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B&W FUEL COMPANY

1989 FUEL PERFORMANCE REPORT

NOVEMBER 1990

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CONTENTS

1. INTRODUCTION

This report was prepared for the U.S. Nuclear Regulatory Commission (NRC)in response to their request for a non-proprietary annual summary of in-reactor fuel performance and ongoing development programs for Babcock & Wilcox (B&W, now B&W Fuel Company) designed commercial nuclear fuel. The NRC will use the fuel performance information provided by nuclear fuel suppliers and utilities to publish a comprehensive fuel performance annual report. The NRC report identifies ongoing fuel surveillance programs, summarizes the results from these programs, reports on generic problems that are of concern during the reporting period, and provides a traceable path of references for additional details.[1] Supporting this objective, this report briefly describes the fuel design and fuel development programs in progress at B&W Fuel Company (BWFC, Section 2), summarizes the operational experience and performance of BWFC-designed fuel for calendar year 1989 (Sections 3 and 4), discusses fuel management and operational improvements that either were implemented or were being demonstrated in 1989 (Section 5), and contains a list of references that provide additional, more detailed information in these areas (section 6).

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B&W Fuel Company
Lynchburg, Virginia

B&W Fuel Company 1989 Fuel Performance Report

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Key Words : Pressurized Water Reactor, Fuel Performance, Fuel Surveillance, UO_2 Fuel, Zircaloy-4 Fuel Cladding, Extended Burnup

ABSTRACT

This summary report, prepared at the request of the U.S. Nuclear Regulatory Commission, briefly describes the fuel development and performance improvement programs being conducted by the B&W Fuel Company.* It also reviews the 1989 in-reactor performance of B&W Fuel Company-designed fuel. References to additional, more detailed information are included.

B&W Fuel Company (BWFC), together with the Duke Power Company, Arkansas Power & Light, Sacramento Municipal Utility District, and The Department of Energy, are involved in continuing programs to improve fuel utilization and extend the average burnup of discharged fuel assemblies to 50 GWd/mtU by developing and demonstrating advanced fuel designs. The Mark-GdB lead test assembly (LTA) in 1989 achieved a burnup of 58.3 GWd/mtU, a new record for LWR fuel assemblies. In addition to the extended burnup program, other programs continue to obtain high burnup data on the in-reactor performance of low absorption Zircaloy spacer grids, a fuel-burnable absorber mixture (UO_2-Gd_2O_3), axial blanket fuel, annular fuel pellets, advanced cladding, and annealed guide tubes.

Additional programs are underway leading to full batch reloads of BWFC designs in Westinghouse-designed reactors. These programs include both 15x15 and 17x17 fuel designs. The 15x15 Zircaloy clad LTA designed to replace stainless steel clad fuel assemblies completed its third cycle of irradiation. The 17x17 Mark-BW LA completed its second cycle. Poolside examinations showed these assemblies were in excellent condition, and that the performance verified the base design.

BWFC-designed Zircaloy clad fuel achieved an estimated fuel integrity level of 99.997 % with 257,712 rods irradiated. Due to debris problems, a large number of stainless steel clad fuel rods failed in service. This incident in which 450 fuel rods failed shows the potential degrading effects of debris on fuel rods. BWFC has developed and is implementing debris resistant fuel designs. Improvements in fuel design have resulted in generally decreasing coolant activity levels. The average coolant iodine for 1989 is one fourth of the 1980 level. During this span, the fuel lifetime and economy expected from the fuel have increased.

* The B&W Fuel Company is a partnership between Babcock & Wilcox and the American subsidiaries of a French consortium.

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1. INTRODUCTION

This report was prepared for the U.S. Nuclear Regulatory Commission (NRC)in response to their request for a non-proprietary annual summary of in-reactor fuel performance and ongoing development programs for Babcock & Wilcox (B&W, now B&W Fuel Company) designed commercial nuclear fuel. The NRC will use the fuel performance information provided by nuclear fuel suppliers and utilities to publish a comprehensive fuel performance annual report. The NRC report identifies ongoing fuel surveillance programs, summarizes the results from these programs, reports on generic problems that are of concern during the reporting period, and provides a traceable path of references for additional details.[1] Supporting this objective, this report briefly describes the fuel design and fuel development programs in progress at B&W Fuel Company (BWFC. Section 2), summarizes the operational experience and performance of BWFC-designed fuel for calendar year 1989 (Sections 3 and 4), discusses fuel management and operational improvements that either were implemented or were being demonstrated in 1989 (Section 5), and contains a list of references that provide additional, more detailed information in these areas (section 6).

2. FUEL DEVELOPMENT PROGRAMS

Ongoing joint programs among B&W Fuel Company (BWFC), The Department of Energy (DOE), the Electric Power Research Institute (EPRI), and utilities emphasize improving fuel utilization and fuel performance in pressurized water reactors (PWRs). Included in the majority of these programs, which are described briefly in this section, are on-site nondestructive, hot cell nondestructive and destructive post-irradiation examinations. The hot cell examinations are conducted in B&W's Lynchburg Research Center.

