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Fuel Performance Annual Report for 1980

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Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute

Prepared for
**U.S. Nuclear Regulatory
Commission**

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ABSTRACT

This annual report, the third in a series, provides a brief description of fuel performance in commercial nuclear power plants. Brief summaries of fuel surveillance programs and operating experience, fuel performance problems, and fuel design changes are provided. References to additional, more detailed, information and related NRC evaluation are included.

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1.0 INTRODUCTION

Important feedback for safety considerations and licensing procedures is produced by monitoring the in-reactor performance of nuclear fuel in commercial light-water power reactors. Members of the public, governing and advisory bodies, and the U.S. Nuclear Regulatory Commission (NRC) staff have expressed interest in a publicly available summary of in-reactor fuel performance. As a result, a series of annual reports, of which this is the third, has been implemented to provide such a summary. The first was NUREG-0633 (Ref. 1), which covered the period through calendar year 1978. The second, NUREG/CR-1818 (PNL-3583) (Ref. 2), covered calendar year 1979.

The first report (Ref. 1) noted that the Atomic Energy Commission (AEC) and then the NRC have requested operating nuclear reactor fuel performance details through the reporting requirements of Regulatory Guide 1.16 (Ref. 3). However, over the years the material covered in these reports has changed. In the 1971 version of the guide, the NRC requested that a summary of fuel performance characteristics be included in semiannual operating reports and that special topical reports be used for fuel inspection details. By 1975 though, only the abnormal degradation of fuel cladding and the indication of failed fuel were reportable items.^(a) The reporting requirements were further reduced in 1977: only abnormal degradation of fuel cladding was to be included and the requirement for an annual operating report was eliminated. Also, normal operation surveillance results, generic problems, and design

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(a) A report published in 1981, NUREG/CR-1380 (PNL-3325) (Ref. 4), elaborates on the reporting of abnormal degradation and fuel failures. The threshold for what constitutes abnormal degradation is not uniform and remains a matter of opinion. Therefore, the degree of reported degradation is not uniform. The definition of failed fuel is tied to the functional, legal, and detection requirements on the fuel. The designation of fuel as failed depends on which functional requirement is not met (safety, commercial, design), whether or not there is a legal contingency on that requirement (Technical Specification, fuel warranty, design basis), and which indicator is used (coolant or off-gas activity, sipping, strain, or deflection). Thus, the definition can vary from outage to outage and from reload to reload for each utility as the considerations change.

trends are not addressed in the NUREG series of reports entitled "Nuclear Power Plant Operating Experience" (Refs. 5-10).

As a result, the primary intent of this report series is to identify on-going surveillance programs, summarize the results from those programs, and report on generic problems that are of concern during the reporting period. In preparing the reports, we attempt to provide a traceable path of references so that the reader can acquire a greater level of detail than is included in the annual summary.

This report, though focusing on fuel operating experience during calendar year 1980, includes some overlap with the previous year. For those problems first encountered prior to 1980, the pre-1980 information will be included for the sake of continuity. In addition, information received or action taken in early 1981 will be included if pertinent to the discussion of problem areas.

2.0 FUEL SURVEILLANCE PROGRAMS AND OPERATING EXPERIENCE

Basic requirements of fuel surveillance programs, followed by a summary of current programs and operating experience, are presented in this section.

2.1 FUEL SURVEILLANCE REQUIREMENTS

It is indicated under Section 4.2, Fuel System Design, of the Standard Review Plan (Ref. 11) that plans for testing, inspection, and fuel surveillance should be submitted and reviewed for each domestic nuclear power plant. Such plans should include preirradiation verification of cladding integrity, fuel system dimensions, fuel enrichment, burnable poison concentration, and absorber composition. Postirradiation surveillance plans are dependent on whether the fuel design is an existing or new design, and if the fuel exhibited any unusual behavior or characteristics. These plans are then referenced and/or summarized in the plant's safety analysis report (SAR). Reference 12 is an example of a required fuel surveillance program.

Typical fuel assembly parameters^(a) and operating conditions for current commercial light-water reactor (LWR) fuel rod designs for use in pressurized water reactors (PWRs) and boiling water reactors (BWRs) are summarized in Table 1. The newer fuel rod designs are those with smaller fuel rod diameters and more fuel rods per assembly. In a fuel assembly that employs fuel rods of a newer design, the total assembly power is maintained while the individual fuel rods are operated at lower linear heat generation rates and temperatures. This design change is expected to aid in improving the irradiation behavior of commercial LWR fuel by reducing fission gas release from the fuel and by reducing the mechanical interaction between the fuel and cladding. For example, sample calculations indicate that shift from a 7 x 7 to an 8 x 8 rod array in a BWR fuel assembly results in a reduction in fission gas release of about 15% at

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(a) The terms "fuel assembly" and "fuel bundle" are used interchangeably by the nuclear industry. Generally the former term is associated with fuel for PWRs and the latter term with fuel for BWRs.

TABLE 1. Typical Fuel Assembly Parameters

VENDOR	B&W	B&W	C-E	C-E	<u>W</u>	<u>W</u>	<u>W</u>	ENC	ENC	GE	GE	GE
Fuel Rod Array	15x15	17x17	14x14	16x16	14x14	15x15	17x17	15x15	8x8	7x7	8x8	8x8 R
Reactor Type	PWR	PWR	PWR	PWR	PWR	PWR	PWR	PWR	BWR	BWR	BWR	BWR
Assemblies per Core	177	205	217	177	121	193	193	193	560	764	560	560
Fuel Rod Locations Per Assembly	225	289	196	256	196	225	289	225	64	49	64	64
Fuel Rods Per Assembly	208	264	176	236	179	204	264	204	60	49	63	62
Empty Locations Per Assembly	17	25	5	5	17	21	25	21	4	NONE	1	2
Rod Pitch, mm (in.)	14.4 (0.568)	12.7 (0.501)	14.7 (0.580)	12.9 (0.5063)	14.1 (0.556)	14.3 (0.563)	12.6 (0.496)	14.3 (0.563)	16.3 (0.842)	18.7 (0.738)	16.3 (0.640)	16.3 (0.640)
System Pressure, MPa (psia)	15.2 (2200)	15.5 (2250)	15.5 (2250)	15.5 (2250)	15.5 (2250)	15.5 (2250)	15.5 (2250)	15.5 (2250)	7.14 (1035)	7.14 (1035)	7.14 (1035)	7.14 (1035)
Core Average Power Density, kW/liter	91.4	107.3	78.5	96.4	95.6	98.1	104.7	98.1	40.57	50.732	50.51	49.15
Average LHGR, (a) kW/M (kW/ft)	20.3 (6.20)	18.8 (5.73)	20.0 (6.09)	17.5 (5.34)	20.3 (6.20)	22.0 (6.70)	17.8 (5.44)	22.0 (6.70)	15.2 (4.63)	23.1 (7.049)	17.9 (5.45)	17.7 (5.38)
Axial Peak LHGR, in an Average Rod, W/M (kW/ft)	24.41 (7.44)	22.57 (6.88)	24.00 (7.31)	21.00 (6.41)	24.36 (7.44)	26.40 (8.04)	21.36 (6.53)	26.40 (8.04)	18.24 (6.02)	27.72 (9.16)	21.48 (7.09)	21.24 (6.99)
Max. Peak LHGR, kW/M (kW/ft)	53.0 (16.16)	49.9 (15.20)	53.5 (16.3)	42.7 (13.0)	56.8 (17.3)	61.7 (18.8)	44.6 (13.6)	51.9 (15.83)	47.6 (14.5)	60.2 (18.35)	44.0 (13.4)	44.0 (13.4)
Max. Fuel Temp., °C (°F)	2340 (4245)	2090 (4155)	2140 (3890)	1880 (3420)	2260 (4100)	2340 (4250)	1870 (3400)	2200 (3997)	2040 (3700)	2440 (4430)	1830 (3325)	1890 (3435)
Core Average Enrichment, wt% ²³⁵ U	3.00	2.67	2.35	2.36	2.90	2.80	2.60	3.02	2.65	2.19	1.80	1.99
Max. Local Exposure, MWd/MTU (b) GJ/kgU (c)	55,000 4752	55,000 4752	50,000 4320	55,000 4752	50,000 4320	50,000 4320	50,000 4320	47,500 4104	35,000 3024	40,000 3456	40,000 3456	45,000 3888
Cladding Material	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-2	Zry-2	Zry-2	Zry-2

TABLE 1. (contd)

VENDOR	B&W	B&W	C-E	C-E	W	W	W	ENC	ENC	GE	GE	GE
Fuel Rod Length, m (in.)	3.904 (153.688)	3.864 (152.125)	3.71 (145.9)	4.09 (161.02)	3.87 (152.36)	3.80 (149.7)	3.85 (151.6)	3.86 (152.0)	3.99 (156.92)	4.09 (161.1)	4.09 (161.1)	4.20 (185.4)
Active Fuel Height, m (in.)	3.602 (141.8)	3.632 (143)	3.47 (136.7)	3.81 (150)	3.66 (144)	3.66 (144)	3.65 (143.7)	3.66 (144)	3.66 (144)	3.66 (144)	3.71 (146)	3.81 (150)
Plenum Length, m (in.)	0.298 (11.27)	0.242 (9.52)	0.22 (8.6)	0.25 (10.00)	0.18 (6.99)	0.21 (8.2)	0.16 (6.3)	0.17 (6.8)	0.27 (10.63)	0.41 (16.0)	0.36 (14.0)	0.25 (10.0)
Fuel Rod OD, mm (in.)	10.922 (0.430)	9.627 (0.379)	11.18 (0.440)	9.70 (0.382)	10.72 (0.422)	10.72 (0.422)	9.50 (0.374)	10.77 (0.424)	12.74 (0.5015)	14.30 (0.563)	12.52 (0.493)	12.27 (0.483)
Cladding ID, mm (in.)	9.576 (0.377)	8.407 (0.331)	9.86 (0.388)	8.43 (0.332)	9.48 (0.3734)	9.48 (0.3734)	8.36 (0.329)	9.25 (0.364)	10.91 (0.4295)	12.68 (0.499)	10.80 (0.425)	10.64 (0.419)
Cladding Thickness, mm (in.)	0.673 (0.0265)	0.610 (0.0240)	0.660 (0.026)	0.635 (0.025)	0.617 (0.0243)	0.617 (0.0243)	0.572 (0.0225)	0.762 (0.030)	0.914 (0.036)	0.813 (0.032)	0.864 (0.034)	0.813 (0.032)
Diametral Gap, micron (mil)	213.4 (8.4)	198.1 (7.8)	216 (8.5)	178 (7.0)	190 (7.5)	190 (7.5)	165 (6.5)	190 (7.5)	254 (10.0)	305 (12.0)	229 (9.0)	229 (9.0)
Fuel Pellet Diameter, mm (in.)	9.362 (0.3686)	8.209 (0.3232)	9.64 (0.3795)	8.26 (0.325)	9.29 (0.3659)	9.29 (0.3659)	8.19 (0.3225)	9.06 (0.3565)	10.66 (0.4195)	12.37 (0.487)	10.57 (0.416)	10.41 (0.410)
Fuel Pellet Length, mm (in.)	15.240 (0.600)	9.525 (0.375)	16.51 (0.650)	9.91 (0.390)	15.24 (0.600)	15.24 (0.600)	13.46 (0.530)	6.93 (0.273)	8.13 (0.320)	12.70 (0.500)	10.67 (0.420)	10.41 (0.410)
Fuel Pellet Density, %TD	95	95	94.75	95	94	95	95	94	95	95	95	95

(a) LHGR = Linear heat generation rate.

(b) MWd/MTU = number of megawatt days of thermal energy released by fuel containing one metric ton (10^3 kg) of heavy-metal atoms (e.g., U = uranium).

