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QUALIFICATION OF THE B&W MARK B FUEL ASSEMBLY FOR HIGH BURNUP

Third Semi-Annual
Progress Report:
July-December 1979

Prepared for
The United States
Department of Energy

Babcock & Wilcox

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for the Period
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T. A. Coleman
Project Engineer

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Qualification of the B&W Mark B Fuel Assembly
for High Burnup — Third Semi-Annual Progress
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ABSTRACT -

Five Babcock & Wilcox-designed Mark B (15 × 15) pressurized water reactor fuel assemblies were irradiated to extended burnups in Duke Power Company's Oconee Unit 1 reactor. An assembly average burnup of 40,000 MWd/mtU, which is about 29% greater than previous discharge burnups at Oconee 1, was attained. The irradiation and examinations of these assemblies are part of a Department of Energy-sponsored program to improve uranium utilization by extending the burnup of light water reactor fuel.

The nondestructive examination of these five assemblies, which have been irradiated for four fuel cycles, was begun. Data obtained included fuel assembly and fuel dimensions, water channel spacings, fuel rod surface deposit samples, and holddown spring preload forces. Visual examination of the assemblies indicated that good fuel performance was maintained through four cycles of irradiation.

This report, covering the July-December 1979 period, is the third semi-annual progress report for the program. Efforts during this period included performance tracking of the extended burnup assemblies, nondestructive examination of the four-cycle assemblies, and analytical work to identify and quantify the factors that limit fuel assembly lifetime.

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1. INTRODUCTION

A major new constraint was imposed on the nuclear fuel cycle by the decision of the U.S. government to indefinitely defer reprocessing spent fuel and recycling plutonium and uranium. For a number of years, the traditional practice in the light water reactor (LWR) industry has been to discharge fuel after batch average burnups of 25,000-33,000 MWd/mtU. The discharge batch average burnup of 33,000 MWd/mtU had been established through economic optimization studies which assumed spent fuel reprocessing to reclaim and re-use residual fissile materials in spent fuel. This burnup target of ~33,000 MWd/mtU has been demonstrated over the past decade to be conservative from a mechanical performance standpoint and to give ample assurance of cladding integrity and safe operating performance.

In the absence of reprocessing and recycling, however, conventional LWR fuel management strategies no longer represent optimum resource utilization. The industry must now assume that the residual fissile materials in spent fuel cannot be reclaimed and reused. This currently imposed "once-through" fuel cycle has created an economic incentive to minimize uranium requirements.

One of the more straightforward and most readily employable means of improving uranium utilization in LWRs in the near term is to increase discharge batch average burnup. Engineering projections have indicated that uranium utilization improvements of 15 to 20% can be achieved in pressurized water reactors (PWRs) when discharge batch average burnup is increased from about 33,000 to 45,000-50,000 MWd/mtU. Lower fuel cycle costs and fewer spent fuel assemblies to be stored are additional significant benefits from increasing discharge batch average burnup.

The Department of Energy (DOE) has initiated research, development, and demonstration efforts involving Duke Power Company (Duke), Arkansas Power & Light Company (AP&L), and Babcock & Wilcox (B&W), with a goal of improved uranium utilization, primarily through successful operation of PWR fuel assemblies to extended burnups. The uranium utilization improvement effort was divided into

two separate but related programs. In the program reported herein, Duke and B&W are seeking to demonstrate that the batch average burnup of current PWR assemblies can be safely increased from about 33,000 to 38,000 MWd/mtU. This program involves no design changes in current fuel assemblies, but rather will extend the fuel performance data base, paving the way for wide-scale implementation of higher batch average burnups beginning as early as 1981. This burnup extension will allow substantial improvements (about 5%) in uranium utilization to be realized, beginning within 2 to 3 years.

In the second program, AP&L and B&W are undertaking the development and demonstration of an improved fuel assembly that is capable of batch average burnups of 45,000 to 50,000 MWd/mtU.¹⁻³ The AP&L/B&W program is expected to lead to full-scale implementation of higher-burnup fuel assemblies by the late 1980's. An additional 10 to 15% improvement in fuel utilization should result.

This report is the third semi-annual progress report for the DOE/Duke/B&W program (contract No. DE-AC02-78ET34212).^{4,5} It covers progress from July 1 through December 31, 1979. This report covers work under the three technical tasks that makeup the program: Task 1 - High Burnup Characterization of the Mark B Fuel Assembly, Task 2 - Operational Limits of a Mark B Core, and Task 3 - Evaluation of PWR Fuel Failure Mechanisms for Extended Burnups.

2. SUMMARY

Implementation of extended burnup and realization of the subsequent uranium utilization improvement are the primary objectives of this program. To support these objectives, fuel performance data will be collected from PWR fuel assemblies at burnups in excess of 40,000 MWd/mtU. These data will be used to justify the routine irradiation of PWR assemblies to extended burnups.

This report, describing work performed during the last six months of 1979, is the third semi-annual progress report for the program. Efforts included performance tracking of five PWR assemblies during their fourth cycle of irradiation, nondestructive examination of the assemblies after completion of their fourth cycle, studies of the characteristics of extended-burnup fuel cycles and analysis of high-burnup leaker assemblies to determine their life-limiting failure mechanism.

A major milestone in the program was achieved with successful completion of the fourth cycle of irradiation for five PWR assemblies. An assembly average burnup of 40,000 MWd/mtU was attained. This discharge burnup is 29% greater than previous assembly discharge burnups at Oconee Unit 1, the host reactor for the irradiation.

Nondestructive examinations of the four-cycle assemblies were begun. Examinations completed included visual observations, measurement of fuel assembly and rod lengths, and collection of surface deposits. Other tests partially completed include holddown spring load-deflection tests, water channel spacing measurements, and fuel rod diameter profilometry. Reduction and analysis of the collected data are in progress.

In general, the fuel assemblies looked good with no abnormalities observed. Based on preliminary results, crud (fuel rod surface deposits) levels were low; the fuel assemblies grew by approximately 0.15 inch, which is consistent with previous growth data; and the spring rate for the holddown springs showed no significant change. Thus, a qualitative assessment of the fuel assemblies'

characteristics at the end of four cycles of irradiation indicates good fuel performance.

Referencing previously completed extended-burnup fuel cycle designs⁵, evaluations normally considered in reload licensing were begun. Physics parameters, such as reactivity coefficients and kinetics characteristics, were generated for extended-burnup fuel cycles. These parameters are used in safety, control, and dose evaluations and were generally similar to values for currently licensed reloads with the exception of more negative end-of-cycle (EOC) moderator coefficients. Further safety evaluations will be required to assess the impact of the more negative coefficients. Increased soluble boron concentrations required to extended burnup cycles do not necessitate modifications to plant hardware (for discharge average burnups up to 48,500 MWd/mtU, the highest burnup considered) but merely require changes in the Technical Specifications. The generation of radiological source terms for use in dose evaluations of extended-burnup fuel cycles was completed. These source terms are larger than previous values, but the doses calculated for the various hypothetical accidents are expected to be well below limits stipulated in 10 CFR 100.

The individual fuel assembly and fuel rod power histories from the previously completed extended-burnup fuel cycles involving batch average discharge burnups as high as 48,500 MWd/mtU were compiled for use in mechanical and thermal evaluations of high-burnup fuel performance. Using these data, mechanical analyses indicated that cladding collapse for the standard Mark B (15 × 15) fuel rods is not expected to be the limiting design criterion for extended-burnup. Cladding strain is acceptable for fuel pellet burnups higher than 74,000 MWd/mtU.

A review and detailed analyses of loss-of-coolant accident (LOCA) limits for extended-burnup fuel in lowered-loop 177-fuel assembly plants were begun. Fuel rod dimensions, internal pressures, and fuel temperatures were generated for the LOCA analyses from thermal evaluations of extended-burnup fuel rod power histories.

Fuel integrity checks conducted at the end of three cycles indicated that three of sixteen candidate assemblies for a fourth cycle of irradiation were leakers. Visual examination of the leakers showed no defects; thus, an investigation to identify the fuel rod failure mechanism was begun. A review of manufacturing records for these assemblies revealed no significant variance

in their as-built characteristics compared to their sister assemblies. A review of fuel assembly operating history showed that the fuel experienced a higher than average (5.66 kW/ft) operating history. The potential impact of this history on fuel integrity is being evaluated.

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3. PROGRAM DESCRIPTION

Since the last report period⁵, the program scope has been expanded to include an additional task, "Evaluation of PWR Fuel Failure Mechanisms for Extended Burnup." Thus, the current program comprises three tasks which include both fuel demonstration irradiations and examinations, and analytical studies of extended burnup. The specific objectives of each task and a description of the technical work scope with a breakdown by subtask are presented in the following paragraphs. The status of each subtask is also given.

3.1. Task 1 — High Burnup Characterization of the Mark B Fuel Assembly

3.1.1. Objectives (Status)

1. To develop a fuel cycle design for Oconee 1 cycle 5 that irradiates selected fuel assemblies to burnups in excess of 38,000 MWd/mtU and meets cycle energy requirements. (Completed)
2. To characterize the high-burnup lead assemblies prior to their fourth cycle of operation. (Completed)
3. To secure an operating license for Oconee 1 cycle 5. (Completed)
4. To characterize the high-burnup lead assemblies after their fourth cycle of irradiation. (In progress)

3.1.2. Technical Work Scope (Status)

1. Subtask 1A — Fuel Cycle Design — Design Oconee 1 cycle 5 to meet standard fuel cycle design criteria. Use the PDQ07⁶ computer code with two dimensional (X-Y), quarter-core geometry and a pin-by-pin representation to model the core. Design cycle 5 based on a cycle 4 lifetime of 235 EFPD. A cycle 5 lifetime of 320 EFPD was specified by Duke Power Company. Select five fuel assemblies from Oconee 1 for a fourth cycle of irradiation with a target burnup of 38,000 MWd/mtU. (Completed)
2. Subtask 1B — Licensing — License Oconee 1 cycle 5 as a standard uranium reload in compliance with the requirements outlined in the USNRC document, "Guidance for Proposed License Amendments Relating to Refueling." (Completed)

3. Subtask 1C — Poolside Characterization — Using nondestructive measurement techniques and B&W post-irradiation examination equipment installed in the Oconee 1 and 2 spent fuel pool, characterize the five fuel assemblies at the end of their third cycle. Include visual examination, gross gamma scans, water channel spacing measurements, dimensional measurements, and measurements of holddown spring load-deflection characteristics. (Completed)
4. Subtask 1D — Irradiation — Follow the operation of Oconee 1 cycle 5. (Completed)
5. Subtask 1E — Nondestructive Testing (Poolside) — Perform nondestructive examination of the five four-cycle fuel assemblies at the end of Oconee 1 cycle 5. Include spacer grid spring load-deflection measurements and crud sampling in addition to the measurements carried out in subtask 1C. (In progress)
6. Subtask 1F — Destructive Testing (Hot Cell) — Ship one of the five four-cycle fuel assemblies to B&W's Lynchburg Research Center hot cell facility for further detailed nondestructive and destructive examination.

3.2. Task 2 — Operational Limits of Mark B Core

3.2.1. Objectives (Status)

1. To define the maximum reliable operating limits for the B&W standard Mark B (15 × 15) fuel assembly. (In progress).
2. To identify constraints that limit allowable fuel assembly and component burnup and quantify the burnup limit. (In progress)

3.2.2. Technical Work Scope (Status)

1. Subtask 2A — Fuel Cycle Design — Analyze extended-burnup fuel cycles. Perform fuel cycle calculations using the PDQ07 diffusion theory code and B&W's standard two-dimensional reactor model to develop high-discharge-burnup fuel cycles. The fuel cycle calculations will yield enrichment requirements, fuel loadings, power distributions, fuel burnup data, selected control rod worths, and isotopics. (Completed)
2. Subtask 2B — Nuclear, Mechanical, and Thermal-Hydraulic Analyses — Generate key physics parameters, such as moderator and Doppler reactivity coefficients, for use in reactor safety and control analyses. Analyze fuel assembly mechanical and thermal-hydraulic behavior at high burnup. Evaluate

fuel temperatures, fuel rod internal pressure, cladding creep collapse, fatigue, stress, and strain. (In progress)

3. Subtask 2C — Safety Analyses and ECCS Evaluation — Assess the relative impact of extended burnup on plant safety analyses. Include group and single rod withdrawals, rod drop, rod ejection, four-pump coastdown, feedwater and steam line breaks, and anticipated transients without scram. Evaluate dose rates for extended-burnup fuel. Develop LOCA limits for extended-burnup fuel. Identify the worst time in life for LOCA and generate specific linear heat rate limits as functions of height in the core. (In progress)
4. Subtask 2D — Control Analyses — Conduct soluble boron shutdown analyses to determine the boron requirements of extended-burnup fuel cycles. Review reactor load change capability and control scheme requirements. (In progress)
5. Subtask 2E — System Evaluation — Review the results from subtask 2D above to ensure compatibility with the capabilities of the auxiliary fluid and chemical addition systems. (In progress)

3.3. Task 3 — Evaluation of PWR Fuel Failure Mechanisms for Extended Burnup

3.3.1. Objectives (Status)

1. To identify the fuel failure mechanisms experienced by high-burnup PWR fuel assemblies. (In progress)
2. To expand the fuel performance data base at assembly burnups of about 31,000 MWd/mtU. (In progress)

3.3.2. Technical Work Scope (Status)

1. Subtask 3A — Irradiation History — Compile assembly and fuel rod powers and burnups for the selected leaker assembly. Review assembly operating history to identify power transients of interest. Model these transients to analyze changes in local power experienced by the assembly. Calculate fission gas release, fuel rod void volume, rod internal pressure, cladding ovality, fuel temperatures, and fuel swelling for the leaker assembly. (In progress)
2. Subtask 3B — Fuel Examinations — Perform a visual examination of the leaker assemblies. Based on visual observations and the irradiation history, select a leaker for detailed examination in the Lynchburg Research

Center (LRC) hot cell. Develop a detection system for locating the failed rod within the leaker assembly. Extract six rods from the assembly to undergo the following examinations. (In progress)

- Visual examination
- Rod length measurement
- Eddy-current scans
- Profilometry
- Isotopic and gross gamma scans
- Fission gas release measurement
- Metallographic examination
- Fuel density determination
- Scanning electron microscopy and x-ray analysis of cladding ID surface deposits

3. Subtask 3C — Evaluation of Results — Compare measured data with analytical predictions of fuel behavior at 31,000 MWd/mtU. Determine possible failure mechanisms and identify factors leading to failure. Define design changes or operational criteria that could minimize the potential for fuel failure.

