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Mr. Michael Tokar
Core Performance Branch
Division of System Integration
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Tokar:

The information you requested for the NRC's "Fuel Performance Annual Report" is attached. Table 1 has been changed to reflect the current B&W design. Table 2 and the writeup cover B&W's fuel performance programs that were underway in 1979. Table 3 is a summary of the B&W fuel performance for 1979.

If you have any further questions in this area contact me at the following number 804-384-5111, Ext. 2817.

Very truly yours

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II. Status of Fuel Surveillance Programs

A. B&W Fuel Designs

Typical B&W fuel assembly design parameters for the current 15x15 and 17x17 fuel rod arrays are presented in Table 1. Also shown are the former design values as contained in NUREG-0633. The current values reflect the change to 95% TD fuel, resulting in a reduction in active fuel height in the 15x15 array, and a 3800 MWt design core of the 17x17 array.

B. New Fuel Design Commitments

A new fuel design can be introduced into a power reactor in one of two ways: (1) as the initial full core, or (2) regionally as a reload batch. Before the actual implementation of new designs, lead test assemblies (LTA's) are usually introduced for initial application and verification testing. After the LTA's have demonstrated reliable operating performance, the reload batch or first core of a new design is implemented.

C. B&W Fuel Performance Programs

A large number of fuel performance programs were underway in 1979. All of the programs are directed towards the improvement of fuel utilization and the development of zero-defect fuel. Included in the majority of these programs, which are described briefly below, are both on-site nondestructive and hot-cell destructive post-irradiation examinations. The destructive examinations are conducted in B&W's hot-cell facility and also include extensive nondestructive examinations in the hot-cell. A summary of the major B&W fuel performance programs in 1979 is contained in Table 2.

1. B&W/Duke 15x15 Fuel Surveillance Program (Oconee 1)

The Oconee 1 fuel surveillance program for the 15x15 fuel assembly array was completed in 1979 with the acquisition of third-cycle destructive data. Significant operating performance information was obtained through the first three cycles of Oconee 1 operation with assembly average burnups of 26,500 MWd/mtU. The information thoroughly verified the soundness of the 15x15 design. Fuel performance results obtained after the first cycle of irradiation are summarized in references 1 and 2.

2. DOE/Duke/AP&L/B&W Fuel Utilization Improvement Program (Oconee 1 and ANO-1)

A significant milestone was accomplished in 1979 on Phase 1 of the Fuel Utilization Improvement Program, also called the Extended Burnup Program. Five current-design 15x15 assemblies completed their fourth cycle of irradiation in the Oconee 1 reactor with burnups to 40,000 MWd/mtU. This burnup is approximately 30% greater than previously achieved at Oconee 1. Visual examinations of the five assemblies revealed no abnormalities. Preliminary evaluations of the nondestructive examination results indicate that the fuel performed well.

The Fuel Utilization Improvement Program, which began in 1978, is a joint effort among the Department of Energy (DOE), Duke Power Company (Duke), Arkansas Power & Light (AP&L) and B&W to improve the efficiency of uranium utilization in light water reactors. The program is divided into two phases. Phase I involves the DOE, Duke, and B&W and has the objectives of qualifying the current-design 15x15 assembly for higher burnup (~40,000 MWd/mtU) and identifying fuel life limiting phenomena. Phase II involves the DOE, AP&L and B&W. It has as its objectives the design, development, and licensing of an advanced design 15x15 PWR assembly having a design burnup of 50,000 MWd/mtU and improved fuel utilization.