(c) GJ/kgU = gigajoule/kilogram of heavy metal (e.g., U = uranium).

high burnups (e.g., the release value drops from 26% to 13% at 30,000 MWd/MTU and from 46% to 30% at 45,000 MWd/MTU).^(a)

2.2 FUEL SURVEILLANCE PROGRAM AND EXPERIENCE SUMMARY

Table 2 contains an up-dated listing of the major fuel surveillance programs for each fuel vendor. Additional information on the programs and on a number of the associated fuel design changes is provided in Section 4.0. Depending on the program, nondestructive and/or destructive examination techniques are used. Some of these programs are jointly sponsored with participation by the fuel vendor, utility, utility groups, and the Electric Power Research Institute (EPRI). Listing of a given program under a vendor heading is not, therefore, intended to imply that that vendor is the sole sponsor. Project details shown in Table 2 reflect the NRC's current understanding of the programs and are subject to change. The following subsections provide a brief synopsis of each vendor's program and EPRI's involvement.

2.2.1 Babcock & Wilcox Company (B&W)

A summary of Babcock & Wilcox fuel rod experience, from the startup of their first reactor (Oconee-1) in April 1973 through December 1980 is shown in Table 3 (Ref. 13), which shows that over 587,000 fuel rods have been irradiated. A total of 305,568 fuel rods were irradiated in 1980.^(b) While B&W has reported relatively good fuel performance, several plants issued Licensee Event Reports in 1980 on iodine spiking (see Section 3.2.4), which implies that fuel rod failures are associated with those events.

(a) MWd/MTU = number of megawatt days of thermal energy released by fuel containing one metric ton (10^6 g) of heavy-metal atoms (e.g., U = uranium).

(b) The fuel information by B&W for the previous annual report (Ref. 2) showed the following: 1) 94 leaker rods were estimated to be in cores at the end of 1979 (that number was subsequently revised downward to 78 as shown in Table 3), 2) 333,440 rods were irradiated and 80 leaker rods were estimated to have been generated in 1979, and 3) the number of leaker rods in cores at the end of 1978 was estimated to be 49. Leaker refers to a fuel rod that is releasing fission products to the reactor coolant through a breach in its cladding (Ref. 13).

TABLE 2. Major Fuel Performance Programs: Status Through 1980

Vendor	Fuel Type(a)	Power Plant	Planned No. of Operating Cycles	Scheduled Completion	Interim Inspections To Date
Babcock & Wilcox	15 x 15	Oconee-1	5	1983	2
	15 x 15(b)	Arkansas-1(c)	3	1990	None
	15 x 15(d)	Rancho Seco	3	1986	None
	15 x 15(e)	Oconee-2	4	1980	Completed
	15 x 15(f)	Oconee-2	3	1981	3
	15 x 15(g)	Oconee-2	3	1985	None
	17 x 17 (LTAs)	Oconee-2	3	1981	3
Combustion Engineering	14 x 14	Calvert Cliffs-1	5	1982	Completed
	14 x 14	Fort Calhoun			
	14 x 14	Fort Calhoun		1982	
	16 x 16	Arkansas-2(h)		1983	
	16 x 16	Arkansas-2		1984	
Exxon Nuclear	15 x 15	H. B. Robinson-2		1980	Completed
	8 x 8	Oyster Creek		1979	
	8 x 8	Oyster Creek		1983	
	rods	Big Rock Point		1982	
General Electric	7 x 7 (MO ₂)	Quad Cities-1		1982	(i)
	8 x 8 (LTAs)	Monticello	5		
	8 x 8 (LTAs)	Quad-Cities-1	4		
	8 x 8R (LTAs)	Peach Bottom-2	4	1983	
	8 x 8R (LTAs)	Vermont Yankee	5		
	LTAs	Quad Cities-1	5		
	LTAs (barrier cladding)	Quad Cities-1	1		
	LTAs (extended burnup)	Monticello	5	1982	
Westinghouse	15 x 15	Zion-1/Zion-2	5	1984	
	17 x 17	Trojan	5	1984	
	17 x 17 (LTAs)	Surry-2	4	1981	
	17 x 17 (OFA-Demo)	Farley-1			
	17 x 17 (FA-Demo)	Salem-2			
	17 x 17 (OFA-Demo)	Beaver Valley-1			
	MO ₂	R. E. Ginna			

- (a) LTA = lead test assembly, MO₂ = mixed oxide (UO₂-PuO₂) fuel, R = retrofit fuel design, OFA-Demo = Demonstration Optimized Fuel Assemblies.
 (b) Lead test assemblies of an advanced 15 x 15 extended burnup design.
 (c) Arkansas Nuclear One-Unit 1 (Arkansas-1)
 (d) Current-design 15 x 15 assemblies containing axially-blanketed fuel columns.
 (e) Current-design 15 x 15 assemblies with special Zircaloy cladding materials and EPRI creep collapse specimen clusters.
 (f) Current-design 15 x 15 assemblies with lifted rods and cladding having a known spiral eccentricity.
 (g) Current-design 15 x 15 assemblies utilizing low absorption spacer grid material (Zircaloy-4).
 (h) Arkansas Nuclear One-Unit 2 (Arkansas-2)
 (i) The fuel bundles are now discharged.

TABLE 3. Summary of Babcock & Wilcox Fuel Rod Performance for 1980(a)

Fuel Rod Type	15 x 15	17 x 17
1. Cumulative Number of Rods Irradiated Through Dec. 1980:	586,054	1,056
a. Maximum Rod-Average Burnup, MWd/MTU	41,600	28,210
b. Mean Rod-Average Burnup, MWd/MTU	18,650	26,560
2. Total Number of Rods Irradiated in 1980	304,512	1,056
3. Number of Irradiated Rods In-Core on Dec. 31, 1980:	266,448	528
a. Maximum Rod-Average Burnup, MWd/MTU	32,430	28,210
b. Mean Rod-Average Burnup, MWd/MTU	14,740	27,130
4. Number of Rods Discharged in 1980:	61,360	528
a. Maximum Rod-Average Burnup, MWd/MTU	39,240	27,940
b. Mean Rod-Average Burnup, MWd/MTU	24,900	26,000
5. Estimated Number of(b) Leaker Rods:		
a. In-Core on Jan. 1, 1980	78(c)	-
b. Discharged in 1980	10	-
c. In-Core on Dec. 31, 1980	77	-
d. Generated in 1980	9	-
e. Cumulative through Dec. 1980	216	-

(a) Connecticut Yankee (Haddam Neck) and Three Mile Island-Unit 2 (TMI-2) are excluded from this tabulation.

(b) Estimated from equilibrium coolant radio-iodine behavior during full-power operation (iodine-131, iodine-133, and iodine-135 activity levels). The relatively small percentage of 17 x 17 type fuel (LTAs) operating with 15 x 15 type fuel precludes differentiation of probable leakers among the two fuel types.

(c) Revised from the value reported for Dec. 31, 1979, in NUREG/CR-1818 PNL-3583. The revised value is based upon identification of leaker assemblies in one reactor and correcting for their above-average power density.

With the exception of fuel rods in four lead test assemblies (LTAs), the majority of the fuel rods irradiated were in fuel assemblies of the standard, 15 x 15, Mark B design. The operating status (as of December 31, 1980) for B&W plants is listed in Table 4 (Ref. 13). The relatively small fraction of 17 x 17 type fuel (LTAs) operating with the 15 x 15 type fuel precludes differentiation of probable leakers among the two fuel types. The number of leaker rods was estimated by B&W from the equilibrium coolant radioiodine (iodine-131, iodine-133, and iodine-135) behavior during full-power operation. Because of the uncertainties associated with the location and nature of probable leakers, the number of leaker rods stated in Table 3 represents a best-estimate of the status of fuel rod integrity.

The B&W method of coolant radioactivity analysis to assess the leaker level of operating fuel is described in Reference 14. In applying this method, only data obtained during conditions of equilibrium-iodine, full power operation are considered. Also, it is assumed that the leaker fuel rods are operating at core average conditions and that the plant purification system is operating at a known constant condition. Although these assumptions are

TABLE 4. Babcock & Wilcox Operating Plant Status as of December 31, 1980

Reactor	Current Cycle	Maximum Assembly Burnup, MWd/MTU	
		In Core	Discharged
Oconee-1	6	26,300	40,000
Oconee-2	5	26,100	33,700
Oconee-3	5	23,900	31,940
TMI-1 ^(a)	4	32,400	32,200
Arkansas-1 ^(b)	4	30,740	33,220
Rancho Seco	4	31,180	37,730
Crystal River-3	3	20,870	22,390
Davis-Besse-1	2	16,960	14,780

(a) Cycle 5 start-up activities at Three Mile Island, Unit 1 (TMI-1) were terminated on March 30, 1979. The unit has remained in cold shut-down.

(b) Arkansas Nuclear One-Unit 1 (Arkansas-1).

generally valid, errors can result if the local heat rate in the leaker rods is above or below the core average. Thus, uncertainties in the location of the leaker rods, the nature of the cladding defects and their axial location along a rod preclude making an absolute correlation of coolant activity levels to the number of leaking rods in any one batch. However, B&W believes that, on the average, the leaker correlation provides a useful indication of fuel performance to date. In all sipping tests and visual examinations conducted to date, the radiochemistry projections reportedly correlate well with observed results. Consequently, B&W feels that the leaker estimates are correct.

More specific information on the number of leaker rods was not available for this report because fuel assembly leak testing (sipping) was performed at only one reactor site, Rancho Seco, in 1980. Of 60 fuel assemblies discharged from Rancho Seco after three cycles, 43 were sipped as part of the "Load Following and Fuel Sipping Program" conducted jointly by Nuclear Fuel Industry Limited of Japan, the Sacramento Municipal Utility District, and B&W. Sipping (using the wet technique) was conducted during the refueling outage at the end of Cycle 3 to verify the fuel cladding integrity of three-cycle fuel assemblies that were in the core during periods in which Rancho Seco operated in a load-follow mode. Only one assembly was identified as a leaker. Only one leaker rod on the periphery of that assembly was identified by subsequent visual inspection of 1400 peripheral rods in 25 fuel assemblies (Refs. 13 and 15). Visual inspection of the leaker rod revealed a pinhole leak at one location and about 10 cm (4 in.) of cladding missing (a "portion" of several pellets was visible) at another location (Refs. 15 and 16). Slight swelling and discoloration were also found on an adjacent fuel rod in another assembly. Iodine-131 and -133 levels in the reactor coolant were found to be normal. Only insignificant amounts of fuel were dispersed from the fuel rod. The initiating cause (or primary failure mechanism) of this cladding defect has not been identified. The associated fuel assembly was in the reactor core during the first three cycles of reactor operation. A review and evaluation of the coolant radiochemistry behavior for the Rancho Seco Unit shows that the initial cladding breach did not occur during the third cycle of operation, but most likely occurred during the first cycle. Deterioration of the defect area has erased all physical traces of the primary failure mechanism. With only one fuel rod

within the batch identified as a leaker, it is apparent that the initial cladding defect was a random, low probability event which is not attributable to a systematic type of failure mechanism. B&W projects that this was an isolated hydriding failure due to random contamination. A careful study of this failure is being made by the licensee, and a copy of the report has been requested by the NRC.

During 1980, a total of 20 broken hold-down springs (see Section 3.2.3) were found at Crystal River-3, Davis Besse-1, and Oconee-1 and iodine spiking (see Section 3.2.4) occurred at Crystal River-3 and Davis Besse-1.

Information is available on the B&W fuel surveillance programs shown in Table 2. The B&W/Duke^(a) 15 x 15 Fuel Surveillance Program in Oconee-1 was completed in 1979 and is described in References 17-19. The DOE^(b)/Duke/AP&L^(b)/B&W Extended Burnup Programs involve irradiations in Oconee-1 and Arkansas-1.^(c) Recent results from and plans for the DOE/Duke/B&W extended Burnup Program are presented in References 20-22; other reports issued from the program are References 23-27. Recent information on the DOE/AP&L/B&W Extended Burnup Program is in Reference 28; other reports published under the program are References 29-32. The DOE/SMUD^(d)/B&W Axial Blanket Fuel Design and Development Program in Rancho Seco is described in References 33 and 34. Information on the EPRI/Duke/B&W Cooperative Program on PWR fuel rod performance in Oconee-2 is contained in References 35-42. The B&W/Duke Fuel Rod Bow Program in Oconee-2 is described in Reference 43. The B&W/Duke Low Absorption Grid Program in Oconee-2 is discussed in References 44 and 45. Reference 46 provides information on the B&W/Duke 17 x 17 Lead Test Assembly (LTA) Demonstration Program in Oconee-2.

