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Fuel Performance Annual Report for 1983

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Prepared by W. J. Bailey/PNL
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Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory
Commission

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Fuel Performance Annual Report for 1983

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ABSTRACT

This annual report, the sixth in a series, provides a brief description of fuel performance during 1983 in commercial nuclear power plants. Brief summaries of fuel design changes, fuel surveillance programs, fuel operating experience, fuel problems, high-burnup fuel experience, and items of general significance are provided. References to additional, more detailed information and related NRC evaluations are included.

CONTENTS

ABSTRACT.....	iii
1.0 INTRODUCTION.....	1
2.0 FUEL SURVEILLANCE REQUIREMENTS.....	3
3.0 FUEL DESIGN CHANGES AND SUMMARY OF FUEL SURVEILLANCE PROGRAMS.....	7
4.0 FUEL OPERATING EXPERIENCE.....	15
4.1 BABCOCK & WILCOX COMPANY.....	15
4.2 COMBUSTION ENGINEERING, INC.....	19
4.3 EXXON NUCLEAR COMPANY, INC.....	19
4.4 GENERAL ELECTRIC COMPANY.....	23
4.5 WESTINGHOUSE ELECTRIC CORPORATION.....	27
4.6 ELECTRIC POWER RESEARCH INSTITUTE.....	27
5.0 PROBLEM AREAS OBSERVED DURING 1983.....	35
5.1 PROBLEMS IN 1983 THAT ARE SIMILAR TO THOSE IN 1982.....	35
5.1.1 Debris-Induced Fretting.....	35
5.1.2 Iodine Spiking or Radioactive Gas Release.....	37
5.1.3 PWR Fuel Assemblies Damaged by Baffle Jetting.....	53
5.1.4 Cladding Degradation.....	54
5.1.5 BWR Fuel Assemblies Loaded Out of Proper Orientation.....	55
5.1.6 Control Rod Guide Tube Support Pins.....	56
5.1.7 PWR Hold-Down Springs.....	57
5.1.8 Closing of PWR Rod Shoulder Gap.....	59
5.1.9 Deformation of PWR Spent Fuel Storage Rack.....	60

5.2	NEW PROBLEMS IN 1983.....	60
5.2.1	Fuel Rods in a PWR Optimized Fuel Assembly Damaged by Fretting.....	60
5.2.2	Cladding Wear on PWR Rod Cluster Control Assemblies.....	61
5.2.3	Binding of Components and Top Nozzle Fretting in PWR Fuel Assemblies.....	61
5.2.4	Fuel Investigations at Two B&W Reactors.....	62
5.2.5	Misloaded Pellets in a PWR Fuel Assembly.....	63
5.2.6	PWR Fuel Assembly and RCCA Damaged During Placement of Upper Internals in the Reactor Vessel.....	65
5.2.7	PWR Fuel Assemblies Fall or are Dropped.....	65
5.2.8	Heat Flux Spike at a BWR.....	66
5.2.9	Control Rods Inserted in Reverse Order at a BWR.....	67
5.2.10	Overflowing or Partial Draining of Water from Refueling Cavity and/or Spent Fuel Storage Pool.....	67
5.2.11	Other Problems.....	68
5.3	OLD PROBLEMS THAT DID NOT RECUR OR THAT WERE SOLVED.....	75
5.3.1	BWR Fuel Rod Breakage During Visual Inspection.....	75
5.3.2	Hydriding of PWR Fuel Rods.....	75
6.0	SUMMARY OF HIGH-BURNUP FUEL EXPERIENCE.....	77
7.0	ITEMS OF GENERAL SIGNIFICANCE.....	81
8.0	REFERENCES.....	83

FIGURES

1	Schedule for EPRI Fuel Performance Surveillance Program.....	11
2	Exposure of Irradiated ENC Fuel as of December 31, 1983.....	24
3	Assemblywise Performance of Westinghouse Zircaloy-Clad Fuel.....	31
4	Westinghouse Zircaloy-Clad Fuel Experience As of December 31, 1983.....	33
5	Domestic BWR Fuel Burnup Experience.....	78
6	Domestic PWR Fuel Burnup Experience.....	79

TABLES

1	Typical Fuel Assembly Parameters.....	4
2	Major Fuel Performance Programs: Status Through 1983.....	8
3	Various Fuel Rod Types That Have Been or Are Being Irradiated in Big Rock Point Reactor Under DOE's Fuel Performance Improvement Program.....	12
4	1983 Performance Summary for B&W-Supplied Fuel Rods.....	16
5	Operating Status of B&W Fueled Reactors.....	17
6	Summary of Burnup Experience Through December 31, 1983, for B&W- Supplied Fuel.....	18
7	Summary of Burnup Experience Through December 31, 1983, for B&W- Supplied Stainless Steel-Clad Fuel.....	19
8	Summary of Combustion Engineering Fuel Irradiated and/or Discharged in 1983.....	20
9	Combustion Engineering Burnup With All-Zircaloy Assemblies: Status as of December 31, 1983.....	22
10	Typical Equilibrium Iodine-131 in the Primary Coolant of Combustion Engineering Reactors During 1983.....	23
11	Summary of Exxon Nuclear Fuel Assembly Distribution, Fuel Rod Performance, and Gadolinia Fuel Experience, as of December 31, 1983.....	25
12	Summary of ENC's 1983 Fuel Failures and Repairs.....	26
13	General Electric Fuel Experience Summary.....	26
14	Summary of General Electric Lead Test Assembly Surveillance Programs.....	28
15	Assemblywise Burnup Distribution of Westinghouse Zircaloy-Clad Fuel Rods: Status as of December 31, 1983.....	29
16	Westinghouse Fuel Performance Status Report - 1983.....	32
17	Technical Specifications for Primary Coolant Activity.....	38
18	Iodine Spiking Events.....	41
19	Data on Iodine Spiking Events at Calvert Cliffs-2 in 1983.....	42

20	Data Associated with LER 83-002 at Davis Besse-1.....	43
21	Iodine Spiking Event on May 28, 1983, at Millstone-2.....	45
22	Iodine Spiking Event on Feb. 19, 1983, at Millstone-2.....	45
23	Iodine Spiking Event on Dec. 31, 1982, at Millstone-2.....	45
24	Iodine Spiking Event on Nov. 18, 1983, at North-Anna-1.....	47
25	Iodine Spiking Event on Sept. 28, 1983, at North-Anna-1.....	47
26	Iodine Spiking Event on July 23, 1983, at North-Anna-1.....	47
27	Iodine Spiking Event on July 10, 1983, at North-Anna-1.....	48
28	Iodine Spiking Event on June 6, 1983, at North-Anna-1.....	48
29	Iodine Spiking Event on May 28, 1983, at North-Anna-1.....	48
30	Iodine Spiking Event on May 9, 1983, at North-Anna-1.....	49
31	Iodine Spiking Event on April 16, 1983, at Prairie Island-2.....	50
32	Iodine Spiking Event on Jan. 8, 1983, at St. Lucie-1.....	51
33	Iodine Spiking Event on Dec. 15, 1983, at Surry-1.....	52
34	Iodine Spiking Event on Nov. 22, 1983, at Surry-1.....	53
35	Iodine Spiking Event on Feb. 7, 1983, at Surry-1.....	53

1.0 INTRODUCTION

Monitoring the in-reactor performance of nuclear fuel in commercial light-water power reactors yields important feedback for safety considerations and licensing procedures. 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 sixth, 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 third, NUREG/CR-2410 (PNL-3953) (Ref. 3), covered calendar year 1980. The fourth, NUREG/CR-3001 (PNL-4342) (Ref. 4), covered calendar year 1981. The fifth, NUREG/CR-3602 (PNL-4817) (Ref. 5), covered calendar year 1982.

As noted in the first report (Ref. 1) of this annual series, 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. 6). However, over the years the material covered in these reports has changed. The 1971 version of the guide 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 abnormal degradation of fuel cladding and an indication of failed fuel were reportable items.^(a) 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. In May 1982, the NRC proposed amending its regulations to improve the information received in Licensee Event Reports (LERs) from nuclear power plant licensees (Refs. 7-11). The new regulation became effective on Jan. 1, 1984. Normal operation surveillance results, generic problems, and design trends are not addressed in the NUREG series of reports entitled "Nuclear Power Plant Operating Experience" (Refs. 12-18). Results of plant operating experience are also screened by the Electric Power Research Institute (Refs. 19 and 20).