2.1. BWFC Fuel Designs

In 1989, BWFC made no changes to the design parameters of the Mark B, Mark C and 15X15 stainless steel clad fuel rod array assemblies. A debris resistant lower end fitting is in development for the Mark-BW17 design. The Mark-BW15 fuel assembly is scheduled to have a debris resistant fuel rod similar to that used in the Mark-B8. Table 2-1 contains typical BWFC fuel assembly design parameters for the current 15X15 (Mark B) and 17X17 (Mark C) Zircaloy clad fuel rod arrays in B&W-designed reactor systems. Table 2-2 provides the current design parameters for the BWFC replacement fuel for Westinghouse designed reactors: 17X17 (Mark-BW17) zircaloy clad, 15X15 stainless steel clad, and 15X15 (Mark-BW15) zircaloy clad fuel rod arrays. The Mark-BW zircaloy clad designs for Westinghouse-designed reactors represent the latest additions to the BWFC product line. They are discussed along with the Mark B8 design in more detail in the following sections.

2.1.1. Mark B8 Design

The Mark B8 fuel assembly is based on the standard Mark B fuel assembly with design features added which allow for easy field reconstitution, provide protection against debris induced fretting failure, and allow for high burnup. To permit field reconstitution the upper end fitting was made easily removable. The connection between the upper inconel spacer grid and the upper end fitting was modified to allow for removal of the upper end fitting. To provide more room for fuel rod growth, the upper to lower end fitting distance was increased. This increase was accomplished by shortening the lower end fitting and lengthening the guide and instrument tubes to retain the same overall assembly length.

To provide protection against debris, the length of the lower fuel rod end plug was significantly increased with most of the length being solid stock. Also, the position of the lower spacer grid was lowered so that the solid portion of the lower end plug extends through it. This arrangement will trap debris that can become lodged below the spacer grid at the solid portion of the end plug. Thus, fretting wear cannot breach the cladding barrier.

2.1.2. Mark-BW17 Design

The Mark-BW17 is a BWFC-designed fuel assembly completely compatible with Westinghouse 17X17 standard (STD) and optimized (OFA) fuel assemblies. The Mark-BW is designed for use in Westinghouse reactors while providing standard BWFC-design features such as floating spacer grids, thicker fuel rod clad, and a double fuel rod plenum. The major design parameters are listed in Table 2-2.

Four lead assemblies (LAs) began irradiation in November 1987 in cycle 5 of the McGuire Unit 1 reactor. The first cycle of exposure was completed in October 1988. A poolside nondestructive examination of the LAs was conducted in November 1988. The assemblies were in excellent condition with performance trends meeting or being enveloped by the design assumptions. The LAs were reinserted for a second cycle of irradiation in McGuire Unit 1 cycle 6, which started in November 1988 and finished in February 1990. A second poolside examination was performed after cycle 6, and the results showed that the base Mark-BW LAs were in excellent condition after 27.7 GWd/mtU.

2.1.3. Mark-BW 15 Design

The Mark-BW15 design for the Westinghouse reactor systems was developed as an upgrade for the existing stainless steel clad rod design. The design utilizes the same structural cage while providing significant uranium utilization advantages over the existing stainless steel clad design. The base fuel rod design will be changed to a design similar to the Mark-B8 with a solid, long lower end plug extending into the bottom spacer grid. This design change will be implemented in 1991. The major design parameters are listed in Table 2-2.

Lead test assemblies (LTAs) completed their third cycle of irradiation in cycle 15 of Connecticut Yankee in September 1989. Nondestructive examinations confirmed the performance of the assemblies through three cycles of operation. An Echo 330 examination showed that no leaking fuel rods were present in the LTAs even though debris had damaged many of the stainless clad fuel assemblies in cycle 15. After the second cycle, the upper end fittings (UEFs) on the four LTAs were replaced to allow for more fuel rod growth. Growth measurements after three cycles showed a fuel rod growth margin of .45 inches. The results of the third cycle poolside examination verified the design assumptions for the full batch design. Full batch implementation will start with cycle 17 in 1991.

2.2. Fuel Performance Programs

A summary listing of the major fuel performance programs underway in 1989 is contained in Table 2-3. The objectives and status (12/89) of these programs are presented in the following paragraphs.

2.2.1. DOE/Duke/AP&L/BWFC/ Extended-Burnup Programs (Oconee 1 and ANO-1)

The Extended-Burnup Programs, which began in 1978, are joint efforts among the DOE, Duke Power Company (Duke), Arkansas Power & Light (AP&L), and BWFC to achieve improvements in the nuclear fuel cycle by extending the useful lifetime of light water reactor (LWR) fuel assemblies, thus, realizing the benefits of reduced spent fuel generation and fuel cycle costs. The DOE/Duke/BWFC Program focused on qualifying the early design (1973) 15x15 assembly for higher burnup (~40 GWd/mtU) and identifying fuel life-limiting phenomena. Included in this high-burnup qualification program was the collection of fuel performance data to burnups of 50 GWd/mtU. An additional phase within the DOE/Duke/BWFC Program used state-of-the-art extended burnup technology to develop a fuel design (15x15) that incorporated a urania-gadolinia (fuel-burnable absorber) mixture. The DOE/AP&L/BWFC Program applied the knowledge gained from the former program to design, develop, and irradiate an advanced 15x15 assembly to a burnup of 57.3 GWd/mtU.