4. TASK 1 — HIGH BURNUP CHARACTERIZATION OF THE MARK B FUEL ASSEMBLY

B. A. Guthrie, J. T. Mayer, T. D. Pyecha

4.1. Introduction

The major efforts conducted under Task 1 during this report period consisted of (1) performance tracking of five high-burnup assemblies during their fourth cycle of irradiation and (2) the initiation of nondestructive examinations of these five assemblies. A major milestone in the extended-burnup program was achieved with the completion of the fourth cycle of irradiation of the five high-burnup assemblies in Oconee 1. Four of these assemblies attained assembly average burnups of 40,000 MWd/mtU, which is 29% greater than any previously discharged fuel from Oconee 1. The results from these efforts are reported herein under subtask 1D, "Irradiation," and subtask 1E, "Nondestructive Testing (Poolside)."

Duke Power Company's Oconee Unit 1 is a B&W-designed PWR with a core composed of 177 Mark B (15 × 15) fuel assemblies. A cutaway version of a typical Mark B fuel assembly is shown in Figure 4-1. The structural cage of the assembly comprises 16 control rod guide tubes (CRGTs) of cold-worked, stress-relieved, Zircaloy-4 permanently attached to the upper and lower end fittings. The Inconel 718 spacer grids form the structural link between the guide tubes and the fuel rods (cold-worked, stress-relieved, Zircaloy-4 clad), but are free to move with the fuel rods as the rods grow. For reference, the six intermediate spacer grids are numbered sequentially from 1 to 6 (from top to bottom), and the midspan levels are numbered evenly from 2 to 14 as shown in Figure 4-2. Fuel assembly lift resulting from coolant flow is prevented by the helical holddown spring in the upper end fitting of the assembly.

Each fuel assembly consists of a 15 × 15 lattice arrangement with 16 positions occupied by guide tubes and one position, the center lattice site, occupied by an incore instrumentation tube. The remaining 208 lattice positions contain

fuel rods. A schematic cross section of a typical fuel assembly displaying the fuel rod numbering convention is presented in Figure 4-3.

The fuel in the rods is low-enriched UO_2 in the form of dished-end, chamfered, cylindrical pellets (sintered). The nominal design parameters and operating conditions of Oconee 1 fuel are summarized in Table 4-1.

Table 4-1. Oconee 1 Nominal Fuel Design Parameters

Reactor Core

Core power rating, MWr	2568
Core average linear heat rate, kW/ft	5.66
Core inlet coolant temperature, F	555
Core outlet coolant temperature, F	606
Core operating pressure, psia	2200

Fuel Assemblies and Fuel Rods

Number of fuel assemblies in core	177
Number of fuel rods per fuel assembly	208
Number of CRGTs per assembly	16
Fuel rod outside diameter, in.	0.430
Fuel tubing (cladding) thickness, in.	0.0265
Fuel rod pitch, in.	0.568
Fuel assembly pitch, in.	8.587

Fuel

Material	UO_2 (sintered)
Fuel pellet length, in.	0.70
Fuel pellet diameter, in.	0.3670
Active fuel length, in.	141.0
Fuel density, % theoretical	95.5

Figure 4-1.
Mark B Fuel Assembly

Figure 4-2. Mark B Fuel Assembly Showing
Numbering Convention

Figure 4-3. Fuel Assembly Cross Section Displaying Numbering Conventions

4.2. Subtask 1D - Irradiation

Five standard PWR fuel assemblies - 1D13, 1D26, 1D42, 1D45, 1D55 - were reinserted in the Oconee 1 cycle 5 core for their fourth cycle of operation in October 1978.⁴ The location of these assemblies in the cycle 5 core is shown in Figure 4-4. During this report period, these assemblies completed their fourth cycle of irradiation fulfilling a major milestone of the extended-burnup program.

Four of these high-burnup lead assemblies attained assembly average burnups of approximately 40,000 MWd/mtU making them the highest burnup B&W assemblies to date - 29% higher than the highest assembly burnup previously discharged at Oconee 1. The center fuel assembly achieved a slightly lower discharge burnup of 36,000 MWd/mtU. The measured end of cycle assembly average burnups are given in Figure 4-5.

Cycle 5 of Oconee 1 attained initial criticality on October 25, 1978, and was terminated on November 22, 1979, after reaching a cycle exposure of 303.5 EFPD. The reactor power history for cycle 5 is presented in Figure 4-6.

Cycle 5 was operated in a "feed-and-bleed" mode. In this mode all full-length control rod assemblies (CRAs) are fully withdrawn from the core at full power except for one bank of CRAs (designated group 7), which is about 90% withdrawn during full power operation. The cycle 5 core of Oconee 1 also contained axial power shaping rods (APSRs, designated group 8), which are CRAs having an absorber section only 3 ft long in the lower quarter of the rod. The APSRs were approximately 20-25% withdrawn during steady-state, full-power operation. As shown in Figure 4-7, the four-cycle assembly in the center core location (H8) was beneath a safety CRA during cycle 5, and the other four high-burnup assemblies did not contain CRAs.

The core power distribution during cycle 5 was monitored continuously by self-powered neutron detectors located in 52 of the core's 177 fuel assemblies. Measured and PDQ07-calculated⁶ assembly relative powers are compared in Figure 4-8. The agreement between measured and calculated powers was good for the four-cycle assemblies - within 2.2% for the four assemblies on the core axes and 3.2% for the center assembly.

Cycle 5 reactor operations were modeled as closely as practical using the three-dimensional nodal computer code FLAM3.⁷ Power distribution calculations were

performed at various stages throughout cycle 5 with such input parameters as CRA positions, core power level, and boron, each conforming to actual reactor conditions. Representative axial power distributions from FLAM3 for assembly 1D42 are shown in Figure 4-9. Figure 4-10 presents data for assemblies 1D13, 1D26, 1D45, and 1D55. Axial burnup distributions calculated by FLAM3 are shown in Figures 4-11 and 4-12 for assembly 1D42 and assemblies 1D13, 1D26, 1D45, and 1D55, respectively.

Reactor Coolant Activity

^{131}I , ^{133}I , and ^{135}I coolant activities were monitored during cycle 5 of Oconee 1 operation. Figures 4-13, 4-14, and 4-15 show plots of these activities throughout the cycle. Figures 4-16 and 4-17 display the corresponding ratios of ^{131}I to ^{133}I and ^{131}I to ^{135}I . The reactor power history is also included in these figures.

The data in Figures 4-13 through 4-17 show considerable scatter primarily due to fluctuations in the reactor power level. The phenomenon of a sudden increase in measured iodine activity (iodine spiking) occurs when the reactor trips or is shut down with small leakers present in the core. It can also occur — to a lesser extent — following less severe power transients. Each iodine spike routinely manifests itself over a period of one to four days, with the spike peaking approximately 4 to 8 hours following a power change.

Although iodine spikes during power transients signal the presence of leaking fuel rods, no absolute correlation of the coolant activity levels to the number of leaking fuel rods in the core can be made because of uncertainties in the location and nature of the leakers. For example, the size, type, and axial location of the leaks as well as the power level and fuel region of the leaking rod(s) are all unknown. These uncertainties are further compounded by other factors, such as variations in core power level, coolant flow rate, and coolant cleanup rate. However, the equilibrium activity levels of ^{131}I (and to a lesser extent ^{133}I and ^{135}I) do provide a useful indication of the relative leak condition of the core under steady-state, high power (>95%) reactor operation. In addition, the ratios of ^{131}I to ^{133}I and ^{131}I to ^{135}I provide an indication of the type of defects in the core. If these ratios are high (greater than about 0.6 and 1.0, respectively), a significant delay is indicated in the release of fission product iodine from the fuel to the coolant,

implying the presence of small, pin-hole size leaks. Similarly, if the ratios are low (less than about 0.6 and 1.0, respectively), the delay in iodine release to the coolant is short, indicating that a significant contribution to the coolant iodine levels may be due to the fission of fuel exposed directly to the coolant (such as large leaks or significant levels of tramp fuel on core surfaces).

The general trend in the equilibrium ^{131}I data obtained during high power operation is masked somewhat in Figure 4-13 by the iodine spiking resulting from fluctuations in reactor power. However, it is reasonably apparent that the equilibrium ^{131}I data exhibit a slight increase at mid-cycle (about $0.08\ \mu\text{Ci/g}$) and near the end of the cycle (about $0.13\ \mu\text{Ci/g}$ average) in comparison to early in the cycle (about $0.07\ \mu\text{Ci/g}$). These trends are also apparent in the ^{133}I and ^{135}I activities shown in Figures 4-14 and 4-15. The cycle 5 equilibrium ^{131}I activities averaged $0.09\ \mu\text{Ci/g}$, which is approximately 2.8% of the design basis activity ($3.2\ \mu\text{Ci/g}$) for ^{131}I reactor coolant activity.⁸ For cycles 1, 2, 3, and 4 the equilibrium ^{131}I coolant activities averaged 0.11, 0.02, 0.09, and $0.06\ \mu\text{Ci/g}$, respectively.

The equilibrium iodine ratios for ^{131}I to ^{133}I varied from approximately 0.55 to 0.99, averaging approximately 0.81 during cycle 5 operation, while the equilibrium ^{131}I to ^{135}I ratios varied from 0.85 to 1.48, averaging 1.00 (refer to Figures 4-16 and 4-17). Iodine ratios in this range are indicative of iodine release from the fuel column gap after a moderate delay, implying the presence of small leaks. Analysis of the ^{134}Cs to ^{137}Cs ratio data indicates that the majority of the leaks are probably contained within the three-cycle (batch 5) assemblies. The integrity of batch 5 assemblies and the five four-cycle (batch 4) assemblies will be verified later by leak testing (sipping).

Figure 4-4. Oconee 1, Cycle 5 Full Core Loading Diagram

A						7	7	7	7	7					
B			7	7	7	E7 5	R3 6	E9 5	7	7	7				
C		7	M2 6	C6 5	F7 5	L2 6	M5 5	L14 6	F9 5	C10 5	M14 6	7			
D	7	B11 6	O13 6	L1 6	N3 6	D5 5	7	D11 5	N13 6	L15 6	O3 6	B5 6	7		
E	7	F3 5	A10 6	O9 5	K1 6	N2 6	O5 1D55 4D	N14 6	K15 6	K3 5	A6 6	F13 5	7		
F	7	7	G6 5	C12 6	A9 6	K13 5	D6 5	N8 5	D10 5	O7 5	A7 6	C4 6	G10 5	7	7
G	7	G5 5	B10 6	E4 5	B12 6	F4 5	7	K8 5	7	F12 5	B4 6	E12 5	B6 6	G11 5	7
H	7	H15 6	M11 5	7	M13 1D13 4D	H12 5	H9 5	K14 1D42 4D	H7 5	H4 5	E3 1D45 4D	7	E5 5	H1 6	7
K	7	K5 5	P10 6	M4 5	P12 6	L4 5	7	C8 5	7	L12 5	P4 6	M12 5	P6 6	K11 5	7
L	7	7	K6 5	O12 6	R9 6	G9 6	N6 6	D8 5	N10 5	G3 5	R7 6	O4 6	K10 5	7	7
M	7	L3 5	R10 6	G13 5	G1 6	D2 6	C11 1D26 4D	D14 6	G15 6	G7 5	R6 6	L13 5	7		
N	7	P11 6	C13 6	F1 6	D3 6	N5 5	7	N11 5	D13 6	F15 6	C3 6	P5 6	7		
O	7	E2 6	O6 5	L7 5	F2 6	E11 5	F14 6	L9 5	O10 5	E14 6	7				
P			7	7	7	M7 5	A8 6	M9 5	7	7	7				
7						7	7	7	7	7					
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15

G5 ← Previous Cycle Location
 1D55 ← Assembly Ident.
 4D ← Batch No.