(a) A report published in 1981, NUREG/CR-1380 (PNL-3325) (Ref. 21), elaborates on the reporting of abnormal degradation and fuel failures. The threshold for what constitutes abnormal degradation is not uniform throughout the industry. Therefore, the degree of degradation reported has not been 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 as these considerations change. Definitions of fuel damage, failure, and coolability, as these terms are applied in the NRC's review of fuel system designs, are provided in Section 4.2, Fuel System Design, of the NRC's Standard Review Plan (Ref. 22).

As a result, the primary intent of this report series is to summarize fuel design changes, fuel surveillance programs, fuel operating experience, fuel system problems--especially generic ones--that are of concern during the reporting period, high-burnup fuel experience, and items of general significance. 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 1983, includes some overlap with the previous year. For those problems first encountered prior to 1983, some pre-1983 information will be included for the sake of continuity. In addition, information received or action taken in early 1984 will be included if pertinent to the discussion of problem areas.

2.0 FUEL SURVEILLANCE REQUIREMENTS

Section 4.2, Fuel System Design, of the Standard Review Plan (Ref. 22) requires that plans for testing, inspection, and surveillance of fuel be submitted and reviewed for each domestic nuclear power plant. The 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). Ref. 23 contains a supplementary fuel surveillance program appropriate for new fuel designs.

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 shifting 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 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).^(b)

-
- (a) The terms "fuel assembly" and "fuel bundle" are used interchangeably by the nuclear industry, although generally the former term is associated with fuel for PWRs and the latter term with fuel for BWRs. A BWR fuel assembly consists of a fuel bundle and the open-ended channel that encloses the bundle.
- (b) 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).

TABLE 1. Typical Fuel Assembly Parameters

Vendor	B&W	B&W	B&W	C-E	C-E	W	W	W	ENC	ENC	ENC	GE	GE	GE
Fuel Rod Array	15x15	15x15	17x17	14x14	16x16	14x14	15x15	17x17	15x15	8x8	9x9	7x7	8x8	8x8 R
Reactor Type	PWR	PWR	PWR	PWR	PWR	PWR	PWR	PWR	PWR	BWR	BWR	BWR	BWR	BWR
Assemblies per Core	157	177	205	217	177	121	193	193	193	560	724	764	560	560
Fuel Rod Locations Per Assembly	225	225	289	196	256	196	225	289	225	64	81	49	64	64
Fuel Rods Per Assembly	204	208	264	176	236	179	204	264	204	60	80	49	63	62
Empty Locations Per Assembly	21	17	25	20	20	17	21	25	21	4	1	None	1	2
Rod Pitch, mm (in.)	14.3 (0.563)	14.4 (0.568)	12.8 (0.502)	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 (0.642)	14.5 (0.572)	18.7 (0.738)	16.3 (0.640)	16.3 (0.640)
System Pressure, MPa (psia)	13.9 (2015)	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)	7.07 (1026)	7.14 (1035)	7.14 (1035)	7.14 (1035)
Core Average Power Density, kW/liter	82.3	91.4	107.3	78.5	96.4	95.6	98.1	104.7	98.1	40.57	46	50.732	50.51	49.15
Average LHGR, ^(a) kW/M (kW/ft)	18.2 (5.55)	20.3 (6.20)	18.8 (5.73)	20.0 (6.09)	18.2 (5.54)	20.3 (6.20)	22.0 (6.70)	17.8 (5.44)	22.0 (6.70)	15.2 (4.63)	12.1 (3.68)	23.1 (7.049)	17.9 (5.45)	17.7 (5.38)
Axial Peak LHGR, in an Average Rod, kW/M (kW/ft)	21.7 (6.62)	24.4 (7.44)	22.6 (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)	17.5 (5.34)	27.72 (9.16)	21.48 (7.09)	21.24 (6.99)
Max. Peak LHGR, kW/M (kW/ft)	50.3 (15.30)	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)	37.7 (11.5)	60.2 (18.35)	44.0 (13.4)	44.0 (13.4)
Max. Fuel Temp., °C (°F)	2010 (3650)	2340 (4245)	2090 (4155)	2140 (3890)	1880 (3420)	2260 (4100)	2340 (4250)	1870 (3400)	2200 (3997)	2040 (3700)	2040 (3705)	2440 (4430)	1830 (3325)	1890 (3435)
Core Average Enrichment, wtt ²³⁵ U	4.0 ^(b)	3.30 ^(b)	3.15 ^(b)	3.89 ^(b)	2.36	2.90	2.80	2.60	3.02	2.65	2.8	2.19	1.80	1.99
Max. Local Exposure Mwd/MTU ^(c) GJ/kgU ^(d)	55,000 4752	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	55,000 4752	40,000 3456	40,000 3456	45,000 3888
Cladding Material ^(e)	304SS	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-2	Zry-2	Zry-2	Zry-2	Zry-2

TABLE 1. (contd)

Vendor	B&W	B&W	B&W	C-E	C-E	W	W	W	ENC	ENC	ENC	GE	GE	GE
Fuel Rod Length, m (in.)	3.22 (126.7)	3.90 (153.7)	3.88 (152.7)	3.71 (145.9)	4.09 (161.0)	3.87 (152.4)	3.80 (149.7)	3.85 (151.6)	3.86 (152.0)	3.99 (156.9)	3.99 (157.2)	4.09 (161.1)	4.09 (161.1)	4.20 (185.4)
Active Fuel Height, m (in.)	3.061 (120.5)	3.602 (141.8)	3.632 (143.0)	3.47 (136.7)	3.81 (150)	3.66 (144)	3.66 (144)	3.65 (143.7)	3.66 (144)	3.66 (144)	3.69 (145.24)	3.66 (144)	3.71 (146)	3.81 (150)
Plenum Length, m (in.)	0.122 (4.81)	0.298 (11.72)	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)	0.24 (9.37)	0.41 (16.0)	0.36 (14.0)	0.25 (10.0)
Fuel Rod OD, mm (in.)	10.719 (0.422)	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)	10.77 (0.424)	14.30 (0.563)	12.52 (0.493)	12.27 (0.483)
Cladding ID, mm (in.)	9.881 (0.389)	9.576 (0.377)	8.407 (0.331)	9.75 (0.384)	8.43 (0.332)	9.48 (0.3734)	9.48 (0.3734)	8.36 (0.329)	9.25 (0.364)	10.91 (0.4295)	9.25 (0.364)	12.68 (0.499)	10.80 (0.425)	10.64 (0.419)
Cladding Thickness, mm (in.)	0.419 (0.0165)	0.673 (0.0265)	0.610 (0.0240)	0.711 (0.028)	0.635 (0.025)	0.617 (0.0243)	0.617 (0.0243)	0.572 (0.0225)	0.762 (0.030)	0.914 (0.036)	0.762 (0.030)	0.813 (0.032)	0.864 (0.034)	0.813 (0.032)
Diametral Gap ^(f) , micron (mil)	165.1 (6.5)	213.4 (8.4)	198.1 (7.8)	190.5 (7.5)	178 (7.0)	190 (7.5)	190 (7.5)	165 (6.5)	190 (7.5)	254 (10.0)	190 (7.5)	305 (12.0)	229 (9.0)	229 (9.0)
Fuel Pellet Diameter, mm (in.)	9.715 (0.3825)	9.362 (0.3686)	8.209 (0.3232)	9.56 (0.3765)	8.26 (0.325)	9.29 (0.3659)	9.29 (0.3659)	8.19 (0.3225)	9.06 (0.3565)	10.66 (0.4195)	9.06 (0.3565)	12.37 (0.487)	10.57 (0.416)	10.41 (0.410)
Fuel Pellet Length, mm (in.)	11.633 (0.458)	15.240 (0.600)	9.525 (0.375)	11.43 (0.450)	9.91 (0.390)	15.24 (0.600)	15.24 (0.600)	13.46 (0.530)	6.93 (0.273)	8.13 (0.320)	10.41 (0.410)	12.70 (0.500)	10.67 (0.420)	10.41 (0.410)
Fuel Pellet Density, %TD ^(g)	95	95	95	95	95	94	95	95	94	95	94.5	95	95	95

(a) LHGR = Linear heat generation rate.

