.076-mm Low Oxygen Zirconium Liner on Zircaloy-2) Tested in 337°C, 40 Pa Iodine.

1000X

Figure 3-2-14 Scanning Electron Micrograph Montage Showing the Fracture Surface of LZ7-S13 (0.076-mm Crystal Bar Zirconium Liner on Zircaloy-2) Tested in 337°C, 40 Pa Iodine

Figure 3.2-15. Comparison of the Relative Performance (in CERT) of EBR-II-1 Reference and Sponge Zirconium Lined Specimens and EBR-II-2 Reference, Sponge Zirconium Lined Specimens, and Crystal Bar Lined Specimens Tested in Gaseous Iodine.

NOTE: Arrows indicate test terminated before maximum load was attained.

Figure 3.2-16. Slug Load and Strain as a Function of Time in Typical RH Expanding Mandrel Tests of Irradiated Reference and Barrier Tubing from the EBR-II-2 Irradiation. (a) Reference (b) Crystal Bar Zr-Liner (c) Low-Oxygen Sponge Zr-Liner.

Figure 3.2-17. Photomicrographs of Microcracks Penetrating into Liners of LZ5-S14 (0.076-mm Low Oxygen Sponge Zirconium Liner on Zircaloy-2) and LZ7-S14 (0.076-mm Crystal Bar Zirconium Liner on Zircaloy-2) Tested (RH) in 337°C, 40 PA Iodine.

It should be noted that for one crystal bar lined specimen (LZ7-S14), a transverse view revealed small ($<20\text{ }\mu\text{m}$) cracks that had penetrated into the liner in arrays around the full circumference of the specimen. Figure 3.2-18 presents a schematic of a transverse view which maps these cracks. It appears that many of the cracks are associated with small imperfections in the liner. The pattern of microcracks in Figure 3.2-18 is similar to that reported previously for the crystal bar zirconium lined SRP-3/35 after in-reactor ramp testing. In contrast, the sponge zirconium liner does not contain the cracking pattern observed in the crystal bar liner, but instead, has similar sized cracks constrained to the mandrel segment interface regions (see Figure 3.2-17).

To assess the performance of the irradiated Zr-liner tubing to strains greater than the nominal 2.0%, one specimen of each liner type (sponge and crystal bar) was ramped (loaded) to 2.6% and 3.1%, respectively. Post-test visual examination revealed no evidence of through wall cracks on the outer surface of either specimen. Figure 3.2-19 shows a transverse view of the sponge lined specimen that was ramped to 2.6%. Small ($<30\text{ }\mu\text{m}$) cracks in the liner are visible which are similar to the cracks observed in the sponge liner that was ramped to a lesser extension of $\sim 2.0\%$ (Figure 3.2-17).

3.2.2.2.3 Conclusions

The expanding mandrel tests conducted on reference and zirconium lined cladding irradiated in the EBR-II support the following conclusions:

CERTS

1. Both crystal bar and sponge zirconium lined cladding exhibit improved mechanical performance over reference cladding when tested under simulated PCI conditions.
2. In flowing iodine, the reference cladding experienced a maximum load, achieved relatively small strain of ~ 0.8 to 1.8% at maximum load and displayed through wall cracks.
3. In general, both types of zirconium lined cladding did not reach a maximum load at plastic strains of ~ 2.4 to 3.0% .
4. Small incipient cracks formed without penetrating the zirconium liners at strains of 2.4% and 2.6% for crystal bar and sponge, respectively.
5. Based on post-test metallographic examinations, cracks penetrated the zirconium liners for plastic strains greater than about 2.9% for sponge and 2.8% for crystal bar.
6. Within the range investigated, there is no clear effect of irradiation temperature (~ 373 to 397°C) or fast fluence. (~ 1.4 to $5.9 \times 10^{21}\text{ n/cm}^2$, $E > 1\text{ MeV}$) on the susceptibility of reference or zirconium-lined cladding to stress corrosion cracking in iodine.

Figure 3.2-18. Schematic of a Transverse View of Specimen LZ7-S14 (0.076-mm Crystal Bar Zirconium Liner on Zircaloy-2) Showing Location of Microcracks in the Liner Around the Circumference.

Figure 3.2-19. Photomicrograph of Microcracks Penetrating into the Liner of LX7X-S16 (0.076-mm Low Oxygen Sponge Zirconium Liner on Zircaloy-2) Tested in 337°C, 40 Pa Iodine.

RH Tests

1. Under simulated control blade withdrawal conditions (ramp and hold in flowing iodine), crystal bar and sponge zirconium lined cladding showed resistance to cracking whereas all reference cladding was embrittled and failed by through wall cracks.
2. Zirconium lined cladding maintained its resistance to cracking up to the highest strains attained, namely 3.2-3.3%.
3. Post-test metallography of zirconium lined specimens revealed microcracks ($<30\ \mu\text{m}$) that did not penetrate through the liner.
4. Arrays of microcracks ($<20\ \mu\text{m}$) developed around the entire circumference in the crystal bar lined specimen and predominantly at the mandrel segment interfaces in the sponge lined specimens.

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3.3 SUBTASK II.2 LICENSING TESTS (T. C. Rowland, L. D. Noble)

Reactivity initiated accident (RIA) tests to assess the relative performance of reference and barrier fuels have been conducted in the Nuclear Safety Research Reactor (NSRR) in Japan through arrangements with the U.S. Nuclear Regulatory Commission (NRC) and the Japan Atomic Energy Research Institute (JAERI). Results of 16 tests at ambient conditions on six reference rods, five copper barrier and five zirconium liner, were reported previously.² Five additional tests have been performed on two reference rods, two zirconium liner rods and one copper barrier fuel rod.⁶

The most recent test results confirm the previous results, i.e., that barrier fuel rods, either zirconium liner or copper plated, do not perform any differently than reference fuel rods in RIA tests. The failure threshold was the same for all of the fuel types (~ 270 cal/g), (Figure 3.3-1) the energy deposition at which fuel is dispersed was about 390 cal/g, and the thermal to mechanical energy conversion ratio was also equivalent when the fuel was dispersed. Therefore, to date, the performance of Zr-liner and Cu-barrier fuel rods is equivalent to that of conventional fuel in an RIA.

Additional tubing for both barrier and reference fuel rods has been sent to the Nuclear Safety Research Reactor for additional RIA tests at both ambient and at BWR hot standby conditions. In addition, irradiated reference and barrier fuel rods are being furnished to EG&G Idaho, Inc. for Power Burst Facility (PBF) tests which are planned for 1981. The PBF tests were to include both RIA and operational transient (OPTRAN) tests. The NRC recently deferred indefinitely all further RIA tests. Disposition of this fuel awaits a decision on a new test matrix to be proposed by EG&G.

Figure 3.3-1. Comparison of Failure Threshold for Reference and Barrier Clad Fuel Rods

3.4 SUBTASK II.3 FUEL IRRADIATION TESTS

3.4.1 Segmented Rod Irradiation Tests (J. H. Davies, E. Rosicky, L. D. Albert, E. L. Esch and S. R. Schutz)

3.4.1.1 Bundle Status

The irradiation status of the three segmented test rod (STR) assemblies in the segmented rod program (SRP) is updated in Table 3.4-1. During the current reporting period the Monticello bundle (SRP-2) was reconstituted. Nineteen (19) segments were retrieved for ramp testing in 1980 (see Section 3.4.1.3).

3.4.1.2 1979 Power Ramp Tests

The 1979 Ramp Test campaign in the R2 Test Reactor at Studsvik was described previously,¹ and preliminary results were reported. In the present report period those preliminary findings have been confirmed.

One interesting additional observation was a negative deflection (spike) in the power output of Rod SRP-2/24 about three minutes after a ramp, but before activity release. Positive power spikes have been discussed previously^{1,2}; they are believed to be primary indications of rod defects and are taken as evidence that fission product release is not always simultaneous with cladding crack penetration. Studsvik personnel interpret the spikes in terms of sudden changes in the gap heat conductance due to coolant ingress. The positive or negative sign of the deflection depends on the initial helium pressure. Positive spikes have been observed only on defected rods with 1 atm of helium. Rod SRP-2/24, which exhibited a negative spike, was filled with 22 atm of helium.

The principal conclusions drawn from the 1979 test remain unaltered, i.e.

- Reducing the pre-ramp standardization time from 300 hours to 6 hours did not significantly affect ramp test performance.
- Although Cu-barrier fuel (nonbonded) outperforms reference fuel, it is not immune to failure at high burnups.
- All barrier fuel subjected to a single large power step (Ramp Sequence B) at low to medium exposures (to 10.5 MWd/kg U) survived. (See page 2-6 for Reference.)
- A Cu-Zr intermetallic layer between barrier and cladding clearly degrades the performance of Cu-barrier fuel.

The rods were all neutron radiographed and visually examined at Studsvik. The neutron radiographs confirmed the ramp test results; getter hydriding was observed in all the rods labelled "failed" and was not observed in those labelled "sound."

Visible evidence of defects in the failed rods was generally rather limited. The longest crack was about 12 mm and most defects were characterized as X-marks or white deposits.

Twenty-seven (27) of the 36 ramp tested rods (of the 1979 test series) were received at Vallecitos in April, 1980, and selected rods are now being examined.