Location of thrice-burned batch 4D assemblies

Figure 4-5. Measured Assembly Burnups, MWd/mtU — Oconee 1 EOC 5

Figure 4-6. Oconee 1 Cycle 5 Power History

Figure 4-7. Oconee 1, Cycle 5 Control Rod Locations

x ← Group Number

Group	No. of rods	Function
1	8	Safety
2	8	Safety
3	9	Safety
4	8	Safety
5	8	Control
6	8	Control
7	12	Control
8	8	APSRs
Total	69	

[illegible]

Figure 4-9. FLAM3 Axial Power Distribution for Assembly, 1D42

* Normalized to average assembly power.

Figure 4-10. FLAM3 Axial Power Distribution for Assemblies
1D13, 1D26, 1D45, and 1D55

* Normalized to average assembly power.

Figure 4-11. FLAM3 Axial Burnup Distribution for Assembly 1D42

Figure 4-12. FLAM3 Axial Burnup Distribution for Assemblies 1D13, 1D26, 1D45, and 1D55

Figure 4-13. Oconee 1 Cycle 5 Coolant Activity (^{131}I)
and Reactor Power History

Figure 4-14. Oconee 1 Cycle 5 Coolant Activity (^{133}I)
and Reactor Power History

I-133

Figure 4-15. Oconee 1 Cycle 5 Coolant Activity (^{135}I)
and Reactor Power History

Figure 4-16. Oconee 1 Cycle 5 Coolant Activity ($^{131}\text{I}/^{133}\text{I}$) and Reactor Power History

Figure 4-17. Oconee 1 Cycle 5 Coolant Activity ($^{131}\text{I}/^{135}\text{I}$)
and Reactor Power History

4.3. Subtask 1E — Nondestructive Testing (Poolside)

This subtask includes end-of-cycle (EOC) 5 poolside measurements on five four-cycle fuel assemblies. The work scope is identical to that conducted at EOC-4⁴ except that crud sampling and grid spring testing were included in the EOC-5 work scope. These tests will obtain fuel performance data near 40,000 MWd/mtU for comparison with previous data obtained near 30,000 MWd/mtU. The data will be used to quantify the changes in key fuel rod and assembly parameters and will also be used as input in selecting one of the assemblies for destructive examination and one for a potential fifth cycle of irradiation.

4.3.1. Poolside Operations

Examination of the five assemblies (1D13, 1D26, 1D42, 1D45, and 1D55) began immediately after the EOC-5 fuel shuffle in December 1979. Tests completed to date are shown in Table 4-2 along with the total work scope. The remaining tests (approximately half the total work scope) will be completed in early 1980.

Figure 4-18 shows one of the four-cycle assemblies being examined in the line scan tester (LST) which is the heart of the post-irradiation examination (PIE) equipment, located in the Oconee Unit 1 and Unit 2 spent fuel pool. It comprises a strongback frame and various test heads for diameter profilometry, assembly bow, and water channel spacing measurements. An assembly is positioned in the LST with its lower end fitting resting in a support socket at the bottom of the LST and its upper end fitting restrained within the confines of a cylinder at the top. The bottom of the LST is about 26 feet below the surface of the water. The bottom support socket is a turntable which permits fuel assembly (FA) rotation in the LST so that any face of the assembly is accessible to the measuring heads. The measuring heads are mounted on a cart which travels vertically to scan the axial profile of the assembly. The measuring head for diameter profilometry is calibrated before data collection begins. The probe used for water channel spacing measurements is also calibrated before each pass through the assembly. Precision for these systems is estimated to be ± 0.000104 and ± 0.00007 inch, respectively, for the two systems at the 1 σ level. Controls for the measuring heads and the data acquisition packages are located topside at the spent fuel pool. Figure 4-19 shows this area while work is in progress.

Except for crud sampling, measurement techniques have been described in previous reports.^{4,9} Crud sampling operations were recorded on videotape and were conducted with the FA suspended from the hoist. With the TV camera positioned to view the area to be sampled, the crud sampling tool was brought into contact with the fuel rod. The tool itself consists of a steel knife-edge scraper with a curved blade machined to conform to about one-third of the fuel rod circumference. This blade is attached to a handling pole and manually moved up and down to remove the crud. The removed crud flakes were drawn into a suction hose positioned near the tip of the blade and drawn up to poolside (along with pool water) into a 4-liter flask. The water and crud mixture was then passed through a 1- μ filter. The particulate, along with spent fuel pool water background samples, were returned to the Lynchburg Research Center for analysis using gamma and x-ray spectroscopy. Periscope photographs of the sampled areas were taken to determine the fuel rod area sampled.

Since the crud sampling operation can produce scratches on the fuel rod surface, which might preclude further service, such sampling is normally done only on assemblies that have reached the end of their service life. Since one of the five four-cycle assemblies may be reinserted for a fifth cycle, it was necessary to select from the five those two assemblies least likely to be reinserted. The criterion for the choice was fuel assembly length. Prior to crud sampling, the lengths of all four of the higher-burnup assemblies were measured. The two longest assemblies (1D13 and 1D55) were chosen for crud sampling, leaving the lower-burnup assembly (1D42) and the two shorter high-burnup assemblies (1D26 and 1D45) as candidates for a fifth cycle. Crud samples were taken at various axial locations on one rod each from 1D13 and 1D55.

4.3.2. Results and Discussion

A preliminary qualitative assessment of the available results from the high-burnup assemblies (1D13, 1D26, 1D42, 1D45, 1D55) indicates that the assemblies have performed well through four cycles of irradiation. The measurement campaign is near completion, and a comprehensive analysis of the measured data is currently in progress. The available preliminary results are discussed below.

4.3.2.1. Visual Inspections

Television examination of all four sides of each assembly showed no abnormalities. Crud (corrosion product) deposit patterns were similar to those observed

after three cycles of operation and were typical of patterns commonly seen on PWR fuel. An area of assembly 1D13 at the end of three and four irradiation cycles is shown in Figures 4-20 and 4-21 along with closeup views of the rods. Differences in color between Figures 4-20 and 4-21 are not significant due to the effects of variations in background lighting. However, crud bands occurring at a frequency corresponding to the fuel pellet lengths can be observed in both sets of photographs.

Pictures taken during the crud sampling operations on assembly 1D55 at various axial locations (Figures 4-22 through 4-26) display areas of uniform color, variations in crud patterns from area to area, and "banding" at pellet interfaces similar to observations at the end of three irradiation cycles.

4.3.2.2. Crud Sampling

Although no isotopic or quantitative analyses of the crud samples have been performed as yet, two observations indicate that the crud amount is less than that previously observed on one- and two-cycle assemblies from Oconee 1.^{10,11} First, the amount of material on the filters appeared to be significantly smaller than previous samples; second, the radiation levels of the collected samples were lower.

4.3.2.3. Fuel Assembly Length and Holddown Spring

Preliminary analyses of length and holddown spring data showed that the observed irradiation-induced changes during the assemblies' fourth cycle were as anticipated. The four higher-burnup assemblies grew approximately 0.20 inch or less during their fourth cycle, which is consistent with data obtained previously.⁵ Holddown spring preload on the three assemblies measured decreased by 20 to 40 pounds; the spring rate itself showed no significant change.

Table 4-2. Oconee 1 EOC-5 Nondestructive Examination

<u>Measurement</u>	<u>Planned scope</u>	<u>Status</u>
1. TV examination	Videotaped examination of all four faces on all five assemblies	Completed all
2. Periscope photography	Repeat photos from previous cycle, photograph crud sample areas	Completed all
3. Holddown spring test	All five assemblies	Completed 1D26, 1D42, 1D55
4. Water channel spacing	All mid-plane spacings on all five assemblies	Completed 1D26 1D42, 1D55
5. Diameter scans	Full-length scans on 28 rods from each assembly	Completed 1D26 1D42, 1D55
6. Assembly length and grid locations	All four faces of each assembly	Completed all
7. Crud sampling	Selected rods from two assemblies	Completed all
8. Gross gamma scans	All corner rods on each assembly	Incomplete
9. Grid spring tests	Selected cells from two assemblies	Incomplete
10. Leak test (sipping)	All five four-cycle assemblies	Incomplete

Figure 4-18. Assembly in Line Scan Test Frame

4-18a

4-18b

Figure 4-19. Poolside Operation on Four Cycle Assemblies

Figure 4-20. Typical Areas on Assembly 1D13

4-20a

4-20b

Figure 4-21. Assembly 1D13 — Face B Between
Top Span and Grid 4

4-21a

4-21b

4-21c

Figure 4-22. Assembly 1D55 — Face D Between Upper
End Skirt and Grid 1

Figure 4-23. Crud Sample Area on Assembly 1D55 Face D
Between Grids 1 and 2

Figure 4-24. Crud Sample Area on Assembly 1D55 Face D
Between Grids 2 and 3

Figure 4-25. Crud Sample Area on Assembly 1D55 Face D
Between Grid 6 and Lower End Skirt

Figure 4-26. Closeup of Crud Sample Area on Assembly 1D55 Face D
Between Grid 6 and Lower End Skirt

5. TASK 2 — OPERATIONAL LIMITS OF MARK B CORE

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5.1. Introduction

The simplified fuel cycle models used to scope the uranium utilization benefits of extended burnup do not provide information on such important core operational parameters as power distributions, control rod worths, or reactivity coefficients. Design and licensing requirements pertaining to these core parameters can conceivably introduce undesirable constraints on extended-burnup fuel cycles. Therefore, during the last report period⁵, fuel management plans resulting in discharge batch average burnups from about 34,000 to 48,500 MWd/mtU were developed as a basis for detailed core physics, mechanical, thermal-hydraulic, safety, control, and accident evaluations normally considered in reload licensing applications.

Five fuel management plans for the Oconee-class plants were selected for study to determine the technical feasibility of designing fuel cycles with relatively high discharge burnups. The five selected combinations of reload batch size and fuel cycle length were as follows:

1. 80-FA feed, 460 EFPD cycle @ 2772 MWt, feed and bleed.
2. 68-FA feed, 460 EFPD cycle @ 2772 MWt, feed and bleed.
3. 60-FA feed, 460 EFPD cycle @ 2772 MWt, feed and bleed.
4. 60-FA feed, 497 EFPD cycle @ 2568 MWt, rodded.
5. 36-FA feed, 292 EFPD cycle @ 2772 MWt, feed and bleed.

To obtain data of generic applicability to B&W 177-FA plants, four of the fuel management evaluations were conducted at the maximum licensed power level of 2772 MWt. The 2772-MWt plants operate in a "feed-and-bleed" mode, which primarily utilizes soluble boron to control reactivity changes. In this mode, control rods are almost fully withdrawn from the core during full-power operation. For comparison, a fuel management plan for operation at 2568-MWt in the

"rodded" mode (corresponding to the operation of several B&W reactors) was also developed. In the rodded mode, a bank of control rods is inserted deeply during full-power operation.

The "low-leakage" or lumped burnable poison (LBP) shuffle scheme was used in the extended-burnup fuel management evaluations because a substantial improvement in uranium utilization (about 4%) can be achieved with this scheme. In the LBP shuffle scheme, fresh FAs containing LBP clusters are interspersed in the core interior with the fuel of highest burnup. The lowest-burnup fuel from the previous cycle is placed on the core periphery. Radial core neutron leakage is thereby reduced relative to out-in fuel management, in which fresh fuel is loaded on the core periphery.

The major efforts conducted under Task 2 during this report period consisted of (1) the generation of nuclear input for subsequent control and accident evaluations, (2) cladding creep collapse and strain analyses, (3) calculation of fission product inventories, (4) a burnup sensitivity study for loss of coolant accidents, and (5) an assessment of soluble boron requirements for extended-burnup cycles. The results from these efforts are reported herein under subtask 2B, "Nuclear, Mechanical and Thermal-Hydraulic Analyses"; subtask 2C, "Safety Analyses and ECCS Evaluation"; and subtasks 2D and 2E, "Control Analyses and System Evaluation."

5.2. Subtask 2B — Nuclear, Mechanical and Thermal-Hydraulic Analyses

5.2.1. Nuclear Analyses

5.2.1.1. General

The physics parameters that reflect the unique characteristics of high-burnup cores must be evaluated and compared to those for standard designs to ensure licenseability and compatibility with control systems. The work completed to date and not included in previous reports^{4,5} is described in section 5.2.1.2.

This section discusses reactivity coefficients, reactivity deficits, neutron kinetics parameters, and fuel assembly batch burnup and power fraction data. These parameters are part of the total nuclear input for safety, control, and radiation evaluations of high-burnup designs. The safety evaluation is in progress and will determine whether licensing changes will be needed for extended burnup. Control analyses will determine whether plant control systems are compatible with extended-burnup core reactivity requirements. Radiation analyses will determine the radiological consequences of extending fuel burnup limits.