(b) Reload batch average enrichment.

(c) 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).

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

(e) Type 304 stainless steel (304SS), Zircaloy-4 (Zry-4), and Zircaloy-2 (Zry-2)

(f) Diametral gap = cladding ID - pellet diameter.

(g) Theoretical density (TD) of stoichiometric UO_2 is 10.96 g/cm³.

5/6

3.0 FUEL DESIGN CHANGES AND SUMMARY OF FUEL SURVEILLANCE PROGRAMS

Typical fuel reload orders today for a BWR and a PWR have warranties for average batch burnups of about 31,000 MWd/MTU and 36,000 MWd/MTU, respectively (Ref. 24). Earlier, design and warranty burnups for BWR and PWR fuel were lower and were based on limited fuel performance data and not on optimized fuel cycle costs (Refs. 25 and 26). In considering higher burnups for fuel, the industry's focus has been on four principal areas of concern: corrosion of the cladding exterior surface (especially in the new plants with higher temperature coolant), release of fission gas from the fuel, failure of fuel rods due to pellet-cladding interaction, and distortion of fuel rods and fuel assemblies. A small number of fuel assemblies have reached burnups of 50,000 MWd/MTU or higher, and none of those areas of concern have appeared to be a potentially serious problem with those assemblies. However, there continues to be a need for additional data of a confirmatory nature, both for steady-state and transient test conditions.

A wide variety of fuel design changes are being studied by domestic fuel vendors, the Electric Power Research Institute (EPRI), and the U.S. Department of Energy (DOE). Table 2 summarizes the major fuel surveillance programs currently in progress and includes information on a number of the associated fuel design changes being studied. Additional information on some of the 15 X 15 fuel assemblies in Table 2 is in Refs. 27-38. The schedule for EPRI's fuel surveillance projects is given in Figure 1. Table 3 shows the fuel types being irradiated in Big Rock Point. The NRC recently reviewed (Ref. 39) the operating history of Big Rock Point.

No design parameter change was made in 1983 for the standard B&W fuel types. The status of B&W's major fuel performance programs (Ref. 40) is shown in Table 2. B&W, in cooperation with Duke Power Company (Duke), Arkansas Power & Light (AP&L), Sacramento Municipal Utility District (SMUD), and DOE, is involved with programs to improve fuel utilization by extending the average burnup of LWR fuel assemblies to approximately 50,000 MWd/MTU and by developing and demonstrating advanced fuel designs. In addition to the extended burnup program, other programs include demonstrating the in-reactor performance of low absorption Zircaloy spacer grids, a fuel-burnable poison mixture ($\text{UO}_2\text{-Gd}_2\text{O}_3$), axial blanket fuel, annular fuel pellets, advanced claddings, and annealed guide tubes. B&W's annual summary (Ref. 40) provides detailed information on the DOE/Duke/B&W Extended-Burnup Program in Oconee-1, the DOE/AP&L/B&W Extended-Burnup Program in Arkansas-1, the DOE/SMUD/B&W Axial Blanket Fuel Design and Development Program in Rancho Seco, the B&W/Duke Low Absorption Grid Program in Oconee-1 and -2, and the B&W/Duke Advanced Cladding Pathfinder Program in Oconee-2. Additional information on these five programs may be found in 36 reports listed in Ref. 40; five of those reports (Refs. 41-45) were issued in 1983 and two (Refs. 46 and 47) in 1984. More information on B&W extended-burnup fuel in the DOE-sponsored programs is in Refs. 48-50. Information is also available on the Fuel Surveillance Project (Refs. 25 and 36) and the Duplex Fuel Manufacturing Study (Ref. 25) at B&W that are sponsored by EPRI.

TABLE 2. Major Fuel Performance Programs: Status Through 1983

Vendor	Fuel Type ^(a)	Power Plant	Planned No. of Operating Cycles	Scheduled Completion	Interim Inspections to Date
Babcock & Wilcox	15x15	Oconee-1	5	1985	3
	15x15 ^(b)	Arkansas-1 ^(c)	3	1988	1
	15x15 ^(d)	Rancho Seco	3	1986	1
	15x15 ^(e)	Oconee-2	4	Completed	4
	15x15 ^(f)	Oconee-2	3	Completed	3
	15x15 ^(g)	Oconee-2	1	Completed	1
		Oconee-1	3	1987	1
	17x17 ^(h)	Oconee-2	3	Completed	3
	15x15 ⁽ⁱ⁾	Oconee-1	4	1990	None
Combustion Engineering	14x14 ^(j)	Fort Calhoun	6	1983	--
	16x16 ^(k)	Arkansas-2 ^(l)	3	1983	--
	16x16 ^(m)	Arkansas-2	3	1984	--
	16x16 ⁽ⁿ⁾	Arkansas-2	4	1987	--
	14x14 ^(o)	Calvert Cliffs-1	5	1988	--
Exxon Nuclear	15x15	Robinson-2	5	Completed	--
	17x17 ^(p)	Cook-2	--	--	--
	(q)	Prairie Island-2	4	--	--
	(q)	Prairie Island-1	--	--	--
	8x8	Oyster Creek	5	Completed	--
	9x9	Dresden-2	--	--	--
	60 rods ^(r)	Big Rock Point	--	Completed	--
	20 rods ^(r)	Big Rock Point	--	--	--
	plus FPIP and MO ₂ rods				
General Electric	8x8 LTAs ^(s)	Monticello	(7) ^(t) (u)	Completed	--
			(6) ^(t) (u)	Completed	--

TABLE 2. (contd)

Vendor	Fuel Type ^(a)	Power Plant	Planned No. of Operating Cycles	Scheduled Completion	Interim Inspections to Date
	8x8 LTAs ^(v)	Peach Bottom-2	(4)(t)(w)	Completed	--
			(5)(t)(w)	Completed	--
	P8x8R LTA ^(x)	Peach Bottom-3	(4)(t)	--	--
	LTAs ^(y)	Quad Cities-1	(3)(t)	--	--
	1981 LTAs ^(z)	Browns Ferry-3	(1)(t)	--	--
	1983 LTAs ^(aa)	Peach Bottom-3	--	--	--
Westinghouse	17x17 ^(bb)	Trojan	(5)	1984	--
	17x17	Farley-1	(3)(t)	--	3
	(OFA-Demo) ^(cc)				
	17x17	Salem-1	(3)(t)	--	3
	(OFA-Demo) ^(cc)				
	17x17	Beaver Valley-1	(2)(t)	--	2
	(OFA-Demo) ^(cc)				
	14x14	Point Beach-2	(2)(t)	--	2
	(OFA-Demo) ^(dd)				
	MO ₂ ^(ee)	R. E. Ginna	4	--	

- (a) LTA = lead test assembly, MO₂ = mixed oxide (UO₂-PuO₂) fuel, R = retrofit fuel design, OFA-Demo = Demonstration Optimized Fuel Assembly, FPIP = Fuel Performance Improvement Program.
- (b) LTAs of an advanced, extended-burnup design.
- (c) Arkansas Nuclear One-Unit 1 (also known as ANO-1).
- (d) Current-design assemblies containing axially-blanketed fuel columns.
- (e) Current-design assemblies with special Zircaloy cladding materials and EPRI creep collapse specimen clusters. See Refs. 27-36.
- (f) Current-design assemblies with lifted rods and cladding having a known spiral eccentricity spiral eccentricity in wall thickness. See Refs. 37 and 38.

