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QUALIFICATION OF THE B&W MARK B

FUEL ASSEMBLY FOR HIGH BURNUP

**Second Semi-Annual Progress Report
for the Period
January-June 1979**

**Thomas A. Coleman
Project Engineer**

MASTER

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by

**BABCOCK & WILCOX
Power Generation Group
Nuclear Power Generation Division
P. O. Box 1260
Lynchburg, Virginia 24505**

**BABCOCK & WILCOX
Power Generation Group
Research and Development Division
Lynchburg Research Center
Lynchburg, Virginia**

Submitted by

**BABCOCK & WILCOX
Contract Research Division
P. O. Box 835
Alliance, Ohio**

**DUKE POWER COMPANY
P. O. Box 33189
Charlotte, North Carolina 28482**

Babcock & Wilcox

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Principal Authors

T. A. Coleman
H. H. Davis
M. R. Eastburn
B. A. Guthrie
S. S. Kere
J. G. Larson
J. T. Mayer
L. W. Newman
T. D. Pyecha
N. Smith

Babcock & Wilcox
Power Generation Group
Nuclear Power Generation Division
Lynchburg, Virginia

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ABSTRACT

Five Babcock & Wilcox-designed Mark B (15 × 15) pressurized water reactor (PWR) fuel assemblies are being irradiated to extended burnups in Duke Power Company's Oconee Unit 1 reactor. This effort is part of a Department of Energy-sponsored program to improve uranium utilization by extending the burnup of light water reactor fuel. These fuel assemblies were nondestructively examined after three cycles of irradiation in the Oconee Unit 1 reactor prior to their reinsertion for a fourth cycle of operation (as Oconee 1 cycle 5). Data obtained included fuel assembly and fuel rod dimensions, water channel spacings, holddown spring preload forces, and fuel column axial gap and stack lengths. The data were within design limits and generally followed projected behavior.

This report, covering the January-June 1979 period, is the second semi-annual progress report for the program. Efforts during this period included reduction and analysis of the post-irradiation examination data, performance tracking of the extended burnup assemblies during their fourth cycle, and analytical work to identify and quantify those factors limiting fuel assembly lifetime. The Oconee Unit 1 reactor operated at a 94% capacity factor during this report period. At the end of the period, four of the extended burnup assemblies had average burnups of 37,700 MWd/mtU, and the other one had an average burnup of 33,890 MWd/mtU. Irradiation of the fuel assemblies will continue during the next report period.

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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 15-20% could 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 a reduced number of spent fuel assemblies to be stored result from the increase in discharge batch average burnup.

The Department of Energy (DOE) has initiated research, development, and demonstration efforts involving Duke Power Company (Duke), Arkansas Power and Light Company (AP&L), and Babcock & Wilcox (B&W), with a goal of demonstrating improved uranium utilization, primarily through the successful operation of PWR

fuel assemblies to extended burnups. The uranium utilization improvement effort was divided into two separate but interrelated 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 increased safely 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 (~5%) in uranium utilization to be realized, beginning within 2-3 years.

In the second program, AP&L and B&W are undertaking the development and demonstration of an improved fuel assembly which is capable of batch average burnups of 45,000-50,000 MWd/mtU. 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 second semi-annual progress report for the DOE/Duke/B&W program (contract No. ET-78-C-02-4711). It covers progress from January 1 through June 30, 1979. This report covers work under both of the technical tasks that make up the program: Task 1 - High Burnup Characterization of the Mark B Fuel Assembly, and Task 2 - Operational Limits of a Mark B Core.

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. Efforts during this report period included detailed analyses of post-irradiation examination data obtained at burnups of about 30,600 MWd/mtU, performance tracking of high burnup lead assembly irradiation, and development of extended burnup fuel cycles.

Detailed analyses were conducted on post-irradiation examination (PIE) data from high-burnup lead assemblies. The data were obtained at an average burnup of 30,566 MWd/mtU prior to insertion of the assemblies in Oconee 1 cycle 5 for their fourth cycle of operation. The results, reported herein, indicate that the assemblies performed well through three cycles of operation. All the data were within design limits and agreed well with predicted performance values.

Visual examinations of peripheral fuel rods revealed crud patterns typical of PWR fuel.¹ Fuel assembly and fuel rod growth averaged 0.36 and 0.51% $\Delta L/L$, respectively, consistent with previous PIE results. Ripples in the fuel rod diameter profilometry traces, which correlated with fuel pellet lengths, indicated that pellet-cladding contact had occurred in several corner rods in three of the five assemblies examined. Fuel rod creepdown and fuel rod ovality were low, averaging 0.0038 and 0.0023 inch, respectively. Water channel closure was low in magnitude and compared favorably with previous results from other three-cycle assemblies. Holddown spring preload force data indicated that the holddown spring net operating force remains relatively constant as a function of incore residence time. Only eight axial gaps (none more than 0.1 inch) were found in the 20 fuel rods gamma scanned. Average fuel column stack length exhibited no net change after three cycles of irradiation, which was in good agreement with predicted result.

Irradiation of the high-burnup lead assemblies in Oconee 1 cycle 5 continued. A 94% capacity factor was maintained during the 6-month report period, resulting in a cycle burnup of 226 effective full power days (EFPD), and assembly average burnup of 37,700 MWd/mtU for four assemblies and 33,890 MWd/mtU for the other lead assembly. The end of cycle 5 is projected to occur in the fourth quarter of 1979. Nondestructive and destructive examinations of the high-burnup lead assemblies are planned after their discharge.

Extended burnup fuel cycle designs satisfying standard nuclear design criteria for uranium reloads were completed. These fuel cycles form the base on which multi-disciplinary evaluations normally considered in reload licensing applications will be performed. The multi-disciplinary evaluations will be initiated during the next reporting period.

3. PROGRAM DESCRIPTION

As currently structured, the program scope comprises irradiation and examination of five PWR fuel assemblies at burnups in excess of 40,000 MWd/mtU and analytical work to identify and quantify those factors limiting fuel assembly lifetime. The program is divided into two tasks: Task 1 — High Burnup Characterization of the Mark B Fuel Assembly, and Task 2 — Operational Limits of a Mark B Core.

The specific objectives of each task with breakdown by subtask and a description of the technical workscope are presented in the following paragraphs.

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

3.1.1. Objectives

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.
2. To characterize the high-burnup lead assemblies prior to their fourth cycle of operation.
3. To secure an operating license for Oconee 1 cycle 5.
4. To characterize the high-burnup lead assemblies after their fourth cycle of irradiation.

3.1.2. Technical Workscope

1. Subtask 1A — Fuel Cycle Design — Design Oconee 1 cycle 5 to meet standard fuel cycle design criteria. Use the PDQ07² 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.

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."
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 hold-down spring load-deflection characteristics in the examination.
4. Subtask 1D - Irradiation - Follow the operation of Oconee 1 cycle 5.
5. Subtask 1E - Nondestructive Testing (Poolside) - Perform nondestructive examination of the five four-cycle fuel assemblies at the end of cycle 5 of Oconee 1. Include spacer grid spring load-deflection measurements and crud sampling in addition to the measurements carried out in subtask 1C.
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

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

3.2.2. Technical Workscope

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.

2. Subtask 2B — Nuclear, Mechanical, and Thermal-Hydraulic Analyses — Generate key physics parameters, such as moderator temperature and Doppler 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.
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 loss-of-coolant-accident (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.
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.
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.

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

T. D. Pyecha, B. A. Guthrie, J. T. Mayer,
H. H. Davis, J. G. Larson

4.1. Introduction

The detailed quantitative analyses of the NDT PIE data obtained during the EOC 4 refueling outage were completed during the current report period. The design and licensing activities for the high-burnup lead assemblies were described in a previous report³ under Subtasks 1A and 1B. A description of the PIE operations, on-site NDT methods, and a preliminary quantitative assessment of the PIE results for the high-burnup lead assemblies were also reported previously under Subtask 1C.³

The major efforts conducted under Task 1 during this report period consisted of (1) reduction and analysis of nondestructive (NDT) PIE data obtained on the high-burnup lead assemblies during the Oconee 1 end-of-cycle 4 (EOC 4) refueling outage, and (2) performance tracking of these lead assemblies during their fourth cycle of operation. These results are reported herein under Subtask 1C, "Poolside Characterization," and Subtask 1D, "Irradiation."

4.2. Subtask 1C — Poolside Characterization (Nondestructive Examination)

4.2.1. Fuel Assembly Nomenclature Description

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 (cold-worked, stress-relieved, Zircaloy-4). 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 4-1.