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	13.9	16.2	April 30, 1980
	Mid-Top	19.9	23.3	
	Mid-Bottom	19.7	21.4	
	Bottom	15.9	19.8	
	Bundle Avg.	17.3		
SRP-2 (Monticello)	Top	12.3	17.7	April 30, 1980
	Mid-Top	19.0	26.7	
	Mid-Bottom	21.2	29.6	
	Bottom	18.2	25.4	
	Bundle Avg.	17.7		
SRP-3 (Millstone)	Top	14.0	18.8	April 30, 1980
	Mid-Top	21.2	25.9	
	Mid-Bottom	23.9	27.4	
	Bottom	21.2	24.2	
	Bundle Avg.	20.1		

3.4.1.3 1980 Power Ramp Tests

Twenty-seven (27) rods have been selected for ramp testing in 1980. Eight (8) of these rods have been characterized and shipped to Studsvik; they will be tested in August. The remainder were retrieved from SRP-2 (Monticello) in February and are currently undergoing pre-ramp characterization; this includes visual examination, profilometry, gamma scanning, neutron radiography and pulsed eddy current testing. These rods are scheduled to be tested in November. A general description of all 27 rods is provided in Table 3.4-2. Note that for the first time segment locations (Top, Middle or Bottom) during incubation in the segmented rod assemblies are given.

Top and Bottom segments differ from Middle segments in exhibiting burnup gradients. Axial burnup profiles are determined during pre-ramp characterization by ^{137}Cs gamma scanning. Typical scans for Top, upper Middle, lower Middle and Bottom segments are shown in Figures 3.4-1 to 3.4-4. Peak to average (P/A) burnup ratios are indicated in the figure captions. Each fuel column (and gamma scan) is subdivided in ten equal length axial nodes (Figure 3.4-5). This facilitates data handling and more precise definition of ramp tests. Most previously ramped rods were middle segments and were adequately described by their average burnups. The need to refine definitions will be apparent when the test specification is outlined later in this section.

Table 3.4-2
DESCRIPTION OF FUEL RODS TO BE TESTED IN 1980

Segment Identification	Axial Location in Bundle	Design Feature^(a)	Cladding Wall (mm)	Diam Gap (mm)	He Pressure (MPa)	Ave BU^{(b)(c)} (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 Figures 3.4-1 through 3.4-4 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.

Figure 3.4-1. ^{137}Cs Gamma Scan Top Segment ($P/A = 1.25$)

Figure 3.4-2. ^{137}Cs Gamma Scan Upper Middle Segment ($P/A = 1.08$)

Figure 3.4-3. ^{137}Cs Gamma Scan Lower Middle Segment ($P/A = 1.11$)

Figure 3.4-4. ^{137}Cs Gamma Scan Bottom Segment ($P/A = 1.22$)

Figure 3.4-5. Diagram of Fuel Rod Segment

The 1980 Test Specification has been considerably revised to meet program objectives. The principal new feature is the addition of a Ramp Sequence C, which is a single step ramp at a 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 performed using a ^3He neutron attenuation device and previous experience has shown that ramp rates are limited to about $19 \text{ kW} \cdot \text{m}^{-1} \cdot \text{min}^{-1}$ ($5.8 \text{ kW} \cdot \text{ft}^{-1} \cdot \text{min}^{-1}$). To meet Sequence C ramp rate requirements, the test will be performed with the peak power near the lower end of a fuel column by raising the fuel rod 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 will subject its lower end to a substantial, predetermined power shock at a ramp rate, which reasonably simulates control blade withdrawal ramp rates. Thus, where specified, some of the rods will be double ramped. Besides providing a critical test of the barrier concepts, Sequence C ramps double the number of results per fuel rod at minimal incremental cost. Specified Ramp Sequences A, B' and C are shown in Figures 3.4-6, 3.4-7, 3.4-8 and typical power distributions in double ramps are shown schematically in Figure 3.4-9 from data generated by the staff of Studsvik Energiteknik AB.

The specified test matrix is summarized in Table 3.4-3. Several additional comments and words of explanation are necessary:

- A new definition, Ramp Type, has been coined. This describes the type of segment being tested, i.e., Top, Middle, or Bottom and the location of the power peak during testing, i.e., middle or bottom of the rod. This new definition describes each test more accurately and facilitates specification of double ramps.
- Ramp Sequence A is the standard "staircase" ramp to 59 kW/m (18 kW/ft), except that the hold time at the terminal peak power level has been reduced from six to four hours. This change is justified by prior test data that show 90 % of all defects were detected in the first hour following a ramp step and the longest reported delay time was 146 minutes.
- Ramp Sequence B' differs from the 1979 Sequence B mainly in the terminal peak power, which is variable. One of the objectives of the current test series is to define the Reference failure threshold for single step (B') ramps. Thus, in some cases, final specification of terminal peak power will depend on the results of prior tests.
- Specific test objectives are:
 1. Define Reference failure threshold for Sequence B' ramps.
 2. Qualify Ramp Sequence C.
 3. Extend definition of reference failure curve to higher burnups.
 4. Test barrier fuel at lead burnups.
 5. Test capability of barrier fuel to withstand very fast ramps (C Ramps).
 6. Compare performance of sponge and crystal bar at low to moderate burnups.

Table 3.4-3
1980 RAMP TEST MATRIX

1st Batch of Rods

Rod Identification	Design Feature	Ave. BU (MWd/kgU)	Ramp Seq.	Ramp Type	Terminal Peak Pwr. ^(a) (kW/ft)		Objective ^(c)
					1st Ramp	2nd Ramp ^(b)	
SRP-3/52	Copper-on-Oxide	5.9	A C	Mm Mb	18	16	4 2, 5
SRP-3/51	Copper-on-Oxide	7.8	B' C	Mm Mb	16	16 (14)	4 2, 5
SRP-3/49	Reference	9.7	B' C	Tm Tb	16	14 (16)	1 2
SRP-3/48	Reference	12.2	B' C	Bm Bb	14 (16)	14	1 2
SRP-3/53	XB Liner	14.9	A C	Tm Tb	18	14	4 5
SRP-3/50	XB Liner	21.4	A C	Bm Bb	18	14	4 5
SRP-2/35	Reference	17.1	B' C	Bm Bb	14	14	1 2
SRP-2/36	Reference	16.4	B' C	Bm Bb	14 (12)	14 (12)	1 2

2nd Batch of Rods

SRP-2/53	Reference		B' C	Tm Tb	14	14	1 2
SRP-2/55	Reference		B' C	Tm Tb	14 (12)	14 (12)	1 2
SRP-2/48	Reference		B' C	Mm Mb	14 (12)	14 (12)	1 2
SRP-2/47	Reference		B' C	Mm Mb	14 (12) (10)	14 (12) (10)	1 2
SRP-2/54	Reference		A	Bm	18	—	3
SRP-2/52	Reference		A C	Bm Bb	14 (12) ^(d)	14 (12)	3 2
SRP-2/39	Copper-on-Oxide		A C	Mm Mb	18	14	4 5

Table 3.4-3
1980 RAMP TEST MATRIX (Continued)

2nd Batch of Rods

Rod Identification	Design Feature	Ave. BU (MWd/kgU)	Ramp Seq.	Ramp Type	Terminal Peak Pwr. ^(a) (kW/ft)		Objective ^(c)
					1st Ramp	2nd Ramp ^(b)	
SRP-2/38	Copper-on-Oxide		A	Mm	18		4
			C	Mb		14 (16)	5
SRP-2/41	Copper-on-Oxide		A	Bm	18	—	4
SRP-2/42	Copper-on-Oxide		A	Tm	18		4
			C	Tb		14 (16)	5
SRP-2/40	Cu Barrier (nb)		A	Tb	18	—	4
SRP-2/37	Cu Barrier (nb)		B'	Bm	16		4
			C	Bb		14	5
SRP-2/45	XB Liner		A	Mm	18		6
			C	Mb		14 (16)	5
SRP-2/44	Sponge Liner		A	Mm	18		6
			C	Mb		14 (16)	5
SRP-2/43	XB Liner		A	Bm	18		6
			C	Bb		16	5
SRP-2/46	Sponge Liner		A	Tb	18	—	6
SRP-2/51	Reference		B'	Tm	16		1
			C	Tb		16	2
SRP-2/50	RX Liner		A	Mm	18		6
			C	Mb		16	5
SRP-2/49	XB Liner		A	Mm	18		6
			C	Mb		16	5

^(a) Selection of terminal peak power contingent on results of prior tests in the series. Alternate values in parentheses.

^(b) Second ramp performed only if rod survives first ramp.

^(c) See text, Section 3.4.1.3.

^(d) Truncated Sequence A.

Figure 3.4-6. Ramp Sequence A

Figure 3.4-7. Ramp Sequence B'

Figure 3.4-8. Ramp Sequence C

Figure 3.4-9. Double Ramp Test — Relative Power Distributions

3.4.1.4 Destructive Post-Irradiation Examination

The non-destructive post-irradiation examination (PIE) of the twelve rods ramp tested at Studsvik in 1978 was described previously.¹ Six rods were selected for destructive PIE. These are listed in Table 3.4-4. The main objectives of the destructive examination were: to investigate the causes of the pulsed eddy current (PEC) signals in the sound rods, SRP-2/1 and SRP-3/35; to determine the thickness of any Zr-Cu intermetallic layer in SRP-2/1; to characterize the state of the Cu-barrier and the Cu-Zr interface in the failed barrier rod, SRP-3/33.