Under Phase I, five assemblies were extensively characterized after their third cycle of irradiation (~31,000 MWd/mtU burnup) in Oconee 1 cycle 4 (reference 3) to obtain baseline reference data. The five assemblies were then reinserted for a fourth cycle to achieve an expected assembly average burnup of 40,000 MWd/mtU. Included in this phase are an extensive nondestructive examination of each fuel assembly after its fourth cycle of irradiation, a destructive examination of one assembly after its fourth cycle, and a destructive examination of one three-cycle leaker assembly having a burnup of 31,000 MWd/mtU. Also proposed is a fifth cycle irradiation in Oconee 1 cycle 7 of one of the five assemblies to achieve a burnup of 50,000 MWd/mtU. The fifth cycle irradiation, if approved, will be followed by extensive nondestructive and destructive examinations. The three-cycle leaker and four-cycle destructive examinations are scheduled to begin in 1980. The proposed fifth cycle destructive examination is tentatively planned for 1983. Coupled with B&W's other fuel surveillance programs, the performance data obtained from these examinations will provide a substantial data base for extending the burnup limitation on PWR fuel assemblies and for developing an advanced-design PWR assembly. Other reports issued under this phase of the program are references 4 and 5.

The development of an advanced-design 15x15 assembly is included under Phase II of the Fuel Utilization Improvement Program.

This phase also includes an investigation of optimum fuel burnup with respect to fuel cycle costs and parametric studies of the effects of changes in fuel rod design parameters on fuel behavior. In the first stage of Phase II, four 15x15 lead test assemblies (LTA's) having a design capability of 50,000 MWd/mtU assembly average burnup are being fabricated for insertion in ANO-1 cycle 5. (Start-up is currently scheduled for 1981). These LTA's incorporate relatively short lead-time 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) annular fuel pellets. Also incorporated in these LTA's are several segmented fuel rods to provide high burnup fuel that can be used in subsequent test reactor experiments. In the second stage of Phase II, changes requiring a longer lead-time are being incorporated in the design of four additional LTA's that are scheduled for insertion in ANO-1 cycle 6. Poolside nondestructive examinations of both sets of LTA's are planned after each cycle of irradiation. Detailed destructive and nondestructive examinations in B&W's hot-cell facilities are also planned for each design upon completion of irradiation. Program completion is scheduled for 1988.

3. <u>DOE/SMUD/B&W Axial Blanket Fuel Design and Development Program</u> (Rancho Seco)

A new program undertaken in 1979 jointly by the DOE, Sacramento Municipal Utility District (SMUD), and B&W is the Axial Blanket Fuel Design and Development Program. The motivation behind this program is to reduce neutron leakage, thereby yielding an improvement in neutron economy and a resultant uranium savings.

The design concept involves the replacement of approximately six inches of enriched uranium in the neutronically unimportant regions at the top and bottom of the fuel column with a lower enrichment material, such as natural uranium or tailings. In this program, four 15x15 LTA's having axially-blanketed fuel columns will be inserted in cycle 5 of Rancho Seco (SMUD) in 1981. Batch implementation of the axial blanket concept is scheduled for cycle 6, continuing through cycle 8 until a full core is in place. Two gadolinia traveling detectors will be installed (one in blanketed fuel and the other in non-blanketed fuel) and used to establish comparative axial power shapes during reactor operation. Program completion is scheduled for 1986.

4. <u>EPRI/Duke/B&W Cooperative Program on PWR Fuel Rod Performance</u> (Oconee 2)

The final stage of the EPRI, Duke Power Company, and B&W Cooperative Program on Fuel Rod Performance (RP-711-1) neared completion with the fourth cycle of irradiation of the last creep collapse specimen (CCS) cluster in the Oconee 2 reactor. This program was initiated in 1972 to investigate various fuel and cladding design variables that influence pellet-cladding mechanical interactions and in-reactor performance. The program emphasis was redirected in 1976 to characterize the long term in-reactor irradiation creep, irradiation growth, and post-irradiation mechanical properties of zircaloy-4 cladding. Four different types of zircaloy cladding, each having significantly different physical and mechanical properties, were included in the program which is divided into creep collapse testing(7) and PWR demonstration(6) phases.

In the creep collapse testing phase, four full-length CCS clusters of 16 rods (unfueled) each were inserted in Oconee 2 cycle 1 and irradiated in the guide tube positions of four peripheral fuel assemblies in symmetric core locations. One CCS cluster is to be discharged after each cycle of irradiation and examined destructively. The results through three cycles are documented in references 8, 9, 10, 11, and 12. This phase of the program is scheduled for completion in 1980.