A cutaway version of a typical Mark B (15 × 15) fuel assembly is shown in Figure 4-2. The structural cage of the assembly consists of 16 control rod guide tubes (cold-worked, stress relieved, Zircaloy-4) permanently attached to the upper and lower end fittings. The spacer grids (Inconel 718) form the structural link between the guide tubes and the fuel rods (cold-worked, stress relieved, Zircaloy-4 cladding) but are free to move with the fuel rods as the rods grow. Fuel assembly lift due to coolant flow is prevented by the helical holddown spring contained in the upper end fitting of the assembly.

Six intermediate spacer grids and two end grids are utilized on each assembly. The midspan level and spacer grid locations are shown schematically in Figure 4-3. For reference, the six intermediate spacer grids are numbered sequentially from 1 to 6 from top to bottom, and the water channel midspan levels are evenly numbered from 2 to 14.

4.2.2. Assembly Power Histories

The five assemblies selected for extended burnup were assemblies 1D13, 1D26, 1D42, 1D45, and 1D55. They were initially loaded into the Oconee 1 reactor after its first cycle of operation as members of batch 4B; they were re-designated as batch 4D for their fourth cycle of irradiation in Oconee 1 cycle 5. The fuel contained in batch 4D had an initial average enrichment of 3.20 wt % ^{235}U . The pre-irradiation fuel column length (the mean of the as-built fuel rod lot averages) of these five assemblies was 141.20 inches.⁴

The core locations of the extended burnup fuel assemblies during their previous incore residence and for cycle 5 of Oconee 1 are given in Figure 4-4. Four of the batch 4D assemblies (1D13, 1D26, 1D45, and 1D55) were from the same group of symmetric core locations (e.g., L-14 during cycle 2 of Oconee 1).

During cycle 2, these four assemblies were in a core position that contained control rod assemblies assigned to group 6, which is used to control xenon and power transients. The group 6 rods were about 85 to 95% withdrawn for approximately the first 237 EFPD of cycle 2. From 237 EFPD to the end of cycle 2 at 292 EFPD, the group 6 rods were fully withdrawn during full-power operation. Table 4-1 is a tabulation of both the power histories and assembly burnups for these assemblies during cycle 2. The effect of the group 6 control rod assembly insertion is evident in Figure 4-5, which presents representative

axial power distributions for assembly 1D13 and its symmetric assemblies. During cycles 3 and 4 of Oconee 1, these assemblies contained control rod assemblies assigned to the safety bank. Safety control rod assemblies are fully withdrawn during power operation. Therefore, the power histories and burnups of these assemblies were not directly influenced by the presence of full-length control rod assemblies. However, assembly 1D13 and its symmetric counterparts were adjacent to the axial power shaping rods (APSRs), which are control rod assemblies that have only a three foot section of absorber in the lower fourth of the rod. The APSRs are used to control the reactor axial power shape, and they remain in the active core region during power operation. The APSRs, which were typically 5 to 25% withdrawn, tended to flatten the axial power shape of these assemblies during cycles 3 and 4 of Oconee 1. Tables 4-2 and 4-3 give both the power histories and assembly burnups for 1D13, 1D26, 1D45, and 1D55 during cycles 3 and 4, respectively. Representative axial power shapes are displayed in Figures 4-6 and 4-7. For cycle 4 of Oconee 1, all full-length control rod assemblies were essentially withdrawn from the core; thus, the axial power shape of the batch 4D assemblies did not change significantly over cycle 4. The calculated axial burnup profile at the end of each cycle for assembly 1D13 and its symmetric is presented in Figure 4-8. The assembly average burnups at the end of cycles 2, 3, and 4 of Oconee 1 for these assemblies were 11,710, 23,770, and 31,160 MWd/mtU, respectively.

During cycles 2, 3, and 4 of Oconee 1, the fifth extended burnup assembly (1D42) was located in core positions that did not contain control rod assemblies. However, during cycle 3 this assembly was adjacent to the APSRs and to a group 7 control rod assembly. Group 7 control rod assemblies are used to control xenon and power transients. They were only 15 to 25% withdrawn over the first 240 EFPD of cycle 3. After 240 EFPD, group 7 was about 85% withdrawn. These control rod assemblies strongly influenced the axial power shape for assembly 1D42 during cycle 3. The power histories and assembly burnups for this assembly during cycles 2, 3, and 4 of Oconee 1 are given in Tables 4-1, 4-2, and 4-3, respectively. Representative axial power shapes are given in Figures 4-9, 4-10, and 4-11 for each cycle. The calculated axial burnup profile at the end of each cycle for assembly 1D42 is presented in Figure 4-12. The assembly average burnups at the end of cycles 2, 3, and 4 for this assembly were 9,350, 21,040, and 28,210 MWd/mtU, respectively.

All of the axial power distributions were calculated using the three-dimensional nodal code FLAME⁵ with one node per assembly in the radial direction and 32 nodes per assembly in the axial direction. The actual reactor operations during each cycle were modeled as closely as practicable.

4.2.3. Measurement Techniques

The five re-inserted assemblies were visually examined and characterized with respect to the following: (1) fuel assembly and fuel rod growth (fuel assembly length measurements), (2) cladding creepdown and ovality (fuel rod diameter profilometry), (3) fuel rod bow (water channel spacing measurements), (4) hold-down spring force (spring load-deflection), and (5) fuel column axial gap and stack lengths (gamma scans), and (6) fuel integrity (sipping). The estimated precision of the various measurements is shown in Table 4-4.

The measurement techniques used to collect data on fuel assembly appearance, mechanical properties, and physical dimensions have been described in previous reports.^{1,3}

4.2.4. Results and Discussion

4.2.4.1. Fuel Assembly Leak Tests

A fuel assembly leak-check (sipping) test was conducted on 57 three-cycle (batch 4) assemblies to obtain a large data base and to establish a confident baseline activity for sound assemblies. Sixteen of these assemblies, representing two core locations, had been designated as candidates for a fourth cycle of irradiation.

Results — The results of the sipping tests of the 57 assemblies are summarized in Figure 4-13. Of the 16 candidate assemblies, three were determined to be leakers based on the level of cesium-137 activity in their water samples: 1D20, 1D31, and 1D54. The latter was one of five assemblies originally selected for a fourth cycle, so it was replaced in the program by 1D55. Among the remaining 41 assemblies, one additional leaker (1D49) was found.

Discussion — A visual examination of the leaking assemblies 1D20 and 1D54 was conducted. No indications of a defect on the visible portions of peripheral rods were evident.

The present examination program has been expanded to include a comprehensive examination of the leaking assemblies to determine the cause of rod failure and whether a life-limiting phenomenon is present. This expanded program includes both a more thorough visual inspection of the four leakers and a detailed destructive examination of one assembly. This examination will also enlarge the fuel performance data base at burnups of approximately 30,000 MWd/mtU.

4.2.4.2. Visual Examinations

Ten fuel assemblies (five primary and five alternate candidates) were visually examined on all four sides using underwater television. All such examinations were videotaped. In addition, an optical periscope examination was conducted on assemblies 1D26 and 1D55, along with portions of 1D13, 1D21, and 1D42. Selected areas were photographed during the periscope examination.

Results — All of the assemblies were generally similar in appearance, exhibiting the typical crud patterns seen in previous B&W fuel examinations. A common surface feature of the crud was a series of narrow (~0.1 inch wide) circumferential bands (see Figure 4-14). These bands were seen on all assemblies and were generally confined to the higher-burnup region of the assemblies between grids 3 and 6, although some were observed as far down as the center of the bottom grid-span. The bands were evenly spaced axially at approximately pellet-length intervals, and were apparently located at fuel pellet interfaces. Another crud pattern commonly seen was a wide horizontal band (or bands) near the top of the fuel column and extending across the assembly face. These bands were at least 2 inches wide. No assessment of the relative crud thickness across the various crud patterns could be made.

Minor physical damage was observed on fuel assemblies 1D21 and 1D42, both of which had passed the sipping test. Assembly 1D21 had a minor tear on the corner of grid 1 and an indentation on peripheral rod A13 just below grid 3 (displayed in Figure 4-15). Since the depth of the indentation could not be determined accurately at poolside, the assembly was not reinserted for a fourth cycle; assembly 1D26 was selected as a substitute. Assembly 1D42 had a small dent in the edge of the upper grid skirt near corner rod D15 (Figure 4-16). Close examination revealed that the edge of the skirt was not touching the cladding surface. Thus, assembly 1D42 was reinserted for a fourth cycle.

Discussion — The surface crud pattern seen on the fuel rods are not uncommon. They are considered to be variations in crud color or composition possibly resulting from variations in coolant flow and heat flux. Flow distribution around spacer grids causes local crud pattern variations. A possible relationship of crud formation to variations in heat flux is illustrated by the wide patterns seen above the fuel column, where the heat flux varies considerably even though the cladding temperature and coolant flow are fairly constant. Since the interface region between two fuel pellets can have slight heat flux variations due to the small fuel void created by the pellet end-dish and chamfer, narrow circumferential crud bands may result at the pellet interfaces.