TABLE 2. (contd)

- (g) Current-design assemblies utilizing low-absorption spacer grid material (Zircaloy-4).
- (h) Two of these four LTAs are reconstitutable.
- (i) Gadolinia LTAs of an advanced, extended-burnup design.
- (j) SAVFUEL demonstration assemblies and standard-design, high-burnup program.
- (k) Burnable poison irradiation program.
- (l) Arkansas Nuclear One-Unit 2 (also known as ANO-2).
- (m) Standard surveillance program.
- (n) Advanced fuel design and standard-design, high-burnup program.
- (o) Advanced fuel design LTAs.
- (p) ENC's first reload of 17x17 fuel.
- (q) TOPROD program involves assemblies incorporating axial blankets of natural urania at each end and includes five characterized assemblies at Prairie Island-2 and a full reload at Prairie Island-1.
- (r) Sixty rods discharged in February 1982; 20 of those rods are to be recharged in a hydrid assembly that includes FPIP and MO₂ fuel rods.
- (s) Eight extended-burnup LTAs (two groups).
- (t) Cycles completed through December 1983.
- (u) Four bundles now discharged. See Table 14.
- (v) Four extended-burnup LTAs (two groups).
- (w) Two bundles now discharged. See Table 14.
- (x) One bundle.
- (y) Four bundles (barrier cladding).
- (z) Eight bundles with improved design features.
- (aa) Four bundles with improved design features.
- (bb) Includes 62 assemblies: 60 standard assemblies, one especially characterized removable rod assembly, and one characterized standard assembly.
- (cc) Two assemblies
- (dd) The four assemblies completed their second cycle of irradiation in 1983. Subsequent examination showed one assembly had nine failed fuel rods. The other three assemblies were in good condition and were returned to the core.
- (ee) The four assemblies have completed three cycles of irradiation.

FIGURE 1. Schedule for EPRI Fuel Performance Surveillance Program (Ref. 26)

TABLE 3. Various Fuel Rod Types^(a) That Have Been or Are Being Irradiated in Big Rock Point Reactor Under DOE's Fuel Performance Improvement Program (Refs. 51 and 52)

<u>Fuel Type</u>	<u>Description</u>
Standard	Standard ENC "product-line." Solid, cylindrical, dished-end-fuel pellet. The enrichment was increased from 2.52% to 2.69% to compensate for the reduced mass.
Reference	Solid, cylindrical, dished-end, chamfered-corner fuel pellet.
Annular	Cylindrical, flat-ended, chamfered-corner fuel pellet with a central hole equivalent to 10 vol% of a solid, undished pellet. The enrichment was increased from 2.52% to 2.69% to compensate for the reduced mass.
Vipac	Packed-particle fuel composed of high-density, annular fuel shards, produced by high-energy pneumatic compaction (Dynapak). The rods are pressurized within 400 to 500kPa of helium, except in the lower segments of the segmented rods.
Sphere-pac	Packed-particle fuel composed of high-density, spherical particles produced by the sol-gel process. ^(b) The rods are pressurized with 400 to 500 kPa of helium.
Coated-cladding	Cladding coated with Dag 4 ^(c) graphite.
Pressurized	Rods pressurized with 0.45 MPa of helium.
Reference-coated	Reference fuel pellets combined with coated cladding.
Annular-coated	Annular fuel pellets combined with coated cladding.
Reference-pressurized	Reference fuel pellets in a pressurized rod.
Annular-coated-pressurized	Annular fuel pellets combined with coated cladding in a pressurized rod.

(a) The irradiation tests include full-length and segmented fuel rods. All rods are clad with cold-worked and stress-relieved Zircaloy-2 cladding.

(b) Produced at the Oak Ridge National Laboratory.

(c) Product of Acheson Colloids Corp., Port Huron, Michigan.

Lead control elements containing Ag-In-Cd and manufactured by B&W and Westinghouse are being evaluated in an EPRI-sponsored program (Ref. 25); the B&W material is in Oconee. Axial Power Shaping Rods (APSRs) are involved in two B&W programs (end-of-cycle withdrawal and gray APSRs) that are in progress to increase reactor operating efficiency and availability (Ref. 40). The gray APSR design was successfully implemented in the Rancho Seco Cycle 6 core (Ref. 53) and will be used in upcoming cycles in Oconee-1, -2, and -3 and Crystal River-3.

The status of C-E's fuel surveillance programs (Ref. 54) is listed in Table 2. Information is available (Ref. 25) on these two EPRI-sponsored projects at C-E: "PWR Fuel Surveillance," which involves fuel at Arkansas-2, Calvert Cliffs-1, and Maine Yankee, and "Uniform Waterside Corrosion of Zircaloy-Clad Fuel Rods" (the final report, EPRI NP-2789, was issued in December 1982). Details on the EPRI-sponsored surveillance program in Calvert Cliffs-1 are also in Refs. 55 and 56. Data on the DOE-sponsored programs in Fort Calhoun and Arkansas-2 are also provided in Refs. 55, 57, and 58.

The fuel surveillance programs being conducted by ENC are included in Table 2. ENC's annual summary (Ref. 59) provides detailed information on their fuel development programs. Included are the BWR Extended Burnup Demonstration Programs (jointly sponsored by DOE, ENC, Consumers Power Company and GPU Nuclear) in Big Rock Point and Oyster Creek, the PWR High Burnup Demonstration Programs in reactors in the U.S. (Robinson-2) and Europe, and the BWR 9 x 9 Program that ENC, in conjunction with EPRI, Empire State Electric Energy Research Corporation (ESEERCO), and Commonwealth Edison, is conducting with four 9 x 9 lead test assemblies in Dresden-2 (Ref. 25). Additional information is available on the ENC/Carolina Power & Light program in Robinson-2 (Ref. 60), the EPRI/ENC/GPU Nuclear project in Oyster Creek (Refs. 25 and 61), the EPRI/ENC project on mixed oxide irradiations (Ref. 25), and the DOE/Consumers Power/ENC/PNL program in Big Rock Point (Refs. 51, 52, and 62-68).

The status of GE's fuel surveillance program (Ref. 69) is shown in Table 2 (and also in Table 14 in Section 4.0). The fuel surveillance program adopted by GE and accepted by the NRC is described in Refs. 70-73. Additional information is also available on the EPRI-sponsored project involving mixed oxide (UO_2 - PuO_2) fuel that was irradiated in Oyster Creek and Quad Cities-1 (Refs. 25, 26, and 74), the EPRI-sponsored project in Peach Bottom-2 (Refs. 25, 26, and 74), the DOE-sponsored program with extended burnup fuel in Monticello (Refs. 26, 74, and 75), the DOE/Commonwealth Research Corporation/GE program on barrier fuel in Quad Cities-1 (Refs. 76 and 77), the DOE/Commonwealth Edison/GE large-scale demonstration with zirconium barrier fuel at Quad Cities-2 (Ref. 78), and the DOE/Tennessee Valley Authority/GE program (Ref. 79). Additional information on the zirconium-lined cladding used on barrier fuel may be found in Refs. 80 and 81.

The status of the Westinghouse fuel surveillance programs (Ref. 82) is shown in Table 2. More detailed information is available on the EPRI-sponsored

project in Zion-1/Zion-2 (Refs. 25, 26, 74, and 83-86); the EPRI-sponsored project in Trojan (Refs. 25 and 83-85); the DOE-sponsored Extended Fuel Burnup Demonstration Program in Surry-2 (Refs. 83-88); the demonstration Optimized Fuel Assemblies in Farley-1 (Refs. 83-85 and 89), Salem-1 (Refs. 83-85), Beaver Valley-1 (Refs. 83-85), and in Point Beach-2 (Refs. 83 and 84); and the mixed oxide ($\text{UO}_2\text{-PuO}_2$) fuel in Ginna (Refs. 83 and 84). Information on the DOE-sponsored program involving the irradiation of Westinghouse fuel rods in the CEN BR-3 reactor is provided in Ref. 86. Lead control elements containing Ag-In-Cd and manufactured by B&W and Westinghouse are being evaluated in an EPRI-sponsored program (Ref. 25); the Westinghouse material is from Point Beach-1. Information on Westinghouse burnable absorber studies may be found in Ref. 90 and on a DOE-sponsored generic study on extended burnup in PWRs (uses a Westinghouse PWR, Zion-1, as a basis) in Ref. 91.

The current status of the EPRI fuel performance surveillance program is described in a recent report (Ref. 25). The schedule for the EPRI program is shown in Figure 1. Additional information on the program is available in Refs. 26, 92, and 93. Two recent EPRI reports (Refs. 94 and 95) contain information on design changes associated with BWR fuel.

Table 3 (Refs. 51 and 52) shows that a wide variety of fuel types is being irradiated in Big Rock Point. Additional information is also available in Refs. 39, 51, and 62.