2.2.2 Combustion Engineering, Inc. (C-E)

As shown in Table 5, Combustion Engineering fuel was irradiated in all eight of their reactors during 1980 (Ref. 47). Six of the reactors use a

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(a) Duke Power Company (Duke).

(b) U.S. Department of Energy (DOE); Arkansas Power & Light Company (AP&L).

(c) Arkansas Nuclear One-Unit 1 (Arkansas-1).

(d) Sacramento Municipal Utility District (SMUD).

TABLE 5. Summary of Combustion Engineering Fuel Irradiated and/or Discharged in 1980

Reactor (fuel cycles)	Fuel Batch	No. of Assemblies		No. of Fuel Rods		Batch-Averaged Burnup, Mwd/MTU	
		Irradiated	Discharged	Irradiated	Discharged	as of Dec. 1980	at Discharge
Calvert Cliffs-1 (Cycles 4 and 5)	B	1	1	162	162	---	43,000
	D	72	71	12,672	12,496	32,300	30,500
	E	72	20	12,672	3,520	21,200	21,900
	F	72	---	12,672	---	11,300	---
	G	92	---	16,192	---	<100	---
Calvert Cliffs-2 (Cycle 3)	B	1	---	164	---	34,200	---
	C	68	---	11,632	---	33,900	---
	D	84	---	14,784	---	22,000	---
	E	64	---	11,264	---	10,300	---
Fort Calhoun (Cycles 5 and 6)	D0	17	16	2,992	2,816	40,300	33,400
	D1	4	4	704	704	---	30,200
	E6	16	16	2,816	2,816	29,100	25,000
	E1	16	16	2,816	2,816	---	27,800
	F0	20	---	3,520	---	25,100	---
	F1	16	---	2,816	---	24,900	---
	G	44	---	7,744	---	15,000	---
Maine Yankee (Cycles 4 and 5)	E16	61	60	9,760	9,600	23,900	29,700
	F0	12	12	1,968	1,968	---	33,200
	G0	16	---	2,816	---	28,200	---
	G41	4	---	688	---	29,400	---
	G42	12	---	2,064	---	29,700	---
	H0	40	---	7,040	---	27,200	---
	I0	48	---	8,448	---	16,000	---
	I4	24	---	4,128	---	19,000	---
Millstone-2 (Cycles 3 and 4)	B	6	5	984	820	18,500	33,800
	C	68	68	11,632	11,632	---	31,300
	D	72	---	12,672	---	21,500	---
	E	72	---	12,672	---	12,000	---
Palisades (Cycle 4)	D	60	---	12,960	---	24,900	---
St. Lucie-1 (Cycles 3 and 4)	B	21	21	3,444	3,444	---	28,300
	C0	40	39	7,040	6,864	29,200	25,400
	C1	12	12	1,968	1,968	---	29,000
	C2	16	16	2,624	2,624	---	30,000
	D0	40	---	7,040	---	22,100	---
	D1	20	---	3,520	---	23,100	---
	E1	28	---	4,928	---	15,300	---
	E6	40	---	7,040	---	12,100	---
	F1	48	---	8,448	---	6,300	---
	F6	40	---	7,040	---	4,700	---
Arkansas-2(a) (Cycle 1)	A	61	---	14,396	---	10,100	---
	B	60	---	13,440	---	9,700	---
	C	56	---	12,808	---	7,000	---

(a) Arkansas Nuclear One-Unit 2.

14 x 14 fuel assembly design; Palisades uses a 15 x 15 design; Arkansas-2^(a) uses a new 16 x 16 design that is the precursor to Combustion Engineering's standardized System 80® design. Information on the performance of C-E fuel is provided in References 48-51. C-E fuel surveillance programs are listed in Table 2. Reference 52 provides information on the EPRI/C-E surveillance programs. The status of burnup experience, as of December 31, 1980, with C-E, all-Zircaloy assemblies is presented in Table 6. A total 1,802 assemblies or a total of 315,141 fuel rods have been discharged. There were 1,271 active fuel assemblies, which contained a total of 235,052 fuel rods.

Table 7 shows the typical equilibrium iodine-131 concentrations in the primary coolants of C-E reactors during 1980. For this report, C-E indicated they were not going to convert the iodine concentrations to estimated numbers of failed fuel rods because of the uncertainties involved (e.g., radionuclide escape rate models are so variable).^(b) During 1980, iodine spiking (see Section 3.2.4) occurred at Arkansas-1 and St. Lucie-1. The latter had an extended period of power operation during 1980 with a nominal level of fuel leakage (Ref. 53). On January 15, 1981, low level cladding failure was noted at Palisades (Ref. 54), which contained fuel supplied by Combustion Engineering, Inc., and by Exxon Nuclear Company (see Section 2.2.3). Minor fuel cladding degradation was known to exist at Maine Yankee in 1979 and 1980 (Refs. 55-57) and the cause of those fuel failures (15 to 25 fuel rods are believed to have failed) is discussed in Section 3.1.3.1.

2.2.3 Exxon Nuclear Company, Inc. (ENC)

Exxon Nuclear continues to be the sole domestic supplier of both BWR and PWR fuel. Because it is not a power plant vendor, ENC provides reload fuel. Although their reload fuel designs are basically similar to those of the fuel being replaced, ENC performs its own design analysis, and their fuel has its

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(a) Arkansas Nuclear One-Unit 2 (Arkansas-2).

System 80® is a registered trademark of Combustion Engineering, Inc.

(b) In the previous annual report (Ref. 2), it was indicated that C-E estimated, based on the level of iodine-131 in the reactor coolant (data in Ref. 48), that 18 rods out of 225,000 (corresponds to a defect rate of less than 0.01%) had failed as of February 1, 1979.

TABLE 6. Combustion Engineering Burnup Experience with All-Zircaloy-Assemblies:
Status as of December 31, 1980

Fuel Assembly Batch-Averaged Burnup, MWd/MTU	Active Fuel Assemblies with Pressurized Fuel Rods		Discharged Fuel Assemblies with Pressurized Fuel Rods		Discharged Fuel Assemblies with Non- Pressurized Fuel Rods	
	No. of Fuel Assemblies	No. of Fuel Rods	No. of Fuel Assemblies	No. of Fuel Rods	No. of Fuel Assemblies	No. of Fuel Rods
0 to 3,900	92	16,192	0	0	0	0
4,000 to 7,900	144	28,296	6	1,048	0	0
8,000 to 11,900	305	60,220	25	3,696	208	40,500
12,000 to 15,900	184	32,384	157	27,632	190	35,351
16,000 to 19,900	72	12,576	250	43,452	24	3,840
20,000 to 23,900	222	39,044	164	28,884	0	0
24,000 to 27,900	136	26,336	371	62,752	0	0
28,000 to 31,900	46	8,032	369	61,516	0	0
32,000 to 35,900	69	11,796	37	6,308	0	0
36,000 to 39,900	0	0	0	0	0	0
40,000 to 43,900	1	176	1	162	0	0
Totals	1,271	235,052	1,380	235,450	422	79,591

TABLE 7. Typical Equilibrium Iodine-131 Concentrations in the Primary Coolant of Combustion Engineering Reactors During 1980

Reactor	Fuel Cycle	Beginning of Cycle	End of Cycle	1980 Equilibrium Conditions	
				Power, %	Iodine-131, $\mu\text{Ci/g}$
Calvert Cliffs-1	4	07/10/79	10/17/80	100	0.004 to 0.009
	5	12/21/80	04/82 ^(a)	NA ^(b)	NA ^(b)
Calvert Cliffs-2	3	12/06/79	01/17/81	100	0.003 to 0.008
Fort Calhoun	5	12/15/78	01/18/80	98	0.02
	6	06/08/80	09/81 ^(a)	100	0.03 to 0.06
Maine Yankee	4	08/25/78	01/11/80	62 (100 est)	0.03 (0.15 est)
	5	03/09/80	05/08/81	100	0.003 to 0.005
Millstone-2	3	05/18/79	08/16/80	100	0.003 to 0.006
	4	10/20/80	01/82 ^(a)	100	0.006
Palisades	4	05/24/80	08/81 ^(a)	100	0.04 to 0.1 ^(c)
St. Lucie-1	3	06/08/79	03/15/80	100	0.01
	4	05/07/80	10/81 ^(a)	100	0.02 to 0.06
Arkansas-2 ^(d)	1	12/06/78	03/28/81	100	0.10 to 0.15

(a) Target end-of-cycle date.

(b) Not available (NA).

(c) Predominately non-C-E supplied fuel.

(d) Arkansas Nuclear One-Unit 2.

own unique design features. For example, ENC cladding is slightly thicker than others (see Table 1), and all ENC fuel assemblies are reconstitutable.^(a)

Performance of ENC fuel is summarized in Reference 58. Table 8 provides a summary of ENC fuel exposure in 1980 (Ref. 59). There were 2445 fuel assemblies in domestic and foreign reactor cores (1338 in BWRs and 1107 in BWRs) as of December 31, 1980. During 1980, a total of 494 fuel assemblies were discharged from domestic and foreign reactors (374 from BWRs and 120 from PWRs). According to ENC, there were no failures of ENC fuel in 1980. In October 1981, ENC indicated that there was no further word yet regarding the Palisades fuel (see Section 2.2.2), which was being examined at that time.

ENC fuel surveillance programs are shown in Table 2. Reference 44 provides information on the EPRI/ENC programs.

2.2.4 General Electric Company (GE)

Table 9 provides a summary of General Electric's BWR/2-5 experience with 8 x 8 fuel as of January 1981, which shows that less than 0.02% of the 1,239,000 fuel rods^(b) in operating and discharged 8 x 8 fuel bundles are estimated to have failed (Ref. 60).^(c) Currently, over 93% of all GE fuel operating in commercial reactors is of the 8 x 8 fuel bundle design.

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- (a) The only B&W fuel assemblies that are designed to be reconstitutable and that were in reactors in 1980 are the two Mark CR (17 x 17) LTAs in Oconee-2. All of the C-E fuel assemblies under irradiation in 1980 were reconstitutable, except for one Batch-C fuel assembly in Cycle 4 at St. Lucie-1. All GE fuel bundles are reconstitutable. Westinghouse will commence delivery of fuel assemblies with reconstitutable design features in the first quarter of 1982.
- (b) As of January 1, 1980, the cumulative number of fuel rods loaded (in 8 x 8, 8 x 8R, and prepressurized 8 x 8R type fuel bundles) into BWRs was 1,026,414 (Ref. 2).
- (c) In the previous annual report (Ref. 2), the GE estimates for rod failure rates for 7 x 7, 7 x 7R, 8 x 8, and 8 x 8R plus 8 x 8R (prepressurized) fuel types as of January 1, 1980 were 1.01%, 0.066%, 0.16%, and 0.002%, respectively. In May 1979, it was reported by GE (Ref. 61) that the estimated rod failure rates for 7 x 7, 7 x 7R, 8 x 8, and 8 x 8R fuel types were 0.98%, 0.043%, 0.028%, and 0%.

TABLE 8. Summary of Exxon Nuclear Fuel Exposure in 1980(a)

Reactor	Fuel In-Core as of Dec. 31, 1980		Fuel Discharged in 1980	
	No. of Assemblies	Maximum Assembly Burnup, MWd/MTU	No. of Assemblies	Maximum Assembly Burnup, MWd/MTU
BWRs:				
Barsebaeck	162	10,700	0	--
Big Rock Point (extended burnup)	64	34,900 (rod avg.)	4 Rods	34,900 (rod avg.)
Big Rock Point	80	28,300	18	23,000
Dresden-1	66	4,500	0	--
Humboldt Bay	122	12,600	0	--
Kah1	18	11,600	0	--
KRB	0	--	183	21,800
La Crosse	70	12,000	12	13,700
Oskarsham	260	22,100	1	15,400
Oyster Creek	560	28,300	160	25,300
PWRs:				
Biblis-A	76	25,500	1	10,000
O. C. Cook	193	26,000	63	31,900
Fort Calhoun	40	5,000	0	--
H. B. Robinson-2	157	41,700	56	40,400
Kewaunee	68	15,900	0	--
LOFT	9	400	0	--
Maine Yankee	72	7,000	0	--
Palisades	144	30,000	0	--
Prairie Island-1	80	16,000	0	--
Prairie Island-2	40	9,300	0	--
R. E. Ginna	104	24,600	0	--
Tihange-1	48	7,600	0	--
Yankee Rowe	76	23,000	0	--

(a) There were no failures of Exxon Nuclear in 1980.