Plant operating flexibility is also a significant factor in determining the practicality of extended-burnup cycles. The ability to carry out various load-following operations can be restricted by limits on allowable control rod positions and axial imbalance. These limits are discussed in detail in reference 12. Peaking-related limits are derived by comparing calculated core power distributions with initial condition peaking criteria. Of these peaking limits, the linear heat generation rates for LOCAs are usually the most restrictive and are given primary consideration. Operating limits not related to peaking include shutdown margin rod insertion limits and ejected rod insertion limits. The work completed to date on operating limits is also described in section 5.2.1.2.

5.2.1.2. Results and Discussion

Physics Parameters — Although the detailed safety, control and radiation analyses have not been completed, the potential impact of the high-burnup fuel cycle designs on licenseability and control requirements can be assessed by comparing parameters from high-burnup designs with those of a standard design. Cycles 3 and 4 of Arkansas Nuclear One, Unit 1 (ANO-1), were chosen as standard designs for comparison to an extended-burnup fuel cycle. The extended-burnup

cycle was the 60-FA feed, rodded operation, 497-EFPD case. Cycle 3 of ANO-1 was an annual fuel cycle representative of three-batch, out-in fuel management. Cycle 4 of ANO-1 is an 18-month cycle with LBP fuel management. Each cycle was designed with a control rod bank fully inserted during the first portion of cycle depletion. All three designs use geometrically identical Mark B 177-FA cores. The following paragraphs discuss the results of the parameter comparisons.

Reactivity Coefficients — Moderator and Doppler reactivity coefficients for the equilibrium cycle of each of the five extended-burnup fuel management schemes are presented in Table 5-1. The coefficients for the 60-FA feed rodded design are compared to those for ANO-1 cycles 3 and 4 in Table 5-2. End of cycle (EOC) moderator coefficients for the extended-burnup designs are somewhat more negative than those for current Mark B reloads and in fact exceed FSAR values for B&W 177-FA plants. For reactivity insertions such as rod ejection or rod withdrawal incidents, the more negative moderator coefficient at EOC results in a more negative power coefficient and lessens the severity of the power excursion. However, for overcooling transients such as the steamline break incident, the more negative moderator coefficient increases the positive reactivity inserted by cold water and results in a more limiting accident. The acceptability of a more negative moderator coefficient will be assessed during the detailed safety evaluation of extended-burnup cycles. The Doppler coefficients for the extended-burnup cycles are not significantly different from those of current reload designs.

Reactivity Deficits — Moderator and Doppler reactivity deficits for each of the five extended-burnup fuel management schemes are presented in Table 5-3. A reactivity deficit is defined as the change in core reactivity resulting from decreasing reactor power and temperature from hot full power to hot zero power conditions (580 to 532F moderator temperature). These deficits were obtained from two-dimensional PDQ07 calculations. The total three-dimensional power deficit was obtained by combining the moderator and Doppler reactivity deficits with the reactivity addition due to flux redistribution (the axial shift of flux toward the top of the core when power is reduced). The three-dimensional power deficits for the 60-FA feed rodded design are compared with those from ANO-1 (cycles 3 and 4) in Table 5-4. The three-dimensional power deficits, particularly at EOC, tend to increase with increasing burnup

primarily because of the more negative moderator coefficient. The flux redistribution component of the three-dimensional power deficit also increases in the later cycles.

Neutron Kinetics Parameters — Effective delayed neutron fractions at the beginning and end of a cycle are compared in Table 5-4 for ANO-1 cycle 3, ANO-1 cycle 4, and the transition cycles of the 60-FA feed rodded design. The tendency for the delayed neutron fraction to decrease with increasing burnup is offset by the higher enrichments used in the extended-burnup cycles. The higher feed enrichments also tend to reduce the prompt neutron lifetime by several μ s. The prompt neutron lifetime in ANO-1 cycle 4 ranged from 24 to 27 μ s; in the 60-FA feed, feed-and-bleed equilibrium cycle it ranged from 22 to 25 μ s.

Batch Burnup and Power Fraction Data — Included in the nuclear input to radiation analyses are batch average and maximum assembly burnups, batch average relative power densities, and batch and isotopic power fractions. These parameters for the 60-FA feed rodded design are compared to those for ANO-1 cycles 3 and 4 in Table 5-5. An increase in the fraction of an assembly's power produced from plutonium fissions relative to the power produced by uranium fissions was expected for extended burnup. However, the higher feed enrichments in the extended-burnup designs tended to offset the expected increase in plutonium power fraction.

FLAME3 Tableset Development — Power peaking is calculated with the three-dimensional nodal code FLAME3⁷ to evaluate the effects of extended-burnup cycles on operating limits. Because relatively high ^{235}U enrichments and fuel burnups were used in the extended-burnup cycles, the FLAME3 input data tablesets were expanded to include enrichments from 2.0 to 4.5 wt % ^{235}U and burnups up to 80,500 MWd/mtU. Table 5-6 lists the important core parameters and their fitting ranges incorporated into the FLAME3 model.

The FLAME3 code and tablesets were also modified to include plutonium buildup produced by moderator and fuel temperature history effects. The addition of these history effects was needed to model the relative axial buildup of plutonium in the assemblies which have high burnup. The local reactivity dependence on the moderator and fuel temperature history effects are shown in Figures 5-1 and 5-2, respectively. The reactivity differences between the top and bottom core regions caused by moderator temperature history effects can create power

offset* differences. Several cycles have been depleted both with and without these new tablesets and 1 to 2% more positive offsets were observed when the temperature effects were included.

The detailed three-dimensional studies to establish reactor operating limits with the updated FLAME3 model are currently in progress.

5.2.1.3. Summary

The physics parameters for extended-burnup fuel cycles were generated for use in safety and reactor control analyses. In general, the physics parameters were similar to values for currently licensed reloads, although in some cases they did exceed the range of values normally encountered. The most significant parameter was the more negative moderator coefficient for extended-burnup cycles, the acceptability of which must be determined by detailed safety evaluations.

5.2.2. Mechanical Analyses

5.2.2.1. General

The objective of this subtask is to determine the burnup at which the mechanical limits for the B&W Mark B fuel assembly design are reached. During the present report period, cladding strain and creep collapse analyses were performed. These analyses were based on the five fuel cycle designs described in section 5.1 of this report.

Fuel assembly power histories from the five fuel cycle designs were examined to determine the most limiting assemblies for creep collapse and cladding strain calculations. The most limiting assemblies are those with the highest power throughout life and with the longest incore residence time. Fuel assemblies with incore residences of two, three, four, and five cycles were analyzed. Due to the evolution of the Mark B fuel rod design over the years, four existing variations of the Mark B fuel rod design were used in conjunction with the most limiting assembly power histories to develop generic enveloping analyses.

$$\text{*offset} = \frac{\text{power in top half of core} - \text{power in bottom half of core}}{\text{total power}}$$

These designs are representative of currently operating Mark B fuel. The basic differences in the designs are the initial pellet diameters and densities and the fuel stack length (see Table 5-7). All cladding dimensions are identical.

5.2.2.2. Results and Discussion

Cladding Creep Collapse Analyses — Cladding creep under external pressure is a phenomenon that must be evaluated in fuel rod design, particularly for extended-lifetime fuel. Cladding creep is a function of neutron flux, temperature, and applied stress. Ovality changes in fuel rod cladding due to these phenomena need to be investigated. The CROV computer code was used to perform the fuel rod creep collapse analyses.¹³ CROV predicts the conditions necessary for collapse of a fuel rod and the resultant time to collapse. Comparison of the predicted time to collapse and the planned operating lifetime of the rod determines acceptability.

The TACO2 computer code¹⁴ was used to generate fuel rod internal pressures and cladding temperatures at various time steps for each of the four Mark B designs listed in Table 5-7. These data were input to the CROV code. Zero fission gas release was assumed in order to provide a conservative evaluation of fuel rod internal pressures for creep collapse analyses.

Analyses were performed using the minimum initial fill gas pressure allowed by manufacturing specifications. Preliminary results showed that of the four design variations considered, 95% TD fuel exhibited the greatest resistance to creep collapse. Creep collapse resistance improved because less fuel densification occurs with the 95% TD fuel; thus, this fuel rod operates at a higher internal pressure than one with lower density fuel. (All designs have the same initial fill gas pressurization.) Further refinements are being made in the analyses in order to provide specific collapse times for each design. It is expected that cladding creep collapse will not be the limiting mechanical concern for high-burnup Mark B fuel assemblies because of the high prepressure of the Mark B design.

Cladding Strain Analysis — The fuel pellet/cladding gap sizes were optimized to balance pellet-induced cladding strain and fuel temperatures. During fuel rod operation, the effective pellet/cladding gap decreases until hard pellet/cladding contact is reached. At this point, pellet-induced cladding strain occurs. Calculations were performed on the four Mark B fuel rod designs given

in Table 5-7 to confirm that the uniform transient diametral strain does not exceed a 1.0% strain criterion.

Pellet/cladding strain was simulated using the TACO2 code, which includes a model that is a function of burnup and time-averaged linear heat rate. The code incorporates the following effects for gap closure:

1. Thermal expansion
2. Irradiation growth of the cladding
3. Fuel densification/swelling
4. Cladding creep
5. Cladding elastic deformation

A conservative power history, including transient conditions, was used for each design so that worst-case circumferential uniform strain (transient-induced deformation) was calculated.

The Mark B cladding strain analyses modeled a maximum fuel rod average burnup of 62,000 MWd/mtU. Maximum cladding transient strain was calculated for the maximum pellet burnup location (74,000 MWd/mtU). Using conservative fuel rod dimensions and transient conditions, calculated cladding strain values were within design limits.

5.2.2.3. Conclusions

Based on the results of the mechanical analysis of cladding strain and the preliminary results for cladding creep collapse, the standard Mark B fuel rod design is acceptable for cladding strain to fuel rod average burnups of 62,000 MWd/mtU. Although analyses are not finalized, it is certain that creep collapse results for higher density fuel will be better than for the lower density fuel of the Mark B design. Development of specific creep collapse limits requires further evaluations.

5.2.3. Thermal-Hydraulic Analyses

5.2.3.1. General

Development of fuel parameters and thermal data required for a 177-fuel assembly, lowered-loop LOCA analysis was performed during the current report period. The bottom of the steam generator in lowered-loop plants is at about the same elevation as the reactor vessel, whereas the steam generators in raised-loop plants are at a higher elevation relative to the reactor vessel. B&W's operating plants are of the lowered-loop design. This section describes

analytical models and methods used in generating the LOCA analysis input. Section 5.3 of this report gives the results of the LOCA analyses conducted to date.

The present B&W emergency core cooling system (ECCS) evaluation model¹⁵ utilized fuel parameters and inputs generated with the TAFY computer code.¹⁶ TACO2¹⁴, which is B&W's latest fuel rod performance code, employs modeling techniques not contained in the TAFY code. The fission gas release model is a function of burnup, open porosity, and temperature, whereas the model in TAFY is solely a function of temperature. The effects of fuel densification kinetics as well as cladding creep have been incorporated into TACO2. The gap conductance in TACO2 between the fuel OD and cladding ID includes the effect of pellet cracking/relocation. The TACO2 computer code was submitted to the NRC in January 1979 for review and approval for licensing applications.

5.2.3.2. Methods

Data for the LOCA analyses were generated for three assembly power levels: (1) maximum design power (hot assembly), (2) 10% reduction of the maximum design power level, and (3) core average fuel rod power. For generic applicability to 177-FA plants a core power of 2772 MWt was used. The same fuel rod power history was used for the fuel rod internal pressure and temperature calculations for both the 100 and 90% maximum design power cases. "Average" rod calculations were performed for a hypothetical fuel rod operating at the core average linear heat rate throughout its operating history.

The initial fuel parameters for the LOCA analyses were conservatively calculated using an enveloping fuel rod power history and a worst-case axial power shape. These conservatisms result in an over-prediction of fuel rod power versus burnup, which in turn increases the predicted release of fission products. The additional gaseous fission products in the void volume of the rod degrade gap conductance, resulting in elevated temperature and increased internal pressure. The application of these conservatisms is discussed below.

Fuel Rod Power History — A conservative fuel rod power history was used for fuel thermal analyses to calculate fuel rod internal pressure and temperature. The shape of this design history was obtained by enveloping a composite plot of individual fuel rod histories so that no single fuel rod in the core would experience a power history this severe.

Previous extended-burnup analyses were performed with a fuel rod design power history based on fuel cycle designs that were optimized to relatively low burnup values (38,000 MWd/mtU fuel rod average). The power history for the previous analyses was derived by expanding its shape proportionately to a burnup of 60,000 MWd/mtU. Both the original design power history and the expanded history are shown in Figure 5-3.

Several extended-burnup fuel cycle designs have since been developed (see section 5.1), and fuel rod power histories were obtained from those designs using an enveloping technique; a composite plot is shown in Figure 5-4. The power histories based on the extended-burnup fuel cycle studies (Figure 5-4) were compared to the expanded design shape (Figure 5-3) and found to represent lower power throughout life except for a few points beyond a burnup of 40,000 MWd/mtU.