4.0 FUEL OPERATING EXPERIENCE

The following subsections provide synopses of domestic fuel operating experience. The six subsections include information from a) the five fuel vendors: Babcock & Wilcox Company, Combustion Engineering, Inc., Exxon Nuclear Company, Inc., General Electric Company, and Westinghouse Electric Corporation, and b) the Electric Power Research Institute. While overall fuel operating experience continues to be excellent there are sporadic events involving damage to or failure of fuel, and those events are discussed in Section 5.0. Historical information on fuel failure rates in BWRs and PWRs was provided in the Appendix of the previous annual report (Ref. 5). It should be noted that the definition of failed fuel is not uniformly applied (see footnote in Section 1.0), and in many cases the number of fuel failures is inferred from indirect evidence, and that in some cases only directly observed failures are counted (Ref. 96).

4.1 BABCOCK & WILCOX COMPANY (B&W)

B&W has issued the annual summary for 1982 (Ref. 40) of in-reactor fuel performance and ongoing development programs for B&W-designed commercial nuclear fuel. B&W has irradiated a total of 770,240 fuel rods during the 11-year operating history of B&W-designed reactors. The fuel assemblies with Zircaloy-clad fuel rods that have been irradiated have been of the B&W 15 x 15 Mark B design with the exception of four 17 x 17 lead test assemblies (Mark C and Mark CR). Only 30 new "leaking"^(a) Zircaloy-clad fuel rods were detected among the 336,128 rods irradiated during 1983--a fuel integrity of 99.991% for operation of Zircaloy-clad fuel rods within this period. B&W-supplied, stainless steel-clad fuel irradiated in 1983 maintained 100% integrity. See Section 5.0 for information on fuel failures. One fuel assembly completed its fifth cycle of irradiation at Oconee-1 in June 1983 and attained a burnup of 50,600 MWd/MTU--this was the highest fuel assembly burnup achieved by B&W fuel in 1983. B&W-designed reactors are now using 15- to 18-month cycles with the lumped burnable poison shuffle scheme to achieve projected burnups of 32,500 to 36,500 MWd/MTU.

A summary of B&W fuel rod experience, from the startup of their first reactor (Oconee-1) in April 1973 through December 1983, is provided in Table 4. The operating status of B&W-designed reactors is shown in Table 5. The burnup experience for B&W-supplied fuel is summarized in Table 6. The burnup experience for B&W-supplied, stainless steel-clad fuel is summarized in Table 7. Tables 4-7 are from Ref. 40. The status of B&W's fuel performance programs is presented in Section 3.0.

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

TABLE 4. 1983 Performance Summary for B&W-Supplied Fuel Rods^(a)
(December 31, 1983)

Fuel Rod Type	Stainless Steel-Clad Fuel	Zircaloy-Clad Fuel:	
		Fuel Assembly Rod Array Type 15 x 15	17 x 17
1. Cumulative Number of Rods Irradiated Through Dec. 1983:	74,256	769,184	1,056
a. Maximum Rod Average Burnup, MWd/MTU	38,600	51,400	36,400
b. Mean Rod-Average Burnup, MWd/MTU	26,800	21,200	30,100
2. Total Number of Rods Irradiated in 1983:	42,840	336,128	0
3. Number of Irradiated Rods In Core on Dec. 31, 1983:	32,028	283,712	0
a. Maximum Rod-Average Burnup, MWd/MTU	33,200	34,200	0
b. Mean Rod-Average Burnup, MWd/MTU	18,100	14,400	0
4. Number of Rods Discharged in 1983:	10,812	78,416	0
a. Maximum Rod-Average Burnup, MWd/MTU	37,400	51,400	0
b. Mean Rod-Average Burnup, MWd/MTU	33,500	28,000	0
5. Estimated Number of Leaker Rods Generated in 1983 ^(b)	0	30	-

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

(b) Estimated from equilibrium coolant radioiodine behavior during full-power operation (iodine-131, iodine-133, and iodine-135 activity levels), using the method described in Ref. 97. Because of the uncertainties associated with the location and nature of probable leakers, the number of leaker rods represents a best-estimate of the fuel integrity status.

TABLE 5. Operating Status of B&W Fueled Reactors (as of December 31, 1983)

Reactor	Reactor Cycle	Maximum Assembly Burnup, MWd/MTU	
		In-Core Fuel	Fuel Discharged to Date
Oconee-1	8	30,800	51,400
Oconee-2	7	23,100	36,200
Oconee-3	7	32,900	35,500
Three Mile Island-1 (a)	5	25,200	32,300
Arkansas-1 (b)	6	28,700	36,500
Rancho Seco	6	28,200	38,300
Crystal River-3	5	20,000	29,900
Davis Besse-1	4	25,400	32,800
Connecticut Yankee (c)	12	31,900	37,100

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

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

(c) Haddam Neck

TABLE 6. Summary of Burnup Experience Through
December 31, 1983, for B&W-Supplied Fuel^(a)

Fuel Assembly Batch-Averaged Burnup, MWd/MTU	Assemblies In Core on Dec. 31, 1983		Assemblies Discharged in 1983		Assemblies Discharged Through Dec. 31, 1983	
	No. of Assemblies	No. of Rods	No. of Assemblies	No. of Rods	No. of Assemblies	No. of Rods
0 to 3,900	172	35,776	0	0	0	0
4,000 to 7,900	329	68,432	0	0	4	832
8,000 to 11,900	48	9,984	0	0	103	21,424
12,000 to 15,900	261	54,288	0	0	136	28,288
16,000 to 19,900	149	30,992	53	11,024	148	30,784
20,000 to 23,900	265	55,120	9	1,872	254	52,832
24,000 to 27,900	179	37,232	124	25,792	949 ^(b)	197,504
28,000 to 31,900	13	2,704	162	33,606	576	119,808
32,000 to 35,900	0	0	0	0	135 ^(c)	28,192
36,000 to 39,900	0	0	28	5,824	29	6,032
40,000 to 43,900	0	0	0	0	3	624
44,000 to 47,900	0	0	0	0	0	0
48,000 to 51,900	0	0	1	208	1	208
TOTALS	1,416	294,528	377	78,416	2,338	486,528

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

(b) Includes two nonreconstitutable, 17 x 17 lead test assemblies (Mark C).

(c) Includes two reconstitutable, 17 x 17 lead test assemblies Mark CR).

TABLE 7. Summary of Burnup Experience Through December 31, 1983, for B&W-Supplied Stainless Steel-Clad Fuel

Fuel Assembly Batch-Averaged Burnup, MWd/MTU	Assemblies In Core on Dec. 31, 1983		Assemblies Discharged in 1983		Assemblies Discharged Through Dec. 31, 1983	
	No. of Assemblies	No. of Rods	No. of Assemblies	No. of Rods	No. of Assemblies	No. of Rods
0 to 3,900	0	0	0	0	0	0
4,000 to 7,900	36	7,344	0	0	0	0
8,000 to 11,900	16	3,264	0	0	0	0
12,000 to 15,900	0	0	0	0	0	0
16,000 to 19,900	40	8,160	0	0	0	0
20,000 to 23,900	12	2,448	0	0	2	408
24,000 to 27,900	0	0	0	0	0	0
28,000 to 31,900	45	9,180	1	204	21	4,284
32,000 to 35,900	8	1,632	52	10,608	172	35,088
36,000 to 39,900	0	0	0	0	12	2,448
40,000 to 43,900	0	0	0	0	0	0
TOTALS	157	32,028	53	10,812	207	42,228

4.2 COMBUSTION ENGINEERING, INC. (C-E)

The performance of C-E fuel in 1983 is described in a recent letter (Ref. 54). There were 333,700 fuel rods irradiated during 1983. The reliability for C-E fuel in the year 1983 was 99.98%. See Section 5.0 for information on the fuel failures. As of December 31, 1983, C-E has irradiated a total of 780,872 fuel rods. A summary of C-E fuel rods and assemblies irradiated and/or discharged and the batch-averaged burnups achieved in 1983 is presented in Table 8. The status of C-E burnup experience with all-Zircaloy assemblies is shown in Table 9. The equilibrium iodine-131 activity concentrations in the primary coolants of the C-E reactors during 1983 are listed in Table 10. Tables 8-10 are from Ref. 54. The status of C-E's major fuel performance programs is presented in Section 3.0.