3.4.1.5 Fission Gas Release

All three sound (unfailed) rods were punctured for fission gas samples. These were analyzed by gas chromatography to determine the relative amount of fission gas and gamma counted for ⁸⁵Kr. A summary of the reported results is presented in Table 3.4-5. The absence of hydrogen in all three samples provides further confirmation of rod integrity.

Table 3.4-4
RODS SELECTED FOR DESTRUCTIVE EXAMINATION

Rod Identification	Design Feature	Average Burnup ^a (MWd/kg-U)	Maximum Peak Power ^b		Time at Maximum Power ^c	Test Results
			(kW/m)	(kW/ft)		
SRP-2/11	Reference	15.4	45.9	14	0 ^d	Failed
SRP-2/14	Reference	14.6	45.9	14	30 s	Failed
SRP-2/1	10 μ m Cu-Barrier (nonbonded)	12.4	59.1	18	12 h	Sound
SRP-3/33	5 μ m Cu-Barrier (nonbonded)	16.7	52.5	16	39 min	Failed
SRP-3/35	Zr-Liner (crystal bar)	16.6	59.1	18	12 h	Sound
SRP-3/36	Zr-Liner (crystal bar)	15.3	59.1	18	12 h	Sound

^a Burnups calculated from ¹³⁷Cs gamma scans by comparison to known standard

^b Nominal value after last ramp step

^c Failure times based on loop activity level

^d Failed during ramp step to peak power shown

Table 3.4-5
FISSION GAS ANALYSES FOR THREE SOUND RODS

	SRP-2/1	SRP-3/35	SRP-3/36
• Orig. Helium Pressure (atm)	1.07	17.3	17.3
• Puncture Gas Pressure (atm)	3.05	20.0	19.83
• Puncture Gas Volume (STPcm ³)	39.24	263.27	260.67
• Puncture Void Volume (cm ³)	13.78	14.15	14.15
• Gas Chromatography Volume Percent Gases:			
Hydrogen	ND ^a	ND	ND
Oxygen	ND	ND	ND
Nitrogen	ND	ND	ND
Argon	0.08	ND	ND
Methane	ND	ND	ND
Carbon Monoxide	ND	ND	ND
Carbon Dioxide	ND	ND	ND
Krypton	7.39	1.75	1.53
Xenon	56.91	14.02	12.22
Helium	Bal.	Bal.	Bal.
Xe/(Xe + Kr)	0.8850	0.8893	0.8886
• Gamma Count ⁸⁵ Kr (atoms/cm ³) ^b	1.46×10^{17}	3.7×10^{16}	3.3×10^{16}

^a ND = None detected

^b Corrected for decay

Fission gas release fractions were calculated and are presented in Table 3.4-6 along with those from comparable rods tested earlier in the GE Test Reactor (GETR)³. The Table also shows the pellet location (radial distance) at which equiaxed grain growth and columnar grain growth occurred. These locations were determined from metallographic sections taken at peak power locations. The grain growth data from both sets of fuel rods are quite similar, but the fission gas release fractions from the rods tested in the R2 reactor are consistently lower than those from the rods tested in GETR. At best, part of this difference can be explained qualitatively in terms of the different neutron flux profiles in the two reactors. The GETR flux profile is somewhat broader and flatter than that of R2, so that at identical peak powers, the integrated rod powers in GETR (hence fission gas release fractions) are somewhat higher than in R2.

The grain growth data indicate that both sets of rods experienced comparable peak powers. The fission gas release fractions are not so directly comparable, but it is clear that the rods tested in R2 did not experience higher powers than those tested in GETR. Therefore, the higher permanent cladding strains observed on the R2 rods (Table 3.4-5 of Reference 1) cannot be explained by higher powers. The observation may be due to greater fuel relocation prior to ramping in the higher burnup rods tested in R2.

Table 3.4-6
SUMMARY OF FISSION GAS RELEASE FRACTIONS AND GRAIN GROWTH DATA
FROM RAMP TESTED RODS

Test Reactor	Rod Identification	Design Feature	Ave ^(b) Burnup (MWd/kg U)	Peak ^(c) Power (kW/m)	Time at Peak Power (h)	Fission Gas Release Fraction	UO ₂ Grain Growth in Peak Power Region ^{d e}	
							r_c/r_o	r_e/r_o
R2	SRP-2/1	10 μ m Cu-barrier	13.5	58.2	12	0.109	0.43	0.60
R2	SRP-3/35	Zr-liner	19.3	58.6	12	0.117	0.37	0.60
R2	SRP-3/36	Zr-liner	18.6	58.6	12	0.105	0.37	0.48
GETR	SRP-3/13	5 μ m Cu-barrier	9.1	59.1	24	0.170	0.46	0.55
GETR	SRP-3/14	5 μ m Cu-barrier	10.8	59.1	6	0.187	0.51	0.57
GETR	SRP-3/15	5 μ m Cu-barrier	10.5	59.7	24	0.157	0.30	0.55
GETR	SRP-3/17	Zr-liner	8.9	59.1	6	0.115	0	0.31
GETR	SRP-3/18	Zr-liner	11.2	59.1	24	0.168	0.44	0.57
GETR	SRP-3/19	Zr-liner	10.2	59.4	6	0.180	0.50	0.56

Notes:

(a) All rods fabricated with 0.23 mm diametral gap, 0.71 mm wall and 17 atm of He fill gas, except SRP-2/1 which had a 0.86 mm wall and was filled with 1 atm He

(b) Burnup is based on Nd measurement

(c) R2 powers measured calorimetrically to $\pm 5\%$

GETR powers measured neutronically to $\pm 10\%$

(d) r_c = radius of columnar grain growth in peak power region

r_e = radius of equiaxed grain growth in peak power region

r_o = radius of pellets before irradiation (10.87 mm for all rods, except SRP-2/1 where $r_o = 10.57$ mm)

(e) GETR rod grain growth radii re-estimated by current observer

3.4.1.6 Burnup Analysis

Chemical burnup analyses (neodymium) were performed on the three sound rods and the results are shown in Table 3.4-7. The measured burnups on the Zr-liner rods, SRP-3/35 and SRP-3/36 were substantially higher than the average burnups previously reported (cf Table 3.4-4), which were based on the results of ^{137}Cs gamma scans. Part of the difference can be attributed to the axial peak burnup location of the burnup samples and to the additional exposure during the standardization phase of the R2 tests. However, even when credit is taken for these factors (Table 3.4-8), the SRP-3/35 and SRP-3/36 burnups were underpredicted by 9.4 and 15.8 percent, respectively. It is believed now that the imprecision in the burnup predictions arose because insufficient cooling time was allowed prior to performing the ^{137}Cs gamma scans. The Zr-liner rods plus two Cu-barrier rods, SRP-3/33 and SRP-3/34, were retrieved from the SRP-3 bundle and characterized on a very tight schedule to meet the shipping deadline. From theoretical considerations it has been shown that a rod should be cooled for nine months to obtain an accuracy of $\pm 5\%$ on burnup by the ^{137}Cs gamma scan method. All the other rods in the test matrix met this criterion; note the good agreement between predicted and measured burnups on SRP-2/1.

Because rod powers in R2 are determined calorimetrically, the underprediction of rod exposures has a negligible effect on test results. However, the amended burnup values increase confidence in the performance of the Zr-liner design, because PCI resistance has been demonstrated at higher exposures than previously thought.

Table 3.4-7
BURNUP ANALYSES FOR THREE SOUND RODS

SRP	Axial Location (inches)	Measured Burnup (Neodymium Method) (MWd/kg U)	Uncertainty (1 σ)* (MWd/kg U)
2/1	12.55-13.05	14.5	± 0.74
3/35	12.35-12.85	20.2	± 0.59
3/36	12.40-12.90	19.0	± 0.34

* Burnup uncertainty reflects the precision of the average burnup based on the three neodymium isotopes measured.

Table 3.4-8
COMPARISON OF CALCULATED AND MEASURED BURNUPS

Rod Identification	Design	Burnup (MWd/kg U)				
		Power Reactor		R2 Reactor	Total	Measured
		Avg ^a	Peak ^a	Peak ^b	Peak	Peak ^c
SRP-2/1	Cu-Barrier	12.4	13.3	0.80	14.1	14.5 $\pm$ 0.74
SRP-3/35	Zr-liner	16.6	17.4	0.88	18.3	20.2 $\pm$ 0.59
SRP-3/36	Zr-liner	15.3	15.6	0.42	16.0	19.0 $\pm$ 0.34

^a From ^{137}Cs gamma scans

^b From R2 Reactor data

^c From Nd isotopic measurements

3.4.1.7 Metallography

3.4.1.7.1 Zr-Liner Rods, SRP-3/35 and SRP-3/36

Reviewing the metallography reported previously,¹ it was shown that the PEC signal in the sound Zr-liner rod, SRP-3/35, was probably caused by a fabrication artifact. This was a thin sliver of zirconium which had mechanically bonded to the liner inner surface. A more important observation was the formation of a large number of microcracks at the cladding inner surface. The typical appearance of these microcracks in the as-polished condition is shown in Figure 3.4-10. Additional metallographic sections were selected and the number of microcracks was correlated (Table 3.4-9) with local cladding strain and nodal power. A permanent strain threshold of ~ 0.2 percent was established for microcrack formation.