4.2.4.3. Fuel Assembly and Rod Length Measurements

Fuel assembly length and intermediate spacer grid locations were measured on all four faces of each of the five extended-burnup assemblies. Similar measurements were also taken on five alternate assemblies and the results included in the data base. All measurements were corrected for the effects of tape stretch and thermal expansion. Since the measuring tape was read to an accuracy of ± 0.024 inch at the reference edges, the maximum expected error in the corrected measurement was less than ± 0.048 inch.

Two types of growth data were obtained from these measurements: (1) fuel assembly (guide tube) growth, and (2) assembly average fuel rod growth (total axial strain). Assembly growth was obtained directly from the locations of reference positions at each end of the fuel assembly. Rod growth, however, was obtained indirectly from measurements of fuel assembly spacer grid locations as described below.

Differential growth between the fuel assembly (guide tubes) and its fuel rods is accommodated by the spacer grids moving with the fuel rods and sliding along the guide tubes. Since the fuel rods rest on the grillage of the lower end fitting, thermal expansion and irradiation-induced differential growth of the fuel rods is accumulated upward from the rod bottoms into the designed gap between the top of the fuel rods and the grillage of the upper end fitting. Thus, each successive spacer grid from the bottom of the fuel assembly is displaced upward from its original position (relative to the bottom of the assembly) by an increasing amount as rod growth accumulates, providing a measure of assembly average fuel rod growth.

Fuel Assembly Growth Results — The growth of the five extended-burnup assemblies is summarized in Table 4-5. The average observed length change was 0.36% $\Delta L/L$ at an assembly average fast neutron fluence ($E_n > 1$ MeV) of 5.11×10^{21} nvt. These results and those obtained from previous measurements on other Mark B fuel assemblies are displayed in Figure 4-17. The growth curve shown on this figure is the result of a linear regression analysis incorporating the growth results obtained to date.

Fuel Rod Growth Results — Assembly average fuel rod growth for the five extended-burnup assemblies are summarized in Table 4-6. The average of the tabulated fuel rod growth was 0.51% $\Delta L/L$ at an assembly average fast neutron fluence ($E_n > 1$ MeV) of 5.11×10^{21} nvt. These results and the fuel rod growth results obtained to date from poolside (nondestructive) and hot cell (destructive) examinations are displayed in Figure 4-18.

Linear regression analyses were performed for two cases: (1) using all of the results shown, and (2) using only the results from the hot cell examinations in which fuel rod lengths were measured directly and more accurately. The regression analysis for the second case yielded the curve displayed in Figure 4-18. It is judged to describe fuel rod growth more accurately. The regression analysis for the first case yielded a curve of approximately the same slope but a slightly greater growth for a given fluence.

Discussion — Comparison of the fuel assembly and fuel rod growth curves (Figures 4-17 and 4-18) indicates that adequate allowance exists in the fuel assembly design to accommodate differential growth between the fuel assembly and its fuel rods. Thus, fuel rod growth is not a limiting condition for extending fuel assembly operation. However, fuel assembly growth results indicate that assembly growth, which closes the gap between the assembly upper end fitting and the reactor internals, will likely impose an exposure limit on fuel assembly operation. The examinations to be conducted after the fourth cycle of irradiation will quantify this limitation on extending fuel assembly lifetime.

4.2.4.4. Fuel Rod Diameter Profilometry

The diameters of a number of peripheral fuel rods were determined between the intermediate spacer grids of the five extended burnup assemblies using single-orientation, line-scan profilometry.

Orthogonal line-scans were obtained only on the four corner rods of each assembly. All of the profilometry data were digitized at 0.05-inch chart intervals (corresponding to 0.20-inch actual fuel rod distances) using a minicomputer with an interactive plotter. From the digitized data, fuel rod mean diameter, creepdown, and "average maximum ovality" were determined as a function of axial position along each rod. (These terms are defined in the appendix of this report.) Reduced profilometry traces were redrawn by the minicomputer, overlaying the orthogonal scans on the same axes for comparison. As an example, reduced traces of the corner fuel rods of assembly 1D26 are displayed in Figure 4-19.

Note that the "average maximum ovality" obtained in the examinations is defined as the 4σ range (from -2σ to $+2\sigma$) of the mean diameter (i.e., a 95% confidence interval about the mean diameter), rather than the conventionally defined peak-to-peak ovality ($OD_{\max} - OD_{\min}$) at a given transverse plane. A comparison of single-orientation line-scan profilometry and rotating step-scan profilometry measurements obtained during previous post-irradiation hot cell (destructive) examinations indicates that the statistically determined average maximum ovality is a reasonable approximation of the conventionally defined peak-to-peak ovality, the estimated relative error being less than 25%.

Cladding Creepdown Results — Table 4-7 is a summary of average mean diameter and average diametric creepdown values for the peripheral rods of each fuel assembly. The quantities, listed by grid-span location, are the averages of the mean values of the rods measured in their respective assemblies. The creepdown values are based on an as-built cladding lot average diameter of 0.4299 inch. The average measured creepdown of the fuel rods in assembly 1D42 was 3.2 mils (assembly average burnup of 28,210 MWd/mtU); the average of the other batch 4D assemblies was 3.8 mils (31,160 MWd/mtU). The maximum average creepdown observed between spacer grids was 4.5 mils.

A distinct axial variation in creepdown was noted on the fuel rods and is displayed in Figure 4-20. The minimum observed creepdown occurred at the top of the fuel column, while the maximum occurred below the midpoint of the assembly between spacer grids 4 and 6. This trend in the creepdown is similar to that observed on other fuel assemblies during previous post-irradiation examinations.

The original strip chart traces were also examined closely for indications of contact between fuel pellets and cladding. This pellet-cladding contact was inferred from small ripples in the profilometry traces shown in Figure 4-21. The ripples correlated with fuel pellet length, particularly in regions where orthogonal traces exhibited little or no difference in rod diameter (low ovality). Many indications were observed on the four corner rods of assembly 1D55, extending over the region from grid 2 to grid 6. Similar indications were noted on three of the four corner rods of assembly 1D26. Assembly 1D45 exhibited fewer indications, and these were concentrated near the peak creepdown region between grids 5 and 6. No indications were evident on the fuel rods of assembly 1D13.

Cladding Ovality Results — The average maximum ovalities measured for the peripheral rods of each assembly are presented in Table 4-8. The values, listed by grid-span location, are the averages of all rods measured in their respective assemblies. Average maximum ovalities ranged from 0.0015 to 0.0029 inch with an overall mean of 0.0023 inch. The largest peak ovality determined from the orthogonal line-scans of the corner rods was approximately 0.005 inch. The nominal ovality of the batch 4 as-built cladding lots was 0.0002 ± 0.0002 (2σ) inch. The average maximum ovality of the rods of each fuel assembly as a function of grid-span position is displayed in Figure 4-22. No distinct axial variation in average maximum ovality is evident.

Discussion — The axial distribution of fuel rod creepdown (shown in Figure 4-20) peaked below the midpoint of the fuel column in the region of maximum burnup, rather than near the top of the fuel column in the region of higher temperature. This implies the greater dependence of creepdown on burnup than on temperature. It should be noted, however, that the interaction of ovalization and creepdown could cause the maximum creepdown to occur at a location removed from the area of maximum creep rate. This effect could result from two-point contact of the cladding with the fuel stack during more rapid ovalization, thereby changing the local stress state in the cladding and retarding further ovalization and creepdown. Since ovalization is expected to occur more rapidly at higher cladding temperatures, this effect would be more pronounced in the upper region of the fuel rod. Thus, the observed creepdown would be less in this region than in the lower portions of the rod. Indications of this effect were seen in previous hot-cell examinations. This area will be examined in greater detail in the forthcoming destructive examination.

4.2.4.5. Water Channel Spacings

The water channel spacings between fuel rods were measured on each of the five extended-burnup assemblies. This included full characterization of four of the assemblies (1D13, 1D42, 1D45, and 1D55). Measurements on the fifth assembly (1D26) were taken only at midspan level 12 (between spacer grids 5 and 6, as shown in Figure 4-3) where maximum rod bow has historically been observed. All water channel spacing measurements were taken at the axial midspans between spacer grids except those taken at level 2. Because of equipment limitations, measurements for this level were taken approximately 3 inches below the midspan, which lies between the upper end grid and the first intermediate spacer grid.