In order to introduce conservatism into the LOCA analysis, the expanded design power history from Figure 5-3 was used with minor adjustments to envelop the extended-burnup fuel cycle design histories. Figure 5-5 compares the final power history used for the LOCA analysis with the envelope derived from the extended-burnup fuel cycles.

Pellet-to-Cladding Gap — Fuel rod temperature is directly affected by the pellet-to-cladding gap. To determine the pellet-to-cladding gap, a statistical study of typical Mark B cladding IDs has been conducted and the results combined with the lower tolerance limit specification for the fuel pellet OD. This statistical gap resulted in a 100°F rise in the predicted peak fuel temperature relative to using a nominal gap; however, the corresponding increase in effective void volume caused a reduction in predicted EOL fuel rod internal pressure. Since the reduction in pressure (due to void effects) is non-conservative for some analyses that depend on rod internal pressure, the nominal rather than the statistical gap was used in the pressure studies. In addition, sensitivity studies demonstrated that for smaller gaps, high pressure could not be experienced because of the compensating decrease in fuel temperature.

Axial Power Shape — Localized power increases due to plant maneuvering were considered in the thermal performance calculations. A large variety of core power distributions is possible during maneuvering, and it is not feasible in fuel performance evaluations to analyze all possible perturbations to the core power distribution. Therefore, an axial flux shape that is conservative relative to the power experienced by a typical Mark B fuel rod was chosen for

generation of LOCA analysis input. To remove any concern as to the number of times the fuel rod could experience a perturbation of this nature, a 1.67 cosine shape was held constant throughout life for the pressure calculations. Maneuvering power peaks are generally localized in a section of the fuel rod approximately 2-feet long. The symmetric 1.67 cosine shape maintained more of the fuel at a higher power level, resulting in greater fuel temperatures and increased pressures.

5.2.3.3. Results

The LOCA analysis inputs for fuel rod internal pressure and temperature at the highest power level analyzed are shown in Figures 5-6 and 5-7, respectively. The end-of-life fuel rod internal pressure was 2760 psia. This does not imply a change to the mechanical design criterion, which requires fuel rod internal pressure to remain below system pressure. The LOCA analysis input was generated with conservative methods in order to provide margin for future minor modifications to the fuel and/or cycle design that may result from experience gained in this program.

Table 5-1. Moderator and Doppler Reactivity Coefficients for Equilibrium Cycles

Feed batch size	Core power, MWt	Operating mode	Moderator coefficients, $10^{-4} \Delta\rho/^{\circ}\text{F}^{(a)}$			Doppler coefficients, $10^{-5} \Delta\rho/^{\circ}\text{F}^{(a)}$		
			HZP, BOC	HFP, BOC	HFP, EOC	HZP, EOC	HFP, BOC	HFP, EOC
80	2772	FB ^(a)	+0.25	-0.53	-3.16	-1.72	-1.53	-1.69
68	2772	FE	+0.18	-0.62	-3.28	-1.77	-1.59	-1.74
60	2772	FE	+0.18	-0.62	-3.31	-1.81	-1.63	-1.73
60	2568	R	+0.13	-0.64	-3.24	-1.79	-1.63	-1.80
36	2772	FB	-0.34	-1.24	-3.25	-1.84	-1.57	-1.67

(a) FB: feed and bleed, R: rodged, HZF: hot zero power, HFP: hot full power, BOC: beginning of cycle, EOC: end of cycle.

Table 5-2. Comparison of Reactivity Coefficients for ANO-1 Cycle 3, ANO-1 Cycle 4, and 60-FA Feed, Rodded Equilibrium Cycle

	ANO-1, cycle 3	ANO-1, cycle 4	60-FA feed, rodged equil. cycle
Moderator coefficients, $10^{-4} \Delta\rho/^{\circ}\text{F}$			
HZP, BOC	+0.26	+0.34	+0.13
HFP, BOC	-0.66	-0.48	-0.64
HFP, EOC	-2.62	-2.78	-3.24
Doppler coefficients, $10^{-5} \Delta\rho/^{\circ}\text{F}$			
HZP, BOC	-1.96	-1.84	-1.79
HFP, BOC	-1.47	-1.57	-1.63
HFP, EOC	-1.61	-1.71	-1.80

Table 5-3. Moderator and Doppler Deficits

Feed batch size	Core power, MWt	Operating mode	Cycle ^(a)	Moderator deficit, % $\Delta\rho$		Doppler deficit, % $\Delta\rho$	
				BOC	EOC	BOC	EOC
80	2772	FB	Transition 1	0.324	1.232	1.246	1.238
			Equilibrium	0.324	NA	1.292	NA
68	2772	FB	Transition 1	0.334	1.312	1.288	1.276
			Equilibrium	0.390	1.357	1.316	1.204
60	2772	FB	Transition 1	NA	1.309	NA	1.258
			2	NA	1.323	NA	1.201
			3	0.364	1.351	1.275	1.162
			Equilibrium	0.394	1.322	1.303	1.214
60	2568	R	Transition 1	0.415	1.253	1.203	1.186
			2	NA	1.300	NA	1.181
			Equilibrium	NA	1.236	NA	1.259
36	2772	FB	Transition 1	0.685	1.280	1.272	1.232
			Equilibrium	0.709	1.336	1.207	1.116

(a) Transition N designates first, second, or Nth cycle of indicated feed batch size. Equilibrium designates the equilibrium cycle following transition cycles of a give feed batch size. NA: Not available.

Table 5-4. Comparison of Selected Physics Parameters for ANO-1 Cycle 3, ANO-1 Cycle 4, and 60-FA Feed, Rodded Transition Cycle

	ANO-1, cycle 3	ANO-1, cycle 4	Rodded transition cycle
Cycle length, EFPD	294	387	466
Core average initial enrichment, w/o ^{235}U	2.88	2.95	3.45
Burnup, MWd/mtU			
BOC	10,205	8,394	10,941
EOC	19,405	20,505	25,524
Power deficit, % $\Delta\rho$			
BOC	1.67	1.78	2.10
EOC	2.89	3.43	3.59
Effective delayed neutron fraction			
HFP, BOC	0.00591	0.00617	0.00606
HFP, EOC	0.00524	0.00517	0.00514

Table 5-5. Comparison of Nuclear Input to Radiation Analysis for ANO-1 Cycle 3, ANO-1 Cycle 4, and 60-FA Feed, Rodded Equilibrium Cycle

Batch	No. of assemblies	Initial enrichment w/o ^{235}U	Burnup, MWd/mtU		Batch relative power	Batch power fraction	Isotopic power fraction	
			Batch average	Maximum assembly			U	Pu
<u>ANO-1, Cycle 3</u>								
1B	5	2.06	20,207	23,748	0.70	0.02	0.37	0.63
3	60	3.05	30,460	32,808	0.96	0.33	0.40	0.60
4	56	2.64	17,935	19,957	1.10	0.35	0.52	0.48
5	56	3.01	8,959	12,704	0.97	0.30	0.72	0.28
Core	177	2.88	19,405	--	1.00	1.00	0.53	0.47
<u>ANO-1, Cycle 4</u>								
1C	1	2.06	24,977	24,977	0.60	0.01	0.32	0.68
4	56	2.64	28,486	32,915	0.88	0.28	0.36	0.64
5	56	3.01	18,904	28,289	0.83	0.26	0.53	0.47
6	64	3.19	15,209	16,656	1.25	0.44	0.62	0.38
Core	177	2.95	20,634	--	1.00	1.00	0.52	0.48
<u>60-FA Feed, Rodded Equilibrium Cycle</u>								
8	57	4.11	46,004	55,713	1.03	0.33	0.36	0.64
9	60	4.11	31,384	39,381	0.72	0.25	0.49	0.51
10	60	4.11	19,976	22,781	1.25	0.42	0.63	0.37
Core	177	4.11	32,225	--	1.00	1.00	0.51	0.49

Table 5-6. FLAME3 Fitting Ranges of Important Core Parameters

<u>Parameter</u>	<u>Range</u>
Fuel enrichment, wt % ^{235}U	2.0 to 4.5
LBP enrichment, wt % B_4C	0.0 to 2.0
Moderator temperature, °F	532 to 620 @ 2200 psia
Fuel temperature, °F	600 to 2400
Burnup, MWd/mtU	0 to 80,500
Boron, ppm	0 to 1500

Table 5-7. Mark B Fuel Rod Designs

<u>Parameter</u>	<u>I</u>	<u>II</u>	<u>III</u>	<u>IV</u>
Fuel pellet initial density (nom.), % TD	93.5	94.0	94.0	95.0
Fuel pellet OD (mean specif.), in.	0.3700	0.3697	0.3695	0.3685
Undensif. active fuel length (nom.), in.	142.60	143.50	142.25	141.75
Fuel rod OD (nom.), in.	0.430	0.430	0.430	0.430
Fuel rod ID (nom.), in.	0.377	0.377	0.377	0.377

Figure 5-1. K_{∞} Dependence on Moderator Specific Volume
During Depletion

Figure 5-2. K_{∞} Dependence on Fuel Temperature During Depletion

Figure 5-3. Expanded Design Envelope Used for
Extended Burnup Analyses

Figure 5-4. Fuel Rod Power Histories for Extended-Burnup Fuel Cycle Designs

Figure 5-5. Fuel Rod Power History Envelopes

Figure 5-6. Fuel Rod Internal Pressure Vs Burnup

Figure 5-7. Fuel Rod Average Temperature at
18.0 kW/ft. Vs Burnup

5.3. Subtask 2C — Safety Analysis and ECCS Evaluation

During the design of each reload core, FSAR accident analyses are examined with respect to the characteristics of each reload to determine its effects on accident analyses and to ensure that thermal performance during hypothetical transients is not degraded. Under the extended-burnup program, the characteristics of extended-burnup cores are being compared to the parameters assumed during the Final Safety Analysis Report (FSAR) accident evaluations to identify any areas that may be affected. This review considers three areas: kinetics-dependent accidents, dose evaluations, and the LOCA. Efforts to date have included calculation of fission product inventories for dose evaluations and a burnup sensitivity study for the LOCA. Progress in these areas is reported in the following sections.

5.3.1. Safety Analysis

5.3.1.1. General

Extending fuel burnup will increase the fuel fission product inventory, which will affect the calculated doses for the accidents analyzed in the FSAR. Burnup factors that can affect fission product activities are longer fuel residence times and increased power contribution from plutonium fissions.

Burnup data covering batch average discharge burnups ranging from 34,000 to 48,500 MWd/mtU were used to determine fission product activities. The fuel burnup data enveloped a wide range of burnup parameters, including feed batch size, cycle length, batch power, and the fraction of power produced by both uranium and plutonium, as described in section 5.1. Fuel assembly average burnups of up to 55,000 MWd/mtU were considered.

5.3.1.2. Results and Discussion

The several sets of high-burnup nuclear data corresponding to various feed batch sizes were reviewed to determine the one set that would produce the highest activities for the fuel, the fuel rod gap region, the reactor coolant system, and a single assembly and thus result in the highest FSAR accident doses. The fission product activities for each set of fuel data were calculated by a digital computer code which solves the differential rate equations for fission product buildup within the fuel. The code considers 181 isotopes in 80 decay chains with a maximum chain length of five isotopes. The activity can

be calculated for up to 100 time intervals. For each time interval, the thermal core power, the thermal neutron flux, and the fraction of power produced by ^{235}U and ^{239}Pu are input.

To ensure that the high-burnup fuel data represent an upper bound, uncertainty factors were applied to the data. The isotopic plutonium and uranium power fractions were increased by a factor of 1.12. The 12% component simulates an increase of about 15% in the discharge burnup (cycle length). The thermal neutron flux was increased by a calculational uncertainty factor of 1.10.

Core Fission Product Activity — The fission product inventory in the fuel was calculated using the calculational methods described in section 11.1 of the FSAR and the adjusted fuel data described in the preceding section. The following assumptions were also used:

1. Equilibrium cycle thermal core power of 2772 MWt.
2. Operation with no defective fuel assemblies, so that the maximum activities are calculated.

Table 5-8 lists the fuel fission product activities to be used in the dose evaluation of high-burnup fuel.

Fuel Rod Gap Region Fission Product Activity — The fission product inventory in the fuel rod gap region (volume between the uranium fuel pellets and the fuel rod cladding) was based on the same calculational assumptions as for the fuel inventory except that an escape rate coefficient was used to represent the fraction of activity produced in the fuel that is released into the gap region. The following escape rate coefficients were used:

Element	Escape rate coeff, s^{-1}
Xe, Kr	1.0×10^{-7}
I, Br, Cs, Rb	2.0×10^{-8}
Mo, Te, Se, Nb, Tc, Ru, Pd, Ag, Sb, Rb	4.0×10^{-9}
Sr, Ba, Cd	2.0×10^{-10}
All others	1.0×10^{-11}

The fuel rod gap fission product activities to be used in the dose evaluations of high-burnup fuel are also listed in Table 5-8.