In the DOE/Duke/BWFC Program, five circa 1973-design 15x15 assemblies were extensively characterized on-site after their third cycle of irradiation (~31 GWd/mtU burnup) to obtain baseline fuel performance data.[2] The five assemblies were reinserted for a fourth cycle, achieving assembly average burnups of about 40 GWd/mtU. On-site nondestructive examinations of these high-burnup assemblies were completed in 1980.[3] Hot cell examinations of rods from a three-cycle sibling assembly and one of the four-cycle assemblies verified excellent fuel performance to burnups of 40 GWd/mtU and did not reveal any performance phenomena that would preclude higher burnups.[4]

One of the 40 GWd/mtU assemblies underwent a fifth cycle of irradiation and achieved a cumulative burnup of 50.2 GWd/mtU. Extensive on-site examinations were completed on this assembly in 1985[5], and the hot cell examination on 16 fuel rods from the high burnup assembly was completed in 1986.[6] These examinations showed the fuel performed excellently through five cycles of irradiation. Coupled with BWFC's other fuel surveillance programs, the performance data obtained from these examinations have provided a substantial data base for extending fuel burnup in PWRs. Other reports issued under the high burnup qualification phase of the DOE/Duke/BWFC Program are References 7 through 15.

The urania-gadolinia fuel phase of the DOE/Duke/BWFC Program was established to provide the technology to design, verify, and license a UO_2-Gd_2O_3 extended-burnup core. Urania-gadolinia is being developed as a fuel-burnable absorber mixture for PWR application to control reactivity and power peaking in the more highly enriched extended-burnup fuel. Urania-gadolinia should eliminate the need for separate burnable absorber rods and increase the flexibility of fuel management plans. Five urania-gadolinia LTAs fabricated in this program incorporated state-of-the-art fuel performance and fuel utilization features. The design features include urania-gadolinia fuel, annular fuel pellets, annealed guide tubes, Zircaloy-4 intermediate spacer grids, and a removable upper end fitting. These five assemblies completed their first cycle of irradiation in 1984. The poolside examination showed these LTAs to be in excellent condition.[27] Seventeen fuel rods from one LTA were removed and shipped to the Lynchburg Research Center for hot cell examinations. The hot cell examination scope of work has been completed and the results are to be published. The parameters measured and fuel/cladding characteristics examined displayed the expected trends and were consistent with the data base for Mark B fuel. The remaining four LTAs were reinserted for their second and third cycles of irradiation in Oconee 1 cycles 9 and 10, which were completed in 1986, and 1987, respectively. On-site examinations were conducted on these four LTAs after both cycles, and their performance characteristics were excellent for this burnup of 47.6 GWd/mtU.[28] One LTA was reinserted for a fourth cycle. This assembly achieved a burnup of 58.3 GWd/mtU. The urania-gadolinia program is scheduled for completion in 1990. Eleven semi-annual progress reports for this phase of the DOE/Duke/BWFC Program have been published.[16-26]

In the DOE/AP&L/BWFC Program, one of four 15x15 LTAs designed for high burnup (Mark BEB; Mark B extended burnup) reached a burnup of 57.3 GWd/mtU in August 1988. These LTAs incorporated design changes which include: (1) increased fuel rod plenum volume, (2) decreased fuel rod initial fill-gas pressure, (3) thicker fuel rod cladding, (4) fully annealed Zircaloy-4 guide tubes, and (5) several fuel rods containing annular fuel pellets. Also incorporated in these extended-burnup LTAs are several segmented fuel rods to provide high-burnup fuel that could be used in subsequent test reactor experiments. The four Mark BEB LTAs completed their third cycle of irradiation in September 1986. The Mark BEB assemblies average burnup for their first, second, and third cycles was 18, 33, and 47 GWd/mtU, respectively. On-site examinations of the LTAs after these cycles verified excellent fuel performance. Hot cell examinations of fuel rods from one, three cycle Mark BEB assembly were completed in December 1989. Results are consistent with those from previous BWFC efforts. Reports issued under the DOE/AP&L/BWFC Program are References 29 through 40, and the use of LTAs in ANO-1 is addressed in the applicable reload reports. AP&L and BWFC elected to reinsert one Mark BEB assembly for a fourth cycle of irradiation. This assembly was placed in ANO-1 cycle 8, which began in December 1986.

2.2.2. DOE/SMUD/BWFC Axial Blanket Fuel Design and Development Program (Rancho Seco)

The Axial Blanket Program was undertaken in 1979 jointly by the DOE, Sacramento Municipal Utility District (SMUD), and BWFC.[41-49] The objective of this program was to demonstrate reduced neutron axial leakage, thereby yielding better neutron economy and uranium savings, without affecting either the reactor operating capabilities or existing fuel hardware. The design involved replacing approximately six inches of enriched uranium with natural uranium in the neutronically less important regions at the top and bottom of the fuel column.

In this program, four 15x15 axial blanket LTAs were designed, fabricated, and inserted in cycle 5 of Rancho Seco. Gadolinium movable detectors were installed in selected core locations (one for blanketed fuel and the other for non-blanketed fuel) to monitor axial power shapes during reactor operation to confirm the nuclear analytical models. In their first and second cycles of irradiation, the four LTAs achieved burnups of approximately 12 and 20 GWd/mtU, respectively. With the shutdown of Rancho Seco, the LTAs completed irradiation with a burnup of 29.2 GWd/mtU.

The first batch of forty axial blanket assemblies were irradiated in Rancho Seco cycle 6 and at the end of the cycle in March 1985, had achieved a burnup of 14.4 GWd/mtU.[46] For Rancho Seco cycle 7, a feed batch of 56 fresh axial blanket assemblies was implemented. The Rancho Seco cycle 7 core contained 100 axial blanket assemblies. At shutdown, the first and second batches had a burnup of 20.5 and 10.6 GWd/mtU, respectively. The axial blanket assemblies performed as expected.