The second phase consists of a demonstration irradiation of fueled cladding using the four types of zircaloy-4 cladding material in combination with two different types of fuel pellets. Fifty-six demonstration fuel rods were inserted in core 1 of Oconee 2 and were located in peripheral positions of two fuel assemblies in symmetric core locations. The irradiation program for the fueled rods was completed after two cycles of irradiation, achieving an assembly average burnup of 24,500 MWd/mtU. The nondestructive poolside results have been reported in references 8 and 10. The destructive results are currently being compiled for publication.

5. B&W/Duke Fuel Rod Bow Program (Oconee 2)

Four specially constructed and extensively characterized 15x15 fuel assemblies and two characterized standard 15x15 assemblies for comparison are undergoing irradiation in the Oconee-2 reactor to examine the effects of lifted (non-seated) fuel rods and the pitch of cladding eccentricity on fuel rod bow. The lifted fuel rods are held in place by the intermediate spacer grids so that the rod bottoms are above the lower support plate of the lower end fitting. Irradiation of the two lifted rod assemblies began in cycle 2 of Oconee 2. The assemblies are currently undergoing their third and final cycle of irradiation. The cladding of the fuel rods used in the spiral eccentricity investigation was manufactured so that the pitch of the spiral eccentricity in wall thickness was greater than 60 inches. Irradiation of the two spiral eccentricity assemblies began in cycle 3 of Oconee 2 and are undergoing their second cycle of irradiation. The results of the nondestructive examinations conducted after each cycle of irradiation indicate no significant difference from the standard 15x15 assemblies in either rod growth or rod bow. The differences, if they arise, are expected to occur late in life during the third cycle of irradiation. Program completion is scheduled for 1981.

6. <u>B&W/Duke Low Absorption Grid Program</u> (Oconee 2)

In addition to the DOE contracted Fuel Utilization Improvement Program, B&W has an in-house fuel utilization improvement program to incorporate low absorption grids on the 15x15 fuel assembly. The Low Absorption Grid Program consists of replacing the current intermediate spacer grid material, Inconel, with zircaloy to reduce parasitic neutron absorption. One demonstration assembly was inserted in cycle 5 of Oconee 2 in 1980 and will be examined after each of the planned three cycles of irradiation. Program completion is scheduled for 1983.

7. <u>B&W/Duke 17x17 LTA Demonstration Program</u> (Oconee 2)

The B&W 17x17 fuel assembly design was introduced in 1976 with the insertion of two 17x17 LTA's in Oconee 2 cycle 2. Two additional 17x17 LTA's having reconstitutable lower end fittings began their irradiation in cycle 3 of Oconee 2. The first two LTA's are completing their third and final cycle of irradiation, while the other two are in their second cycle. Nondestructive examinations conducted after each cycle indicate that the assemblies are performing excellently. Program completion is scheduled for 1981.

REFERENCES:

1. Post-Irradiation Examination of Oconee 1 Fuel, Cycle 1 Nondestructive Test Phase, BAW-1519, Babcock & Wilcox, Lynchburg, Virginia, June 1979.

2. Post-Irradiation Examination of Oconee 1 Fuel, Cycle 1 Destructive Test Phase, BAW-1535, Babcock & Wilcox, Lynchburg, Virginia, July 1979.

3. Pyecha, T. D., H. H. Davis, J. T. Mayer, B. A. Guthrie, III, and J. G. Larson, Nondestructive Examination of Oconee 1 Fuel Assemblies after Three Cycles of Irradiation, COO-4711-20 (BAW-1553), Babcock & Wilcox, Lynchburg, Virginia, September 1979.

4. Oconee 1, cycle 5 Design Report, BAW-1520, Babcock & Wilcox, Lynchburg, Virginia, May 1979.

5. Coleman, T. A., et al, Qualification of the B&W Mark B Fuel Assembly for High Burnup - First Semi-Annual Progress Report, July-December 1978, BAW-1546-1, Babcock & Wilcox, Lynchburg, Virginia, August 1979.