Reactor Coolant System Fission Product Activity — The fission product inventory in the reactor coolant system was calculated in the same manner as the gap activity except that all of the activity released to the gap region was assumed to leak instantaneously into the coolant. The following additional assumptions were used:

1. Operation with 1% failed fuel. All fuel cladding failures were assumed to occur during the fuel's last cycle in core, developing leaks at the beginning of the fuel cycle of interest and leaking continuously during the cycle.
2. Continuous reactor coolant circulation through the purification demineralizer at an average rate of one reactor coolant volume per day ($1.157 \times 10^{-5} \text{ s}^{-1}$) with a zero removal efficiency for Te, Y, Mo, Cs, and all noble gas isotopes. All other elements were assumed to be removed with a 95% removal efficiency.
3. Purification efficiency of 99% for the bleed system for all elements during the first 90% of the fuel cycle. After that time, when the deborating demineralizers are used for bleed stream processing, the removal of all radionuclides from the bleed system ceases.

Table 5-9 lists the reactor coolant system fission product activities for operation with 1% failed fuel.

Single Assembly Fuel and Gap Fission Product Activity — Fission product activity in the fuel and rod gap regions of a single fuel assembly were calculated in a manner similar to that used for the core and fuel gap activities with the additional following assumptions:

1. An assembly power of 30 MWt was used. This power level corresponds to an equivalent assembly peaking factor of 1.92 (ratio of 30 MWt to the average assembly power).
2. The assembly contained no defective fuel rods, so that the maximum fission product activities were obtained.

The resultant fission product activities in the fuel and gap regions of a single fuel assembly are listed in Table 5-10.

RELOAD Computer Code — The RELOAD computer code is being developed to facilitate evaluation of the dose impact of extended-burnup fuel. RELOAD is designed to

provide (1) the accident doses for reload cores containing fuel batches or sub-batches of high-burnup fuel, (2) a comparison of the reload accident doses with the corresponding FSAR accident doses, and (3) a comparison of the reload doses with the dose limits specified in 10 CFR 100. The code, currently undergoing final testing, will calculate accident doses for the following major FSAR accidents:

1. Fuel handling accident
2. Failure of the waste gas decay tank
3. Steam line break accident
4. Control rod ejection accident
5. Loss-of-coolant accident (LOCA)
6. Maximum hypothetical accident (MHA)
7. Steam generator tube failure

5.3.1.3. Conclusions

Fission product inventories for the fuel, the fuel rod gap region, the reactor coolant system, and a single fuel assembly were calculated for extended-burnup fuel using conservative calculational assumptions. The activities are to be used in the RELOAD code to derive radiological source terms which, in turn, are used to calculate thyroid and whole body doses from hypothetical accidents.

5.3.2. ECCS Evaluation

5.3.2.1. General

As discussed in section 5.2.3 of this report, the B&W ECCS evaluation model used in previous licensing submittals¹⁵ employed both TAFY generated-input and computer code methods.¹⁶ In that evaluation fuel temperatures were not represented as burnup-dependent, while fuel rod internal pressures and cladding oxide layer thicknesses were. This representation results in over-conservatism because fuel temperatures actually decrease with burnup after attaining their peak value very early in life (~60 MWt/mtU). A more realistic LOCA evaluation incorporating the burnup dependency of fuel temperatures as calculated by the TACO2 computer code is being performed under this subtask.¹⁴

Efforts during this report period included an upgrade of the ECCS evaluation model to allow TACO2 input and the initiation of a burnup sensitivity study, which will characterize the burnup for which the peak cladding temperatures

after a LOCA would be at a maximum. Following completion of the burnup study, a three-break mini-spectrum for Mark B fuel is to be performed, as required by NRC regulations, to show that the break spectrum behavior is unaltered by the evaluation model change. These analyses, coupled with the burnup sensitivity study, will result in LOCA limits for extended-burnup Mark B fuel.

5.3.2.2. Methods

To determine the effect of burnup on peak cladding temperature (PCT) during a LOCA, the extended-burnup sensitivity study used the break configuration established as the worst case in reference 17. This was an 8.55-ft² double-ended break at the pump discharge with a discharge coefficient of 1.0. Maximum peak cladding temperatures result for this break.

Several computer codes are used to evaluate the effects of a LOCA: CRAFT2¹⁸, REFLOOD¹⁹, THEAT1-B²⁰, and CONTEMPT²¹. CRAFT2 models reactor coolant system thermal-hydraulic behavior and core power generation during the blowdown phase of a LOCA. REFLOOD simulates the reactor coolant system behavior during core refill and reflood. CONTEMPT calculates reactor building internal pressure. THEAT1-B is used with output from CRAFT2, REFLOOD, and CONTEMPT to determine the peak cladding temperature.

The logic employed to identify the worst time in life for a LOCA is described below. The general behavior of peak cladding temperature (PCT) versus burnup is illustrated in Figure 5-8 for three parameters (fuel temperature, cladding oxide layer thickness, and fuel rod internal pressure), which also vary as a function of fuel burnup. The fuel temperature effect reflects the rise in fuel temperature at low burnups associated with fuel densification and a gradual decrease in fuel temperature with fuel swelling and cladding creepdown as burnup increases. The buildup in oxide layer thickness with increasing burnup is shown by a decrease in PCT. The increase in fuel rod internal pressure with burnup is displayed by the increase in PCT with burnup to a hypothetical point at which it is theorized that sufficient flow blockage could occur to yield uncontrolled PCT.

The cumulative effect of the three parameters is also shown in Figure 5-8. Two discontinuities exist in the cumulative effect PCT versus burnup curve (Figure 5-8d). They are caused by the effects of the fuel rod internal pressure, resulting in cladding rupture. The second discontinuity would produce

an uncontrolled cladding temperature transient. Thus, there would appear to be a limitation on the maximum burnup that can be tolerated for the LOCA.

The reasons for discontinuities in the cumulative effect PCT versus burnup curve can be seen in Figure 5-9, in which the solid curve represents cladding temperature versus time during a LOCA. The dashed lines represent the temperature at which the cladding will rupture for some given initial burnup and corresponding fuel rod internal pressure. When the solid and dashed curves intersect, cladding rupture occurs. The dashed lines numbered 1 through 5 reflect increasing burnup and the buildup of fuel rod internal pressure. Examining Figure 5-9, it is seen that as the fuel rod pressure increases between curves 1 and 2, all cladding ruptures occur after the end of blowdown (EOB). Since there is no effect on the blowdown portion of the transient, the oxide layer increase will decrease PCT (see Figure 5-8b). However, as the fuel rod internal pressure is increased from the value that yields curve 2 to that which yields curve 3, the time for cladding rupture will abruptly change (from about 23 to about 13 seconds) and will cause flow blockage during blowdown. The additional resistance caused by the flow blockage results in decreased core flow, which decreases heat transfer and thus causes a marked increase in PCT. Between curves 3 and 4, the increased oxide layer effect is sufficient to offset the effect of increased fuel rod pressure — thus, PCT decreases. Between curves 4 and 5, however, another abrupt change — caused by increasing fuel rod internal pressure — occurs in the time to fuel rupture, and another discontinuity is encountered in the PCT versus burnup curve. Due to the extremely early time for this rupture (about 3 seconds), only a small fraction of the initial stored energy in the fuel is removed and the cladding temperature becomes uncontrollable during reflooding of the core.

Based on this logic, the following cases were selected for evaluation during the burnup sensitivity study:

- Maximum fuel temperature.
- 40,000 MWd/mtU fuel rod average burnup.
- 58,000 MWd/mtU fuel rod average burnup.

The maximum fuel temperature spike occurring during densification at BOL was chosen because the initial stored energy of the fuel is at its peak. This peak is important because the stored energy of the fuel is the primary driver for the cladding temperature response. The time in life when the hot fuel rod

reaches a burnup of 40,000 MWd/mtU was chosen because the fuel rod internal pressure used in the LOCA evaluation was approximately the reactor coolant system operating pressure. End-of-life (EOL = 58,000 MWd/mtU) core conditions were selected for analysis because it is at this time that fuel rod internal pressure is at its maximum. The burnup sensitivity studies for these cases are currently in progress.

Table 5-8. Extended-Burnup Fuel and Fuel Rod Gap
Fission Product Activities^(a)

Isotope	Activity, Ci	
	Fuel	Gap
Kr83m	9.60+06 ^(b)	0.17+04
Kr85m	2.24+07	5.11+04
Kr85	8.65+05	7.76+05
Kr87	4.05+07	2.67+04
Kr88	5.69+07	8.27+04
Xe131m	5.16+05	7.52+04
Xe133m	3.20+06	9.44+04
Xe133	1.33+08	8.54+06
Xe135m	3.54+07	1.30+04
Xe135	2.17+07	2.03+05
Xe138	1.25+08	1.52+04
I131	8.66+07	1.71+06
I132	9.27+07	1.68+05
I133	1.34+08	2.93+05
I134	1.73+08	1.76+04
I135	1.37+08	9.51+04

(a) Based on data from fuel cycles with discharge batch average burnups of up to 48,500 MWd/mtU.

(b) $9.60+06 = 9.60 \times 10^6$.

Table 5-9. Reactor Coolant System Fission Product Activities
Vs Time for Base-Loaded Operation of 177-FA Plant
at 2772 MWt With 1.0% Failed Fuel

Isotope	Activity, $\mu\text{Ci/gm}^{(a)}$					
	4 EFPD	100 EFPD	200 EFPD	300 EFPD	400 EFPD	469 EFPD
Kr83m	5.35-01 ^(b)	5.37-01	5.14-01	5.37-01	3.57-01	5.37-01
Kr85m	2.39	2.40	2.29	2.40	2.40	2.41
Kr85	7.16-01	1.30+01	1.93+01	1.90+01	1.34+01	1.64+01
Kr87	1.25	1.26	1.20	1.26	1.26	1.26
Kr88	3.87	3.89	3.71	3.89	3.89	3.89
Xe131m	3.45-01	2.96	3.06	3.00	2.86	3.09
Xe133m	9.72-01	4.25	4.08	4.24	4.20	4.27
Xe133	4.04+01	3.91+02	3.76+02	3.89+02	3.81+02	3.95+02
Xe135m	5.05-01	5.05-01	4.88-01	5.05-01	5.05-01	5.06-01
Xe135	8.70	8.31	8.31	8.30	8.29	8.31
Xe138	7.14-01	7.14-01	6.88-01	7.14-01	7.14-01	7.14-01
I131	3.61	6.69	6.69	6.69	6.69	6.69
I132	1.78	6.95	6.71	6.94	6.85	6.99
I133	5.77	6.26	6.00	6.26	6.26	6.26
I134	7.89-01	7.89-01	7.61-01	7.89-01	7.89-01	7.89-01
I135	3.24	3.24	3.12	3.24	3.24	3.24

(a) Based on data from fuel cycles with discharge batch average burnups of up to 48,500 MWd/mtU.

(b) 5.35-01 = 5.35×10^{-01} .

Table 5-10. Extended-Burnup Single Assembly Fuel and Fuel Rod Gap Fission Product Activities^(a)

Isotope	Activity, Ci	
	Fuel	Gap
Kr83m	9.19+04 ^(b)	1.12+02
Kr85m	2.03+05	4.65+02
Kr85	1.10+04	1.03+04
Kr87	3.56+05	2.35+02
Kr88	5.09+05	7.39+02
Xe131m	5.78+03	8.42+02
Xe133m	3.10+04	9.15+02
Xe133	1.29+06	8.28+04
Xe135m	3.91+05	1.41+02
Xe135	2.59+05	2.30+03
Xe138	1.31+06	1.59+02
I131	9.70+05	1.91+04
I132	9.78+05	1.75+03
I133	1.29+06	2.83+03
I134	1.84+06	1.86+02
I135	1.45+06	1.01+03

(a) Based on data from fuel cycles with discharge assembly average burnups of up to 55,000 MWd/mtU.

(b) $9.19+04 = 9.19 \times 10^6$.

Figure 5-8. Peak Cladding Temperatures as a Function of Burnup

a) Fuel Temperature Effect

b) Oxide Layer Effect

c) Fuel Rod Internal Pressure Effect

d) Cumulative Effect

Figure 5-9. Effect of Fuel Rod Internal Pressure on Rupture Time

5.4. Subtasks 2D and 2E - Control Analysis and System Evaluation

5.4.1. General

The five extended-burnup fuel cycle designs described in section 5.1 were analyzed for their compatibility with B&W 177-FA NSS fluid system capabilities. The following areas were investigated:

1. The capability of the design to meet the boric acid storage requirements for reactor shutdown imposed by Technical Specifications.
2. The boric acid volumes required for refueling.
3. Short-term demineralized water requirements.
4. Reactivity control requirements for safe shutdown with respect to existing requirements.

The result of the investigations was that the B&W 177-FA plant designs were found to be compatible with the requirements imposed by extended-burnup fuel cycles. The remainder of this section discusses more fully the requirements imposed and the cycle-to-cycle variations encountered.