During the current period similar microcracks were observed at a high strain location in SRP-3/36. After a light etch the sample was examined under polarized light, which delineated grain structure. It was judged that 80 percent of the microcracks were intergranular (Figure 3.4-11).

Another section was taken from the high cladding strain region of SRP-3/35. The fuel was removed and the cladding ring, which was about 8 mm long, was longitudinally slit in half. One part was mechanically flattened and mounted for examination of the inner surface by scanning electron microscopy (SEM). A fairly dense array of cracks covered the entire surface (Figure 3.4-12). The jagged nature of the cracks plus apparent grain facets, visible in the more open cracks, support other evidence that the microcracks in the Zr-liner were predominantly intergranular. The axial length of most cracks was in the range of 300–400 μm . This SEM specimen was subsequently mounted and examined in cross-section. Although the mechanical flattening has opened up the cracks, they had not propagated any further through the liner.

The nature and cause of the Zr-liner microcracks are still under discussion. A few microcracks were noted in the destructive examinations of SRP-3/17, SRP-3/18, and SRP-3/19, but were believed to be fabrication related. These rods survived ramp tests to ~ 18 kW/ft (59 kW/m) in the GETR at burnups of 7.9, 10.1 and 8.8 MWd/kg U, but as noted previously,¹ the maximum permanent cladding strains were only 0.1, 0.2 and 0.2 percent, respectively, i.e., at the threshold for microcrack formation (see Table 3.4-9). Features similar to microcracks were noted in CC-22A and CC-22B,⁵ two Zr-liner rods tested in the Collapsed Clad test series (Accelerated PCI Tests). These rods survived ramps to 22.6 and 22.1 kW/ft (74.1 and 72.5 kW/m) at burnups of 5.3 and 4.7 MWd/kg U, respectively. Permanent cladding strains up to ~ 2 percent were measured at ridge locations in these thin cladding, tight gap rods and two types of flaw in the zirconium liner were reported. Some had the appearance of grooves or scratches in the soft liner material, but others were small, tight defects, very similar in appearance to the microcracks in the SRP rods. The depth (~ 12 μm) and distribution of these microcracks did not appear to be a function of nodal burnup or peak power. Neither were they an obvious function of cladding strain in the range 0.6 to 2.0 percent. The cladding used in both the CC and SRP rods was fabricated from the same batch of crystal bar zirconium liner tubing, although the final heat treatment in the former was a stress relief anneal and the liner had a smaller grain size than in the SRP cladding which had a recrystallization anneal.

It was noteworthy that the microcracks in both the CC and SRP rods showed no tendency to propagate through the zirconium liner, and it has been hypothesized¹ that their depth of penetration (~ 12 μm) is related to the range of fission fragments and/or oxygen diffusion into the zirconium liner.

The appearance (Figure 3.4-10) and thickness (1–4 μm) of oxide patches on the cladding inner surface of SRP-3/35 and SRP-3/36 were similar to those reported previously. The thickness of the corrosion film on the outer surface was measured at several axial locations, and hydrogen concentrations in the cladding were estimated from hydride precipitate densities. Typical values are presented in Table 3.4-10, including some from lower exposure Zr-liner rods measured earlier.⁴

Table 3.4-9
ZIRCONIUM LINER MICROCRACKING VERSUS LOCALIZED DIAMETRAL CLADDING STRAIN

Fuel Rod and Met Section^a	Nodal Peak Power (kW/ft)	Average Diam. Strain (%)	Ridging Diam. Strain (%)	Total Diam. Strain (%)	Number of Microcracks
3/35-E	18.0	0.53	0.16	0.69	110-148 ^c
3/35-F1	17.8	0.45	0.20	0.65	115-293 ^c
3/35-G	17.1	0.53	0.20	0.73	110
3/35-H2	16.0	0.43	0.20	0.63	170
3/35-H3	15.4	0.32	0.12	0.44	83
3/35-H5	14.2	0.20	0.20	0.40	41
3/35-H6	13.3	0.08	0.12	0.20	9
3/35-I	9.3	None	None	None	None
3/35-K ^b	—	None	None	None	None
3/36-B	17.8	0.47	0.12	0.59	135

^a See Reference 1 Figure 3.4-8

^b Plenum (non-fueled) region

^c Several planes examined

Table 3.4-10
Zr-LINER CLADDING CORROSION AND HYDROGEN PICKUP

Rod Identification	Avg. Burnup^a (MWd/kg-U)	Waterside Corrosion Film Thickness (μm)		Estimated H₂ Content^c (ppm)
		Average^b	Range	
SRP-3/35	16.6	12.2	9.9-18.6	80-120
SRP-3/36	15.3	11.4	7.3-14.7	30-120
SRP-3/17	7.9	10.4	7.7-12.1	80-120
SRP-3/18	10.1	10.3	7.9-15.0	110-150
SRP-3/19	8.9	8.2	5.3-11.6	60-80

^a Uncorrected per Section 3.4.1.6

^b Average of 24 measurements at different circumferential locations

^c Range reflects concentration gradient across cladding

C578-07
500X

Figure 3.4-10. Typical Appearance of Microcracks in Zr Liner Cladding

Figure 3.4-11. Predominantly Intergranular Microcracking in Zr Liner

a) 20650

300X

AXIAL DIRECTION

b) 20651

1000X

Figure 3 4 12 Scanning Electron Microscopic Examination of Inner Surface of Flattened Zr Liner Cladding Sample (SRP-3/35) Showing Microcrack Features

Estimated hydrogen concentrations were comparable with previous results, which gave corrosion hydrogen pickup fractions in the range of 0.24–0.31. As noted before, in the absence of a strong temperature gradient across the cladding (i.e., in a plenum section), the corrosion hydrogen tended to concentrate in the zirconium liner adjacent to the liner-Zircaloy interface (Figure 3.4-13), but the hydride layer was over twice as thick as previous observations, i.e., thicker than would be predicted from the increase in local corrosion film thickness alone (~30 percent thicker). This hydrogen concentration effect is substantially reversed by the temperature gradient in fueled sections (Figure 3.4-14).

It was also noted (Figures 3.4-13 and 3.4-14) that in some locations the thickness of the zirconium liner was in the range 1.5–1.8 mils (38–46 μm).

3.4.1.7.2 Reference Rods, SRP-2/11 and SRP-2/14

Two sections from SRP-2/11 and one from SRP-2/14 were examined in areas where PCI defects were indicated in the visual examination and confirmed by PEC signals. A through-wall crack was found in the former rod (Figure 3.4-15) and cracks penetrating 80 percent of the wall were seen in the latter (Figure 3.4-16). Some crack opening and necking at the crack tip can be discerned in Figure 3.4-16. This causes the characteristic X-mark features that are sometimes visible on a defective rod, marking the locations of PCI cracks.

3.4.1.7.3 Cu-Barrier Rods, SRP-2/1 and SRP-3/33

Metallographic sections were examined from locations where PEC signals indicated through-wall defects in the failed rod, SRP-3/33, and a flaw or other feature on the inner surface of the cladding in SRP-2/1, which was sound. Typical PCI cracks were located in the former (Figure 3.4-17), indicating that Cu-barrier fuel may not be immune to this failure mechanism, if the test is sufficiently severe. Extensive delamination of the thin (5 μm) copper layer had occurred (Figure 3.4-18) and there were large quantities of extraneous copper plus a Cu reaction phase in the fuel-cladding gap. These features are a consequence of failure and coolant ingress.

No significant features were found in the sound rod, SRP-2/1, to account for the PEC signals. The general appearance of the 10 μm copper layer was good (Figure 3.4-19) with no observable delamination. As in previous examinations, embedded fuel particles were a common feature with occasional instances of local cladding oxidation where a particle had penetrated the copper layer and contacted Zircaloy (Figure 3.4-20). There were several areas where the copper layer appeared to have been thinned by interaction with the fuel (Figure 3.4-21) and it is suspected that these were responsible for the PEC signals.

A light etch (Figure 3.4-22) revealed the Cu-Zr intermetallic layer between barrier and cladding, which is barely discernible in the as-polished condition. The thickness of this layer was estimated to be ~1.2 μm , comparable to that in lower exposure rods examined previously. Similarly, in those regions of the SRP-3/33 sections, where the copper had not delaminated, there was no evidence of substantial intermetallic layer formation.

3.4.2 Fuel with Cladding Perforation (J. H. Davies)

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.^{1,2} 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.¹ The locations of the rods in the test rig and their relative powers are shown in Figure 3.4-23. During the current period the irradiation was completed and the fuel rods were examined visually and by neutron radiography. The results are hereby presented and discussed.