TABLE 9. General Electric BWR/2-5 Fuel Experience Summary (January 1981)

	All 8x8
Total rods in core or discharged	1,239,000 ^(a)
Estimated sound rods, %	99.98
Lead assembly exposure, MWD/MTU	40,000
Peak linear heat generation rate, kW/m (kW/ft)	44.0 (13.4)

(a) Cumulative

The overall failure rate for fuel rods irradiated in 1980 was less than 0.01%. GE stated that they were unable to supply an estimate of the percent of failed rods during calendar year 1980 for each plant because individual licensees own the fuel.

Experience with GE BWR fuel through January 1981 is summarized in Reference 60. Information on GE fuel performance is also described in Reference 61.

As of January 1981, more than 2,038,000 GE production fuel rods are now in or have completed operation in 41 commercial BWRs in the U.S. and abroad. In addition, over 950 developmental irradiations have been completed or are underway.

It was reported in 1980 that the failure of 75 rods at Vermont Yankee was associated with a metallic incursion in the feedwater (see Section 3.1.3.2). Some gadolinia fuel rods (the NRC estimates less than 100) were found to be underpressurized (see Section 3.2.2). Iodine spiking occurred in 1980 at Brunswick-2 (see Section 3.2.4).

GE depends on results from fission product monitoring, fuel sipping, and site inspections, as well as detailed fuel inspections and postirradiation examinations, to develop its data base on BWR fuel.

GE, as well as other fuel vendors, uses lead test assemblies to confirm design adequacy or to provide early warning of negative design features. A summary of current surveillance programs with GE lead test assembly and extended burnup bundles is given in Table 10 and also in Table 2. Reference 52 provides information on the EPRI/GE programs.

TABLE 10. Summary of General Electric Lead Test Assembly and Extended Burnup Bundles

Program	Reactor	No. of Bundles	Completed Cycles of Operation	Average Burnup At Last Outage, MWd/MTU	Objectives
8x8 Lead Test Assemblies	Monticello	1	5	40,000	Lead 8x8 Performance, Extended Burnup
	Quad Cities-1	1(a)	4	29,000(a)	Lead 8x8 Performance
8x8R Lead Test Assemblies	Peach Bottom-2	4	3	27,000	Lead 8x8R Performance, Extended Burnup
	Vermont Yankee	2(a)	5	21,000(a)	Lead 8x8R Performance
Lead Test Assemblies	Quad Cities-1	5(b)	5	31,000	Extended Burnup, Fuel Performance
Lead Test Assemblies (Barrier Cladding)	Quad Cities-1	4		12,000	Barrier Clad
Lead Test Assemblies (Extended Burnup)	Monticello	4(c)	5	34,000(c)	Extended Burnup

(a) Bundle(s) now discharged.

(b) Three bundles now discharged.

(c) One bundle discharged. The remaining three bundles include the 8x8 lead test assembly.

2.2.5 Westinghouse Electric Corporation (W)

Westinghouse issues an annual report describing the operating experience of their fuel in domestic and foreign reactors. Reference 62 provides the detailed information on Westinghouse fuel experience up to December 31, 1980. A total of 901,700 Zircaloy-clad fuel rods are currently operating in 35 commercial PWRs in the U.S. and abroad. Those rods plus previously discharged fuel brings the total number of Westinghouse Zircaloy-clad fuel rods to 1,996,400. Table 11 presents a burnup summary for the fuel rods discharged and being irradiated through 1980.^(a) Average burnup for the discharged fuel is 24,500 MWd/MTU and average burnup for all fuel is 20,600 MWd/MTU (this burnup is lower because fresh, unburned fuel is included in that number).

Table 12 provides a Westinghouse fuel performance summary on a plant-by-plant basis. The data on peak burnup in Table 12 reflect an average of the high burnup fuel assemblies (batch) in a region. The values for percent of design basis activity release rates for Jose' Cabrera, Indian Point-2, and Ko-Ri in Table 12 are higher than the values for the other plants but Westinghouse indicates that they are still considerably less than the Technical Specification limit.

Baffle jetting resulted in the failure of two fuel rods in 1980 (see Section 3.1.1). Grid strap damage was observed on 89 fuel assemblies in 1980 (see Section 3.1.2). Cracking of control rod guide tube support pins was found in 1980 at a foreign PWR (see Section 3.2.1). During 1980, 40 cracked or broken hold-down springs were noted at two foreign PWRs (see Section 3.2.3).

Westinghouse fuel surveillance programs are shown in Table 2. For individual discharged fuel assemblies, the highest burnups have been in the range of 39,000 to 46,500 MWd/MTU. As part of the EPRI/W high burnup program, four high burnup demonstration assemblies have been irradiated in Zion-2 (Cycle 4 core) to a burnup of 46,500 MWd/MTU. The burnups of these four assemblies will be extended to about 55,000 MWd/MTU in the Cycle 6 core (starts in early

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(a) Table 11 does not include 3043 fuel rods in 17 fuel assemblies (14 x 14 type) at Ginna. Westinghouse indicated in September 1981 that those rods were inadvertently left out of Table 11 when it was transmitted to the NRC.

**TABLE 11. Westinghouse Zircaloy-Clad Fuel Burnup Status
as of December 31, 1980: Assemblywise Burnup
Distribution of Fuel Rods**

Active Fuel Rods(a)		Total Rods(b)	
Assemblywise Burnup, MWd/MTU	No. of Fuel Rods	Assemblywise Burnup, MWd/MTU	No. of Fuel Rods
0 - 3,900	243,552	0 - 3,900	243,552
4,000 - 7,900	69,328	4,000 - 7,900	72,550
8,000 - 11,900	79,208	8,000 - 11,900	102,919
12,000 - 15,900	104,712	12,000 - 15,900	186,439
16,000 - 19,900	98,900	16,000 - 19,900	342,785
20,000 - 23,900	89,331	20,000 - 23,900	225,368
24,000 - 27,900	116,118	24,000 - 27,900	332,379
28,000 - 31,900	65,194	28,000 - 31,900	298,257
32,000 - 35,900	30,436	32,000 - 35,900	144,099
36,000 - 39,900	995	36,000 - 39,900	41,472
40,000 -	895	40,000 -	3,501

Newly Discharged Fuel Rods(c)		Total Discharged Fuel Rods(d)	
Assemblywise Burnup, MWd/MTU	No. of Fuel Rods	Assemblywise Burnup, MWd/MTU	No. of Fuel Rods
0 - 3,900	0	0 - 3,900	0
4,000 - 7,900	0	4,000 - 7,900	3,222
8,000 - 11,900	2,776	8,000 - 11,900	23,711
12,000 - 15,900	10,207	12,000 - 15,900	81,727
16,000 - 19,900	46,903	16,000 - 19,900	243,885
20,000 - 23,900	5,304	20,000 - 23,900	136,037
24,000 - 27,900	41,964	24,000 - 27,900	216,261
28,000 - 31,900	69,472	28,000 - 31,900	233,063
32,000 - 35,900	38,979	32,000 - 35,900	113,663
36,000 - 39,900	6,207	36,000 - 39,900	40,477
40,000 -	2,427	40,000 -	2,606

(a) Fuel rods not discharged from operating plants.

(b) Active fuel rods plus total number of discharged fuel rods.

(c) Fuel rods discharged during calendar year 1980.

(d) Fuel rods discharged from 1969 through December 1980.

TABLE 12. Westinghouse Fuel Performance Status Report (fourth quarter 1980)

Reactor	Location	Current Cycle No.	Peak Batch Average Burnup as of Dec. 31, 1980, (Mwd/MTU)	Percent of Design Basis Activity Release Rate ^(a)
Jose' Cabrera	Spain	10	31,400	2.8
Beznau-1	Switzerland	10	33,500	0.75
Mihama-1	Japan	2	18,500	(b)
Point Beach-1	U.S.	9	34,500	1.9
Mihama-2	Japan	4	29,600	(b)
Point Beach-2	U.S.	7	36,100	0.04
Surry-1	U.S.	5	24,000	(c)
Turkey Point-3	U.S.	7	33,000	(b)
Surry-2	U.S.	4	32,100	0.08
Zion-1	U.S.	5	36,100	0.63
Indian Point-2	U.S.	5	34,100	2.9
Turkey Point-4	U.S.	7	33,200	(b)
Prairie Island-1	U.S.	6	34,900	≤0.01
Zion-2	U.S.	5	37,100	0.14
Kewaunee	U.S.	6	31,900	≤0.01
Prairie Island-2	U.S.	5	35,100	0.02
Trojan	U.S.	3	28,300	0.26
Indian Point-3	U.S.	3	30,500	(d)
Beaver Valley	U.S.	2	19,300	(e)
Salem-1	U.S.	3	26,900	(d)
Ko-Ri	Korea	2	25,900	4.2
Farley-1	U.S.	3	27,300	0.02
Ohi-1	Japan	2	16,900	(b)
O. C. Cook-2	U.S.	3	28,700	0.33
North Anna-1	U.S.	3	26,600	0.24
Ohi-2	Japan	2	≤2,000	(b)
North Anna-2	U.S.	1	<1,000	(e)
Sequoyah-1	U.S.	1	<1,000	(e)
Ringhals-3	Sweden	1	<1,000	(e)
Salem-2	U.S.	1	<1,000	(b)

(a) Activity release rate calculated from coolant activity averaged over the quarter and presented as percent of that iodine-131 release rate which establishes the basis for design of plant shielding and coolant cleanup system equipment.

(b) No data reported.

(c) Maintenance and inspection or repair during period of report.

(d) Plant refueling during period of report.

(e) Reported information reflects last period of operation or inference from short periods of operation during period of report.

1981) of Zion-1. Reference 52 provides information on the EPRI/W program. References 63 and 64 describe the evaluation of the significant amount of data from Westinghouse high burnup fuel experience.

Region average burnups to 29,000 MWd/MTU have been achieved by 17 x 17 fuel assemblies, which are being irradiated in five plants. Four demonstration 17 x 17 Optimized Fuel Assemblies completed their first cycle of operation in the fall of 1980.

As shown in Table 12, Westinghouse continues the practice of evaluating fuel performance in terms of coolant activity level.^(a) Coolant design basis activity varies somewhat from plant to plant (e.g., because of reactor power and coolant purification rate). However, for comparison purposes, approximately 2 μCi of iodine-131 per gram of coolant water is equivalent to 100% of the coolant design basis. Westinghouse states that the number of defected fuel rods cannot be directly inferred from the coolant radioactivity level. The "coolant activity design basis" is equivalent to the predicted reactor coolant activity concentration, at rated operating power and normal operation of the reactor coolant letdown and cleanup system, resulting from a fuel defect level of one percent at an assumed escape rate coefficient of radioiodine of $1.3 \times 10^{-8} \text{ sec}^{-1}$. The term "defect level" is defined as the equivalent percentage of defective average power density fuel rods expressed as a percentage of reactor rated power contributing radioactivity to the coolant at a specific escape rate through cladding defects. For example, a reactor operating at full power of 3000 MW (thermal) having defective fuel rods generating 30 MW (thermal) would be operating at a fuel defect level of 1%, with a resulting steady-state iodine-131 coolant activity of approximately 2 $\mu\text{Ci/g}$ of coolant water. There is no "standard defect," and the actual release rates from different types of cladding defects can vary significantly. Thus, the actual number of fuel rods that may be defected cannot be defined based on coolant activity. The reported coolant activity levels (Table 12), combined with the burnup experience (Table 11), indicate the

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(a) In the previous annual report (Ref. 2) it was indicated by Westinghouse that iodine-131 activity levels were reported in 23 plants in 1979. Westinghouse estimated that approximately 100 rods failed (corresponds to a defect rate of approximately 0.017%).

reliability and good performance of Westinghouse Zircaloy-clad fuel. According to Westinghouse, in some cases the coolant activity indicates that a small number of cladding defects is present. Except for two fuel rods in peripheral fuel assemblies that failed due to fretting caused by baffle jetting, no cause has been identified yet for the apparent defects.