2.2.3. BWFC/Duke Low Absorption Grid Program (Oconee 1 & 2)

The LTA program for the low absorption grids of Zircaloy-4 for 15x15 fuel assemblies (Mark-BZ) was completed in 1986. Nondestructive examinations of the four Mark-BZ assemblies irradiated in Oconee 1 cycles 7, 8, and 9 revealed that these assemblies performed as expected. Reports relating to this program are given in References 49 through 52. Full batch implementation began in 1984. As of December 31, 1989 a total of 1043 fuel assemblies with Zircaloy-4 grids have been irradiated, with a maximum assembly burnup of 58.3 GWd/mtU.

2.2.4. BWFC/Duke Advanced Cladding Pathfinder Program (Oconee 2)

An advanced-cladding test fuel assembly, "Pathfinder," began irradiation in Oconee 2 in 1983. The assembly contains 12 fuel rods with advanced-design cladding - six have a 2 mil-thick liner of pure zirconium on the cladding inside surface; six are fabricated from beta-quenched, Zircaloy-4 tubing. BWFC expected the advanced fuel cladding to be beneficial both in situations requiring extensive load following and for extremely long-life designs. In addition, "Pathfinder" had a removable end fitting to allow the test rods to be removed either for examination or further performance investigations. The Pathfinder completed its first cycle in 1985, its second cycle in August 1986, and its third cycle in February 1988. Poolside examinations at the end of the first and second cycles are completed and showed that this fuel assembly had performed as expected.[55] Ultrasonic (ECHO 330) examination of the assembly after three cycles showed that no leakers were present. However, the poolside examination after three cycles showed that the beta quenched cladding had higher than expected oxidation.[56] This finding coupled with results from similar projects have led to abandoning further evaluation of beta quench cladding application in PWRs.

Table 2-1. Typical BWFC Fuel Assembly Parameters
(B&W Reactor System)

Fuel Rod Array	15X15	17X17
Cladding Material	Zirc-4	Zirc-4
Reactor Type	PWR	PWR
Assemblies per Core	177	205
Fuel Rods per Assembly	208	264
Empty Locations per Assembly	17	25
Rod Pitch, mm (inch)	14.4 (0.568)	12.8 (0.502)
System Pressure, MPa (psia)	15.2 (2200)	15.5 (2250)
Core Average Power Density, kW/liter	91.4	107.3
Average LHGR, W/cm (kW/ft)	203 (6.20)	188 (5.73)
Axial Peak LHGR of Avg. Rod, W/cm (kW/ft)	244 (7.44)	226 (6.88)
Max. Peak LHGR, W/cm (kW/ft)	530 (16.16)	499 (15.20)
Max. Fuel Temp., °C (°F)	2340 (4244)	2290 (4155)
Fuel Rod Length, cm (inch)	390.4 (153.7)	387.8 (152.7)
Active Fuel Height, cm (inch)	360.2 (141.8)	363.2 (143.0)

Table 2-1. (Continued)

Fuel Rod Array	15x15	17x17
Plenum Length, cm (inch)	29.8 (11.7)	24.2 (9.5)
Fuel Rod O.D., mm (inch)	10.92 (0.430)	9.63 (0.379)
Cladding I.D., mm (inch)	9.58 (0.377)	8.41 (0.331)
Cladding Thickness, mm (inch)	0.673 (0.0265)	0.610 (0.024)
Diametral Gap, microns (mils)	213.4 (8.4)	198.1 (7.8)
Fuel Pellet O.D., mm (inch)	9.362 (0.3686)	8.209 (0.3232)
Fuel Pellet Length, mm (inch)	11.05 (0.435)	9.53 (0.375)
Fuel Pellet Density, % TD	95	95

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Table 2-2. Typical BWFC Fuel Assembly Parameters
(Westinghouse Reactor System)

Fuel Rod Array	17x17	15X15	15X15
Cladding Material	Zirc-4	304/SS	Zirc-4
Reactor Type	PWR	PWR	PWR
Assemblies per Core	193 (157)	157	157
Fuel Rods per Assembly	264	204	204
Empty Locations per Assembly	25	21	21
Rod Pitch mm (inch)	12.6 (0.496)	14.3 (0.563)	14.3 (0.563)
System Pressure, MPa (psia)	15.5 (2250)	13.9 (2015)	13.9 (2015)
Core Average Power Density, kW/liter	82.25	82.25	82.25
Average LHGR, W/cm (kW/ft)	178 (5.43)	181 (5.53)	184 (5.60)
Axial Peak LHGR of Avg. Rod, W/cm (kW/ft)	276 (8.42)	251 (7.66)	255 (7.76)
Max. Peak LHGR, W/cm (kW/ft)	427 (13.0)	476 (14.5)	476 (14.5)
Max. Fuel Temp., °C (°F)	1927 (3500)	2149 (3900)	2149 (3900)
Fuel Rod Length, cm (inch)	384.8 (151.5)	321.8 (126.7)	319.7 (125.9)
Active Fuel Height, cm (inch)	365.8 (144.0)	306.1 (120.5)	301.2 (118.6)

Table 2-2. (Continued)

Fuel Rod Array	17x17	15X15	15X15
Plenum Length, cm (inch)	16.4 (6.4)	12.2 (4.8)	15.9 (6.3)
Fuel Rod O.D., mm (inch)	9.50 (0.374)	10.72 (0.422)	10.72 (0.422)
Cladding I.D., mm (inch)	8.28 (0.326)	9.88 (0.389)	9.35 (0.368)
Cladding, Thickness, mm (inch)	0.610 (0.024)	0.419 (0.0165)	0.686 (0.027)
Diametral Gap, microns (mils)	165 (6.5)	165 (6.5)	178 (7.0)
Fuel Pellet O.D., mm (inch)	8.115 (0.3195)	9.715 (0.3825)	9.17 (0.361)
Fuel Pellet Length, mm (inch)	10.16 0.400)	11.63 (0.458)	10.80 (0.425)
Fuel Pellet Density, % TD	96/95 [a]	95	95

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[a] Design may use either density.