(a) OUTER SURFACE

C581-15

250X

(b) INNER SURFACE

C581-13

250X

Figure 3.4-13. Hydride Concentration in Zircaloy-2 and Zirconium Layer, Plenum Region (SRP-3/35, Etched)

(a) OUTER SURFACE C582-46

250X

(b) INNER SURFACE C582 44

250X

Figure 3.4-14. Hydride Concentration in Zircaloy-2 and Zirconium Layer, Fuel Region (SRP-3/35, Etched)

C601-02

50X

Figure 3.4-15. PCI Defect in Failed Reference Rod, SRP-2/11

C597-03

50X

Figure 3.4-16. PCI Defect in Failed Reference Rod, SRP-2/14

C604-03

250X

Figure 3.4-17. PCI Defect in Failed Cu-Barrier Rod, SRP-3/33

C603-03

250X

Figure 3.4-18. Delamination of Copper Barrier plus Copper Debris in the Failed Rod, SRP-3/33

C598-06

1000X

Figure 3.4-19. Typical Appearance of Copper Barrier in Sound Rod, SRP-2/1

C600-07

500X

Figure 3.4-20. UO₂ Fuel Particle Embedded in Copper Barrier in SRP-2/1

C600-06

500X

Figure 3.4-21. Thinned Region of Cu Barrier in SRP-2/1 (Probable Cause of PEC Signal)

Figure 3.4-22. Zr-Cu Intermetallic Layer Formed Between Barrier and Cladding in SRP-2/1

3.4.2.2 Activity Release Results

The test started on November 22, 1979. It continued through six cycles until March 17, 1980 with an intermission from December 21 to January 17 between the 2nd and 3rd Cycles. The output of the controlling area gamma monitor (R23) and mean rod peak power are plotted versus time in Figures 3.4-24 through 3.4-26. Loop activity rose steadily throughout the first period to twice initial levels followed by a step increase at the start of the second Cycle. This step is attributed to the well-known "spiking" phenomenon⁷ — enhanced release of fission products, especially iodine, following both shutdown and subsequent startup of reactors with defective fuel. The activity level registered by R23 subsequently decreased during the second Cycle and at the end was roughly equivalent to that at the end of the first Cycle. Coolant activity then remained relatively constant throughout the remainder of the experiment except for another release "spike" associated with a reactor scram on January 28. This overall picture of Loop activity history is consistent with the results of the water sample analyses, which are given in Table 3.4-11. There was little change in either total activity or fission product spectrum from Cycle to Cycle. (At least two loop coolant samples were collected and analyzed for each reactor cycle. One sample was taken at a convenient time close to the end of the cycle and another during shutdown).

Fine structure in the data from the Loop activity monitors were read from the strip chart record, which was generated continuously during the course of the experiment. Examples are presented in Figures 3.4-27 through 3.4-30, which provide useful insights into the physical nature of fission product release from defective fuel.

Figure 3.4-27 shows an example of activity "pulsing", which corresponds to small periodic decreases in Loop pressure (from ~8.2 to 8.1 MPa). This effect, which was first reported by Forsyth, et al⁸ and noted in our previous report¹ is probably a major mechanism for release of fission products from defective fuel. The output of the controlling activity monitor (R23) is not linear, but logarithmic; and the pulsing in Figure 3.4-27, for example, represents an increase from ~8 to 11 mR/h following each small pressure step. Activity pulsing was also visible in the Cerenkov detector (R21) output after the built-in delay of a minute or two. The pulsing effect was not always observed, however.

— CORE POSITION B3

— NO SPECIAL ROTATIONAL ORIENTATION
OF THE DEFECT HOLES

RELATIVE POWER DISTRIBUTION,
AVERAGE POWER OF THE FOUR RODS BEING 1.00

<u>ROD No.</u>	<u>DESIGN FEATURE</u>	<u>HOLE SIZE (mm)</u>	<u>RELATIVE POWER*</u>
DTC 2105	REFERENCE	0.15	1.06
DTC 2602	Zr-LINER (CRYSTAL BAR)	0.15	1.06
DTB 2407	Zr-LINER (CRYSTAL BAR)	0.25	0.94
DTB 2513	Zr-LINER (SPONGE)	0.15	0.94

Figure 3.4-23. Rod Positions in the Irradiation Rig

* AVERAGE POWER OF FOUR RODS $\equiv$ 1.00

Figure 3.4-24. Trend of Loop 2 Coolant Activity Through First Two Cycles

Figure 3.4-25. Loop 2 Coolant Activity — 3rd and 4th Cycle

Figure 3.4-26. Loop 2 Coolant Activity — 5th and 6th Cycle

Table 3.4-11
WATER SAMPLE ACTIVITY ANALYSIS — DATA SUMMARY

A. During Operation

Sample Date		3 Dec 79	19 Dec 79	28 Jan 80	11 Feb 80	28 Feb 80	17 Mar 80
Nuclide	Half-Life	Activity (Ci/ml)					
¹³³ Xe	5.29 d	1.44×10^{-4}	1.55×10^{-4}	2.20×10^{-4}	2.24×10^{-4}	1.76×10^{-4}	1.75×10^{-4}
^{133m} Xe	2.26 d	2.87×10^{-6}	2.86×10^{-6}	3.56×10^{-6}	4.73×10^{-6}	3.66×10^{-6}	3.66×10^{-5}
¹³⁵ Xe	9.2 h	9.09×10^{-5}	8.42×10^{-5}	9.42×10^{-5}	10.7×10^{-5}	7.64×10^{-5}	7.36×10^{-5}
^{85m} Kr	4.48 h	1.46×10^{-5}	1.25×10^{-5}	1.20×10^{-5}	1.30×10^{-5}	0.99×10^{-5}	0.95×10^{-5}
⁸⁷ Kr	1.27 h	1.11×10^{-5}	0.96×10^{-5}	1.07×10^{-5}	1.08×10^{-5}	0.76×10^{-5}	0.69×10^{-5}
⁸⁸ Kr	2.80 h	1.28×10^{-5}	1.29×10^{-5}	1.34×10^{-5}	1.30×10^{-5}	1.06×10^{-5}	1.02×10^{-5}

B. Following Shutdown

Sample Date		5 Dec 79	30 Jan 80	14 Feb 80	4 Mar 80
Time After Shutdown (hours)		13	15	30	22.6
Nuclide	Half-Life	Activity (Ci/ml)			
¹³³ Xe	5.29 d	2.25×10^{-6}	2.00×10^{-6}	1.90×10^{-6}	5.97×10^{-8}
^{133m} Xe	2.26 d	—	2.45×10^{-8}	—	—

Figure 3.4-28 shows the effect of depressurization from 8.2 MPa to approximately 3.0 MPa at 180°C during the shutdown at the end of the 4th Cycle. (This was done to allow some work on the Reactor vessel). The R23 monitor registered a seven-fold increase from about 1 mR/hr to ~7 mR/hr.

Figures 3.4-29 through 3.4-31 show the effects of power changes during operation on activity release. In Figure 3.4-29, the change in Reactor power shown is estimated to have resulted in a mean rod power change from 32 to 35 kW/m. A barely discernible increase of ~1 mR/h can be determined in the R23 output on top of the pressure "pulsing" effect. Figure 3.4-30 shows an activity change, as registered on R23, from 14 mR/h to 20 mR/h following a 10-minute negative power transient down to ~24 kW/m. Figure 3.4-31 shows the effect of the Reactor scram and subsequent startup that occurred on January 28. There was little impact on the activity monitors due to the scram itself, but they registered large activity bursts following startup. The R23 level increased from ~11 mR/h to ~45 mR/h and then slowly decreased to pre-scram levels.

3.4.2.3 Non-Destructive Post-Irradiation Examination

The rods were examined visually and by neutron radiography at the Studsvik site. General visual inspection was performed at a magnification of 14× while the rod was slowly rotated and moved axially until a feature of interest was seen. The feature was then further inspected and photographed at a suitable magnification between 2.5× and 72×.

All of the rods had a loose crud layer along one side (about 130° of arc and from 20 mm to ~400 mm above the lower end plug weld). Since there were no special requirements for circumferential orientation of the individual rods, it is not known whether the crudded surfaces were those facing the rig centerline or those facing the surrounding shroud tube. The crud was wiped off and has not been analyzed, but from previous experience in this loop it was believed to be an iron-based crud with a superficial density of ~1 mg/cm². The artificial defects in each rod were easily located and photographed. The holes in DTC 2105 (Reference) and DTB 2513 (Sponge

Figure 3.4-27. Recorder Chart Output — 18 January 1980, Correlation Between Loop Activity and Loop Pressure

Figure 3.4-28. Recorder Chart Output — 13 February 1980, Effect of Depressurization on Loop Activity at Zero Power

Figure 3.4-29. Recorder Chart Output — 31 January 1980, Effect of Small Power Increase on Loop Activity

Figure 3.4-30. Recorder Chart Output — 8 February 1980, Effect of Negative Power Transient on Loop Activity

Figure 3.4-31. Recorder Chart Output 28 January 1980, Reactor Scram

Liner) are shown in Figure 3.4-32 prior to wiping the rods. There were no distinctive differences in appearance among the four rods.

After visual examination the rods were inserted into aluminum capsules and neutron radiographed. The fuel in all four rods appeared quite normal, although central voids were visible in the two rods irradiated in the higher power positions (see Figure 3.4-23). Nothing remarkable was noted around the levels of the holes.