2.2.6 Electric Power Research Institute (EPRI)

The EPRI LWR Core Materials Performance Program began in mid-1974. A series of program-planning documents was initiated in 1975. Reference 52 is the fifth such report and discusses progress in the program during 1979 and 1980 and plans for work in 1981 and 1982. The four previous reports included the detailed planning support documents issued in December 1975 (EPRI SR-25) and January 1977 (EPRI NP-370-SR) and the progress reports issued in January 1978 (EPRI NP-737-SR) and February 1979 (EPRI NP-1024-SR).

When the program was initiated, fuel rod problems were emphasized. However, since 1979 additional attention has been focused on other fuel system and core components, especially control rods, burnable poison rods, and fuel channels for BWRs. Four areas are emphasized in the current program: extending fuel burnup beyond the present 30,000 Mwd/MTU level, decreasing the capacity factor losses caused by fuel-related restrictions on power maneuvering, extending the life of control rods, and extending the life of BWR fuel channels. The program goal is to increase plant availability. In working toward that goal, effort is underway to: 1) determine the response of core component materials, 2) develop a comprehensive fuel performance data base, and 3) develop accurate models and computer codes.

Presently, the EPRI has 22 active projects. The project scopes range from laboratory studies to large-scale tests of both standard and prototypic fuel assemblies. As noted earlier, the EPRI is extensively involved in fuel surveillance projects that are based on standard product line fuel of the type employed in LWRs at domestic commercial nuclear plants. Data have been collected on the detailed performance of fuel assembly components from the cooperative projects between EPRI and each of the five major domestic fuel vendors. In collecting the data under these programs, emphasis was placed on information

related to current industry concerns (e.g., fission gas release, corrosion, and extended burnup) and also on measured properties of traditional interest (e.g., fuel rod diameter and length, fuel rod spacing, and fuel assembly dimensional changes). To avoid difficulties in shipping and waste disposal, poolside examinations were emphasized in these cooperative projects. However, limited hot cell examinations were also conducted.

Reference 52 states that "during 1979 and 1980, several projects produced significant results concerning: 1) power reactor fuel performance, particularly fission gas release, fuel swelling, cladding oxidation, fuel rod growth, and fuel rod bow, 2) test reactor fuel performance of new fuel types, particularly fission gas release and fuel-cladding mechanical interaction, and 3) fuel rod reliability modeling."

References 65-67 are 1980 directories of technical reports issued by EPRI. Included in the lists of reports in those directories were two (Refs. 68 and 69) from EPRI's Nuclear Division that deal with fuel performance in water-cooled reactors.

3.D PROBLEM AREAS OBSERVED DURING 1980

Over the years, fuel problems have been caused by a number of factors including densification, hydriding, fretting and wear, bowing, and pellet-cladding interaction (PCI). NRC staff reports or memoranda have been generated on each item (e.g., Reference 70 on PCI and Reference 71 on guide tube wear). Although several problems that were formerly of concern have been resolved (e.g., fuel densification), some old problems such as baffle jetting, grid strap damage, and crud and corrosion remain, and some new problems have arisen. The following section contains a discussion of 1) problem areas in 1980 that were similar to those in 1979, 2) new problems in 1980, and 3) old problems (1979) that did not reoccur or that were solved.

3.1 PROBLEMS IN 1980 THAT WERE SIMILAR TO THOSE IN 1979

3.1.1 PWR Baffle Jetting

Baffle jetting was one of the problem areas encountered in 1979. Fuel failure caused by baffle jetting was reported in several plants (Ref. 2). At that time, the vendor, Westinghouse Electric Corporation, recommended that both a visual examination and peening of the joints should be performed to alleviate this problem. The vendor recommended that flat-land peening be used on all outside corner baffle joints and that baffle joints should not be welded together.

During 1980, only one plant, Trojan-1, reported the occurrence of baffle jetting (Refs. 72-76). Two fuel rods (one in each of two fuel assemblies) with cladding damage were observed during a preplanned fuel inspection program. The two assemblies were removed from the core. Failure of the cladding on the PWR fuel rods was caused by water-jet impingement on the fuel rods via an enlarged baffle plate joint gap. The baffle joints were peened by both the knife-edge and flat-land peening methods, as recommended by the vendor.

In May 1980, the licensee for Trojan proposed (Ref. 77) to amend the operating license for the plant to permit installation of two fuel assemblies with a maximum of three rods per assembly with dummy (solid) stainless steel

rods in the locations that have been subject to cross flow (baffle jetting). The licensee suggested using coolant chemistry as a criterion for surveillance (Ref. 78). Based on 1) review of the information in Reference 77, 2) required surveillance, and 3) a report of the physics startup tests, the NRC found that the proposed substitution of dummy stainless steel rods was acceptable for Cycle 3 operation (Ref. 79).

The NRC issued a circular in July 1980 about fuel rod damage due to water jetting from baffle plate corners and provided guidelines for recommended actions to all PWR construction permit holders and licensees (Ref. 80). The fuel vendor also performed a sensitivity study of baffle joint gap width to joint jet velocity and established criteria for peening methods. The vendor outlined a baffle joint inspection program and recommended that it should be followed during all refueling operations (until the utility determines that an effective control of baffle jetting has been attained).

As a result of further studies and awareness through vendor and NRC communications, better understanding of the problem was obtained, and the corrective actions appear to have taken effect.

Westinghouse is implementing a design modification at the baffle joints to eliminate this type of failure for plants not yet in operation (Ref. 62).

3.1.2 Westinghouse Grid Strap Damage

During 1979, Salem-1, Salem-2, Indian Point-2, and some other PWRs reported grid strap damage on fuel assemblies (Ref. 2). The grid strap damage was caused by corner-to-corner interaction between adjacent fuel assemblies during fuel handling operations. Fuel assemblies that suffered the damage were either removed from the core or disqualified for reloading into the core. The fuel vendor, Westinghouse, recommended additional precautions to correct the problem. The NRC approved the vendor recommendations and issued IE Circular No. 80-13 (Ref. 81) on this matter to all licensees for PWRs. That circular required all licensees using Westinghouse 14 x 14, 15 x 15, and 17 x 17 fuel to visually inspect the grid straps on discharged fuel assemblies and on fuel

assemblies that are being returned to the core. It also required that the pertinent vendor recommendations on fuel handling precautions be adopted by licensees with Westinghouse fuel.

According to an investigation by Westinghouse (Ref. 82), recent refuelings using the revised fuel handling procedures proposed by Westinghouse have revealed no indication of grid damage. In addition, modifications to the grid corners of all newly fabricated fuel are expected to further reduce the probability of grid damage during fuel handling.

In October 1980, two fuel assemblies at Salem-1 were found during the Cycle 2/3 refueling to have damaged grids (Ref. 62). The damage occurred during insertion prior to Cycle 2 startup. These assemblies (and previously other fuel assemblies) with damaged grids operated satisfactorily, indicating that some degree of damage may be tolerated without attendant fuel rod defects (Ref. 62).

During the Cycle 4/5 fuel shuffle that started in December 1980, grid strap damage was found in a number of fuel assemblies at Indian Point-2; no damage to fuel rod cladding was identified (Refs. 83 and 84). The core was completely discharged and all fuel assemblies were visually inspected, using an underwater TV camera and videotaping system. An updated report (Ref. 85) indicated that 87 of the fuel assemblies were found to have damaged grid straps (19 additional assemblies had trapped debris and were subsequently cleaned). Westinghouse recommended additional fuel assembly handling precautions that minimize the corner interactions between fuel assemblies. There were 49 of these assemblies with damaged grids loaded into the core for Cycle 5. Evaluation of the damage indicated that there will be no adverse effects on the operation in Cycle 5 (Ref. 84).

3.1.3 Crud and Corrosion

3.1.3.1 Crud Deposits on PWR Fuel

Failed fuel assemblies were found at Maine Yankee during the Core 4-5 refueling operation (Ref. 2). Routine inspection (sipping) of all 217 fuel assemblies identified nine assemblies as leakers (Ref. 56). In February 1980, it was indicated that 12 fuel rods (burnups ranged from ~12,000 to

~22,000 MWd/MTU) had been identified as failed and that hydride blisters had been observed on all failed rods (Ref. 86). At that time the cause of failure was conjectured to be pellet-cladding interaction (PCI), which was then followed by secondary hydriding (Ref. 86). All the failed rods were pre-pressurized type fuel rods that contained high-density UO_2 pellets--as the NRC noted (Ref. 86), these rods were not of the same vintage as the rods that were involved in the earlier (3 to 4 years ago) PCI problems at Maine Yankee.

In March 1980, the licensee transmitted additional information (Ref. 87) to the NRC. The nine fuel assemblies, as well as others, were examined visually with the site periscope. Some of the fuel rods were eddy-current tested; profilometry measurements were made on a small number of rods. The licensee believes that a total of 15 to 25 fuel rods failed, of which 11 were positively identified. The failures of these Batch H fuel rods, which occurred at the end of Cycle 4, were most likely caused by external corrosion associated with crud deposition during the January-March 1979 time period. The licensee does not expect further degradation of the fuel during Cycle 5. The fuel vendor issued two reports based on their inspection work (Refs. 88 and 89).

Five of the nine assemblies were returned to the core for additional irradiation after all fuel rods with through-wall penetrations, as determined by visual and eddy-current examinations, were replaced (solid Zircaloy rods and new fuel rods were employed as replacements). The other four assemblies were discharged. The reactor coolant chemistry results showed a step buildup in activity levels to a higher, relatively stable, value, which suggests, according to the licensee, that the fuel rod failure mechanism was such as to limit failures to a limited number of fuel rods and was not of a core-wide degenerative nature (Refs. 56 and 90).

3.1.3.2 Corrosion of BWR Fuel

In the annual report for 1979 (Ref. 2), it was stated that 75 failed rods (71 poison rods, 4 fuel rods) were found in 31 fuel bundles at Vermont Yankee. The failures appeared to be caused by external corrosion. Information submitted by General Electric to the NRC in October 1980 (Ref. 91) indicated that the failures were associated with a metallic incursion in the feedwater. It

was also stated that this has occurred only once in the BWR operating history and is unlikely to reoccur. Attached to that submittal were the vendor's recommendations (dated September 1979) to BWR plants, which should preclude this type of event in the future.

In 1980, a submittal to the NRC for Vermont Yankee Reload 7 did not indicate how the licensee would address and resolve the previously observed phenomenon of waterside corrosion; as a result the NRC transmitted a number of questions to the licensee (Ref. 92). The NRC observed (Ref. 93) that the licensee's response (Ref. 94) appeared to be at variance^(a) with General Electric's conclusions regarding the principal cause and solution for the waterside corrosion (Ref. 91). As a result, the NRC notified the licensee that further discussion of this issue would be required to resolve this apparent divergence of opinion. In response, the licensee stated (Ref. 95) that the failures were "most probably the result of the synergistic effects of variable fuel cladding properties and metallic impurities in the reactor coolant." The major impurity associated with the failures was copper, which appeared to be coming from the copper alloy (admiralty brass) tubing in the main condenser (Ref. 96). Plants without copper-bearing main condenser tubes were thus not expected to experience cladding perforations of this kind.

3.2 NEW PROBLEMS IN 1980

3.2.1 PWR Control Rod Guide Tube Support Pin Cracking

Support pin inspections at a foreign plant revealed stress-corrosion cracking in Westinghouse-supplied support pins. There have been no previous occurrences. Some support pins are susceptible to stress-corrosion cracking due to their low solution heat treatment temperature. The cracked pins were all from one heat treatment lot of material used in foreign reactors. During manufacture, the pins were given a solution heat treatment at temperature below 1255K (1800°F). Testing by the vendor, Westinghouse, shows that a solution

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(a) The licensee stated in Reference 94 that the failures "were the direct result of an anomalous metallurgical condition..." and that "...an excursion in one or more manufacturing processes" was indicated (as the principal underlying cause of the problem).

heat treatment at a temperature of at least 1339K (1950°F) results in support pins that are highly resistant to stress-corrosion cracking.