The reported data were corrected for the effect of the measurement probe, which spreads the fuel rods slightly as it measures the spacings. Fuel rod bow, expressed as the standard deviation (σ) of the corrected water channel spacing measurements, was converted to percent of water channel (gap) closure. A statistical method was used wherein it was assumed that the magnitude of fuel rod bow is described by a normal distribution having a mean of zero and a nominal value of 0.138 inch for pre-irradiation water channel spacing.

Results — Water channel spacing measurements are summarized in Table 4-9.

Statistically, 95% of the water channels have a gap closure (%) less than the values tabulated. Of the fuel assemblies examined, assembly 1D42 exhibits the greater average gap closure of 26.2%. The average of the four fully characterized assemblies is 23.6%.

The worst-span water channel closure observed on all of the assemblies examined to date in B&W post-irradiation examinations is shown in Figure 4-23 as a function of fuel assembly average burnup. While the closure increased approximately linearly with burnup through three cycles of irradiation, an increase in data scatter beyond 23,000 MWd/mtU is apparent.

The axial distribution of fuel rod bow, expressed as % of water channel closure, is displayed in Figure 4-24 for the four fully characterized assemblies. This axial distribution is reasonably consistent with that of other fuel assemblies examined after three cycles of irradiation and generally follows axial fluence profiles with closure increasing in the lower portion of the assemblies.

Discussion — The axial distribution of water channel closure has historically been closely related to the respective assembly fluence profiles, with maximum closure occurring between spacer grids 5 and 6 (midspan level 12). The closure behavior exhibited by the extended burnup assemblies (shown in Figure 4-24) differs slightly from that observed in the past. Although the maximum closure is still skewed toward the bottom of the assemblies, it has shifted slightly toward the middle for assemblies 1D42 and 1D45. For assemblies 1D13 and 1D55, a slight tendency toward a bi-modal axial distribution has started to appear, with maximum closure at midspan levels 6 and 12 and decreasing slightly at levels 8 and 10. This change in axial distribution and the increasing scatter in the worst-span closure data may indicate that a secondary mechanism is beginning to predominate late in life. Examination of the extended burnup assemblies after four cycles of irradiation will provide additional information to assist in determining the manner in which water channel closure occurs and its driving mechanism.

4.2.4.6. Holddown Spring Force

Holddown spring load-deflection curves were obtained for each of the five extended-burnup assemblies to determine holddown spring relaxation. Measured holddown spring preload force and spring rate were used with the measured length of each fuel assembly to calculate the net fuel assembly holddown force at normal operating temperature (operating force).

Results — Spring force measurements and calculated net operating forces are summarized in Table 4-10. Spring preload force decreased an average of 35.5% from the mean as-built preload value of 575 lb. However, the net operating force remained essentially unchanged from its beginning-of-life (BOL) value of 595 lb, having decreased less than 2% after three cycles of irradiation. Net fuel assembly holddown operating force as a function of fuel assembly burnup is displayed in Figure 4-25.

Discussion — Analytical results indicate that the decrease in spring preload force is neither creep-dependent nor is it dependent on fluence. Rather, it is apparently caused by plastic set within the spring, which occurs at cold shutdown. The decrease in spring preload force, however, is countered by fuel assembly growth, which further compresses the spring. Thus, during operation, the net holddown force remains relatively constant with increasing fluence throughout the operating lifetime of the fuel assembly, as shown in Figure 4-25.

4.2.4.7. Gamma Scans

Fuel column axial gap and fuel stack length measurements were obtained on the four corner rods of each of the batch 4D assemblies. The system output of gamma count rate ($E_\gamma > 0.2$ MeV) as a function of axial location (hoist read-out) was plotted on a strip chart recorder. Recorder traces from each scan were used to determine the axial length and location of fuel column gaps as well as the overall length of the fuel column. Gap size measurement precision at the slower scanning speed of 6 inches/minute was approximately 0.05 inch and stack length precision 0.1 to 0.2 inch.

A portion of the gamma scan trace of the CD-corner fuel rod of assembly 1D13 is shown in Figure 4-26 as an example of the sensitivity and resolution of the system. The low amplitude negative peaks occurring at consistent intervals along the trace indicate the normal interfaces between individual pellets. These interface peaks result from the dished voids at the ends of each pellet. The dish, which is only 0.018 inch deep, results in a volume-equivalent cylindrical gap of about 0.012 inch. The negative peak of greater amplitude indicates an axial gap of 0.1 inch between pellets, which is located approximately 1 inch below the top of the fuel column.

Results — Gap size and gap location data for the 20 rods scanned in the five extended burnup assemblies are presented in Table 4-11. Only eight gaps were seen, all less than about 0.1 inch long. The locations of the gaps along the length of the fuel columns were variable, displaying no distinct axial pattern. Values for the fuel column stack lengths (with total gap lengths subtracted) are given in Table 4-12. The overall average stack length of 141.2 inches indicates no net change after three cycles of irradiation.

Discussion — The small number and sizes of the axial gaps observed and their near-random distribution indicate that the fuel performed well through three cycles of irradiation. The zero net change in fuel column average stack length from the fuel rod lot average pre-irradiation value of 141.2 inches is in good agreement with predicted results.

The locations and sizes of these axial gaps will be compared with corresponding values to be obtained at the end of the fourth cycle. The individual stack length values will also be compared to obtain a fourth cycle fuel swelling component.

4.3. Subtask 1D - Irradiation

At the end of the previous reporting period³, Oconee 1 had operated for 56 EFPD in cycle 5, and the burnups on the high-burnup lead assemblies were 29,300 MWd/mtU for the center assembly and 32,800 MWd/mtU for the four assemblies located on the core axes. During the first half of 1979, the current report period, Oconee 1 produced 170 EFPD of power and operated at a 94% capacity factor. As of June 30, 1979, the cycle 5 burnup was 226 EFPD. The burnups of the high-burnup lead assemblies were 33,890 MWd/mtU for the center assembly and 37,700 MWd/mtU for the four assemblies on the core axes. These assemblies have the highest burnup of any B&W assemblies, either operating or discharged.

Core follow results for the Oconee 1 high-burnup lead assemblies continued to exhibit good agreement between measurements and calculations. Figures 4-27 through 4-29 are eighth-core power distribution comparisons of measured assembly relative powers to PDQ07 calculated assembly relative powers. The measured power distributions were obtained from rhodium incore detector signals. The measured powers for the center assembly (core position H-8), which is one of the high-burnup lead assemblies, agrees with PDQ07 calculations within 2.4%. For the other high-burnup lead assemblies (represented by core position H-11), the measured and calculated powers agree within 2.3%.

Analyses of radioiodine activities in the reactor coolant indicate no change in the integrity of the high-burnup lead assemblies since the beginning of cycle 5.

The high-burnup lead assemblies should be discharged for post-irradiation examination in the fourth quarter of 1979.

Table 4-1. Calculated Power Peaking Factors —
Oconee 1 Cycle 2

Exposure, EFPD	Core burnup, MWd/mtU	Assembly burnup, MWd/mtU	Assembly relative power ^(a)	Node maximum relative power ^(b)	Maximum node axial location	Peak pin relative power ^(d)
<u>Assembly 1D42 in core location K-15</u>						
22.5	6268	745	1.006	1.390	29	1.247
65.0	7589	2104	1.092	1.407	52	1.238
103.5	8785	3411	1.067	1.334	47	1.233
131.0	9639	4323	1.046	1.316	38	1.229
156.0	10416	5135	1.037	1.270	38	1.226
184.0	11286	6038	1.026	1.247	34	1.224
217.0	12312	7090	1.007	1.254	34	1.223
235.0	12871	7653	0.984	1.180	106	1.270
252.9	13427	8200	0.975	1.106	34	1.269
273.5 ^(f)	14068	8826	0.968	1.154	25	1.259
<u>Assembly 1D13^(e) in core location L-14</u>						
22.5	6268	963	1.293	1.793	25	1.108
65.0	7589	2700	1.381	1.810	56	1.103
103.5	8785	4352	1.339	1.711	52	1.101
131.0	9639	5496	1.295	1.671	47	1.099
156.0	10416	6502	1.276	1.602	47	1.098
184.0	11286	7613	1.257	1.560	34	1.097
217.0	12312	8902	1.216	1.575	25	1.096
235.0	12871	9582	1.226	1.465	110	1.135
252.9	13427	10264	1.222	1.354	34	1.134
273.5 ^(f)	14068	11047	1.218	1.457	25	1.127

(a) Assembly relative power = assembly power × 177/core power (FLAME calculated).

(b) Node maximum relative power = maximum nodal power × 177 × 32/core power (FLAME calculated).

(c) Inches from bottom of fuel column to center of node of maximum relative power.