Some hydrogen in the cladding of rod DTC 2602 was evident in the neutrographs and hydrogen concentrations were also visible in the cladding of rod DTB 2407, particularly at pellet interface locations; but there was no evidence of "sunburst" type hydride damage in any of the rods. Microdensitometer scans across the radiographs at an arbitrarily selected level (27 cm above the lower end of the fuel) are shown in Figure 3.4-33. Some difference between rod DTC 2602 and the reference rod, DTC 2105, is apparent.

3.4.2.4 Discussion

The artificially defected Zr-liner fuel rods operated successfully for ~82 days at ~35 kW/m (10.7 kW/ft). The estimated burnup is ~3 MWd/kg. Neither activity release data nor post-irradiation examination have given any indication of unusual deterioration. After a steady rise through the first cycle the activity release rate appeared to reach equilibrium and remained relatively constant for the remainder of the irradiation. Locke⁹ has stated that the activity release rate from "failed" fuel (i.e., severe secondary damage) has a doubling time of about 20 days. Moreover, there was no indication of severe secondary damage, and the appearance of the Zr-liner rods was not substantially different from that of the control standard.

It was not the objective of the experiment to study fission product release rates, but these data were collected for monitoring purposes and were available for interpretation and comparison with those from similar experiments.

Radioactive fission product release from defective fuel is usually treated in terms of Release Fraction, R/B , where R is release rate and B is birth rate.

In a closed system like a reactor loop

$$R(\text{Ci/s}) = (\text{Ci/ml})(\text{Coolant Volume})(\lambda_1 + \lambda_2) \quad (1)$$

where

λ_1 is fission product decay constant

λ_2 is the system cleanup or leakage time constant

The quantity λ_2 has not been precisely determined in this experiment, but it can be estimated from the background sample analyses (Table 3.4-11). Except in one case, the effective half life for stripping is ~2 hours, which is consistent with the Studsvik estimate of 60 kg/h coolant circulation flow through the pressurizer, if it is assumed that the fission gas activities are completely removed in this process. A value of $8.75 \times 10^{-5} \text{ s}^{-1}$ was assigned to λ_2 (i.e., effective half life for stripping = 2.2 h.)

The birth rate, B , of a particular fission product isotope is given by:

$$B(\text{Ci/s}) = \frac{(\text{fissions/s}) \times Y \lambda_1}{3.7 \times 10^{10}} \quad (2)$$

where Y is the fission yield.

(a)

72X

↑
COOLANT
FLOW

(b)

72X

Figure 3.4-32. Post-Irradiation Appearance of Artificial Defects in: (a) Zr-liner (Low-Oxygen Sponge Zirconium) Rod, (b) Reference Rod.

Figure 3.4-33. Results of Microdensitometer Scans Across the Radiographs

Precise calculations of the fission product inventories are to be provided by Studsvik. In the meantime, the fission rate has been calculated from first principles assuming a constant mean rod linear power of 34 kW/m and a total fuel length of 1.575 m (4 rods). The fission rate was estimated to be 1.63×10^{15} fission/s.

Calculated fractional release rates (R/B ratios) are presented in Table 3.4-12 and $A/Y\lambda$ (which is proportional to R/B) is plotted vs. λ_1 on log-log paper in Figure 3.4-34. The slope of the graph, $-b$, is a characteristic of the type of fission product release. A value of $b = 1.15$, determined here, is indicative of a fission gas mixture approaching equilibrium. For comparison McDonald and Lipsett¹⁰ reported a b-value between 0.99 and 1.13 for an artificially defected CANDU fuel element operating at 48 kW/m with a 1.2 mm hole opposite the fuel column. Chenebault, et al¹¹ reported a b-value of 1.06 for fuel operating with a hole in the plenum region. In another experiment, where a 15 mm² slit was machined in the fueled part of the cladding during operation at 40 kW/m, a b-value of 0.76 was determined, showing a change in the type of fission product release.

Note the high fractional release (0.52) of ¹³³Xe in Table 3.4-12. There is considerable uncertainty in the determination of this value, mostly due to the uncertainty in λ_2 , the "stripping" time constant. However, the high release fraction is not inconsistent with other experience with defected fuel. Comparable fractional releases for ¹³³Xe, ^{85m}Kr and ⁸⁷Kr were determined¹¹ from operating fuel with plenum defects. It is generally accepted that release rates in defected fuel are one or two orders of magnitude higher than in sound fuel.^{11, 12}

The fuel rods have been shipped back to the Vallecitos Nuclear Center, Pleasanton, California, for destructive examination.

Table 3.4-12
RELEASE FRACTIONS — 1ST CYCLE

Fission Product	Decay Constant (s ⁻¹)	Fission Yield	R ^(a b) (Ci/s)	B ^(c) (Ci/s)	R/B
¹³³ Xe	1.52×10^{-6}	0.0677	2.56×10^{-3}	4.93×10^{-3}	0.52
^{133m} Xe	3.55×10^{-6}	0.00191	5.20×10^{-5}	3.25×10^{-4}	0.16
¹³⁵ Xe	2.10×10^{-5}	0.0663	1.97×10^{-3}	6.67×10^{-2}	0.029
^{85m} Kr	4.30×10^{-5}	0.0131	3.81×10^{-4}	2.70×10^{-2}	0.014
⁸⁷ Kr	1.48×10^{-5}	0.0254	5.23×10^{-4}	1.80×10^{-1}	0.003
⁸⁸ Kr	6.88×10^{-5}	0.0358	4.00×10^{-4}	1.18×10^{-1}	0.003

NOTES:

(a) From Equation 1 assuming coolant mass of 200 kg and $\lambda_2 = 8.75 \times 10^{-5} \text{ s}^{-1}$

(b) Water sample activity data on 3 December 1979 from Table 3.4-11

(c) From Equation 1 assuming fission rate of 1.63×10^{15} fission/sec

Figure 3.4-34. Relationship Between F.P. Release Rate and Decay Constant

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 BARRIER SELECTION

4.2.1 Introduction

An important development in this report period was the selection of the barrier fuel design which will be demonstrated. The selection was made as a recommendation by General Electric to the Program Steering Committee, which has representation from the U.S. Department of Energy, Commonwealth Research Corporation and General Electric. Completion of the selection, including formal approval by the Steering Committee was completed in February, 1980 to allow for a timely start of the fuel design work.

4.2.2 Selection

The barrier fuel selected for the demonstration is of the Zr-liner type. The principal reason for this selection is based on the ability of such fuel to resist failure by pellet-cladding interaction (PCI) and the availability of data on a timely basis to support the demonstration on its projected schedule (Quad Cities Unit 2, Cycles 6 through 9). However, other reasons were also involved in this selection. The following factors contributed to this selection.

1. PCI Resistance

The results of the power ramp tests in the R2 Test Reactor have shown that fuel with a liner of crystal bar zirconium has excellent resistance to PCI at burnup levels to >20.0 MWd/kg U, even in very severe ramp tests to 59 kW/m (18 kW/ft). Subsequent examinations of those tested fuel rods have confirmed the resistance to PCI¹ and have shown no tendency for the PCI resistance of Zr-liner fuel to diminish with burnup, at least to the burnup levels tested to date. Furthermore, these examinations have shown that the power ramp tests are more severe than those in service; the tests have caused large plastic strains ($>0.5\%$ in the circumferential direction) of the zirconium-lined cladding. Despite these large strains and the release of fission products which accompanies the large and sudden increase in power, neither failure nor incipient failure was seen. The excellent PCI resistance of this fuel type is very encouraging.

By contrast, Cu-barrier fuel with copper plated on etched Zircaloy-2 tends to lose its PCI resistance^{1,3} at burnups >14 MWd/kg U. The most recent test reactor results (1979 power ramp tests)¹ support the premise that the mechanism whereby the Cu-barrier fuel loses its good resistance to PCI is by interdiffusion of copper and Zircaloy. Even prior to the start of this program, the possibility for such a wear-out mechanism was recognized, and the Cu-barrier with copper plated on oxidized Zircaloy-2 (Cu-on-oxide) was introduced as a means to mitigate the interdiffusion and to retain good PCI resistance at high burnup.⁵ The test data to date on the Cu-on-oxide barrier fuel are encouraging, and Cu-on-oxide remains a possible PCI remedy.² However, the available test data are at low burnup, and there is no possibility of obtaining data on a timely basis to support the presently projected demonstration.

2. Licensing

Both the Cu-on-oxide and the Zr-liner barrier fuels are considered to be licensable. Both have performed well in simulated loss of coolant accident (LOCA) tests^{1,3} and in the reactivity initiated accident (RIA) tests.⁴ However, the Cu-barrier forms a eutectic phase during LOCA.³ While this metallurgical change did not affect the performance in the LOCA tests, the phenomenon raises questions regarding response to various power transient situations. Therefore, the Zr-liner fuel is considered to have an advantage with regard to licensing of the demonstration itself as well as for future commercial usage.