Domestic plants, including Catawba-1 (Ref. 97), North Anna-1 (Ref. 98), Sequoyah-1 (Ref. 99), and Surry-1 (Ref. 100) were notified by Westinghouse regarding the potential for guide tube support pin cracking. This condition (i.e., cracking) appears at Catawba-1 (Ref. 97). The vendor recommended replacing the pins with ones that have been manufactured with a solution heat treatment at a temperature of at least 1366K (2000°F). All pins will be replaced.

The corrective action appears to have solved the problem, because no new incidents were reported in 1980.

3.2.2 Underpressurized Gadolinia Fuel Rods

There are typically four rods containing urania-gadolinia fuel material in an 8 x 8 BWR fuel assembly (and as many as eight in more recent designs). A number of such gadolinia fuel rods fabricated between June and October 1980 at General Electric's Wilmington facility did not meet specification requirements on helium fill gas concentration and pressure because of a malfunction in a weld station. Some of these underpressurized rods were shipped to Quad Cities-1 for Reload 5, Cycle 6 (Refs. 101-103). The NRC estimated that less than 100 rods are actually involved in the 122 suspect fuel assemblies. There were also some of these underpressurized gadolinia fuel rods in two fuel assemblies that were shipped to Vermont Yankee (Ref. 102).

The problem was evaluated by the NRC, the vendor (General Electric Company), and the licensee (Commonwealth Edison) for Quad Cities-1 (Ref. 102). It was concluded that no unreviewed safety questions are involved and that no safety hazard exists as a result of this problem. The conclusions apply to Quad Cities-1 and to Vermont Yankee.

The licensee for Quad Cities-1 was advised that some degree of on-line and end-of-cycle surveillance of these assemblies is appropriate.

After the weld station was repaired by the vendor, no new incidents have occurred.

3.2.3 PWR Hold-Down Spring Damage

During 1980, information was received by the NRC that broken hold-down springs had been observed on fuel assemblies at three domestic and two foreign PWRs. The function of hold-down springs on fuel assemblies is to provide axial restraint to prevent coolant flow forces from causing movement of the fuel assemblies.

During the May-June 1980 period, broken hold-down springs were detected (by inspection and review of video tapes) at two domestic PWRs, Davis-Besse-1 (Refs. 104-106) and Oconee-1 (Refs. 107-109). At that time, the cause of the problem had not been conclusively defined; it was presumably due to improper material condition and high cycle fatigue. The fuel vendor for the domestic plants, Babcock & Wilcox, was evaluating the problem (Ref. 107).

Ultimately, 1581 springs were examined and a total of 26 broken springs were found (Refs. 13, 108, and 110): 20 failed springs were identified in Cycle 1 fuel assemblies at Davis-Besse-1, five springs (in the 686 fuel assemblies inspected) at Oconee-1, one spring at Crystal River-3, and none at Rancho Seco, Three Mile Island-1, and Arkansas-1. The spring failure at Crystal River-3 is believed to have occurred solely by fatigue (Ref. 110).

The NRC prepared a list of questions regarding the recently detected hold down spring problem and transmitted the list to all Babcock & Wilcox licensees (Ref. 111).

As requested by the NRC letter, Toledo Edison, the licensee of Davis-Besse-1, prepared responses to questions related to this matter (Ref. 106). One response in Reference 106 indicated that six tests had been performed to determine the impact of broken hold-down springs on reactor operation and fuel integrity. It was concluded that long-term operation under the condition of damaged springs would at worst result in damage to some peripheral fuel rods, there is no possibility of damage resulting in non-insertion of control rods,

wear from horizontal vibration will not be a significant problem, and severe vibration will not result for a lifted assembly. Another response in Reference 106 indicated that the cause of the hold-down spring failures was an improper material condition characterized by a coarse outer grain structure. To date, the corrective action recommended by Babcock & Wilcox has been the replacement of all damaged springs with ones made to a current specification. The replacement springs have a uniform fine grain size, which provides increased fatigue resistance.

The NRC prepared a safety evaluation of the issues regarding Babcock & Wilcox fuel assembly hold-down spring failures (Ref. 110). The NRC considered the spring failures to be a generic issue. The NRC review indicates that the predominant mode of failure appears to have been fatigue-initiated cracking, followed by stress-corrosion crack propagation to failure of springs with an improper metallurgical condition (grain size). The NRC believes that there is reasonable assurance that the hold-down spring failures will not occur on as large a scale again, and that neither the potential for a) loss of positive hold-down force, b) loose parts, nor c) interference for normal control rod assembly movement constitute a significant safety hazard. However, because at least one spring failure (at Crystal River-3) appears not to be related to material of improper metallurgical condition (i.e., stress-corrosion cracking), and because some lateral and vertical motion of loose assemblies is theoretically possible under certain (extreme) conditions, the NRC concludes that further surveillance (e.g., video examination) of the assembly hold-down springs should be carried out at the next refueling at each plant. All the licensees have committed to such surveillance.

In 1980, broken hold-down springs were found at two foreign PWRs, which are Westinghouse Upper Head Injection (UHI) type reactors (Ref. 112). Inspection of all 132 hold-down assemblies at the first plant revealed that 31 contained springs that were either cracked or broken; inspection at the second plant disclosed 9 cracked or broken springs out of 132.

The NRC issued a request (Ref. 113) to obtain the concurrence of the foreign utility company associated with one of those PWRs, Ohi-1, in releasing information concerning the defective hold-down springs there that were

manufactured by Westinghouse. A response (Ref. 114) was subsequently received that included the following comments: the problem applies only to UHI plants, it has not been observed in any plant other than Ohi-1 (which lends credence to the UHI factor),^(a) the Ohi-1 springs are "barrel"-shaped whereas springs at domestic Westinghouse plants are not (therefore, the problem may never be encountered in domestic plants), and there was no evidence of loose parts at Ohi-1. Sequoyah and McGuire are the only domestic UHI plants, and, as of the end of 1980, neither plant had begun commercial operation^(b) (Ref. 115). Westinghouse and the licensee (Tennessee Valley Authority) for Sequoyah reviewed the problem and determined that there was "no substantial safety hazard." Subsequently, the NRC requested (Ref. 116) an assessment of the hold-down assembly spring question by the licensee (Duke Power Company) for McGuire. The licensee's response (Ref. 112) indicated that arrangements have been made with Westinghouse to substitute hold-down assembly springs of a new design (discussed below) on all McGuire nonfuel-bearing components that will be used for more than one cycle of operation (thimble plug and secondary source assemblies). Existing hold-down assembly springs will be used on McGuire nonfuel-bearing components that will be used only for the first cycle of operation (burnable poison and primary source assemblies). The Westinghouse safety evaluation (Ref. 112) indicates that the operation of one cycle with the existing hold-down assembly springs does not constitute a safety problem.

Westinghouse is cooperating with the utilities involved to minimize further use of springs of the design that failed. A modified spring design has been developed to minimize the potential for failure by fatigue (Ref. 62).

3.2.4 Iodine Spiking

Iodine spiking (i.e., a temporary increase in coolant iodine concentration) is frequently observed at reactors where leaking fuel rods are present.

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- (a) Later information (Ref. 112) indicated that broken springs were found in a second foreign UHI-type PWR.
- (b) Initial criticality was achieved at Sequoyah-1 on July 5, 1980, and electrical energy was first generated on July 22, 1980 (Ref. 117). The fuel loading date for McGuire-1 was November 1980 (Ref. 118). Initial criticality was attained at McGuire-1 on August 8, 1981, and electrical energy was first generated on September 12, 1981 (Ref. 117).

These temporary increases in iodine concentrations have been observed to occur following shutdowns, start-ups, rapid power changes, and coolant depressurization. Iodine spikes are characterized by a rapid increase in coolant concentration by as much as three orders of magnitude, followed by a return to pre-spike concentrations. The latter characteristic distinguishes the spiking phenomenon from a step-wise permanent increase in coolant activity level caused by the sudden failure of one or more fuel rods.

The NRC has developed Standard Technical Specifications (Table 13) for primary coolant iodine concentrations that make allowance for iodine spikes by permitting temporary excursions (not to exceed 48 hours) above the "equilibrium" concentration limit. For each excursion above the equilibrium limit, a Licensee Event Report is required. Four BWRs (Brunswick-1 and -2, Hatch-2, La Crosse) and approximately one-half of the operating PWRs have this type of technical specification.

During 1980, five reactors reported iodine spikes exceeding the equilibrium iodine limit: one BWR, Brunswick-2, and four PWRs, Arkansas-2, Crystal River-3, Davis-Besse-1, and St. Lucie-1.

The cause of the events is attributed to the presence of a small number of leaking fuel rods and a reactor coolant transient (e.g., reactor shutdown or the subsequent startup). However, in the case of Arkansas-2 (Ref. 119), the licensee attributed the iodine spike to a "crud burst".

On November 13, 1980, the reactor coolant system dose equivalent iodine-131 at Brunswick-2 (Ref. 120) exceeded the equilibrium Technical Specification limit of 0.2 $\mu\text{Ci/g}$ of iodine-131 (dose equivalent) because of a reactor scram and the subsequent increase in fission product inventory coming from a leaking fuel bundle. The fuel bundles were to be sipped during the next refueling outage and the leaking fuel bundles were to be removed from the core. A similar occurrence (and similar corrective action) took place in November 1979 (Ref. 121). The reactor coolant activity also exceeded the equilibrium limit three times in 1980 (as a result of reactor shutdown on December 9 and reactor startups on December 12 and 18) (Ref. 122). The coolant iodine activity was controlled in one event by maintaining a constant coolant

TABLE 13. Technical Specifications for Primary Coolant Activity^(a)

Plant	Type ^(b)	Limits on Coolant Activity		
		Dose Equivalent Iodine-131, ($\mu\text{Ci/g}$)	Other Iso- topes, Value/ $\bar{E}$ ^(c)	Standard ^(d)
Arkansas-1 ^(e)	PWR	3.5	75/ $\bar{E}$	No
Arkansas-2 ^(f)	PWR	1.0	100/ $\bar{E}$	Yes
Beaver Valley-1	PWR	1.0	100/ $\bar{E}$	Yes
Big Rock Point-1	BWR	(g)	(g)	--
Browns Ferry-1,-2, and-3	BWR	3.2	(g)	No
Brunswick-1 and-2	BWR	0.2	100/ $\bar{E}$	Yes
Calvert Cliffs-1 and-2	PWR	1.0	100/ $\bar{E}$	Yes
Cook-1 and-2	PWR	1.0	100/ $\bar{E}$	Yes
Cooper Station	BWR	3.1	(g)	No
Crystal River-3	PWR	1.0	100/ $\bar{E}$	Yes
Davis-Besse-1	PWR	1.0	100/ $\bar{E}$	Yes
Dresden-1	BWR	20.0 ^(h)	(g)	No
Dresden-2	BWR	20.0 ^(h)	(g)	No
Dresden-3	BWR	20.0 ^(h)	(g)	No
Duane Arnold	BWR	1.2	(g)	No
Farley-1	PWR	1.0	--	No
Fitzpatrick	BWR	3.1	(g)	No
Fort Calhoun-1	PWR	2.0	21/ $\bar{E}$	No
Ginna	PWR	3.0	54/ $\bar{E}$	No
Haddam Neck	PWR	(g)	68/ $\bar{E}$	No
Hatch-1	BWR	10	(g)	--
Hatch-2	BWR	0.2	100/ $\bar{E}$	Yes
Indian Point-2	PWR	(g)	60/ $\bar{E}$	No
Indian Point-3	PWR	1.0	100/ $\bar{E}$	Yes
Kewaunee	PWR	(g)	91/ $\bar{E}$	No
La Crosse	BWR	0.2	100/ $\bar{E}$	Yes
Maine Yankee	PWR	1.0	100/ $\bar{E}$	Yes
Millstone-1	BWR	20.0 ^(h)	(g)	No
Millstone-2	PWR	1.0	100/ $\bar{E}$	Yes
Monticello	BWR	5.0	(g)	No
Nine Mile Point-1	BWR	25.0 ^(h)	(g)	No
North Anna-1 and-2	PWR	1.0	100/ $\bar{E}$	Yes
Oconee-1,-2, and-3	PWR	(g)	224/ $\bar{E}$	No
Oyster Creek-1	BWR	8.0 ^(h)	(g)	No