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Table 2-3. Major Fuel Performance Programs

(Status As of December 31, 1989)

Vendor	Fuel Type	Plant	Planned Number of Operating Cycles	Scheduled Completion *	Interim Inspections To Date
BWFC	15 X 15	Oconee-1	5	Completed	3
	15 X 15[a]	ANO-1	4	Completed	3
	15 X 15[b]	Rancho Seco	3	Completed	2
	15 X 15[c]	Oconee-2	4	Completed	4
	15 X 15[d]	Oconee-2	3	Completed	3
	15 X 15[e]	Oconee-2	1	Completed	1
	15 X 15(e)	Oconee-1	3	Completed	3
	17 X 17[f]	Oconee-2	3	Completed	3
	15 X 15[g]	Oconee-1	4	1990	3
	15 X 15[h]	Oconee-2	3	Completed	2
	15 X 15[i]	ANO-1	4	Completed	3

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* Completion of irradiation

(a) LTAs of an advanced, extended-burnup design.

(b) Current-design assemblies containing axially-blanketed fuel columns.

(c) Current-design assemblies with special Zircaloy cladding materials and EPRI creep collapse specimen clusters.

(d) Current-design assemblies with lifted rods and cladding having a known spiral eccentricity in wall thickness.

(e) Current-design assemblies utilizing low absorption spacer grid material (Zircaloy-4).

(f) Two of these four LTAs are reconstitutable.

(g) Gadolinia LTAs of an advanced, extended-burnup design.

(h) Pathfinder LTA with advanced Zircaloy cladding materials.

(i) Same as (a); Additional cycle of irradiation.

Table 2-4 Irradiation Programs for Replacement Fuel Assemblies for Westinghouse Reactors

(Status As of December 31, 1989)

Vendor	Fuel Type	Plant	Planned Number of Operating Cycles	Scheduled Completion *	Interim Inspections To Date
BWFC	15x15[a]	Haddam Neck	3	Completed	3
	17x17[b]	McGuire 1	3	1991	2

* Completion of irradiation

(a) Four Zircaloy-4 clad fuel assembly LTA to replace stainless steel clad fuel assemblies.

(b) Four 17x17 Lead Assemblies (Mark-BW LA)

3. SUMMARY OF PERFORMANCE FOR BWFC-DESIGNED FUEL

BWFC's commercial experience with Mark-B generation nuclear fuel began in April 1973 with initial criticality of Duke Power Company's Oconee Unit 1. During 1989, BWFC-designed fuel was irradiated in eight B&W-designed reactors, and in two Westinghouse-designed reactors. Connecticut Yankee reload fuel is discussed and tabulated separately because of its stainless-steel cladding design. The performance information presented in this section encompasses the period of April 1973 through December 1989.

The fuel burnup status at the end of 1989 is shown in Table 3-1. A summary of burnup experience through 1989 is given in Tables 3-2 and 3-3. The Zircaloy fuel assemblies irradiated are of the BWFC 15X15 Mark B design with the exception of four 17X17 LTAs (Mark C), four Mark-BW lead assemblies (LAs) and four Westinghouse 15X15 Zircaloy Clad LTAs. As the Mark B fuel design has achieved maturity, batch average burnups have increased from 27.0 to 37.0 GWd/mtU with 454 fuel assemblies being discharged with burnups of greater than 36 GWd/mtU. The peak burnup of a discharged fuel assembly in 1989 is 58.3 GWd/mtU.

The performance of BWFC-supplied fuel is summarized in Table 3-4. Over the past seventeen years, B&W-designed reactors have produced an electrical output in excess of 465 million MW-hours, and 1,087,440 fuel rods have been irradiated. An excellent fuel performance record has been maintained with fuel rods that have been subjected to rigorous fuel duty cycles. In 1989 a total of eight leaking* fuel rods were generated out of 257,712 Zircaloy clad rods irradiated. Due to extensive debris damage, a total of 450 leaking fuel rods were generated out of 32,028 stainless steel clad rods irradiated. These estimates are based on coolant chemistry projections and ultrasonic and visual inspection results. This performance represents a fuel integrity level of 99.997 % for operation of Zircaloy fuel rods in 1989.

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* "Leaking" refers to a fuel rod that is releasing fission products to the primary coolant through a breach in its cladding.

Since 1980, the average radioisotopic iodine activity levels in B&W-designed reactors has generally been decreasing. Improved fuel performance coupled with earlier removal of leaking fuel rods has resulted in lower steady state activities. The improved fuel performance is attibuted to better designs and manufacturing methods. The wide use of UT inspection for leaking fuel rods and reconstitutable fuel assemblies in the late 1980's has resulted in most leaking rods being removed from reactor. Typically, leaking rods now will be discharged within one cycle after they are generated. As a combination of both trends, the total number of leaking fuel rods in core is less. The resulting reduction in average coolant iodine can be seen in Table 3-5. Future coolant activities are expected to be significantly lower due to the elimination of debris fretting and spacer grid fretting leaking fuel rods.