5.4.2. Results and Discussion

Technical Specifications - Technical Specifications require that all operating reactors store enough boric acid to bring the reactor coolant system to a cold shutdown condition with a 1% $\Delta\rho$ shutdown margin and no xenon present. Two sources of boric acid are available: The first is the borated water storage tank (BWST). The concentration of boron in the BWST ranges between 1800 and 2270 ppm for the 177-FA operating units. The tank contains much more borated water than is needed for cold shutdown since it serves as the source of water for the high-pressure injection pumps, which supply water to replenish the reactor coolant system inventory following a LOCA.

The second source of borated water is the concentrated boric acid supply, which is stored in either a boric acid addition tank or a concentrated boric acid storage tank. If the concentration of boric acid stored in these tanks is greater than approximately 3.5 wt %, the solution must be heated to prevent the formation of boron crystals. The greater the concentration, the higher the temperature at which the solution must be maintained. The equipment used to maintain the proper temperatures in the tanks and piping of the chemical addition systems is called the "heat tracing" system. In effect, it limits

the concentration of boric acid operating plants can store. Plants with limited storage volume must rely on higher boric acid concentrations.

The required volumes were calculated by assuming that the target shutdown concentration is achieved through a feed-and-bleed process while the plant is in a hot shutdown condition. Three phases constituted the shutdown operation. During the first phase, the critical boron concentration was maintained by filling the hot full power-to-hot shutdown contraction volume with boric acid and demineralized water. This contraction volume resulted from system shrinkage due to the reduction in temperature as the plant went from full power to hot shutdown. During this phase, boric acid was fed to the system while bleed was diverted to the bleed holdup tank. This operation was conducted while the plant was at hot shutdown and continue until the cold shutdown concentration was achieved. In the second phase the plant was brought to a cold shutdown condition.

The third phase consisted of filling the hot shutdown-to-cold shutdown contraction volume with the amount of boric acid and demineralized water that maintained the required cold shutdown concentration. A 10% conservatism was applied to all volumes. (Volumes are referenced to 120F, 150 psia.)

The extended-burnup fuel cycles required more boric acid storage than the present 177-FA fuel cycles. Plants with Standard Technical Specifications currently require the equivalent of 986 ft^3 (referenced to 120F) of 5 wt % boric acid for cold shutdown. For plants with plant-specific technical specifications, the most restrictive cycle design encountered to date required 1147 ft^3 of 4.98 wt % (8700 ppm water). This was for a rodded design. Table 5-11 shows the maximum volume requirements of the five extended-burnup designs. (For simplicity, the volume requirements were rounded to the next highest 25 ft^3 .) The rodded 60-FA feed design had the greatest volume requirement. The technical specification volume requirement for the most restrictive design was 11.2% greater than the rodded plant maximum. For the feed-and-bleed cycles, the technical specification volume for the extended-burnup cycle design was as much as 16.6% greater than the Standard Technical Specification requirement. These differences are not of great concern but do indicate that extended-burnup cycles will have increased soluble boron requirements.

Table 5-12 shows the concentrated boric acid storage volumes available and the range of boric acid concentrations that are permitted by plant system capabilities. Also shown are the boric acid storage volumes that would be required by the technical specifications. It is clear from the table that the only plant that could not meet the storage requirement for the minimum boric acid storage concentration was ANO-1. However, this plant could utilize 7 wt % boric acid in its chemical addition system and thus could conform to the technical specifications.

Adoption of extended burnup cycles will require changes in the technical specifications for all plants regarding the volumes of boric acid to be maintained in storage. Only for ANO-1 will the specification concerning the minimum allowable concentration have to be changed.

Refueling — Refueling volumes for the extended-burnup cycle designs will be greater than those now required. The calculated refueling volumes, which were compiled on a comparative basis, are summarized in Table 5-13. A value of 17 ppm was assumed for an EOL critical boron level prior to refueling. The extended-burnup results were based on the highest refueling concentration required for any cycle in a particular fuel cycle design. In some cases the volume requirement for refueling was greater than the available boric acid storage volume; however, fresh batches of boric acid can be mixed in the boric acid mix tank for subsequent addition to the reactor coolant system as required.

Demineralized Water Requirements — Demineralized water requirements during the first 4 days of a fuel cycle were analyzed. During this time the boron concentration is reduced from the refueling concentration to the zero-day critical concentration. Over the first 4 EFPD of the cycle, the concentration is reduced further to compensate for the buildup of xenon. This represents the short-term water requirement. Subsequent feed-and-bleed operations utilizing demineralized water are required to remove enough boron from the reactor coolant system to compensate for fission product buildup. The fuel cycle depletion (4 days to EOL) volume requirement is primarily dependent on how low the boron concentration must be reduced. The greatest portion of the feed-and-bleed volume is needed during the final 10-15% of the fuel cycle. The plant-to-plant and cycle-to-cycle variation in the EOC critical boron concentration for extended-burnup cycles is expected to be about the same as that for existing fuel cycles. Since the initial (refueling) concentrations for the

extended-burnup cycles were generally higher than those for existing fuel cycles, an extended-burnup cycle would require more demineralized water if the EOC concentrations were identical. The volume required to deborate from 2523 to 2270 ppm is 939 ft³, whereas that needed to deborate from 60 to 40 ppm is 3348 ft³. Thus, the effect of increased initial concentration on the total (refueling to EOC) demineralized water requirement is insignificant when compared to that resulting from variations expected in EOC critical levels.

Table 5-14 shows the refueling concentrations to 4-day critical concentration deboration volumes for both extended-burnup cycles and the maximum cycle requirements encountered to date. The extended-burnup cycles, although requiring higher refueling concentrations, have lower short-term demineralized water requirements because the 0- to 4-day critical boron levels are higher than normal. Thus, less deboration is required.

Safe Shutdown — The limiting safe hot shutdown case occurs when trippable control rod worth provides just enough reactivity to yield the required 1% $\Delta\rho$ shutdown margin after compensation for the moderator and Doppler reactivity deficits. In this case, the reactor coolant soluble boron concentration must be increased to account for long-term xenon decay. Therefore, one measure of the magnitude of the hot shutdown requirements is equilibrium xenon worth.

A comparison of the EOL xenon worths (% $\Delta k/k$) times soluble boron requirements (inverse boron worth, ppm/% $\Delta k/k$) for extended-burnup cycles yielded values similar to those of current reload cycles (see Table 5-15). Thus, hot safe shutdown requirements for extended-burnup cycles are not expected to be significantly different from the requirements for current reload cycles.

Since final ground rules for cold safe shutdown have not been established for operating plants, these requirements were not assessed.

5.4.3. Conclusions

The results of the analyses performed indicate that the 177-FA plant designs are compatible with extended-burnup designs. The minimum allowable boric acid concentration to be maintained in the concentrated boric acid supply system will have to be increased, as will the actual boric acid storage volume required. These requirements can be implemented via Technical Specification changes and have been verified as within plant design capabilities.

Table 5-11. Summary of Shutdown Volumes

Feed concen., wt %	Fuel cycle design	Volume required	Volume required	— Tech Spec — Volume required
		to attain hot shutdown condition with stuck rod, ft ³	to attain hot shutdown condition with all rods out, ft ³	to attain cold shutdown condition with stuck rod, ft ³
3.5	60 FA feed (F/B)	400	1900	1550
	68 FA feed (F/B)	400	1825	1550
	80 FA feed (F/B)	350	1725	1475
	60 FA feed (rodded)	550	2050	1900
	36 FA feed (F/B)	525	1575	1700
4	60 FA feed (F/B)	350	1600	1325
	68 FA feed (F/B)	350	1525	1350
	80 FA feed (F/B)	300	1450	1275
	60 FA feed (rodded)	475	1725	1625
	36 FA feed (F/B)	450	1325	1475
5	60 FA feed (F/B)	300	1200	1050
	68 FA feed (F/B)	275	1175	1050
	80 FA feed (F/B)	250	1100	1025
	60 FA feed (rodded)	375	1300	1275
	36 FA feed (F/B)	350	1025	1150
7	60 FA feed (F/B)	200	800	750
	68 FA feed (F/B)	200	800	750
	80 FA feed (F/B)	175	750	725
	60 FA feed (rodded)	275	900	900
	36 FA feed (F/B)	250	700	800

- Notes:
1. Hot and cold shutdown conditions are 1% $\Delta k/k$ subcritical with no xenon.
 2. (F/B) indicates feed-and-bleed cycle; (R) indicates rodded cycle.

Table 5-12. Comparison of 177-FA Plant Concentrated Boric Acid Storage Capabilities and Maximum Extended-Burnup Requirements

Operating plant	Concentrated boric acid storage capability, ft ³	Range of boric acid concentrations permissible, wt %	Extended-burnup Tech Spec storage requirement	
			For minimum concentration	For maximum concentration
Oconee I, II, III ^(a)	3,000	5-7	1275	900
Crystal River III	2,000	7-12	900	--
Arkansas I	1,000	5-7	1275	900
Rancho Seco	8,021	4	1625	--
Midland I, II ^(a)	3,600	3.5	1900	--
Davis Besse I	13,786	5-7	1275	900
Davis Besse II, III ^(a)	13,786	5-7	1275	900

^(a) Values are per NSS.

Table 5-13. Refueling Volumes — Contraction Volume Method

<u>Fuel cycle design</u>	Maximum refueling concen., ppm	Refueling volume of boric acid required, ft ³				<u>% increase at 5 wt % feed</u>	
		<u>3.5 wt %</u>	<u>4 wt %</u>	<u>5 wt %</u>	<u>7 wt %</u>	<u>Above 1800 ppm case, %</u>	<u>Above 2700 ppm case, %</u>
Present	1800	3626	3045	2360	1686	--	--
Present	2270	4955	4139	3085	2129	30.7	--
60 FA feed (F/B)	2523	5739	4774	3545	2367	50.2	14.9
68 FA feed (F/B)	2440	5475	4562	3392	2289	43.7	10.0
80 FA feed (F/B)	2338	5161	4306	3207	2193	35.9	4.0
60 FA feed (rodded)	2470	5571	4638	3447	2317	46.1	11.7
36 FA feed (F/B)	2113	4494	3762	2808	1981	19.0	-9.0

Table 5-14. Demineralized Water Requirement Comparison
(Refueling to 4-Day Critical Concentration)

<u>Case</u>	<u>Bleed and DW volume, ft³</u>
Max. encountered (refueling core = 1800 ppm)	8,300
Max. encountered (refueling core = 2270 ppm)	10,200
60 FA feed (F/B)	6,254
68 FA feed (F/B)	5,854
80 FA feed (F/B)	5,768
60 FA feed (rodded)	7,297
36 FA feed (F/B)	8,017

Table 5-15. EOL Boron Requirements to Compensate for
Xenon Decay - Limiting Case

<u>Fuel cycle</u>	<u>Required boron level</u>	<u>% difference</u>
Midland	296	0
60 FA feed (F/B)	304	+2.7
68 FA feed (F/B)	298	+0.7
80 FA feed (F/B)	284	-4.1
60 FA feed (rodded)	310	+4.7
36 FA feed (F/B)	287	-3.0

6. TASK 3 — EVALUATION OF PWR FUEL FAILURE MECHANISM FOR EXTENDED BURNUP

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6.1. Introduction

One of the major concerns of extended burnup is an increased potential incidence of fuel failures which result in increased coolant activity levels and additional operational problems. Therefore, a prerequisite for extended burnup is to ensure that extended burnup fuel leakage levels are equal to or less than those currently obtained during the first 30,000 MWd/mtU burnup. To achieve this goal, the defect level of the fuel during its first 30,000 MWd/mtU must be reduced to the lowest possible level. The purpose of the current study is to identify life-limiting failure mechanisms so that they can be eliminated in extended-burnup fuel designs.

During shuffling of three-cycle fuel at Oconee 1, four fuel assemblies had indications of a leaking fuel rod.^{4,5} These assemblies were from the same batch of fuel as the assemblies that were reinserted for irradiation to approximately 40,000 MWd/mtU in Oconee 1 cycle 5.

To better understand the failure mechanism occurring incore, a leaking assembly was selected for destructive examination. The examination has two goals: to identify the cause(s) of the fuel rod defects, and to provide a benchmark at about 31,000 MWd/mtU for the high-burnup assembly (about 40,000 MWd/mtU) to be examined. The data obtained from tests performed can be factored into extended-burnup fuel designs or fuel cycle designs to minimize the potential for fuel failures. Since three of the leaker assemblies were symmetric to the high-burnup assemblies through their first three cycles of irradiation⁸, the hot cell data will enable a direct experimental comparison of the effect of going from 31,000 to 40,000 MWd/mtU.

In performing the hot cell examination, it will be necessary to locate the defective rod or rods within the assembly. A detection technique has been developed based on gamma spectrometric measurement of ⁸⁵Kr activities in the

upper plenum of each rod. A failed rod will have a lower activity than sound rods because most of the fission gas will have escaped from it. Due to the relatively flat radial burnup profile of the subject assemblies, the rod-to-rod variation of released fission gases in sound rods is expected to be small. Although, in this instance, the location of failed rod(s) will be determined at LRC, on-site use of the failed rod location system was considered in the overall design of the equipment. Eventually, the technique could be used to identify failed rods in an assembly and avoid premature discharge of entire assemblies by replacing failed rod(s).