TABLE 13. (contd)

Plant	Type ^(b)	Limits on Coolant Activity		Standard ^(d)
		Dose Equivalent Iodine-131, ($\mu\text{Ci/g}$)	Other Isotopes, Value/ $\bar{E}$ ^(c)	
Palisades	PWR	1.0	100/ $\bar{E}$	Yes
Peach Bottom-2 and-3	BWR	2.0	(g)	No
Pilgrim-1	BWR	20.0 ^(h)	(g)	No
Point Beach-1 and-2	PWR	(g)	162/ $\bar{E}$	No
Prairie Island-1 and-2	PWR	(g)	27/ $\bar{E}$	No
Quad Cities-1 and-2	BWR	5.0	(g)	No
Rancho Seco-1	PWR	(g)	43/ $\bar{E}$	No
Robinson-2	PWR	1.0	100/ $\bar{E}$	Yes
Salem-1	PWR	1.0	100/ $\bar{E}$	Yes
San Onofre-1	PWR	1.0	100/ $\bar{E}$	Yes
Saint Lucie-1	PWR	1.0	100/ $\bar{E}$	Yes
Surry-1 and-2	PWR	1.0	43/ $\bar{E}$	No
Three Mile Island-1	PWR	(g)	130/ $\bar{E}$	No
Three Mile Island-2	PWR	1.0	100/ $\bar{E}$	Yes
Trojan	PWR	1.0	100/ $\bar{E}$	Yes
Turkey Point-3 and-4	PWR	1.0	135/ $\bar{E}$	No
Vermont Yankee-1	BWR	1.1	(g)	No
Yankee Rowe-1	PWR	1.0	100/ $\bar{E}$	Yes
Zion-1 and-2	PWR	(g)	57/ $\bar{E}$	No

(a) As of September 1980.

(b) Pressurized water reactor (PWR), boiling water reactor (BWR).

(c) $\bar{E}$ = average disintegration energy, (MeV).

(d) Standard Technical Specifications:

BWR Activity:

0.2 $\mu\text{Ci/g}$, Dose Equivalent Iodine-131, and 100/ $\bar{E}$ $\mu\text{Ci/g}$, equilibrium

4.0 $\mu\text{Ci/g}$, Dose Equivalent Iodine-131 during iodine spikes

PWR Activity:

1 $\mu\text{Ci/g}$, Dose Equivalent Iodine-131, and 100/ $\bar{E}$ $\mu\text{Ci/g}$, equilibrium

60 $\mu\text{Ci/g}$, Dose Equivalent Iodine-131 during iodine spikes (some plants have higher spiking limits for decreasing power levels).

(e) Arkansas Nuclear One-Unit 1.

(f) Arkansas Nuclear One-Unit 2.

(g) Not available or nonexistent.

(h) Technical Specification stated as total iodine ($\mu\text{Ci/g}$) instead of equivalent iodine-131.

temperature and in two events by limiting reactor power. To eliminate future similar occurrences, the fuel bundles will be sipped by the licensee during the next refueling outage and the leaking fuel bundles removed from the core.^(a)

The reactor coolant system dose equivalent iodine-131 at Crystal River-3 exceeded the equilibrium Technical Specification 3.4.8 limit of 1.0 $\mu\text{Ci/g}$ on December 21, 1979 (Ref. 123), February 8, 1980 (Ref. 124), February 9 and 11, 1980 (Ref. 125), and December 8, 1980 (Ref. 126). The five events were caused by leaking fuel rods in combination with a reactor coolant system transient (i.e., power reduction, plant trip, shutdown, or startup). The event in December 1980 was the nineteenth occurrence reported under this specification. References 123 and 125 indicate that no corrective action was deemed necessary by the licensee as the coolant system purification returned the dose equivalent iodine-131 to within acceptable limits. Reference 124 and 126 indicate that the licensee's corrective action was to increase the reactor coolant purification flow to aid cleanup. In 1980, the Core Performance Branch of NRC (Ref. 127) attempted to learn more about the nature of the fuel failures but has been unable to determine the exact number or types of fuel failures.

On December 3, 1980, the dose equivalent iodine-131 level at Davis-Besse-1 (Refs. 128 and 129) exceeded the equilibrium Technical Specification 3.4.8a limit of 1.0 $\mu\text{Ci/g}$. The cause was a slight leakage of gas through the fuel rod cladding. The licensee also indicated that some leakage is not unusual following a reactor trip (as in this case) when the gases in the fuel rods are further compressed by the contracting cladding. The levels were monitored by the licensee to ensure that the coolant activity decayed off to below limits. There have been no prior reports of high iodine levels at Davis-Besse-1.

During a short (12-hr) shutdown on August 14, 1980, at St. Lucie-1 (Ref. 130), while in a startup mode, the dose equivalent iodine exceeded the equilibrium Technical Specification 3.4.8.a limit of 1.0 $\mu\text{Ci/g}$. After an extended period of power operation with a nominal level of fuel leakage, a dropped control element assembly and the plant shutdown were sufficient

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(a) As of December 31, 1980 (Ref. 115) and August 1, 1981 (Ref. 117), the next refueling outage had not been scheduled yet.

transients to cause the iodine build-up (iodine spiking phenomenon). The dose equivalent iodine-131 level decreased to below the limit after 8.5 hours.

The licensee stated in a January 1980 report (Ref. 56), that they suspected that failed fuel rods were present in Maine Yankee, based on the results of reactor coolant chemistry monitoring through Cycle 4. In 1979, the reactor coolant dose equivalent iodine exceeded the Technical Specification 3.2 limit of 1.0 $\mu\text{Ci/g}$ several times, as noted in Licensee Event Reports 79-010, 79-017, 79-029 (Ref. 55), and 79-030 (Ref. 57). The increases in iodine levels were anticipated during the load reductions (i.e., shutdowns) because of the known existence of minor degradation of fuel rod cladding. The licensee had established a monitoring program to follow the iodine trends and planned to conduct fuel sipping during the next refueling outage (Refs. 55 and 57). The fuel was inspected and the results are described in Section 3.1.3.1.

At Arkansas-1, which does not have the Standard Technical Specifications for coolant activity, the average gross gas release rate during the first quarter of 1980 (Ref. 131) exceeded Environmental Technical Specification 2.4.2.1 and the average gross gas release rate during the third quarter of 1980 (Refs. 132 and 133) exceeded Environmental Technical Specification 2.4.2.3a. Reference 131 indicated that the activity levels were higher than previous levels due to an apparent increase in failed fuel (less than 0.1% failed fuel) and that the fuel should be replaced within two cycles. Reference 132 indicated that the activity was higher than in the past because of approximately 0.08% failed fuel and that the problem will be corrected when the failed fuel is removed from service. Reference 134 indicated that the quarterly releases were significantly impacted by two shutdowns during the third quarter and also the presence of approximately 0.19% failed fuel and that the licensee will make an attempt to identify, replace or repair the failed fuel at the next refueling outage. In February 1981, the licensee inspected the full core of 177 fuel assemblies by wet sipping (a standard gamma-spectrometer was used to identify iodine-131, cesium-134, and cesium-137 isotopes and quantify their activities in the liquid sipping samples) and identified 24 fuel assemblies with leaking fuel rods and tentatively planned to

reinsert five of those assemblies for operation in Cycle 5 in the core outer periphery. The above information is being evaluated by NRC staff (Ref. 135).

Although iodine spiking has not caused serious problems, the NRC and the industry have already taken the initiative to encourage the investigation of the cause and impact of iodine spiking. One area of interest is to predict from iodine spiking when and where fuel failures occur. Babcock & Wilcox reported (Ref. 136) on a study of this subject for EPRI in October 1980. The on-line monitor, a minicomputer-controlled system, used in the study indicates the iodine fission product concentrations in the coolant as a function of time. By installing the monitor at Oyster Creek, it was possible to determine the presence of fuel defects in the core through interpretation of the data.

3.2.5 Alpha Activity in Coolant

Gross alpha activity in the primary coolant at La Crosse (Refs. 137-139) exceeded the Technical Specification level during reactor shutdown. The high alpha activity indication was temporary and did not indicate recent fuel cladding degradation. Reference 138 indicates that this is the fourth occurrence (other three are noted in Licensee Event Reports 80-06, 79-10, and 78-05). The activity is associated with the irradiated fuel material in the reactor coolant system, which is a result of the degraded fuel rod cladding experienced during previous fuel cycles. Also, work was done on the forced circulation system during the shutdown and this resulted in a more general distribution of suspended alpha-bearing material in the coolant. Operation of the purification system for the primary coolant reduced the concentration to an allowable value.

3.2.6 Misoriented Fuel Assemblies

There have been instances of misoriented fuel assemblies in previous years. Three cases occurred in 1980.

A peripheral fuel assembly at Browns Ferry-1 (Ref. 140) was found to be rotated 90 degrees from required orientation during a review of core verification video tapes. During the previous refueling outage at the beginning of Cycle 4, that fuel assembly was improperly oriented. There was no activity release; there was ample margin to operating limits to ensure that limits were not exceeded. A revision of core verification methods is being implemented.

Two fuel assemblies at Browns Ferry-2 (Ref. 141) were found misoriented 90 degrees out of proper orientation during fuel shuffling operations. These two misoriented assemblies were detected during initial reloading verification but were not corrected. The fuel assemblies operated within the bounds of the reload licensing submittal. Core verification procedures will be improved to prevent recurrence.

A fuel assembly at Oyster Creek-1 (Refs. 142 and 143) was found to be mis-oriented 180 degrees during a fuel bundle clip inspection prior to sipping. The assembly was not properly seated in its core cell location. The cause was attributed to personnel error and inadequate procedural controls. Revisions to the procedures were made (e.g., a fuel bundle orientation map is now included). The fuel assembly was inspected by the vacuum sipping method; the results indicated that no cladding failure had occurred.

3.2.7 Other Occurrences

A fuel assembly at Duane Arnold (Ref. 144) was loaded into the cell without having the neighboring control rod fully inserted. The incident was caused by personnel error. The fuel handling procedure is now being revised for clarity, and personnel will be retrained as necessary.

During fuel shuffling operations at Browns Ferry-2 (Ref. 145), fuel was placed in a cell with the adjacent control rod withdrawn. This was caused by a deficient procedure for fuel support casting replacement. The control rod was subsequently inserted and refuel interlocks placed in service. The procedure will be revised to correct the deficiencies.

A fuel assembly at Quad Cities-1 (Ref. 146) was moved prematurely during the end of Cycle 5 core unload. The cause was a misunderstanding of a recent Technical Specification change. Corrective action will be taken so that the use of changes to Technical Specifications will be restricted until either the advance copy or final printed copy is onsite and distributed.

The measured F values (ΔH) at North Anna-1 (Ref. 147) for control rod insertion exceeded design value tolerance because of the presence of a quadrant power tilt, identified through incore flux map analyses, and is believed to be the result of non-symmetric isotopic distribution in the fuel caused by the

Cycle 1 quadrant power tilt. Control rod insertion limits have been revised. An investigation is underway to determine the primary cause of the power tilt.

Two sealed specimens and fuel racks in the spent fuel pool at Kewaunee-1 (Ref. 148) exhibited swelling. The cause was attributed to gaseous products of radiolysis of materials used in boron carbide plates in spent fuel pool canisters and rack specimens. The immediate action taken was to rearrange the spent fuel bundles in a "checkerboard" pattern. An investigation is underway by the licensee to determine long-term corrective actions.