3. Material Availability

Material availability is not a problem for Cu-barrier fuel, but use of crystal bar zirconium must be considered from this perspective. At present, the production capacity for crystal bar zirconium in the United States is estimated to be approximately 10^3 kg/year, and this demonstration would take roughly one year's production if the entire demonstration were of crystal bar Zr-liner fuel. However, it is neither necessary nor desirable to use crystal bar zirconium for all of the demonstration bundles. The data on PCI resistance justifies the demonstration of the fuel with the crystal bar zirconium, but it is prudent to include the low-oxygen sponge Zr-liner fuel so that it can be power ramped when and if the test reactor data prove its PCI resistance at relevant burnup levels. Until that time, they can be operated in the demonstration as normal fuel bundles. Also, fresh low-oxygen sponge Zr-liner bundles can be used as needed to boost the power in the test cells in Cycles 8 and 9 should that prove necessary; the available power ramp test data¹ is already sufficient to support power ramps at low burnup.

4. Fabrication and Quality Assurance

Experience to date indicates that the Zr-liner fuel is easier to fabricate than is the Cu-barrier fuel. Copper plating on oxidized Zircaloy is a new technology, involving fabrication operations which have not been qualified for high volume production. The fabrication of Zr-liner fuel uses conventional equipment and techniques. Thus, in ease of fabrication the Zr-liner fuel has a short term advantage over the Cu-barrier, especially as it relates to the demonstration itself.

5. Impact on Uranium Utilization

The selection of the Zr-liner barrier fuel over the Cu-barrier is consistent with the goal of optimizing uranium utilization. Compared with conventional fuel, the Cu-barrier fuel with a $10\text{ }\mu\text{m}$ copper layer has a reactivity penalty^{4,5} of approximately 0.25%. The Zr-liner barrier fuel has no reactivity penalty.

The reactivity penalty associated with the Cu-barrier fuel impacts all aspects of the total fuel economy, and in particular, it affects adversely, the utilization of uranium resources. The extent of that effect depends on circumstances and any estimate depends on the assumptions made. For example, in fuel designed for target exposure of 28.4 MWd/kg U in a commercial BWR, the reactivity penalty of 0.25% can result in an energy shortfall estimated to be ~ 0.4 MWd/kg U, or $\sim 1.4\%$. To accommodate this energy difference by additional fuel (enrichment held constant) would result in use of $\sim 1.4\%$ more fuel material (U_3O_8) to produce the same amount of energy. This increment amounts to approximately 14.7 kg U_3O_8 per bundle for fuel with average enrichment of 2.75%. However, if the reactivity difference were made up by more initial enrichment, the impact on the use of U_3O_8 can be reduced to roughly 11.5 kg per bundle. Regardless of the assumptions the reactivity penalty associated with the Cu-barrier results inevitably in the less efficient use of uranium resources.

6. Reprocessing

While reprocessing is not an issue presently or in the immediate future, it is a potential concern in the choice of a fuel design for the future. The Cu-barrier fuel presents the fuel reprocessor with a somewhat different chemical situation compared with conventional light water reactor fuel. Thus, an additional cost for reprocessing Cu-barrier fuel might be anticipated. By contrast, the Zr-liner fuel can be reprocessed exactly as is conventional fuel. No cost penalty or difficulties are anticipated with Zr-liner fuel either in reprocessing or in any other operation in the back end of the fuel cycle.

Based on these recommendations, the Program Steering Committee approved the following specific actions for the demonstration:

1. Proceed on the currently projected demonstration schedule (Quad Cities Unit 2, Cycles 6-9) with Zr-liner fuel as the demonstration fuel.
2. Include both crystal bar Zr-liner fuel and low-oxygen sponge Zr-liner fuel which are segregated into their respective fuel bundles. That is, the demonstration fuel bundles will be either of crystal bar or of sponge but the two liner grades are not to be mixed within a single bundle.
3. Fabricate bundles of low-oxygen sponge Zr-liner fuel and of crystal bar Zr-liner fuel in approximately equal numbers.
4. Retain the option to ramp only the crystal bar bundles or both the crystal bar and the low-oxygen sponge bundles, depending on the test reactor ramp test results on sponge Zr-liner test fuel rods as the data become available.

4.3 SUBTASK III.1. DESIGN AND LICENSING OF DEMONSTRATION FUEL (J. A. Baumgartner and C. C. Guitton, NfSED)

4.3.1 Introduction

As was mentioned in Section 2 of this report, the demonstration is planned for Quad Cities Unit 2 starting in Cycle 6. This will require delivery of the 132 bundles of barrier fuel in August of 1981. Specific design, manufacturing and licensing schedules are now being implemented consistent with that delivery date.

4.3.2 Mechanical Design

The mechanical design of the barrier demonstration fuel was defined to be identical to the prepressurized 8 × 8 D-lattice retrofit fuel (P8 × 8R) except for the high-purity zirconium liner metallurgically bonded to the inside surface of the Zircaloy-2 cladding. The overall tube wall thickness including the liner was maintained identical to the P8 × 8R tube wall thickness.

The design and licensing analyses performed showed that the barrier demonstration bundles meet the same thermal-mechanical requirements as those applied to the generic P8 × 8R design.¹³

4.3.3 Nuclear Design

The proposed barrier demonstration was designed to include 132 zirconium liner bundles, 64 of which were scheduled to be power ramped in four separate groups at the end of four successive reactor cycles.

The nuclear configuration of the barrier demonstration bundles was specially designed to provide sufficient power levels during the proposed ramps without adversely affecting the general operating procedures of the demonstration reactor.

The nuclear designs of the barrier bundles were divided in four groups. These groups are represented on Figures 4.3-1 through 4.3-4. The distribution shown on Figure 4.3-4 is also that of the non-barrier fuel bundles used to make up a total fuel reload batch. These particular enrichment distributions were specifically designed for the demonstration.

4.3.4 Licensing

On May 15, 1980 the generic bundle license document for the demonstration fuel bundles was submitted to the U.S. Nuclear Regulatory Commission. That document, if approved, will allow operation of the 132 barrier bundles in a reactor of the type BWR/3.

4.4 SUBTASK III.2. FABRICATION OF DEMONSTRATION FUEL (D. E. Allen, R. E. Donaghy, R. J. Mack — WMD)

4.4.1 Introduction

Work at the Wilmington Manufacturing Department (WMD) is being conducted in several discrete activities as follows:

- Cezus development tubeshells
- Manufacturing development run (MDR)
- Pre-production run (PPR)
- Production run
- Subprograms

4.4.2 Cezus Development Tubeshells

Four low oxygen sponge zirconium lined Zircaloy-2 tubeshells were received from Cezus (formally Ugine Aciers). These tubeshells represented their initial efforts in the production of coextruded tubeshells.

All WMD processing and evaluation efforts are complete; data from the tubes may be summarized as follows:

- Final tube wall thicknesses meet specification.
- Zircaloy-2/zirconium interface was free from bonding defects or delamination.
- Tensile, burst, corrosion, grain size and hydride orientation properties were all acceptable.
- The inner surfaces (zirconium) contained short shallow longitudinal tears (microfissures) at the various stages of tube reduction.

4.4.3 Manufacturing Development Run

Sixty coextruded zirconium lined tubeshells were procured for the manufacturing development run (MDR) studies. Teledyne Wah Chang Albany (TWCA) supplied forty tubeshells (twenty crystal bar and twenty low oxygen sponge zirconium) and Cezus furnished twenty low oxygen sponge lined tubeshells. The work was split into three stages with the original intent of conducting developmental studies during Stages I and II and performing a confirmation run at Stage III.

W-W							
6	5	4*	4	4	4*	4	5
5	4	2	Gd	2	2	2	4
4*	2	2	2	1	1	Gd	2*
4	Gd	2	2	H ₂ O	1	1	2
4	2	1	H ₂ O	1	1	Gd	2
4*	2	1	1	1	1	1	2*
4	2	Gd	1	Gd	1	2	3
5	4	2*	2	2	2*	3	4

ROD TYPE	WT % (²³⁵ U)
1	3.80
2	3.00
3	2.40
4	2.00
5	1.70
6	1.70
Gd	3.00

*TIE ROD

Figure 4.3-1. Cross Section of Barrier Fuel Bundles of 2.63% Average Enrichment for Power Ramp During First Cycle of Operation

W-W							
6	5	4*	4	4	4*	4	5
5	4	2	Gd	2	2	2	4
4*	2	2	2	1	1	Gd	2*
4	Gd	2	2	H ₂ O	1	1	2
4	2	1	H ₂ O	1	1	Gd	2
4*	2	1	1	1	1	1	2*
4	2	Gd	1	Gd	1	2	3
5	4	2*	2	2	2*	3	4

ROD TYPE	WT % (²³⁵ U)
1	3.80
2	3.00
3	2.40
4	2.00
5	1.70
6	1.70
Gd	3.00

*TIE ROD

Figure 4.3-2. Cross Section of Barrier Fuel Bundles of 2.63% Average Enrichment for Power Ramp During Second Cycle of Operation

W-W							
6	7	5*	4	4	4*	5	6
7	Gd1	4	3	Gd2	2	3	5
5*	4	2	1	1	1	Gd2	4*
4	3	1	1	H ₂ O	1	1	2
4	Gd2	1	H ₂ O	1	1	1	2
4*	2	1	1	1	1	Gd2	3*
5	3	Gd2	1	1	Gd2	1	4
6	5	4*	2	2	3*	4	5