6. T. P. Papazoglou, Pre-Irradiation Characterization of Test Specimens in the PWR Demonstration Irradiation Program, LRC-4733-1 (NP), Babcock & Wilcox, Lynchburg, Virginia, November 1975.

7. R. J. Beauregard, Pre-Irradiation Characterization of Test Specimens in the Creep Collapse Program, LRC-4733-2, Babcock & Wilcox, Lynchburg, Virginia, January 1977.

8. H. H. Davis, T. P. Papazoglou, and L. J. Ferrell, Poolside Examination of PWR Demonstration Fuel Assemblies and Creep Specimens-End of Cycle 1, LRC-4733-3, Babcock & Wilcox, Lynchburg, Virginia, May 1977.

9. R. J. Beauregard, T. P. Papazoglou, and L. J. Ferrell, Hot Cell Examination of Creep Collapse and Irradiation Growth Specimens-End of Cycle 1, LRC-4733-4, Babcock & Wilcox, Lynchburg, Virginia, July 1977.

10. H. H. Davis, T. P. Papazoglou, and L. J. Ferrell, Poolside Examination of PWR Demonstration Fuel Assemblies and Creep Specimens-End of Cycle 2, LRC-4733-5, Babcock & Wilcox, Lynchburg, Virginia, August 1978.

11. G. M. Bain and T. P. Papazoglou, Hot Cell Examination of Creep Collapse and Irradiation Growth Specimens-End of Cycle 2, LRC-4733-7, Babcock & Wilcox, Lynchburg, Virginia, May 1979.

12. W. A. Pavinich and T. P. Papazoglou, Hot Cell Examination of Creep Collapse and Irradiation Growth Specimens, End of Cycle 3, LRC-4733-8, Babcock & Wilcox, Lynchburg, Virginia, March 1980.

TABLE 1: Typical Fuel Assembly Parameters

Fuel Rod Array	15x15		17x17	
Reactor Type	PWR		PWR	
Assemblies per Core	177		205	
Fuel Rod Locations per Assembly	225		289	
Fuel Rods per Assembly	208		264	
Empty Locations per Assembly	17		25	
	Former Values	Current Values	Former Values	Current Values
Rod Pitch, MM (inch)	14.4 (0.568)	14.4 (0.568)	12.7 (0.501)	12.7 (0.501)
System Pressure, MPa (PSIA)	15.2 (2200)	15.2 (2200)	15.5 (2250)	15.5 (2250)
Core Average Power Density, kw/liter	90.0	91.4	101.6	107.2
Average LHGR, kw/m (kw/ft)	20.0 (6.105)	20.3 (6.20)	17.8 (5.43)	18.8 (5.73)
Axial Peak in kw/m an Average Rod, (kw/ft)	24.00 (7.33)	24.41 (7.44)	21.36 (6.52)	22.57 (6.88)
Max. Peak kw/m LHGR, (kw/ft)	62.4 (19.03)	53.0 (16.16)	48.4 (14.74)	49.9 (15.20)
Max. Fuel ^oC Temperature, (^oF)	2300 (4170)	2340 (4245)	2020 (3670)	2290 (4155)
Core Average Enrichment, wt% U-235	2.57	2.57	2.96	2.67
Max. Local Exposure, MWD/KG-Metal	55	55	55	55
Cladding Material	Zirc-4	Zirc-4	Zirc-4	Zirc-4
Fuel Rod Length, M (inch)	3.89 (153.13)	3.904 (153.688)	3.86 (152.13)	3.864 (152.125)
Active Fuel M Height, (inch)	3.66 (144)	3.602 (141.8)	3.63 (143)	3.632 (143)

TABLE 1: (Cont'd)