4.3 EXXON NUCLEAR COMPANY, INC. (ENC)

A report summarizing the ENC fuel performance through December 31, 1983, is in preparation and will be published by January 31, 1985. During 1983, there were 4532 fuel assemblies (2211 BWR, 2321 PWR) irradiated and/or discharged. ENC has irradiated a total of 794,282 fuel rods as of December 31, 1983. The overall exposure ranges of BWR and PWR fuel assemblies and rods are shown graphically in Figure 2. Table 11 summarizes the ENC fuel assembly distribution, fuel rod performance, and gadolinia fuel experience as of December 31, 1983. The highest burnups as of the end of 1983 were achieved by lead assemblies being irradiated under two Extended Burnup Demonstration programs. (See Section 3.0 for information on the status of these and other ENC fuel performance programs.) Those burnups were 40,500 MWd/MTU for 32 fuel

TABLE 8. Summary of Combustion Engineering Fuel Irradiated and/or Discharged in 1983

Reactor/ (Fuel Cycle)	Fuel Batch	Number of Assemblies		Number of Fuel Rods		Batch-Averaged Burnup, MWd/MTU	
		Irradiated	Discharged	Irradiated	Discharged	as of December 1983	at Discharge
Arkansas-2 ^(a) / (Cycle 3)	A	9	0	2,124	0	21,300	--
	C	56	0	12,808	0	30,200	--
	D	60	0	14,160	0	21,000	--
	E	52	0	12,272	0	8,200	--
Calvert Cliffs-1/ (Cycles 6 & 7)	B	12	0	1,968	0	18,200	--
	D	21	9	3,696	1,584	23,900	33,500
	E	12	0	2,112	0	24,100	--
	F	49	44	8,624	7,744	26,400	35,400
	G	92	52	15,776	8,736	26,100	28,400
	H	72	0	12,416	0	13,900	--
	J	64	0	11,264	0	500	--
Calvert Cliffs-2/ (Cycle 5)	D	13	0	2,288	0	29,800	--
	F	128	0	21,824	0	26,700	--
	G	76	0	13,152	0	10,000	--
Fort Calhoun/ (Cycle 8)	G	29	0	5,104	0	36,300	--
Maine Yankee/ (Cycle 7)	E	1	0	160	0	27,900	--
Millstone Point-2/ (Cycle 5)	B	1	0	164	0	28,000	--
	E	72	0	12,672	0	34,800	--
Saint Lucie-1/ (Cycle 5)	E	65	0	11,440	0	27,400	--
	F	88	0	14,912	0	20,900	--
	G	64	0	11,024	0	7,000	--

TABLE 8. (contd)

Reactor/ (Fuel Cycle)	Fuel Batch	Number of Assemblies		Number of Fuel Rods		Batch-Averaged Burnup, MWd/MTU	
		<u>Irradiated</u>	<u>Discharged</u>	<u>Irradiated</u>	<u>Discharged</u>	<u>as of December 1983</u>	<u>at Discharge</u>
Saint Lucie-2/ (Cycle 1)	A	73	0	17,228	0	4,500	--
	B	80	0	17,600	0	4,700	--
	C	64	0	14,752	0	3,700	--
San Onofre-2/ (Cycle 1)	A	73	0	17,228	0	6,700	--
	B	80	0	17,600	0	6,900	--
	C	64	0	14,752	0	5,400	--
San Onofre-3/ (Cycle 1)	A	73	0	17,228	0	1,900	--
	B	80	0	17,600	0	1,900	--
	C	64	0	14,752	0	1,700	--

(a) Arkansas Nuclear One-Unit 2 (also known as ANO-2).

TABLE 9. Combustion Engineering Burnup With All-Zircaloy Assemblies:
Status as of December 31, 1983

Fuel Assembly Batch-Averaged Burnup, MWd/MTU	In-Core Fuel Assemblies with Pressurized Fuel Rods		Discharged Fuel Assemblies with Pressurized Fuel Rods		Discharged Fuel Assemblies with Nonpressurized 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	418	92,824	0	0	0	0
4,000 to 7,900	297	67,180	6	1,048	0	0
8,000 to 11,900	128	25,424	25	31,235	208	40,500
12,000 to 15,900	156	26,960	68	9,325	190	35,351
16,000 to 19,900	13	2,132	157	27,036	24	3,840
20,000 to 23,900	92	20,272	181	30,876	0	0
24,000 to 27,900	173	29,726	520	91,844	0	0
28,000 to 31,900	158	30,244	714	125,176	0	0
32,000 to 35,900	66	11,613	407	71,064	0	0
36,000 to 39,900	29	5,104	8	1,408	0	0
40,000 to 43,900	0	0	3	514	0	0
44,000 to 47,900	0	0	0	0	0	0
48,000 to 51,900	0	0	1	176	0	0

TABLE 10. Typical Equilibrium Iodine-131 in the Primary Coolant of Combustion Engineering Reactors During 1983

Reactor	Fuel Cycle	Beginning of Cycle	End of Cycle	1983 Equilibrium Conditions	
				Power, %	Iodine-131, $\mu\text{Ci/g}$
Arkansas-2 ^(a)	3	11/10/82	10/4/83	100	0.01 ^(b) - 0.03 ^(b)
Calvert Cliffs-1	6	6/30/82	10/1/83	100	0.01 - 0.05
	7	11/30/83	5/85 ^(c)	Nonequilibrium Conditions	
Calvert Cliffs-2	5	1/13/83	4/21/84	100	0.03 - 0.17
Fort Calhoun	8	4/4/83	3/4/84	100	0.03 - 0.13 ^(d)
Maine Yankee	7	12/18/82	3/30/84	100	0.001 - 0.002 ^(d)
Millstone Point-2	5	3/11/82	5/28/83	100	0.02 - 0.03 ^(d)
Saint Lucie-1	5	11/29/81	2/26/83	100	0.04 - 0.06
Saint Lucie-2	1	6/2/83	11/84 ^(c)	100	0.02 - 0.13
San Onofre-2	1	7/26/83	9/84 ^(c)	100	0.02 ^(b) - 0.07 ^(b)
San Onofre-3	1	8/29/83	4/85 ^(c)	Nonequilibrium Conditions	

(a) Arkansas Nuclear One-Unit 2 (also known as ANO-2).

(b) Adjusted for two charging pumps.

(c) Projected end-of-cycle date.

(d) Predominately non-Combustion Engineering supplied fuel.

rods at Big Rock Point (BWR) and 47,700 MWd/MTU for one PWR fuel assembly at H. B. Robinson Unit 2. The fuel rod integrity, based on failures that were judged to be from fuel-related causes, continued to be 99.998%. No gadolinia fuel failures were found in 1983. ENC fuel failures observed in 1983 are listed in Table 12. See Section 5.0 for information on fuel failures. Figure 2 and Tables 11 and 12 are from Ref. 59.

4.4 GENERAL ELECTRIC COMPANY (GE)

Table 13 (Ref. 69) provides a summary of GE's BWR/2-6 experience with 8 x 8 fuel through December 1983. This experience includes successful commercial reactor operation of fuel bundles to greater than 45,000 MWd/MTU and peak

FIGURE 2. Exposure of Irradiated ENC Fuel as of December 31, 1983

TABLE 11. Summary of Exxon Nuclear Fuel Assembly Distribution, Fuel Rod Performance, and Gadolinia Fuel Experience, as of December 31, 1983

A. Fuel Assemblies

Reactor Type	Number of Fuel Assemblies				Total Failures, % ^(a)
	In Core as of December 31, 1983	Discharged in 1983	Discharged Prior to 1983	Total	
BWR	1,925	286	599	2,804	0.71
PWR	<u>1,906</u>	<u>415</u>	<u>782</u>	<u>3,069</u>	<u>1.24</u>
Total	3,831	701	1,381	5,873	0.99

B. Fuel Rods

Reactor Type	In-Core Fuel		Discharged Fuel		Total Number of Rods	Failure Rate, % ^(b)	Total Failures, % ^(a)
	Number of Rods	Burnup, Mwd/MTU	Number of Rods	Burnup, Mwd/MTU			
BWR	124,908	40,500	45,802	40,500 ^(c)	170,710	0.008	0.012
PWR	<u>385,765</u>	<u>40,200</u>	<u>237,807</u>	<u>47,700</u>	<u>623,572</u>	<u>0.001</u>	<u>0.013</u>
Total	510,673		283,609		794,282	0.002	0.013

C. Gadolinia Rods

Reactor Type	Number of Gadolinia Rods
BWR	9,753
PWR	<u>2,260</u>
Total	12,013

(a) Failure rates from all causes.