(d) Peak pin relative power = maximum pin power × 208/assembly power (2-D PDQ calculated²).

(e) Assemblies 1D13, 1D26, 1D45, and 1D55 were irradiated in symmetrically equivalent core locations during this cycle. Thus, 1D13 is representative of the four assemblies.

(f) The total exposure for this cycle was 292.2 EFPD.

Table 4-2. Calculated Power Peaking Factors —
Oconee 1 Cycle 3

Exposure, EFPD	Core burnup, MWd/mtU	Assembly burnup, MWd/mtU	Assembly relative power ^(a)	Node maximum relative power ^(b)	Maximum node axial location ^(c)	Peak pin relative power ^(d)
<u>Assembly 1D42 in core location K-12</u>						
32.0	9174	10576	1.226	1.629	20	1.121
59.0	10023	11613	1.206	1.592	20	1.117
97.5	11229	13056	1.185	1.518	20	1.117
121.9	11992	13999	1.235	1.571	25	1.330
165.2	13346	15661	1.212	1.543	25	1.121
189.3	14099	16571	1.204	1.522	25	1.117
203.3	14539	17100	1.193	1.474	25	1.116
245.8	15868	18690	1.220	1.468	83	1.106
273.0	16721	19733	1.218	1.410	92	1.102
308.3	17841	21044	1.210	1.476	43	1.099
<u>Assembly 1D13^(e) in core location K-11</u>						
32.0	9174	13076	1.362	1.615	119	1.065
59.0	10023	14225	1.327	1.538	119	1.065
97.5	11229	15811	1.291	1.472	20	1.065
121.9	11992	16758	1.231	1.467	25	1.077
165.2	13346	18412	1.210	1.446	25	1.071
189.3	14099	19324	1.211	1.436	25	1.069
203.3	14539	19857	1.198	1.392	25	1.068
245.8	15868	21448	1.177	1.257	79	1.093
273.0	16721	22454	1.172	1.307	97	1.087
308.3	17841	23733	1.173	1.370	38	1.083

(a) Assembly relative power = assembly power × 177/core power (FLAME calculated).

(b) Node maximum relative power = maximum nodal power × 177 × 32/core power (FLAME calculated).

(c) Inches from bottom of fuel column to center of node of maximum relative power.

(d) Peak pin relative power = maximum pin power × 208/assembly power (2-D PDQ calculated²).

(e) Assemblies 1D13, 1D26, 1D45, and 1D55 were irradiated in symmetrically equivalent core locations during this cycle. Thus, 1D13 is representative of the four assemblies.

Table 4-3. Calculated Power Peaking Factors —
Oconee 1 Cycle 4

Exposure, EFPD	Core burnup, MWd/mtU	Assembly burnup, MWd/mtU	Assembly relative power ^(a)	Node maximum relative power ^(b)	Maximum node axial location ^(c)	Peak pin relative power ^(d)
<u>Assembly 1D42 in core location K-14</u>						
23.4	10717	21731	0.892	1.064	101	1.174
46.2	11430	22367	0.900	1.065	52	1.163
76.5	12380	23230	0.920	1.082	97	1.153
103.4	13220	24007	0.929	1.069	92	1.142
125.2	13902	24641	0.933	1.090	83	1.134
150.7	14702	25390	0.940	1.057	92	1.126
174.8	15457	26099	0.946	1.070	43	1.120
208.6	16514	27100	0.952	1.044	106	1.113
234.7 ^(f)	17331	27877	0.955	1.060	38	1.106
<u>Assembly 1D13^(e) in core location M-13</u>						
23.4	10717	24462	0.946	1.134	106	1.076
46.2	11430	25136	0.954	1.092	43	1.072
76.5	12380	26044	0.959	1.127	101	1.068
103.4	13220	26849	0.963	1.101	101	1.063
125.2	13902	27505	0.969	1.114	92	1.058
150.7	14702	28279	0.965	1.079	101	1.053
174.8	15457	29009	0.962	1.085	34	1.049
208.6	16514	30030	0.965	1.060	110	1.044
234.7 ^(f)	17331	30819	0.966	1.070	29	1.041

- (a) Assembly relative power = assembly power × 177/core power (FLAME calculated).
- (b) Node maximum relative power = maximum nodal power × 177 × 32/core power (FLAME calculated).
- (c) Inches from bottom of fuel column to center of node of maximum relative power.
- (d) Peak pin relative power = maximum pin power × 208/assembly power (2-D PDQ calculated²).
- (e) Assemblies 1D13, 1D26, 1D45, and 1D55 were irradiated in symmetrically equivalent core locations during this cycle. Thus, 1D13 is representative of the four assemblies.
- (f) The total exposure of this cycle was 245.9 EFPD.

Table 4-4. Post-Irradiation Examination Measurements

<u>Measurement</u>	<u>Estimated precision (1σ)</u>
Fuel assembly length, in.	± 0.048
Fuel rod diameter, in.	± 0.000104
Water channel spacing, in.	± 0.00007
Holddown spring preload force, lb	± 10
Fuel column length, in.	± 0.15
Fuel column gaps, in.	± 0.05

Table 4-5. Summary of Fuel Assembly Growth Results

<u>Fuel assembly</u>	<u>Assembly average fluence (>1 MeV), $\times 10^{-21}$ nvt</u>	<u>Assembly growth, % $\Delta L/L$</u>
1D13	5.20	0.40
1D26	5.20	0.35
1D42	4.71	0.35
1D45	5.20	0.30
1D55	5.20	0.38
Average	5.11	0.36

Table 4-6. Summary of Fuel Rod Growth Results

<u>Fuel assembly</u>	<u>Assembly average fluence (>1 MeV), $\times 10^{-21}$ nvt</u>	<u>Average fuel rod growth, % $\Delta L/L$</u>
1D13	5.20	0.52
1D26	5.20	0.50
1D42	4.71	0.48
1D45	5.20	0.56
1D55	5.20	0.51
Average	5.11	0.51

Table 4-7. Average Mean Diameter and Creepdown by Grid-Span Location

Grid-span loc'n	Fuel assembly				
	1D13	1D26	1D42	1D45	1D55
<u>Top-grid 1</u>					
Diameter, in.	0.4273	0.4273	0.4280	0.4276	0.4276
σ	0.0006	0.0005	0.0003	0.0006	0.0007
Creepdown, (a) mils	2.6	2.6	1.9	2.3	2.3
<u>Grid 1-grid 2</u>					
Diameter, in.	0.4261	0.4261	0.4272	0.4266	0.4265
σ	0.0004	0.0005	0.0002	0.0005	0.0006
Creepdown, mils	3.8	3.8	2.7	3.3	3.4
<u>Grid 2-grid 3</u>					
Diameter, in.	0.4258	0.4257	0.4266	0.4262	0.4260
σ	0.0004	0.0003	0.0002	0.0004	0.0005
Creepdown, mils	4.1	4.2	3.3	3.7	3.9
<u>Grid 3-grid 4</u>					
Diameter, in.	0.4256	0.4255	0.4264	0.4259	0.4259
σ	0.0003	0.0003	0.0002	0.0004	0.0004
Creepdown, mils	4.3	4.4	3.5	4.0	4.0
<u>Grid 4-grid 5</u>					
Diameter, in.	0.4255	0.4254	0.4262	0.4257	0.4258
σ	0.0003	0.0003	0.0004	0.0003	0.0005
Creepdown, mils	4.4	4.3	3.7	4.2	4.1
<u>Grid 5-grid 6</u>					
Diameter, in.	0.4254	0.4255	0.4263	0.4257	0.4257
σ	0.0003	0.0003	0.0003	0.0003	0.0004
Creepdown, mils	4.5	4.4	3.6	4.2	4.2
<u>Grid 6-bottom</u>					
Diameter, in.	0.4259	0.4257	0.4265	0.4258	0.4259
σ	0.0005	0.0004	0.0004	0.0006	0.0005
Creepdown, mils	4.0	4.2	3.4	4.1	4.0
<u>Assembly avg</u>					
Diameter, in.	0.4259	0.4259	0.4267	0.4262	0.4262
σ	0.0012	0.0010	0.0008	0.0012	0.0014
Creepdown, mils	4.0	4.0	3.2	3.7	3.7
n(b)	21	32	12	17	32

(a) All of the listed creepdown values are based on an as-built cladding lot average diameter of 0.4299 inch.

(b) Number of fuel rods measured in each assembly.