	Former Values	Current Values	Former Values	Current Values
Plenum Length, M (inch)	0.29 (11.27)	0.298 (11.72)	0.24 (9.52)	0.242 (9.52)
Fuel Rod O.D., MM (inch)	10.92 (0.430)	10.922 (0.430)	9.63 (0.379)	9.627 (0.379)
Cladding I.D., MM (inch)	9.58 (0.377)	9.576 (0.377)	8.43 (0.332)	8.407 (0.331)
Cladding Thickness, MM (inch)	0.673 (0.0265)	0.673 (0.0265)	0.597 (0.0235)	0.610 (0.0240)
Diametral Gap, Micron (MIL)	178 (7.0)	213.4 (8.4)	203 (8.0)	198.1 (7.8)
Fuel Pellet Diameter, MM (inch)	9.40 (0.370)	9.362 (0.3686)	8.23 (0.324)	8.209 (0.3232)
Fuel Pellet Length, MM (inch)	17.78 (0.700)	15.240 (0.600)	9.53 (0.376)	9.525 (0.375)
Fuel Pellet Density, % TD	94	95	94	95

TABLE 2: Major Fuel Performance Programs
Status Through 1979

Vendor	Fuel Type	Power Plant	Planned Burnup or Operating Cycles	Scheduled Completion	Interim Inspections To Date
Babcock and Wilcox	15x15	Oconee 1	3 cycles	1978	Completed
	15x15	Oconee 1	4 cycles	1980	2
	15x15 (1)	ANO-1	3 cycles	1988	None
	15x15 (2)	Rancho Seco	3 cycles	1986	None
	15x15 (3)	Oconee 2	4 cycles	1980	3
	15x15 (4)	Oconee 2	3 cycles	1981	2
	15x15 (5)	Oconee 2	3 cycles	1983	None
	17x17 (LTA)	Oconee 2	3 cycles	1981	2

FOOTNOTES

(1) Lead test assemblies of an advanced 15x15 extended burnup design.

(2) Current-design 15x15 assemblies containing axially-blanketed fuel columns.

(3) Current-design 15x15 assemblies with special zircaloy cladding materials and EPRI creep collapse specimen clusters.

(4) Current-design 15x15 assemblies with lifted rods and cladding having a known spiral eccentricity.

(5) Current-design 15x15 assembly utilizing low absorption spacer grid material (zircaloy-4).

Table 3. Summary of B&W Fuel Rod Performance for 1979 [1]

Fuel Rod Type:	15x15	17x17
1. Cumulative Number of Rods Irradiated Through Dec. 1979: (2)	525,408	1,056
2. Total Number of Rods Irradiated in 1979 (2)	332,384	1,056
3. Number of Irradiated Rods In-Core on Dec. 31, 1979 (2)	268,736	1,056
a. Maximum Rod - Average Burnup of Rods In-Core, MWd/mtU	[illegible]1,760	23,800
b. Mean Rod - Average Burnup of Rods In-Core, MWd/mtU	13,940	21,900
4. Number of Rods Discharged in 1979	63,648	0
a. Maximum Rod - Average Burnup of Rods Discharged, MWd/mtU	41,037	---
b. Mean Rod - Average Burnup of Rods Discharged, MWd/mtU	22,084	---
5. Estimated Number of Leaking Rods: (3)		
a. In-Core on Dec. 31, 1979	94	(4)
b. Discharged in 1979	35	(4)
c. Generated in 1979	80	(4)

Footnotes:

(1) Conn Yankee and TMI-2 are excluded from this tabulation.

Table 5. (Cont'd)

(2) Oconee-1 and TMI-1 were refueled during this period, but subsequent operation was not resumed in 1979. Therefore, the new fuel is not considered in the irradiated fuel rod tabulation.

(3) The number of leaking fuel rods was estimated from the equilibrium coolant radio-iodine behavior during full-power operation. An absolute correlation of coolant activity levels to the number of leaking fuel rods can not be made because of the various uncertainties in the location and the nature of probable leakers. Thus, the values shown represent only a reasonable indication of the fuel defect status. Only one discharged fuel rod (15x15) was verified as a leaker following 1979 operations during the visual and sipping examinations conducted at one plant.

(4) The relatively small percentage of 17x17 type fuel operating in conjunction with 15x15 type fuel at one plant precluded the differentiation of probable leakers among the two fuel types.