During this report period, efforts were directed toward selecting the leaker assembly based on visual examination and a review of manufacturing records, operational history, and sipping data. In addition, the design and development of a failed rod intra-assembly locating device proceeded. The work completed to date in these areas is discussed in succeeding sections.

6.2. Subtask 3A - Irradiation History

6.2.1. General

The three assemblies considered for hot cell examination were 1D20, 1D31, and 1D54 since they were symmetrically located to the high-burnup lead assemblies for three cycles.⁹ Visual examinations (see section 6.3) failed to reveal any differences that would favor the selection of one assembly over the other. Assembly 1D54 was selected for the destructive examination based on operating, manufacturing, and shipping data.

6.2.2. Results and Discussion

Operating Data - Assemblies 1D20 and 1D54 operated through their first three cycles of irradiation in core locations symmetric to four high-burnup lead assemblies irradiated for a fourth cycle in Oconee 1 cycle 5. These two assemblies have an average burnup of about 31,000 MWd/mtU. The other assembly, 1D31, operated symmetric to the center fourth-cycle assembly and has a burnup of about 28,000 MWd/mtU.

The core locations occupied by these three assemblies during their incore residence are shown in Figure 6-1. Assembly 1D31 had a lower power than assemblies 1D20 and 1D54 during their first burn because 1D31 was a peripheral assembly. During subsequent cycles, the power histories of these three assemblies were not substantially different. Since higher power increases the rate of fission gas release as well as fuel failure susceptibility, the 1D20 and 1D54 power histories would be of more interest. In addition, assemblies 1D20 and 1D54 had a more interesting history with respect to the influence of control rod assignments. The assembly average power histories for these three assemblies are shown in Figures 6-2 and 6-3.

Manufacturing Data - Assembly manufacturing data were reviewed for differences among the three assemblies. Although nothing abnormal was noted, there were a few differences. The fuel in 1D20 was somewhat high in phosphorus content (93 ppm versus an average of about 50). The principal fuel pellet lot in 1D54 had a higher iron content than other lots (133 versus 20-40 ppm). Assembly 1D31 showed nothing of interest. Both 1D20 and 1D54 showed low moisture content. The pertinent parameters for assemblies 1D20, 1D54 and 1D31 are shown in Table 6-1.

Sipping Data — The sipping data from the 57 three-cycle assemblies tested^{4,5} are summarized in Figure 6-4. Even though all four assemblies gave a positive leak indication compared to background counts, 1D20 had a ^{137}Cs activity factor of 10 below the others. Thus, there is a chance (although slight) that 1D20 may not be a leaker, which alone would favor 1D54. In addition, it seems reasonable that a rod with a small defect would have a higher gap inventory, thus giving a higher ^{137}Cs count when sipped. If this is the case, 1D54 should have a smaller (and perhaps younger) defect, which would be less hydrided and more amenable to failure mode analysis.

6.2.3. Summary

Based on the considerations above, assembly 1D54 was chosen for the destructive examination. The primary considerations were the operating history and the sipping data.

The operating history of assembly 1D54 was analyzed using a modified version of the TAC02 computer program¹⁴ in which the conservative licensing models were replaced by more realistic models. The power history used for the analysis was based on measured data and is shown in Figure 6-2. The fuel rod parameters used in the analyses were based on the data in Table 6-1. The predicted fuel rod internal pressures and percent fission gas release are shown in Figures 6-5 and 6-6, respectively. These calculated data will be compared to the hot cell measured data. This comparison will enable the benchmarking of the fission gas release model at 31,000 MWd/mtU so that a more accurate assessment of the effects of going from 31,000 to 40,000 MWd/mtU can be made.

Table 6-1. Manufacturing Data

	Assembly identification		
	1D54 ^(a)	1D20 ^(b)	1D31 ^(a)
Pellet density, % TD			
Average	96.0	95.8, 95.9	96.1
Std dev'n	0.34	0.43 (max)	0.20
Pellet OD, in. (avg)	0.3679	0.3675	0.3674
Fuel rod moisture, ppm			
Average	0.73	0.67-0.70	0.67
Range	0.41-1.53	0.29-1.21	0.22-1.02
Impurities	Nothing peculiar, higher than nor- mal Fe	Nothing peculiar, higher than nor- mal P	Nothing peculiar

(a) Indicates that the fuel was primarily from a single manufacturing unit.

(b) Indicates that the fuel was from two separate manufacturing units.

Figure 6-1. Assembly Shuffle History

Figure 6-2. Power History for Assemblies 1D54 and 1D20

Figure 6-3. Power History for Assembly 1D31

Figure 6-4. Sipping Results for Oconee 1 Batch 4
Fuel Assemblies Histogram

Figure 6-5. Fuel Rod Internal Pressure History
for Assembly 1D54

Figure 6-6. Percent Fission Gas Release for Assembly 1D54

6.3. Subtask 3B - Fuel Examination

6.3.1. General

A visual examination of four leaker assemblies and development of a failed rod intra-assembly locating device were completed under this subtask.

6.3.2. Results and Discussion

Visual Examination - The four fuel assemblies identified as leakers in the sipping campaign were visually examined to determine whether any of the assemblies had a visually observable feature which might suggest that it would be preferable for hot cell examination. The examination was conducted using an underwater TV camera. Although the image quality was such that a small defect might have gone unnoticed, it was good enough to detect a "progressed" defect. No defects were observed in any of the four assemblies examined.

Failed Rod Location System - The location and segregation of fuel assemblies that contain a leaking rod has become a routine operation at many reactor sites by using "sipping" techniques. These techniques are based on collection of a gaseous or liquid sample from the assembly in a depressurized environment and gamma spectrometric analysis of the sample to determine its activity. However, the nondestructive identification of the specific defective fuel rod(s) within the assembly has not been routinely accomplished. In practice, unless the defective rod can be visually observed, its identification can be very difficult since destructive examination of all the fuel rods (about 200) from one assembly is prohibitively expensive. Thus, this subtask was structured to develop a technique and the equipment for locating defective fuel rods in a leaking PWR assembly.

The system selected for locating the leaking rod is based on gamma spectrometric detection of fission gases (in particular, ^{85}Kr) in the upper plenum of the fuel rod. Krypton-85 emits characteristic detectable gamma rays at rates proportional to its concentration in the fission gases. Therefore, the ^{85}Kr activity measured from the plenum of all the intact rods should be very similar. The only exception would be in assemblies where relatively large intra-assembly burnup gradients exist. The burnup gradients in the assemblies being considered here are relatively flat.

A leaking fuel rod will discharge its plenum gas inventory during the cooldown process and equilibrate to system pressure. The ^{85}Kr activity of the leaking rods will be reduced relative to an intact rod. The resulting activity of the leaker, when compared to that of intact rods will thus indicate loss of gas and make the rod a candidate for removal.

Laboratory experiments confirmed the technical feasibility of this approach. Earlier studies involved the detection of ^{133}Xe , a less difficult isotope to detect but of limited application due to its short half-life (5.27 days). The use of ^{85}Kr as the source activity ensures that decay will have a minor influence due to its long half-life (10.72 years). However, the use of ^{85}Kr does have a drawback due to the proximity of its 514 KeV peak to the ^{106}Ru fission product peak at 512 KeV and the 511 KeV position annihilation peak. Until very recently, the discrimination between peaks so close together, particularly when their relative peak height may be quite different, was only possible with the best of laboratory gamma spectrometry system.

A feasibility test was conducted in the LRC laboratory to verify the capability of state-of-the-art commercial equipment to distinguish between the 511, 512, and 514 KeV gammas. The spectral information obtained is shown in Figure 6-7. This figure depicts the relative heights of the undesired 511 and 512 KeV peaks as seen by a detector looking at the side of the plenum of a single rod after its fission gases had been released by puncturing the cladding (no 514 KeV peak present). It also shows these peaks plus the desired 514 KeV ^{85}Kr peak in a spectrum from an unpunctured rod from the same fuel assembly. The difference in the two types of spectra is obvious and is the basis for operation of this system.

These spectra were taken on single rods that were not near the remainder of the fuel assembly. Therefore, they represent a somewhat idealized picture of what the system can be expected to provide. Spectra taken with the collimator colinear with the axis of the rod were unusable because of spectral interference from the fuel column. Thus, the decision to partially withdraw rods from the assembly was made.

The ultimate application of the failed rod location system described above would be in support of a fuel repair operation at an operating reactor. It could be used in conjunction with a fuel assembly design easily accommodating

rod removal and replacement and should involve a minimum of fuel handling operations and interference with other reactor support functions. The performance of the system in this examination will provide insight into modifications that may be required for eventual use at a reactor site.

System Design — The failed rod location system is designed to operate in conjunction with existing equipment in the hot cell transfer canal at LRC. A cask transporter (shown in Figure 6-8) provides the underwater support frame for both the leak locator and the fuel assembly.

In its operating configuration, the system sits on top of the transporter frame and mates with the fuel assembly positioned in the side-loaded fuel assembly support frame. The upper end fitting is removed by previously developed techniques, and the remainder of the assembly is clamped in positive alignment with a base plate on which the remainder of the leak locator is mounted.

The failed rod location system comprises three subsystems mounted on a common base plate:

1. Radiation detector/shield/collimator.
2. X-Y positioner.
3. Rod puller/reinsertion device.

A separate instrumentation console allows the operator to control the system. An umbilical cord containing the electrical and hydraulic lines and liquid nitrogen fill and vent lines connects the console to the underwater components. Figure 6-9 is a conceptual sketch of the underwater system.

The detector chosen for this application is an intrinsic germanium unit manufactured by EG&G-ORTEC (designated as Model 1512-08180-S). It is a coaxial detector with the following characteristics:

Relative photopeak efficiency, %	>8
Resolution, KeV FWHM @ 1332 MeV.	<1.8
Peak to Compton Ratio	>34:1

The detector and its associated preamplifier are coupled to a 1.7-liter dewar flask for cooling. The detector housing is enclosed in a cylindrical lead shield attached to a radiation collimator. The collimator axis is horizontal, and its shape maximizes its selectivity for radiation emanating from the gas

plenum volume of a fuel rod. The collimator, shield, detector, and dewar flask are enclosed in a watertight housing, which is connected to a manifold on the operator's console through a pressurized umbilical tube. The housing is maintained at a positive pressure with respect to the hydrostatic head at the operating depth to ensure against leakage and damage to electrical components.

The fuel rod pulling/reinsertion mechanism is located in fixed alignment with the radiation collimator axis; it provides the means for withdrawing (and reinserting) the rod to be tested to a position where the signal from the rod suffers minimal interference from the remainder of the fuel assembly. The mechanism employs a hydraulically actuated pair of gripper jaws that are machined to fit the upper end caps of the fuel rods. The operator can hydraulically raise and lower the gripper or rotate it about its vertical axis. The stroke of the vertical actuation is variable.

A hydraulically actuated X-Y translation table attached to the base plate supports and properly positions both the detector and the rod puller with respect to a corner rod of the fuel assembly. Indexing mechanisms on each axis of motion allow the device to be moved to the next rod position automatically.

When this movement is complete, the rod pulling mechanism is actuated, the rod is withdrawn to the raised position, and radiation counting is performed. At the end of the counting period, the rod is reinserted, either the X or Y translation stage is actuated (as appropriate), and the rod puller is automatically positioned over the next rod to be tested. The process is repeated until all the rods have been tested.

The instrumentation console houses hydraulic controls and their associated safety features, electronic instrumentation for radiation counting, data storage and display, cryogenic system fittings, and system status instruments. The hydraulic system has interlocks to prevent operations that could damage the fuel assembly under test. For example, the XY translation capability is disabled when the rod puller jaws are closed and in anything other than the full "UP" position, eliminating the possibility of bending a rod. Other unsafe conditions are similarly avoided through the use of interlocks.

An important aspect of the concept adopted is the accessibility of the failed rod location system in the event of a system malfunction. As is normal with prototype systems, the reliability of components and the proper alignment and

adjustment of mechanisms is not certain. Operations can be interrupted for adjustment or repairs at virtually any time. If required, the entire system may be lifted off its mounting on the cask transporter, brought to the water surface, adjusted, and returned to its operating position without disturbing the position of the fuel assembly under test.

6.3.3. Conclusions

A visual examination of four leaker assemblies failed to reveal any differences that would favor the selection of one assembly over the other as the leaker. The design of a failed rod location system for identifying defective fuel rods within a leaker fuel assembly was completed. The design is based on gamma spectrometric analysis for the fission gas ^{85}Kr in the upper plenum of the fuel rod. This design approach was verified by a feasibility test on punctured and unpunctured fuel rods.

Figure 6-7. Gamma Spectra From Irradiated Fuel Rods

Figure 6-8. Cask Transporter

Figure 6-9. Failed Rod Location System

7. REFERENCES

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