Table 4-8. Average Maximum Ovality by
Grid-Span Location

Grid-span location	Fuel assembly				
	1D13	1D26	1D42	1D45	1D55
Top-grid 1	0.0015 ^(a)	0.0022	0.0015	0.0020	0.0019
Grid 1-grid 2	0.0024	0.0023	0.0022	0.0026	0.0023
Grid 2-grid 3	0.0026	0.0021	0.0022	0.0023	0.0021
Grid 3-grid 4	0.0026	0.0022	0.0024	0.0024	0.0021
Grid 4-grid 5	0.0027	0.0024	0.0026	0.0026	0.0021
Grid 5-grid 6	0.0024	0.0021	0.0024	0.0024	0.0021
Grid 6-bottom	0.0026	0.0020	0.0025	0.0029	0.0019
Assembly avg	0.0024	0.0022	0.0023	0.0024	0.0021
n ^(b)	21	32	12	17	32

(a) Average maximum ovality is defined as the 4σ range of the mean diameter, i.e., a 95% confidence interval about the mean diameter. The values are given in inches.

(b) Number of fuel rods measured in each fuel assembly.

Table 4-9. Summary of Fuel Assembly Water Channel Spacing Measurements

Assembly 1D13				Assembly 1D45			
Midspan level	(a) Mean spacing, mils	Rod bow(σ), mils	Gap (b) closure, %	Midspan level	Mean spacing, mils	Rod bow(σ), mils	Gap (b) closure, %
2	137.4	6.8	8.1	2	137.0	5.7	6.8
4	139.9	12.9	15.4	4	139.4	12.3	14.6
6	141.8	22.6	26.9	6	140.7	25.0	29.8
8	141.1	20.5	24.6	8	139.4	28.7	34.2
10	141.6	20.0	23.8	10	141.5	28.6	34.1
12	141.1	21.1	25.2	12	141.3	26.4	31.4
14	140.3	11.6	13.9	14	139.0	15.4	18.4
Avg	140.4	17.5	20.8	Avg	139.8	21.9	26.2

Assembly 1D42				Assembly 1D55			
Midspan level	Mean spacing, mils	Rod bow(σ), mils	Gap (b) closure, %	Midspan level	Mean spacing, mils	Rod bow(σ), mils	Gap (b) closure, %
2	137.6	5.8	6.9	2	137.2	5.8	6.9
4	139.1	9.6	11.5	4	139.1	10.3	12.3
6	141.7	22.0	26.2	6	140.2	21.5	25.7
8	140.4	28.8	34.3	8	141.8	21.6	25.7
10	138.7	31.6	37.7	10	141.6	20.4	24.3
12	142.8	27.1	32.3	12	142.7	21.0	25.0
14	139.5	15.2	18.1	14	140.2	12.2	14.5
Avg	140.0	22.0	26.2	Avg	140.4	17.3	20.6

Assembly 1D26			
Midspan level	Mean spacing, mils	Rod bow(σ), mils	Gap (b) closure, %
12	142.4	22.5	26.8

(a) Refer to Figure 4-3.

(b) Statistically, 95% of the water channels have a gap closure less than the value tabulated.

Table 4-10. Summary of Holddown Spring Force Measurements

<u>Fuel assembly</u>	<u>Spring^(a) rate, lb/in.</u>	<u>Measured preload, lb</u>	<u>Change in^(b) preload, % $\Delta F/F$</u>	<u>Net operating^(c) force, lb</u>
1D13	282	348	-39.5	562
1D26	308	385	-33.0	600
1D42	298	374	-35.0	583
1D45	285	393	-31.6	574
1D55	320	361	-37.2	611
Average	299	372	-35.3	586

(a) The nominal as-built spring rate is 277 lb/in.

(b) Using a mean as-built preload force (F) of 575 lb.

(c) The nominal beginning-of-life value is 595 lb.

Table 4-11. Fuel Column Axial Gap Data From Gamma Scans

<u>Fuel assembly</u>	<u>Rod</u>	<u>Gap size, in.</u>	<u>Gap location</u>
1D13	AB	None	
	BC	None	
	CD	0.1	1 in. from top of fuel column
	AD	0.1	54 in. from bottom
1D26	AB	None	
	BC	None	
	CD	None	
	AD	None	
1D42	AB	None	
	BC	None	
	CD	None	
	AD	None	
1D45	AB	None	
	BC	None	
	CD	None	
	AD	0.1	14 in. from top
		0.1	19 in. from top
1D55	AB	0.08	66 in. from bottom
	BC	0.08	5.5 in. from top
	CD	0.1	33 in. from bottom (a)
	AD	0.05	78 in. from top

(a) Location less accurate due to slippage of paper on chart recorder.

Table 4-12. Fuel Column Stack Length Data
From Gamma Scans

<u>Fuel assembly/rod</u>	<u>Avg burnup, MWd/mtU</u>	<u>Corrected stack length^(a), in.</u>
1D13/AB	29,300	142.0
BC	31,010	141.8
CD	29,840	141.8
AD	30,980	142.1
1D26/AB	29,840	139.7 ^(b)
BC	31,010	140.2
CD	29,300	142.2
AD	30,980	141.4
1D42/AB	25,370	141.5
BC	29,600	141.1
CD	29,930	141.7
AD	26,680	141.9
1D45/AB	29,840	140.7
BC	30,980	141.6
CD	29,300	141.1
AD	31,010	141.1
1D55/AB	29,300	140.1
BC	30,980	140.8 ^(b)
CD	29,840	141.3 ^(b)
AD	31,010	140.7

(a) Measured stack length minus total gap length.

(b) These values are less accurate since they were taken from down-scans.

Figure 4-1. Fuel Assembly Cross Section Displaying Numbering Conventions

Figure 4-2.
Mark B Fuel Assembly

Figure 4-3. Mark B Fuel Assembly Showing Numbering Convention of Water Channel Midspan Levels

Figure 4-4. Locations of Extended Burnup Fuel Assemblies During Their Four Cycles of Operation

Figure 4-5. Axial Power Shape — Oconee 1, Cycle 2:
Assemblies 1D13, 1D26, 1D45, 1D55

*Relative to core average power.

Figure 4-6. Axial Power Shape — Oconee 1, Cycle 3:
Assemblies 1D13, 1D26, 1D45, 1D55

*Relative to core average power.

Figure 4-7. Axial Power Shape — Oconee 1, Cycle 4:
Assemblies 1D13, 1D26, 1D45, 1D55

*Relative to core average power.

Figure 4-8. Axial Burnup Profiles: Assemblies
1D13, 1D26, 1D45, 1D55

Figure 4-9. Axial Power Shape — Oconee 1, Cycle 2:
Assembly 1D42

*Relative to core average power.

Figure 4-10. Axial Power Shape — Oconee 1, Cycle 3:
Assembly 1D42

*Relative to core average power.

Figure 4-11. Axial Power Shape — Oconee 1, Cycle 4:
Assembly 1D42

*Relative to core average power.

Figure 4-12. Axial Burnup Profiles: Assembly 1D42

Figure 4-13. Histogram of Sipping Results for Oconee 1 Batch 4 Fuel Assemblies

Figure 4-14. Photograph Displaying Narrow Circumferential
Crud Bands, Assembly 1D13

Figure 4-15. Photograph Displaying Indentation on Rod A13 of Assembly 1D21

Figure 4-16. Photograph Displaying Dent on Grid Skirt of Assembly 1D42

Figure 4-17. Fuel Assembly Growth as Function of Assembly Average Fluence

Figure 4-18. Fuel Rod Growth as Function of Assembly Average Fluence

Figure 4-19. Assembly 1D26 Corner Fuel Rods Orthogonal Diameter Profilometry Traces

FUEL ASSEMBLY 1D26 RODS A1 & D15

FUEL ASSEMBLY 1D26 RODS D1 & C15

FUEL ASSEMBLY 1D26 RODS C1 & B15

FUEL ASSEMBLY 1D26 RODS B1 & A15

Figure 4-20. Axial Distribution of Fuel Rod Average Diametric Creepdown

Figure 4-21. Typical Full-Size Diameter Trace Segments
(Orthogonal Scans) — Fuel Assembly 1D55

Figure 4-22. Axial Distribution of Fuel Rod Average Maximum Ovality

Figure 4-23. Worst-Span Water Channel Closure as Function of Assembly Average Burnup

Figure 4-24. Axial Distribution of Water Channel Closure

NOTE: The left most data point for each fuel assembly was measured ~3" below midspan.

*95% of the fuel rods have a gap closure less than the plotted value.

Figure 4-25. Holddown Spring Net Operating Force

Figure 4-26. Portion of Gamma Scan Trace — Assembly 1D13, CD-Corner Fuel Rod

Figure 4-27. Power Distribution at 44.4 EFPD — Oconee 1, Cycle 5

Figure 4-28. Power Distribution at 90.4 EFPD — Oconee 1, Cycle 5

*LOCATION OF EXTENDED BURNUP ASSEMBLIES.