(b) Failures not directly attributable to external causes (e.g., plant-related causes such as baffle jetting, fretting from the presence of foreign objects or other off-normal core conditions) and occurring below warranted burnup.

(c) Average of Extended Burnup Demonstration Program fuel rods.

TABLE 12. Summary of ENC's 1983 Fuel Failures and Repairs

<u>Reactor</u>	<u>Number of Fuel Assemblies</u>	<u>Fuel Type</u>	<u>Date Found</u>	<u>Failure Type</u>	<u>Repaired for Core Reinsertion</u>
BWRs:					
Dresden 3	1	UO ₂	11/83	(a)	Yes
PWRs:					
Palisades	1	UO ₂	09/83	(a)	Yes
Palisades	4	UO ₂	09/83	(b)	Yes
Prairie Island-2	1	UO ₂	09/83	(b)	Yes
Tihange-1	3	UO ₂	03/83	(b)	Yes

(a) Possible fabrication related defect.

(b) Failure due to off-normal core conditions or handling.

TABLE 13. General Electric Fuel Experience Summary
(Through January 1, 1984)

	<u>All 8 x 8</u>
Total Rods in Core or Discharged	>2 million ^(a)
Estimated Sound Rods, %:	
In 1983	(b)
Cumulative	(b)

(a) Cumulative.

(b) Proprietary.

pellet exposures of approximately 60,000 MWd/MTU. See Section 5.0 for information on fuel failures. As of January 1, 1984, over 2.0 million GE 8 x 8 fuel type production Zircaloy-clad UO_2 fuel rods were in or had completed operation in commercial BWRs. A summary of GE's lead test assembly and extended burnup bundles is shown in Table 14 (Ref. 69). The status of GE's fuel performance programs is presented in Section 3.0.

4.5 WESTINGHOUSE ELECTRIC CORPORATION (W)

The operational experience in Westinghouse cores with Zircaloy-clad fuel through December 31, 1983, is summarized in a recent report (Ref. 82). This experience commenced with the introduction of Westinghouse Zircaloy-clad fuel in the Jose Cabrera plant in June 1968. At the end of 1983, a total of 1,359,493 Zircaloy-clad fuel rods were in operation, which represents 5758 fuel assemblies. The addition of discharged fuel brings the total number of irradiated Westinghouse Zircaloy-clad fuel rods to 3,171,381. This represents 14,400 fuel assemblies containing 6649 metric tons of UO_2 . The average burnup of the discharged fuel is 25,580 MWd/MTU, and the average burnup of all Westinghouse fuel (in-core plus discharged) is 21,480 MWd/MTU. The highest burnups of discharged fuel assemblies are in the range of 39,000-55,000 MWd/MTU. A total of 519 assemblies have achieved assembly average burnups greater than 36,000 MWd/MTU, of which 53 had burnups greater than 40,000 MWd/MTU. The burnup experience of Westinghouse fuel includes 14 x 14, 15 x 15, 16 x 16, and 17 x 17 fuel assembly designs. Twenty-one plants are operating with the 17 x 17 fuel design. Table 15 provides a burnup summary for Westinghouse Zircaloy-clad fuel discharged and being irradiated through 1983. Figure 3 shows the assemblywise burnup performance of Westinghouse Zircaloy-clad fuel rods. Westinghouse continues to evaluate fuel performance in terms of coolant activity level (i.e., instead of fuel integrity level). Table 16 presents a performance summary for Westinghouse fuel on a plant-by-plant basis and includes data on coolant activity level. See Section 5.0 for information on fuel failures. Westinghouse experience with Zircaloy-clad fuel as of December 31, 1983, is shown in Figure 4. As may be noted in Figure 4 of the previous annual report (Ref. 82), the average coolant activity started to rise during 1981 but decreased during 1982. Tables 15 and 16 and Figures 3 and 4 are from Ref. 82. The status of Westinghouse fuel performance programs is presented in Section 3.0.

4.6 ELECTRIC POWER RESEARCH INSTITUTE (EPRI)

EPRI initiated an LWR program for core materials in mid-1974. The program, which originated with emphasis on fuel problems but since 1979 also includes other core components, involved fuel surveillance projects with each of the domestic fuel vendors. Those projects monitor primarily standard product-line fuel (however, EPRI is also sponsoring studies on new improved fuel design concepts-see Section 3.0). A series of program planning documents was initiated in 1975. Reference 25 is the sixth such report and describes progress in the program between March 1981 and March 1983 and plans for future work.

TABLE 14. Summary of General Electric Lead Test Assembly Surveillance Programs

Program	Reactor	No. of Bundles	Completed Cycles of Operation	Average Burnup at Last Outage, MWd/MTU	Objectives
8x8 Lead Test Assemblies (Extended Burnup- 2 groups)	Monticello	4 4	7 6	42,000-46,000 ^(a) 35,000-40,000 ^(a)	Lead 8x8 perfor- mance, extended burnup
8x8 Lead Test Assemblies (Extended Burnup- 2 groups)	Peach Bottom-2	2 2	4 5	35,000 ^(a) 43,000 ^(a)	Lead 8x8R perfor- mance, extended burnup
P8x8R Lead Test Assembly	Peach Bottom-3	1	4	32,000	Lead P8x8R perfor- mance
Lead Test Assemblies	Quad Cities-1	4	3	31,000	Barrier cladding
1981 Lead Test Assemblies	Browns Ferry-3	8	1	12,000	Improved design features
1983 Lead Test Assemblies	Peach Bottom-3	4	--	--	Improved design features

(a) Bundles now discharged.

TABLE 15. Assemblywise Burnup Distribution of Westinghouse Zircaloy-Clad Fuel Rods: Status as of December 31, 1983

A. Active Fuel Rod Burnup Status^(a):

Assemblywise Burnup, MWd/MTU	Number of Fuel Rods				Totals
	14 x 14 Rods ^(b)	15 x 15 Rods	16 x 16 Rods	17 x 17 Rods	
0 - 3,900	4,296	28,764	18,800	152,592	204,452
4,000 - 7,900	21,587	32,438	47,470	112,200	213,693
8,000 - 11,900	15,680	27,948	0	158,928	202,558
12,000 - 15,900	15,322	15,096	5,640	141,768	177,828
16,000 - 19,900	16,599	38,556	13,395	78,408	146,958
20,000 - 23,900	14,343	39,372	0	61,248	114,963
24,000 - 27,900	15,740	39,168	0	64,680	119,588
28,000 - 31,900	11,814	36,720	0	82,568	111,102
32,000 - 35,900	8,234	26,724	0	3,960	38,918
36,000 - 39,900	3,759	14,688	0	8,184	26,631
40,000 - 43,900	358	2,448	0	0	2,808
44,000 - 47,900	0	0	0	0	0
48,000 - 51,900	0	0	0	0	0
52,000 - 55,900	0	0	0	0	0
56,000 - 59,900	0	0	0	0	0
Totals	127,732	301,920	85,305	844,538	1,359,493

B. Discharged Fuel Rod Burnup Status^(c):

Assemblywise Burnup, MWd/MTU	Number of Fuel Rods				Totals
	14 x 14 Rods ^(b)	15 x 15 Rods	16 x 16 Rods	17 x 17 Rods	
0 - 3,900	0	0	0	0	0
4,000 - 7,900	4,296	0	0	5,280	9,576
8,000 - 11,900	23,449	8,976	0	30,624	63,049
12,000 - 15,900	168,28	49,776	9,400	46,200	122,202
16,000 - 19,900	61,576	93,432	0	175,296	330,304
20,000 - 23,900	53,700	111,996	0	44,618	210,312
24,000 - 27,900	118,858	103,428	0	93,720	316,004
28,000 - 31,900	110,264	194,820	0	77,352	382,436
32,000 - 35,900	134,250	105,060	0	81,778	301,088
36,000 - 39,900	22,912	33,048	0	13,728	89,688
40,000 - 43,900	5,012	1,224	0	0	6,236
44,000 - 47,900	179	0	0	0	179
48,000 - 51,900	0	0	0	0	0
52,000 - 55,900	0	816	0	0	816
56,000 - 59,900	0	0	0	0	0
Totals	551,320	702,576	9,400	548,592	1,811,888