All stainless steel fuel rods are irradiated in the Haddam Neck (Connecticut Yankee) reactor. Due to a debris problem, a large number of leaking fuel rods were generated from debris fretting. The fuel performance for Connecticut Yankee was determined from ultrasonic examination and visual inspection during an extensive reconstitution effort. The number of leaking fuel rods is estimated at 450.

The number of leaking rods identified in Table 3-4 were estimated from equilibrium coolant radio-iodine levels during full-power operation using the method described in Reference 54 or by ultrasonic inspection. Because of uncertainties associated with the location and nature of probable leakers, the number of leaker rods shown represents a best estimate of the fuel integrity status.

The ultrasonic inspection for leaking fuel rods has been widely practiced in the last several years. The BWFC Fuel Company ECHO-330 system provides this type of inspection to the utilities. The method utilizes a Lamb wave to detect the presence of water in the fuel-to-clad gap in individual fuel rods. This method represents a major improvement in detection of leaking fuel rods as it permits a more precise determination of the number of leakers.

To date, nine ultrasonic inspections have been performed at five B&W-designed reactor sites (Arkansas Nuclear One, Unit 1; Oconee 1; Oconee 2; Oconee 3 and Three Mile Island Unit 1), and Connecticut Yankee. The ultrasonic data have revealed that a large uncertainty exists in radiochemistry projections. Ongoing investigations that have resulted from inspections as well as other fuel performance investigations are covered in section 4.2.

Compare with Table 7 (in 1988)

Table 3-1. Operating Status of BWFC-Fueled Reactors
(December 31, 1989)

Reactor	Reactor Cycle	Maximum Assembly Burnup, MWd/mtU: Incore	Maximum Assembly Burnup, MWd/mtU: Discharged to Date
Oconee-1	12	40,595	58,310
Oconee-2	11	34,646	42,820
Oconee-3	12	35,594	42,740
TMI-1	7	33,966	33,863
ANO-1	9	34,972	57,318
Rancho Seco	7	0	38,268
Crystal River-3	7	38,793	35,350
Davis-Besse 1	6	33,690	40,300
McGuire-1[a]	7	27,700	NA
Connecticut Yankee[b]	16	0	36,000 ? less than in 1988?

[a] Westinghouse-designed reactor with four Mark-BW LA's.
[b] In refueling and undergoing fuel assembly reconstitution.

Use as is
Table 3-2 (89)

Table 3-2. Summary of Burnup Experience for BWFC-Supplied Zircaloy Clad Fuel[(a)]
(December 31, 1989)

Fuel Assembly Average Burnup MWd/mtU	Assemblies Incore on Dec. 31, 1989		Assemblies Discharged in 1989		Assemblies Discharged Through Dec. 31, 1989	
	No. of Assy's	No. of Rods	No. of Assy's	No. of Rods	No. of Assy's	No. of Rods
0 to 4,000	52	10,816	0	0	0	0
4,000 to 8,000	104	21,632	0	0	4	832
8,000 to 12,000	44	9,152	56	11,648	159	33,072
12,000 to 16,000	120	24,960	0	0	134	27,872
16,000 to 20,000	224	46,592	32	6,656	192	39,936
20,000 to 24,000	153	31,824	33	6,864	330	68,640
24,000 to 28,000	189	39,312[(d)]	16	3,328	1154[(b)]	240,144
28,000 to 32,000	201	41,808	46	9,568	1057	219,856
32,000 to 36,000	103	21,424	40	8,320	553[(c)]	115,136
36,000 to 40,000	44	9,152	72	14,976	312	64,596
40,000 to 44,000	5	1,040	41	8,528	80	16,648
44,000 to 48,000	0	0	0	0	10	2080
48,000 to 52,000	0	0	0	0	1	208
52,000 to 56,000	0	0	1	208	1	208
56,000 to 60,000	0	0	1	208	2	416
	1,239	257,712	338	70,304	3,989	829,728

(a) Three Mile Island Unit 2 is excluded from this tabulation.

(b) Includes two non-reconstitutable, 17X17 LTA's (Mark C).

(c) Includes two reconstitutable, 17X17 LTA's (Mark CR).

(d) Includes four, 17x17 LAs (Mark-BW).

Table 3-3. Summary of Burnup Experience for BWFC-Supplied Stainless Steel Clad Fuel

(December 31, 1989)

Fuel Assembly Average Burnup (MWd/mtU)	Assemblies Incore [(a)] on Dec. 31, 1989		Assemblies Discharged in 1989		Assemblies Discharged Through Dec. 31, 1989	
	No. of Assy's	No. of Rods	No. of Assy's	No. of Rods	No. of Assy's	No. of Rods
0 to 4,000	0	0	0	0	0	0
4,000 to 8,000	0	0	16	3,264	16	3,264
8,000 to 12,000	0	0	20	4,080	22	4,488
12,000 to 16,000	0	0	20	4,080	66	13,464
16,000 to 20,000	0	0	0	0	12	2,448
20,000 to 24,000	0	0	32	6,528	32	6,528
24,000 to 28,000	0	0	21	4,284	56	11,424
28,000 to 32,000	0	0	10	2,040	58	11,832
32,000 to 36,000	0	0	38	7,752 [(b)]	326	66,504
36,000 to 40,000	0	0	46	0	94	19,176
40,000 to 44,000	0	0	0	0	0	0
	0	0	157	32,028	682	139,128

(a) No fuel assemblies incore as of December 31, 1989. All fuel assemblies are offloaded for debris cleaning, inspection and reconstitution.