POWER	MEASURED	PDQ
EXPOSURE	99.5	100 %FP
GP6	90.4	100 EFPD
GP7	100% WD	OUT
GP8	88% WD	OUT
	24% WD	IN

Figure 4-29. Power Distribution at 193.2 EFPD — Oconee 1, Cycle 5

MEASURED PDQ
99.1 100 %FP
193.2 200 EFPD
GP6 100% WD OUT
GP7 90% WD OUT
GP8 24% WD IN

*LOCATION OF EXTENDED BURNUP ASSEMBLIES.

4-52

5. TASK 2 — OPERATIONAL LIMITS OF MARK B CORE

L. W. Newman, M. R. Eastburn, N. Smith, S. S. Kere

5.1. Introduction

The simplified fuel cycle models that were 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, fuel management plans resulting in discharge batch average burnups from ~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.

During this reporting period, subtask 2A "Fuel Cycle Design," was completed. In this subtask, evaluations were conducted to develop extended-burnup fuel cycles. These fuel cycles form the base on which subtasks 2B through 2E, the multi-disciplinary evaluations normally considered in reload licensing applications, are performed. Efforts on the nuclear analyses for subtask 2B, "Nuclear, Mechanical, and Thermal-Hydraulic Analyses," were initiated during the present reporting period.

5.2. Subtask 2A — Fuel Cycle Design

5.2.1. Introduction

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-fuel assembly (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 operating 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 selected for 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.

5.2.2. Design Methods and Criteria

The diffusion theory program PDQ07² with thermal-hydraulic feedback was used for the detailed fuel management evaluations. Two-dimensional quarter-core geometry was used with one mesh interval per cell pitch. Two-group neutron cross sections used in PDQ07 are fit as functions of several variables, including ²³⁵U enrichment and fuel burnup (MWd/mtU). Because relatively high ²³⁵U enrichments and fuel burnups were projected for the extended-burnup cycles, the cross section library was expanded to 5.25% ²³⁵U enrichment and 80,000 MWd/mtU burnup.

Beginning-of-cycle (BOC) isotopics for four of the five fuel management evaluations were obtained from previously licensed out-in, rodded fuel cycles. Annual cycles employing the out-in shuffle scheme and feed-and-bleed operation

were used to obtain BOC isotopics for the fifth fuel management evaluation, the 36-FA feed case.

For all cases except the 80-FA feed case, the projected equilibrium cycle enrichment was loaded, the cycle was depleted, and cycle lifetime was determined. The first two cycles are transition cycles from the conventional out-in scheme to the low-leakage shuffle scheme, which incorporated high burnup and higher feed enrichments. In actual practice, the feed enrichment of transition cycles would probably be different from that of the equilibrium cycle to make the transition more gradual and to reduce power peaking. For this study the equilibrium enrichment was loaded to reach an equilibrium cycle sooner. The 80-FA feed case used enrichments varied by cycle to demonstrate the reductions in power peaking that could be achieved during transition cycles.

The following design criteria were utilized: maximum radial relative pin power of less than 1.651, 1.0% Δp shutdown margin with the most reactive control rod assumed stuck out of the core, moderator temperature coefficient less than or equal to zero at rated power, and a negative power coefficient in the operating range of 15-100% rated power.

5.2.3. Results and Discussion

The results of primary interest are uranium utilization, power distributions, fuel burnups, and control rod worths. Uranium utilization is a measure of how efficiently the uranium ore is used. It is measured in units of energy per unit of ore, or megawatt-years per short ton of yellowcake. Power distributions are required to establish the reactor operating limits that could be imposed for a given cycle. Fuel burnups and control rod worths are needed to verify that the fuel cycle is within the design limits of the fuel and that adequate shutdown margin is maintained.

The base case against which various options were compared involved three-batch refueling, the LBP shuffle scheme, and annual (292 EFPD) fuel cycles at a power level of 2568 MWt. Discharge batch average burnup for the base case is 27,413 MWd/mtU, and its uranium utilization is 11.73 MWy/2000 lb U_3O_8 (assuming 1% fabrication overage and a 0.5% conversion loss).

Table 5-1 shows the uranium utilization for the five fuel management schemes. As expected, the 36-FA feed case (approximately five-batch refueling) with annual cycles gave the highest uranium utilization, 13.76 MWy/2000 lb U_3O_8 ,

which represents a 17.3% increase over the base case. Of the 18-month fuel cycles, the best utilization was the 60-FA feed case (approximately three-batch refueling). The utilization was 12.54 MWy/2000 U₃O₈, a 6.9% increase over the base case. Thus, a substantial improvement in uranium utilization can be realized from extended burnup in either annual or 18-month fuel cycles relative to annual, three-batch refueling.

The extended burnup cycles resulted in no significantly increased peaking for the equilibrium cycle compared to typical three-batch fuel management. Therefore no major restrictions to nuclear operating limits for extended-burnup equilibrium cycles are expected. The margins in Table 5-2 to the maximum pin power peaks confirm this conclusion. The 80-FA feed case gave the lowest peaking, even in the transition cycles, because the enrichment of fresh fuel introduced during the transition was adjusted to reduce peaking. There are also more fresh FAs to share power with the 80-FA feed case. The detailed fuel management evaluations indicate that power peaking in extended-burnup fuel cycles can be controlled to within the nuclear design criteria currently accepted for light water reactors (LWRs). Transition cycle power peaking will be more difficult to control, but adequate design margin can be maintained by careful selection of fuel enrichment and feed batch size.

Discharge batch average burnups and maximum FA discharge burnups are given in Table 5-3 for the five fuel management schemes. The ratio of maximum assembly burnup to batch average discharge burnup is higher for extended burnup cycles than it is for the base case. Some reduction in the ratio could be achieved by placing more emphasis on equalizing discharge burnups during the fuel cycle design process.

Control rod worths and shutdown margins for extended-burnup fuel cycles were not significantly different from the base case. The equilibrium extended-burnup cycles met the requirement of 1.0% $\Delta\rho$ shutdown margin. Table 5-4 gives shutdown margins for the equilibrium fuel cycles.

5.2.4. Conclusions

Detailed fuel management evaluations of various extended-burnup fuel cycles confirmed the uranium utilization improvements projected by simplified reactor models. Furthermore, extended-burnup cycles can be designed to satisfy standard nuclear design criteria.

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

5.3.1. Nuclear Analyses

5.3.1.1. Introduction

A significant aspect of high-burnup fuel management is the generation of physics parameters for safety and control analyses. These parameters reflect the unique characteristics of high-burnup cores and must be evaluated and compared to those for standard designs to ensure licensability and compatibility with control systems. This section describes work completed to date in this area. It includes PDQ07 deck development, calculated boron and reactivity requirements, and a comparison to a "standard" design.

5.3.1.2. Results and Discussion

PDQ07 Deck Development — The generation of physics parameters requires an accurate representation of nuclide cross sections and their dependency on reactor characteristics. These dependencies include flux distribution in the reactor core, spatial distribution of various nuclides, nonuniform distributions of moderator density, moderator and fuel temperatures, plus many others. Because of the extensive number of physics parameters to be analyzed, a simplified PDQ07 model was used. A homogenized fuel assembly PDQ07 model simplifying the geometric representation of the core was selected. It reduces the required computer memory and permits an increase in accuracy with which complex cross-sectional dependencies over large temperature and burnup ranges can be represented. Table 5-5 lists the important core parameters and their cross-sectional fitting ranges incorporated in the homogenized model. The moderator and fuel temperature ranges have been expanded beyond nominal reactor operating temperatures in order to generate the data needed to analyze safety- and control-related accidents.

Physics Parameters — The physics parameters calculated to date for the five extended burnup fuel cycle designs are boron concentrations, soluble boron worth, and xenon worth. These parameters are part of the total nuclear input for both safety and control 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.

Although the detailed safety and control analyses have not been completed, the potential impact of the high-burnup fuel cycle designs on licensability and control requirements can be assessed by comparing these parameters 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.

Boron Concentration/Boron Worth — Boron concentrations versus cycle lifetime at hot full power, hot zero power (HFP, HZP) and 70F are compared in Figures 5-1, 5-2 and 5-3, respectively, for ANO-1, cycle 3; ANO-1, cycle 4; and the equilibrium cycle of the 60-FA feed rodded design. Figure 5-1 compares critical boron concentrations for nominal HFP operation. The discontinuities at 250 EFPD for ANO-1, cycles 3 and 4, and at 466 EFPD for the extended-burnup design are caused by withdrawing the transient control rod bank. Figures 5-2 and 5-3 compare hot and cold shutdown boron concentrations for the three designs. An overall comparison of selected physics parameters is presented in Table 5-6. A notable trend is apparent from the data shown in Figures 5-1, 5-2, and 5-3 and Table 5-6. As cycle average burnup increases, feed enrichment, soluble boron requirements for reactivity control, and initial LBP loading all increase.