ROD TYPE

WT % (²³⁵U)

1	3.95
2	3.80
3	3.30
4	2.80
5	2.40
6	2.00
7	1.70
Gd1	1.70
Gd2	3.00

*TIE ROD

Figure 4.3-3. Cross Section of Barrier Fuel Bundles of 2.98% Average Enrichment for Power Ramps During Third and Fourth Cycles of Operation

W-W		7	6	5*	5	4	4*	4	5
		6	4	4	3	2	Gd	2	4
		5*	4	Gd	1	1	1	Gd	2*
		5	3	1	1	H ₂ O	1	1	2
		4	2	1	H ₂ O	1	1	1	2
		4*	Gd	1	1	1	Gd	1	2*
		4	2	Gd	1	1	1	1	3
		5	4	2*	2	2	2*	3	4

ROD TYPE

WT % (²³⁵U)

1

3.80

2

3.30

3

3.00

4

2.40

5

2.00

6

1.70

7

1.30

Gd

3.00

*TIE ROD

Figure 4.3-4. Cross Section of Barrier Fuel Bundles and Non-Barrier Fuel Bundles of 2.84% Average Enrichment

4.4.3.1 Stage I

This stage was used largely to investigate the effect of various vendor processing variables and to gain information on the fabricability of the tubeshells. The Stage I tubeshells (fifteen) were processed with the following variables:

- Tubeshells from different positions in the ingot (top, middle and bottom) for both TWCA and Cezus.
- Various vendor pre-coextrusion fitup (billet to liner) clearances.
- Multiple Zircaloy-2 ingots (i.e., two different ingots were supplied by each vendor)

Tube reduction was undertaken using normal procedures but special attention was paid to chamfering and removal of burrs. Upon completion of tube reducing, tubing was evaluated for liner/wall thickness, dimensions, grain size, bondline integrity, and hardness. These attributes were not observed to be sensitive to the process variables delineated above.

Additionally, visual examination (X20) and scanning electron microscopy (SEM) were used to characterize the inner surfaces at all tube reduction stages.

All tube reduction is complete and the tubing has been utilized for the various surface conditioning techniques described in Subprogram A and ultrasonically inspected.

4.4.3.2 Stage II

Nineteen tubeshells were selected representing mainly those ingot positions not tested in Stage I. Again both vendors and all liner types were represented.

Additionally the TWCA tubeshells were divided into two groups; those showing ultrasonic test indications and those free from indications. Blind holes were also drilled in two of the tubeshells in an attempt to mark and track defects as they progress through tube reduction.

Tube reduction was completed without any major problems. At this time, the ultrasonic test results from Stage I tubes became available (see Subprogram E), showing indications of irregularities on the inner surface of the tubing. The Stage II tubes were conditioned on their inside surface in one of the following ways:

- Grit blast followed by dip etch
- Dip etch
- Dip etch followed by grit blast

Ultrasonic inspection of this tubing is currently underway.

4.4.3.3 Stage III

Twenty-five tubeshells are being reduced at this stage. Work is in progress.

4.4.4 Pre-production Run (PPR)

Sixty-five coextruded, zirconium lined tubeshells have been purchased for PPR work. Teledyne Wah Chang Albany is supplying forty-five (twenty-five crystal bar and twenty sponge) zirconium lined tubeshells and Cezus is providing twenty sponge zirconium lined tubeshells.

TWCA tubeshells have been received and are currently undergoing liner thickness measurements (see Subprogram H).

Tube reduction will begin during the third quarter of 1980.

4.4.5 Production

Purchase orders for the following materials have been processed.

TWCA — 129 tubeshells (approx.)
70 tubeshells (approx.)

Cezus — 59 tubeshells (approx.)

Delivery of the tubeshells is scheduled for November 1980.

4.4.6 Subprograms

In addition to the main thrust of assuring a sound manufacturing process for zirconium lined tubing, there are some peripheral issues which have been relegated to subprograms:

A. Inner Surface Finish

Several processes — grit blast, flush etch, dip etch, and combinations of these — are being evaluated with respect to the final dimensions of the zirconium liner and to the resulting surface finish.

B. Chemical Analysis of Liners in Tubeshells

Work is underway to develop working chemical standards for analysis of the oxygen, nitrogen and hydrogen in the liner material. Standards with known amounts of these elements at very low concentrations are not available at present.

C. Chemical Analysis of Liners in Final Tubing

In cooperation with the GE Research and Development Center in Schenectady, New York, several analytical techniques are being evaluated: ion microprobe, x-ray fluorescence, mass spectrometry, optical emission, inert gas fusion, and transmission scanning electronic microscopy with x-ray fluorescence.

D. Tube Shell Defect Tracking

Some defects identified in the tubeshell are being followed and characterized as they are changed by the various tube reductions and anneals to determine which types of defects result in rejectable flaws in the final tubing.

E. Ultrasonic Testing of Tubing

Ultrasonic signals are being evaluated to determine which ones correspond to defects that should be rejected. The Zr-liner tubing tends to have a higher ultrasonic background noise than does conventional tubing. The ultrasonic tests are used to seek out defects on the inner surface of the tubes and to measure tubing dimensions: wall thickness, diameter, and ovality.

F. Nondestructive Measurement of Tubeshell Liner Thickness

An ultrasonic method for measuring the liner thickness of tubeshells has been developed and has proven useful in evaluating the MDR tubeshells.

G. End Plug Welding

Qualification of the procedure for welding end plugs on Zr-liner tubing is in progress. Fabrication of the barrier cladding wall require welding of a dummy end plug so that the outer surface can be autoclave oxidized while protecting the inner surface from the steam. The dummy end plug will then be cut off prior to loading the fuel pellets. The weld and cut-off operations are being qualified for production of the Zr-liner fuel.

H. Tubing Liner Thickness Measurement

An automated optical microscopy arrangement is planned to measure liner thickness of tubing. It involves an automatic metallographic grinding system and an electronic image analyzer. The system has been planned and procedures are being established at this time.

4.5 TASK IV. LEAD TEST ASSEMBLIES (J. R. Thompson, NFSed)

Four barrier fuel lead test assemblies (LTA's) are under irradiation in the core of Quad Cities Unit 1. This set of four LTA's is comprised of one assembly of each of the following types:³

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

Irradiation on the LTA's was begun in February 1979, and to date the bundle average burnup of approximately 11.0 MWd/kg U has been achieved. Plans are in progress for the first interim examination of the LTA's at the scheduled outage in the Fall of 1980. At that time all four assemblies will be inspected visually and by sipping. The two Zr-liner LTA's will be disassembled for a more detailed examination.

5. REFERENCES

1. Demonstration of Fuel Resistant to Pellet-Cladding Interaction PHASE 2, Second Semiannual Report, July-December 1979, compiled by H. S. Rosenbaum, GEAP-25613-2, March 1980.
2. Demonstration of Fuel Resistant to Pellet-Cladding Interaction, PHASE 2, First Semiannual Report, January-June 1979, compiled by H. S. Rosenbaum, GEAP-25163-1, August 1979.
3. Demonstration of Fuel Resistant to Pellet-Cladding Interaction, PHASE 1 Final Report, compiled by H. S. Rosenbaum, GEAP-23773-2, March 1979.
4. Demonstration of Fuel Resistant to Pellet-Cladding Interaction, Second Semiannual Report January-June 1978, compiled by H. S. Rosenbaum, GEAP-23773-1, September 1978.
5. Demonstration of Fuel Resistant to Pellet-Cladding Interaction, First Semiannual Report, July-December, 1977, compiled by H.S. Rosenbaum, GEAP-23773, February 1978.
6. T. Hoshi, K. Iwata, T. Yoshimura, M. Ishikawa, "Fuel Failure Behavior of PCI-Remedy Fuels under Reactivity Inhibited Accident Conditions" JAERI-M8836 (May 1980).
7. Special Session at American Nuclear Society Meeting, "Iodine Spiking during Reactor Transients," Trans. Amer. Nucl. Soc., 28 (1978) 649.
8. R. S. Forsyth, W. H. Blackadder, K. Malen, and G. Roennberg, "Fission Product Release to Loop Water from Intentionally Defected Fuel," IAEA Specialists Meeting on Defected Fuel Performance, Chalk River, Canada (Sept. 1979).
9. D. H. Locke, "Mechanisms of Deterioration of Defected LWR Fuel," IAEA Specialists Meeting, Chalk River, 1979.
10. R. D. McDonald and J. J. Lipsett, "The Behavior of Defected Zircaloy Clad UO_2 Fuel Elements Irradiated at Linear Powers of 48 kW/m in Pressurized Water." IAEA Specialists Meeting, Chalk River, 1979.
11. D. Chenebault, G. Kura, A. Harrer, and J. P. Stura, "Evaluation of Fission Gases and Halogens Released out of Failed Fuel Running at Constant Power and in Power Cycling Regime." IAEA Specialists Meeting, Chalk River, 1979.
12. E. Schuster, F. Garzarolli, K. H. Neeb, and H. Stehle, "Release of Fission Products from Defective Fuel Rods of Light Water Reactors." IAEA Specialists Meeting, Chalk River, 1979.
13. Generic Reload Fuel Application. General Electric Co. NEDO-24011-A-1, July 1979.

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