(b) Includes four Lead Test Assemblies with Zircaloy clad fuel rods.

Use as is - Table 6 (89)

Table 3-4. 1989 Performance Summary for BWFC-Supplied Fuel Rods[a]

(December 31, 1989)

Fuel Rod Type	Stainless Steel	15X15	* (Mark C) 17X17	* (Mark-BW) 17X17
1. Cumulative Number of Rods Irradiated Through Dec. 1989	107,100	1,055,216	1056	1056
a. Maximum Rod-Average Burnup, GWd/mtU	39.2	60.8	36.4	15.5
b. Mean Rod-Average Burnup, GWd/mtU	27.8	27.7	30.1	15.3
2. Total Number of Rods Irradiated in 1989	32,028	257,712		1056
3. Number of Irradiated Rods Incore on Dec. 31, 1989	--	257,712		1056
a. Maximum Rod-Average Burnup, GWd/mtU	--	40.6		15.5
b. Mean Rod-Average Burnup, GWd/mtU	--	21.6		15.3
4. Number of Rods Discharged in 1989	32,028	70,304		
a. Maximum Rod-Average Burnup, GWd/mtU	39.7	60.8		
b. Mean Rod-Average Burnup, GWd/mtU	26.0	25.5		
5. Estimated Number of Leaker Rods Generated in 1989	450(b)	8(c)		

(a) Three Mile Island Unit 2 is excluded from this tabulation.

(b) Based on a combination of ultrasonic inspection and visual inspection during reconstitution. All failures examined had debris wear on cladding near bottom end cap.

(c) Estimated from equilibrium coolant radio-iodine behavior during full-power operation, or UT examination of fuel assemblies.

[illegible]

Table 3-5

Average Steady State Coolant Iodine Activity For B&W Designed Plants

Date	I-131 Activity, uci/gm.
1980	.086
1981	.046
1982	.031
1983	.041
1984	.051
1985	.031
1986	.014
1987	.028
1988	.035
1989	.023

4. PROBLEM AREAS OBSERVED DURING 1989

4.1 Fuel Assembly Holddown Spring Failure

In 1989, 19 broken holddown springs were found at two reactors. The first reactor was Oconee 2 following cycle 10 where eight broken holddown springs were found. The second reactor was Oconee 3 following cycle 11 where eleven broken holddown springs were found. All nineteen broken springs were in fuel assemblies to be reinserted and were replaced.

Although the Mark B design started irradiation in 1973, the first broken holddown springs were not found until the first refueling outage at Davis Besse in May, 1980. At which time, a total of 20 broken holddown springs were found at Davis Besse.

Subsequent inspection in 1980 of 1581 fuel assemblies in-core or in the spent fuel pools at all B&W sites found 26 broken holddown springs. Most of these failures (24) were traced to two heats of material (Inconel X-750) having an anomalous material condition characterized by a casing of coarse grains at the wire surface. Continuing inspection has found broken springs at various plants.

When the first broken holddown springs were found, the potential problems of reactor operation with broken springs were evaluated. The evaluation examined the problem of loose parts, control rod interference, and lifted assemblies. It was determined that broken holddown springs presented no safety concern for continued reactor operation.

Several design changes were made to the holddown spring in a effort to prevent broken springs. The wire diameter was increased and the alloy changed from Inconel X-750 to X-718. Additional process changes were made in the manufacturing of the holddown springs. Further efforts to prevent broken holddown springs are ongoing.

4.2 Fuel Performance

During 1989 a total of eight leaking fuel rods were generated in Zircaloy-4 clad rods. This was out of a total of 257,712 fuel rods irradiated, resulting in an overall fuel integrity level of 99.997%. In only two plants were more than one leaking fuel rod generated. No plant generated more than 3 leaking fuel rods. Investigations into those events have not identified the cause. Poolside examination of leaking fuel rods from previous years shows debris in the core and spacer grid fretting as being the primary causes of leaking fuel.

Stainless Steel Clad Fuel

Connecticut Yankee, Cycle 15

Extensive debris damage occurred to the fuel in Connecticut Yankee cycle 15. Due to the nature of the defects, the extent of the number of leaking fuel rods was not revealed until the core was inspected. During the cycle 14 to 15 refueling the entire core was examined by UT for leaking fuel rods. A total of nine fuel rods were labeled as leaking based on the results of the inspection. Two of these nine leakers were reinserted into the core for Cycle 15.

On the startup of cycle 15 the concentration of I-131 rose with reactor power to approximately .01 uci/ml at 100% full power. With further operation at 100% full power, the I-131 concentration rose slowly to .02 to .03 uci/ml. This increase indicated that additional leaking fuel rods were generated.

The activity levels when corrected for uranium contamination on the cladding indicated that nine leaking fuel rods were present in Cycle 15. Activity remained steady through the remainder of cycle 15. At the end of cycle 15 the reactor was shutdown. A normal shutdown spike was seen. When the system pressure was reduced a very large spike of activity occurred. After the core was off-loaded, fuel assemblies were visually inspected and examined by UT systems for leakers. Approximately 450 leaking fuel rods were found. During reconstitution, all indicated leaking rods and selected adjacent rods in reinsert fuel assemblies were examined. Debris wear marks in the lower end caps and adjacent cladding were seen. Debris generated from repair work during the cycle 14 to 15 refueling had become trapped by the bottom spacer grid. The debris fretted against the cladding, wearing through-wall holes in many cases.