In these fuel cycles, initial fuel excess reactivity is controlled by soluble boron and LBP. The ideal situation is to select an initial LBP loading that will completely deplete during the cycle lifetime and to control the remainder of the excess reactivity with soluble poison. However, a trade-off between the two means of reactivity control can be made. High soluble boron concentrations can present a problem with borated water storage tank size and waste storage requirements. The initial LBP concentration can be increased to reduce soluble boron requirements, but increasing the initial LBP concentration reduces cycle length because of an increased LBP residual at the end of the cycle.

A comparison of HFP inverse boron worths (IBWs) is provided in Figure 5-4. The variations in IBW are caused by neutron spectrum shifts from different initial fuel enrichments and cycle depletion characteristics. The low boron worth, together with increased reactivity requirements of high burnup, results in a greater borated water storage tank requirement.

Xenon Worth — Equilibrium xenon worths for the three cycles are shown as a function of cycle lifetime in Figure 5-5. Changes in the reactivity held by xenon are caused by the variation of the fuel macroscopic cross section (Σ_f) and the macroscopic absorption cross section of xenon (Σ_{aXe}) as a function of burnup and fuel enrichment. In a large reactor with flux levels greater than or equal to 10^{13} n/cm²-s, xenon worth is proportional to Σ_{aXe}/Σ_f . Although both cross sections increase with increasing enrichment and decrease with cycle burnup, their rates of change are not equal. The behavior of the ratio Σ_{aXe}/Σ_f determines the xenon worth trends exhibited in Figure 5-5. Specifically, this ratio increases slightly with burnup for a given fuel cycle design, causing an increase in the xenon worth, but it decreases as the initial fuel enrichment increases. Since extended-burnup cycles require higher fuel enrichments, their xenon worths for a given burnup are less than those of current reloads. The lower xenon worth for the extended-burnup cycles is expected to enhance reactor control because the change in reactivity due to buildup and decay during power changes will be reduced.

5.3.1.3. Conclusions

Preliminary analyses reveals that current B&W operating plants have borated water storage tanks (BWSTs) of adequate size to provide the contraction volumes necessary for plant control. In a few cases the boric acid concentrations in the BWST must be increased. However, the higher soluble boron concentrations will present no foreseen dilution problems for safety-related accidents.

Table 5-1. Uranium Utilization — 177-FA Plant,
15 × 15 Rod Assemblies

Feed batch size	Core power, MWt	Equilibrium cycle length, EFPD	Equilibrium discharge burnup, MWt/mtU	Actual ^(a) uranium utilization MWy/STU ₃ O ₈	Percent change in uranium utilization
59 ^(b)	2568 R	292	27,413	11.73	0.0
80	2772 FB	460	34,382	11.24	-4.2
68	2772 FB	460	40,448	12.13	+3.4
60	2772 FB	460	45,838	12.54	+6.9
60	2568 R	497	45,838	12.52	+6.7
36	2772 FB	292	48,500	13.76	+17.3

(a) Verified by completed PDQ analysis of four fuel cycles.

(b) Base case.

Table 5-2. Percent Margin to Maximum Allowed
Pin Power Peak

Feed batch size	Core power, MWt	Cycle					
		1	2	3	4	5	6
59 ^(a)	2568 R	--	--	--	5.0	--	--
80	2772 FB	15.5	15.5	11.7	8.4	--	--
68	2772 FB	7.0	6.3	3.3	4.9	--	--
60	2772 FB	5.2	6.2	8.5	4.5	--	--
60	2568 R	0.8	0.4	4.4	4.5	--	--
36	2772 FB	5.6	8.7	8.1	9.3	9.5	9.7

(a) Base case.

Table 5-3. Discharge Burnups

<u>Feed batch size</u>	<u>Core power, MWt</u>	<u>Burnup</u>		
		<u>Batch average, MWd/mtU</u>	<u>Maximum assembly, MWd/mtU</u>	<u>Ratio, max:avg</u>
59 ^(a)	2568 R	27,413	30,700	1.12
80	2772 FB	34,770	39,552	1.14
68	2772 FB	40,806	49,967	1.22
60	2772 FB	45,891	54,506	1.19
60	2568 R	45,673	55,713	1.22
36	2772 FB	49,376	51,553	1.04

(a) Base case.

Table 5-4. Shutdown Margin

<u>Feed batch size</u>	<u>Core power, MWt</u>	<u>Transition cycle</u>		<u>Equilibrium cycle</u>	
		<u>BOC</u>	<u>EOC</u>	<u>BOC</u>	<u>EOC</u>
59 ^(a)	2568 R	--	--	2.87	1.45
80	2772 FB	3.02	2.10	3.27	1.91
68	2772 FB	3.43	1.94	3.13	1.70
60	2772 FB	3.48	1.59	3.55	2.20
60	2568 R	2.67	1.80	2.49	1.51
36	2772 FB	1.90	1.26	1.83	1.28

(a) Base case.

Table 5-5. Homogenized Model Fitting Ranges

Parameter	Range ^(a)
Fuel enrichment, wt % ²³⁵ U	2.00 → 5.00
LBP enrichment, wt % B ₄ C	0.01 → 3.00
Moderator temperature, F, three plateaus	70 → 80 @ 15 psig 300 → 320 @ 800 psig 500 → saturation + 50% voids @ 2200 psia
Fuel temperature, F, three plateaus	70 → 80 @ 15 psig 300 → 320 @ 800 psig 500 → 3000 @ 2200 psig
Burnup, MWd/mtU	0 → 81,000

(a) Cross-sectional arrangement allows interpolation off the ends of all ranges except below 70F for the moderator and fuel temperature.

Table 5-6. Nuclear Parameter Comparison for ANO-1
Cycle 3, ANO-1 Cycle 4, and 60-FA Feed,
Rodded Transition Cycle

	<u>ANO-1 cycle 3</u>	<u>ANO-1 cycle 4</u>	<u>Rodded transition cycle</u>
Avg cycle length, EFPD	294	387	466
HFP BOC critical conc, banks 7 and 8 inserted, ppm	1068	1247	1413
BOC IM boron worth, ppm/% Δp	104	117	136
Int LBP wt %/assembly	0	0.29	0.40
Xenon worth, 4 EFPD, % Δp	2.66	2.59	2.49
Core avg initial enrichment, wt % ²³⁵ U	2.88	2.95	3.45

Figure 5-1. Critical Boron Concentration, Nominal HFP Operation

Figure 5-2. Hot Shutdown Boron Concentration, HZP, 1% Shutdown,
No Xenon, ARI/SRO

Figure 5-3. Cold Shutdown Boron Concentration, 70F, 1% Shutdown,
No Xenon, ARI/SRO

Figure 5-4. BOC HFP Inverse Boron Worth

Figure 5-5. Comparison of Xenon Worths for ANO-1 Cycle 3, ANO-1 Cycle 4, 60-FA Feed, Rodded Cycle 5

6. REFERENCES

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- ⁴ T. D. Pyecha, et al., Nondestructive Examination of Oconee 1 Fuel Assemblies After Three Cycles of Irradiation, BAW-1553, Babcock & Wilcox, Lynchburg, Virginia, September 1979.
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APPENDIX
Fuel Rod Diameter Profilometry —
Definition of Terms

1. Average Diameter — the arithmetic mean of all diameter values from a given rod segment (between spacer grids). Rod average diameter is the unweighted mean of the segment means.
2. Average Creepdown — the BOL diameter (0.4299 in.) minus average diameter as defined above.
3. Average Maximum Ovality — the standard deviation of diameter values for a rod segment multiplied by four. Rod average maximum ovality is the unweighted mean of the segment values. This value was chosen because it includes 95% of the diameter values on either side of the mean. It is only intended to provide an approximation of ovality peaks rather than to approximate the average ovality. When compared to hot cell data (obtained using 180° rotating, step-scan profilometry), it compares favorably with average ovality if the ovality is fairly constant, but it is higher than average ovality when there are large fluctuations in the hot cell ovality data.
4. Peak Ovality — the maximum minus the minimum outside diameter measured in the same transverse plane. Approximate peak ovality values can be obtained from orthogonal, single-orientation, line scan traces of corner rods. The measured values are not necessarily the true peak ovality values because the orientation of the profilometry scans may not coincide directly with the major and minor axes of the respective oval. However, since the true peak ovalities cannot be smaller than the corresponding observed values, the measured values indicate the minimum peak ovality of the cladding.