The spent fuel pool level at Davis-Besse-1 (Ref. 149) was found 15.9 cm (6.25 in.) below Technical Specification requirements (there must be 7.0 m or 23 ft of water above the top of the assemblies) due to an improper alarm set point. Water level was restored in 2 hr. The set point was changed.

Deficiencies in documentation of the design and fabrication of the high density spent fuel storage racks for Watts Bar-1 (Ref. 150) were found in an audit of the supplier's quality assurance program. Possible radionuclide release could result from a seismic event which collates the racks. Investigation of this matter is underway.

3.3 OLD PROBLEMS (1979) THAT DID NOT REOCCUR OR THAT WERE SOLVED

3.3.1 PWR Guide Tube Wear

Wear of control rod guide tubes has been observed in discharged PWR fuel assemblies since 1973. The cause of this wear was identified as flow-induced vibration of control rods. A survey of the measures taken by both the industry and the NRC to deal with this matter is described in a recent report (Ref. 71). A summary of the events that occurred during 1979 was presented in Reference 2. The licensing actions during 1979 are discussed in Reference 151.

Several vendors and licensees have taken corrective actions to alleviate the guide tube wear problem in 1980: Combustion Engineering has placed flow-modifying test assemblies in five reactors, performed surveillance on guide tube integrity in five reactors, and concluded that the use of sleeves (recommended in 1979) continued to be a satisfactory preventive method. An owner's group of Babcock & Wilcox reactors was formed and submitted the details of a

surveillance program conducted in Oconee-1 and -3 and Rancho Seco. The program is being evaluated by the NRC. Exxon Nuclear performed surveillance on fuel assemblies discharged from H. B. Robinson-2, and reported that the measured wear did not extend through the guide tube walls.

Because of effective hardware modifications and satisfactory confirmatory surveillance, guide tube wear is considered to be under effective control. However, in 1980 the Core Performance Branch of NRC requested additional information on guide tube wear characteristics for use in reviewing applications for North Anna-2 (Ref. 152), Diablo Canyon-1 and -2 (Ref. 153), McGuire-1 and -2 (Ref. 154), and Sequoyah-1 and -2 (Ref. 155). The NRC prepared updated inputs for the Safety Evaluation Reports (SERs) for North Anna-2 (Ref. 156), Diablo Canyon-1 and -2 (Ref. 157), and Salem-2 (Ref. 158), which closes out the issue of guide tube thimble wear in the operating license applications for those plants. In those references, the NRC staff concluded that the Westinghouse analysis probably accounts for all major variables controlling this wear process. However, because of the complexities and uncertainties in determining a) contact forces, b) surface-to-surface wear rates, c) forcing functions, and d) extrapolations of these variables to other fuel designs (such as the 17 x 17 fuel assembly design in North Anna-1, Diablo Canyon-1 and -2, and Salem-2), the NRC believes it is prudent for each of the applicants to participate in a surveillance plan for the examination of guide thimble wear.

3.3.2 Westinghouse Control Spider Rodlet Failures

The cause of this type of failure was determined by the NRC to be a manufacturing problem and not a generic design error. No failure of this type was reported during 1980.

3.3.3 Other Problems in 1979 That Did Not Reoccur in 1980

Some problems in 1979 (Ref. 2) did not reoccur in 1980. Those problems, plus recent information of interest, are shown below:

- Cracking of Stainless Steel Cladding on PWR Fuel Rods

In February 1979, Haddam Neck (Connecticut Yankee) reported detecting (by sipping and visual inspection during the Cycle 9 refueling

outage) 36 leakers among 48 Core 8, Batch 8 fuel assemblies (Ref. 2). The NRC noted (Ref. 159) that the most surprising result from the postirradiation examination was the observation of several incipient cracks (i.e., cracks that do not extend through the cladding wall) that initiated from the outer surface of the stainless steel cladding, in contrast to typical pellet-cladding-interaction (PCI) cracks in Zircaloy cladding that initiate on the inner surface of the cladding.

Preliminary evaluations and conclusions from the hot cell examination are presented in a November 1980 interim report (Ref. 160). In each of three specimens, 1) the cracks in the cladding originated from the outer surface, 2) the cracks were intergranular in nature, and 3) a fuel chip wedged in the fuel-cladding gap was noted directly across from the crack in the cladding. The investigators "tentatively concluded that high local cladding stresses, caused by the presence of the fuel chip, were the primary cause of cracking in the cladding and that the fuel rod failures may have occurred by the same mechanism. Additional work is in progress to verify this conclusion."

The licensee (Connecticut Yankee Atomic Power Company) evaluated iodine-131 activity trends: activity decreased during Cycle 9 when compared to Cycle 8; Cycle 9 activity was still higher than Cycle 7 and two to ten fuel rods were predicted to be defective (Ref. 161). Although the coolant activity was not limiting, the licensee inspected fuel during the Core 10 refueling outage. All 52 Batch 9 discharged fuel assemblies were sipped: 11.5% were found to be leaking (6 were leaking, 1 was possibly leaking, and 45 were sound). Seventeen of the assemblies (included the seven known or possible leakers) were visually inspected by periscope; the visual inspection was documented with color photographs. Approximately 950 fuel rods were visually inspected and no failures (as exhibited in Core 8, Batch 8 fuel) were found and no failures of the fuel rod cladding similar in visual characteristics to those observed during the inspection of Core 8, Batch 8 fuel were apparent. The licensee

indicated that no one parameter for the Batch 9 fuel assemblies can be identified conclusively as a common parameter affecting the probability of failure; however, burnup seems to have some bearing on the failure probability (e.g., the failures were associated with fuel assemblies that had attained burnups greater than 33,000 MWd/MTU). Further investigations are underway by the licensee.

- BWR Water Rod Lower End Plug Wear

In October 1979, water rod end plug wear was observed at Vermont Yankee (Ref. 2). The NRC issued an information notice (Ref. 162) regarding this wear problem. Water rod lower end plug inspection results are described by General Electric in Reference 163. In 1980, a submittal for Vermont Yankee Reload 7 did not indicate how the licensee would address and resolve this problem; as a result, the NRC transmitted a number of questions to the licensee (Ref. 92). In response, the licensee indicated (Ref. 94) that the lower end plug configuration had been changed "to incorporate the design of earlier fuel types which have not exhibited wear" and that future end plug inspections would be consistent with General Electric's recommendations as part of their generic investigation into the problem.

- BWR Control Blade Cracking

Movable blades comprised of many small rods containing B_4C are used to control power in BWRs. As noted in the previous annual report (Ref. 2), cracks in control rod cladding have been observed recently. However, no Licensee Event Reports involving BWR control rod cracking were submitted to the NRC in 1980. The early estimate of control blade life was ~ 12 cycles, a later estimate was ~ 8 cycles, and, as a result of the recently observed cracking, the current estimate is 6.4 cycles (Ref. 164).

A recent paper (Ref. 165) describes the results of the postirradiation examination of a defective control rod (had cracks in cladding of the B_4C absorber rods) that had been irradiated for eight years in a foreign BWR, KRB(A).

- Extensive Core Damage (Three Mile Island-2)

Three Mile Island-2 (TMI-2) has remained shut down since the accident on March 8, 1979, which caused extensive core damage (Ref. 115). Authority to operate the facility was suspended by the NRC on July 20, 1979, which requires the licensee to maintain the facility in a shutdown condition. As described in the previous report (Ref. 2), the accident prompted a large number of investigations into the accident, the behavior of the reactor during the accident, and related subjects. Summaries of many papers dealing with the accident or related studies have been published recently: there are 55 in Reference 166, 20 in Reference 167, and 3 in Reference 168. The final definition of the damage to the core of TMI-2 will have to wait until a detailed examination of the TMI-2 core and primary system has been performed (Refs. 169-171). The scenario that the NRC believes to be the most likely for the progression of the damage to the TMI-2 core is described in a recent paper (Ref. 172). In 1980, EPRI issued a bibliography of material related to Three Mile Island-2 (Ref. 173).

4.0 FUEL DESIGN CHANGES

Several programs are assessing the commercial acceptability of improved fuel concepts in an effort to solve the pellet-cladding interaction (PCI) problem^(a) and to improve uranium utilization.^(b)

The ultimate objective of a program at General Electric (Refs. 174 and 175) is to demonstrate an advanced fuel design that is resistant to the PCI fuel failure mechanism. Two advanced fuel concepts are being developed for possible demonstration within this program: a) Cu-barrier fuel and b) Zr-liner fuel. Four lead test assemblies of barrier fuel (two of Zr-liner and two of Cu-barrier) were examined at the end of one full cycle of operation in Quad Cities-1 (Table 2). The bundle average burnup was 11,700 MWd/MTU. The program intends to include a total of 132 barrier bundles in the reload for Cycle 6 at Quad Cities-2.

In the Consumers Power Company/ENC/PNL Fuel Performance Improvement Program (Refs. 176-178), an annular-coated-pressurized design and a sphere-pac design are being evaluated by means of steady-state and ramping irradiations and their performance is being compared with a reference fuel design. Four fuel assemblies containing demonstration fuel rods were inserted in Big Rock Point during the spring of 1979 (see "rods" under ENC in Table 2).

As shown in Table 2, irradiation of other fuel assemblies with design changes (e.g., use of an optimized design, low absorption spacer grids, or axially blanketed fuel columns) is underway.

The EPRI programs (Ref. 52) were discussed in Section 2.2.6.

The Department of Energy (DOE) has programs that involve extended burnup demonstrations of current product-line fuel. DOE is sponsoring (or has sponsored) such programs with the following (utility/fuel vendor/reactor):

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- (a) PCI is the principal concern regarding fuel failure at higher burnup with present LWR fuel (Ref. 179). The PCI phenomenon is generic to reactors using Zircaloy-clad UO₂ fuel (Ref. 180).
  - (b) For example, uranium utilization can be improved by having fuel go to higher burnup, by reducing neutron absorption, or by reducing neutron leakage.

- Duke Power Company/Babcock & Wilcox/Oconee-1
- Omaha Public Power District/Combustion Engineering/Fort Calhoun
- Arkansas Power & Light Company/Combustion Engineering/Arkansas-2
- Virginia Electric & Power Company/Westinghouse/Surry-2
- Northern States Power Company/General Electric/Monticello
- Consumers Power Company and GPU/Exxon Nuclear/Oyster Creek and Big Rock Point
- --/Westinghouse/BR-3

Five Mark B (15x15) PWR fuel assemblies designed by Babcock & Wilcox were irradiated to extended burnups in Oconee Unit 1 (Ref. 25). An assembly average burnup of 40,000 MWd/MTU was attained. Efforts during this period included performance tracking of the extended burnup assemblies, nondestructive examination of the four-cycle assemblies, and analytical work to identify and quantify the factors that limit fuel assembly lifetime.

B&W, DOE, and Sacramento Municipal Utility District are participating in an axial blanket fuel demonstration program (Ref. 181), which is part of U.S. effort to improve uranium utilization in LWRs.

The Improved Uranium Utilization Program at General Electric (Ref. 182) includes a Phase I evaluation of BWR design alternatives that have potential to reduce uranium ore requirements in once-through fuel cycles. Emphasis is being placed on developing practical alternatives which can be demonstrated at the earliest possible date in existing operating BWR nuclear power stations. Priority is being given to alternatives which have the maximum payoff, measured by reduction of the nation's uranium resource requirements, on the shortest schedule.

The purpose of one program at Combustion Engineering (Ref. 183) is to demonstrate two techniques for improving fuel utilization in current PWRs: 1) more efficient fuel management, and 2) high burnup. These demonstrations are being performed in the Fort Calhoun reactor. The current Phase I program will be accomplished in two parts: one to demonstrate SAVFUEL (Shimmed And

Very Flexible Uranium Element Loading), which is expected to reduce uranium requirements by two to four percent (2% to 4%), and the other to demonstrate high burnup.

The purpose of another program that is sponsored by DOE at Combustion Engineering (Ref. 184) is to develop gadolinium as a burnable poison in PWR extended-burnup, low-leakage fuel management schemes that achieve improved uranium utilization.

DOE, Virginia Electric and Power Company, and Westinghouse are participating in an Extended Fuel Burnup Demonstration Program (Ref. 185) to improve the utilization of fuel resources in LWRs.

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