Zircaloy-4 Clad Fuel

This section discusses investigations based on UT results which include leakers from Oconee 1, 2 and 3, and TMI-1.

Oconee 2

An investigation into leakers at Oconee 2 by BWFC, the B&W Owners Group (BWOG), and EPRI was instigated from the results of a June 1986 UT examination of Oconee 2 fuel assemblies. Leakers were identified on the periphery and around the instrument tubes of the fuel assemblies. Twelve rods were extracted from four fuel assemblies. These rods include five leaking rods and seven adjacent or symmetrical nonleaking rods. Four of the rods were corner rods, and the remainder were from around the instrument tube. Adjacent or symmetrical rods were examined to investigate possible incipient defects from which the cause of the leakers might be identifiable. The extracted rods were visually examined, eddy-current (EC) scanned for cladding defects, and scanned for diameter and oxide thickness profiles. The corner rod leakers appeared to have been caused by debris or mechanical damage. Poolside data were insufficient to determine a cause of the leaking rods around instrument tube locations.[57] To aid in identifying possible failure mechanisms, four of the nonfailed rods from around instrument tube locations were sent to a hot cell where they were destructively examined.

1989 UT Examinations

During 1989 ultrasonic inspections with the ECHO 330 system were performed at four B&W designed reactor sites. These were Oconee 1, Oconee 2 , Oconee 3 and TMI-1. A total of 904 fuel assemblies were examined. Of that number, 42 fuel assemblies with a total of 57 leaking fuel rods were identified. Of the fifty seven leaking fuel rods, a high percentage were on the fuel assembly periphery.

A selected number of fuel assemblies with the leaking rods on the periphery were examined. During a poolside examination, the peripheral rods were gripped and lifted slightly to view the spacer grid contact sites. On some of the rods examined, though-wall wear marks corresponding to the spacer grid stops were seen. It was evident that many of the peripheral leakers had occurred due to spacer grid fretting. An investigation was started to determine the factor or combination of factors which resulted in leaking fuel rods. The investigation determined that manufacturing variations contributed to the spacer grid fretting. Corrective action was taken to prevent those variations. A program is in place to follow both spacer grid manufacturing and fuel performance to prevent a similar problem from arising.

5. FUEL MANAGEMENT AND OPERATIONS IMPROVEMENTS

Utilities continue to seek improved plant operation, reduced operating costs, reduced fuel costs, reduced fuel flow into storage, and extension of plant lifetime. The role of the vendors is to develop products and services that further these objectives. The trends observed in 1989 and discussed in this section are consistent with this pattern.

5.1 Fuel Utilization

Fuel utilization improvements are derived from improvements in fuel assembly design, fuel cycle design, and burnable poison design. Many significant improvements in fuel assembly design that affect fuel utilization have already been developed and implemented. These improvements included low-absorption structural materials, axial blankets, and extended burnup capability. However, because of the need to reduce the amount of spent fuel going into storage, burnup capability is being increased from 50 to the equivalent of 60 GWd/mtU for a fuel rod. When applied to fuel cycle design, in the form of feed batch size reduction, this will contribute a further reduction in enriched uranium requirements in additional to fabrication and storage savings.

The higher burnups require higher enrichments. The trend is toward enrichments as high as 5.00 wt%. Analyses and, in some cases, physical changes are being made to support the use of the higher enrichments. Manufacturing plants, shipping containers, and storage facilities are all potentially affected.

Average assembly discharge burnup from BWFC-designed plants in 1989 was approximately 37 GWd/mtU. Current feed batch sizes will eventually result in discharge assembly burnups approaching 44 GWd/mtU. Proposed average assembly burnups for fuel assemblies to be loaded in two years are as high as 46 GWd/mtU, and for those to be loaded in four to five years, average burnups approach 50 GWd/mtU.

Fuel cycle design continues to evolve towards very-low-leakage loading schemes in which fuel assemblies are arranged in such a manner as to minimize pressure vessel fluence. Such schemes also tend to improve fuel utilization because of the reduced neutron leakage. However, very-low-leakage fuel cycles with increased enrichments and higher discharge burnups tend to have increased power peaking that reduces thermal margins. Improved burnable absorbers are being developed to compensate for this effect. In addition, fuel assembly improvements that increase thermal margin are being developed, and the technology for assessing the thermal capabilities of the assemblies is being improved.

Fuel utilization is negatively impacted by fuel failure if it results in premature discharge of fuel assemblies from the reactor. In addition to the low residual level of failures due to manufacturing defects, some of the more significant incidents of failure have been caused by debris in the primary system. To mitigate these problems, debris-resistant fuel assembly designs have been developed, quality-control during manufacturing has been further improved, and technologies for locating and replacing failed rods within an assembly have been implemented.

5.2 Cycle length

Planned cycle length continues to increase in plants fueled by BWFC. Originally designed with annual cycles, all plants converted to 18-month refueling, and some continued on to 24-month refueling. Most 18-month cycles originally produced 360 to 420 EFPD. These values have gradually increased, and now, as a result of plant availability improvement programs, at least one utility is planning for 18-month cycles of 465 EFPD, equivalent to a capacity factor including refueling outage time of 85%. Two utilities have converted to 24-month refueling. Current expectations for energy output of these cycles is in the range of 575 to 600 EFPD.

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