TABLE 15. (contd)

C. Total Fuel Rod Burnup Status:

Assemblywise Burnup, MWd/MTU	Number of Fuel Rods				Totals
	14 x 14 Rods ^(b)	15 x 15 Rods	16 x 16 Rods	17 x 17 Rods	
0 - 3,900	4,296	28,764	18,800	152,592	204,452
4,000 - 7,900	25,883	32,436	47,470	117,480	223,269
8,000 - 11,900	39,129	36,924	0	189,552	265,605
12,000 - 15,900	32,148	64,872	15,040	187,968	300,028
16,000 - 19,900	78,175	131,988	13,395	253,704	477,262
20,000 - 23,900	68,043	151,368	0	105,864	325,275
24,000 - 27,900	134,596	142,596	0	158,400	435,592
28,000 - 31,900	122,078	231,540	0	139,920	493,538
32,000 - 35,900	142,484	131,784	0	65,736	340,004
36,000 - 39,900	26,671	47,736	0	21,912	96,319
40,000 - 43,900	5,370	3,672	0	0	9,042
44,000 - 47,900	179	0	0	0	179
48,000 - 51,900	0	0	0	0	0
52,000 - 55,900	0	816	0	0	816
56,000 - 59,900	0	0	0	0	0
Totals	679,052	1,004,496	94,705	1,393,128	3,171,381

(a) Fuel rods currently operating.

(b) The fuel rods are associated with fuel assemblies of the 14 x 14 rod array type.

(c) Fuel rods discharged from 1969 through December 1983.

FIGURE 3. Assemblywise Performance of Westinghouse Zircaloy-Clad Fuel (Representative of all Zircaloy-Clad Fuel Operating and Discharged)

**TABLE 16. Westinghouse Fuel Performance Status Report - 1983
(Through December 31, 1983)**

Reactor	Location	Date of Initial Criticality	Current Cycle No.	Peak Region Average Burnup, MWD/MTU	Iodine-131 Activity In Primary Coolant, (a) $\mu\text{Ci/g}$
Jose de' Cabrera	Spain	6/68	11	30,800	0.034
Beznau-1	Switzerland	6/69	13	33,800	(b)
R. E. Ginna	U.S.A.	11/69	13	26,700	(b)
Mihama-1	Japan	7/70	5	26,000	(b)
Point Beach-1	U.S.A.	11/70	11	36,400	0.0400
Mihama-2	Japan	4/72	4	28,400	(b)
Point Beach-2	U.S.A.	5/72	10	37,500	0.0071
Surry-1	U.S.A.	7/72	7	32,700	0.102 ^(c)
Turkey Point-3	U.S.A.	10/72	8	31,900	0.087
Surry-2	U.S.A.	3/73	7	33,300	0.00013
Indian Point-2	U.S.A.	5/73	6	33,400	0.014
Turkey Point-4	U.S.A.	6/73	9	35,100	0.031
Zion-1	U.S.A.	7/73	7	37,100	0.0098
Prairie Island-1	U.S.A.	12/73	8	35,700	0.0001
Zion-2	U.S.A.	12/73	7	36,700	0.00033
Kewaunee	U.S.A.	3/74	9	33,800	0.0022
Prairie Island-2	U.S.A.	12/74	7	33,100	0.0001
D. C. Cook-1	U.S.A.	1/75	8	33,600	0.00041
Trojan	U.S.A.	12/75	6	34,100	0.042
Milstone-2 ^(d)	U.S.A.	12/75	5	23,000	0.03
Indian Point-3	U.S.A.	4/76	3	34,800	0.0074
Beaver Valley	U.S.A.	5/76	4	32,800	0.0070
Salem-1	U.S.A.	12/76	5	32,700	0.0014
Ko-RI-1	Korea	6/77	5	30,000	0.068
Fareley-1	U.S.A.	8/77	5	32,000	0.014 ^(c)
Ohl-1	Japan	12/77	2	16,900	(b)
D. C. Cook-2	U.S.A.	3/78	4	34,800	0.0018
North Anna-1	U.S.A.	4/78	4	30,700	0.056 ^(c)
Ohl-2	Japan	9/78	3	20,600	(b)
North Anna-2	U.S.A.	6/80	3	26,500	0.0086
Sequoyah-1	U.S.A.	7/80	2	25,800	0.034
Ringhals-3	Sweden	7/80	1	14,700	0.0054
Salem-2	U.S.A.	8/80	2	19,500	0.0009
Almaraz-1	Spain	4/81	1	16,500	0.0045
Fareley-2	U.S.A.	5/81	3	27,100	0.00055
McGuire-1	U.S.A.	6/81	1	15,100	0.0060
Krsko-1	Yugoslavia	9/81	2	18,400	0.00097
Sequoyah-2	U.S.A.	10/81	2	19,300	0.014
Angra-1	Brazil	3/82	1	4,900	0.00016
Ringhals-4	Sweden	8/82	1	12,600	(b)
V. C. Summer	U.S.A.	10/82	1	9,100	0.00051
Ko-RI-2	Korea	4/83	1	7,500	(b)
McGuire-2	U.S.A.	5/83	1	3,900	0.00018
ASCO-1	Spain	6/83	1	6,200	(b)
Almaraz	Spain	9/83	1	2,000	0.00023

(a) Activity release rate is presented as a percent of the design basis activity release rate (i.e., as a percent of the Iodine-131 release that establishes the basis for design of plant shielding and coolant cleanup system equipment).

(b) No data reported.

(c) Increased letdown rate.

(d) Currently fueled by Westinghouse.

FIGURE 4. Westinghouse Zircaloy-Clad Fuel Experience As of December 31, 1983

5.0 PROBLEM AREAS OBSERVED DURING 1983

5.1 PROBLEMS IN 1983 THAT ARE SIMILAR TO THOSE IN 1982

5.1.1 Debris-Induced Fretting

Evaluations of fuel failures at Arkansas-2, Calvert Cliffs-2, Millstone-2, Point Beach-1, and Surry-1 indicate that debris-induced fretting appears to be the major PWR fuel operating experience problem in 1983.

Three plants (Millstone-2, Point Beach-1, and Surry-1) have had a total of about 80 fuel assemblies with fuel rod leakers primarily due to debris-induced fretting (Ref. 82). These plants contained metallic debris and all have had relatively high coolant activity levels. Fifty of the 80 were found at one plant during a 1983 refueling. A large quantity of metallic debris was observed in the fuel assemblies during on-site examinations at this plant. Many of the failed rods had hydride defects, and examinations showed that most of the rods failed due to metallic debris-induced fretting. The utilities have been informed about this potential damage mechanism, and Westinghouse recommended appropriate precautions to be taken to keep the coolant system as clean as possible during repair/maintenance operations.

a. Arkansas-2 (Refs. 98-102)

In January 1980, the licensee determined from primary coolant activity that a limited number of fuel rods had perforated during the initial power ascension program (Ref. 101). It was previously stated (Ref. 102) that the decrease in iodine activity immediately following the initial high sample indicated that a "crud" burst, rather than fuel failures, took place.

Fuel inspections at the end of Cycle 1 identified seven fuel assemblies having perforated fuel and/or burnable poison rods and five different fuel assemblies having fuel assemblies with spacer grid damage (Refs. 98 and 99). The spacer grid damage [attributed to grid-to-grid interaction (i.e., hangup) with adjacent fuel assemblies during fuel shuffling in the reactor vessel] was described in the fuel performance annual reports for 1982 (Ref. 5, p. 345) and for 1981 (Ref. 4, p. 33). The rod perforations in two of the seven assemblies were diagnosed as due to fretting caused by foreign debris that had been trapped below the bottom Inconel grid (Refs. 98 and 99). Causes of the rod failures in the remaining five assemblies remain unknown.

Inspections performed at the end of Cycle 2 revealed 10 damaged fuel assemblies: five having perforated fuel rods, one having a perforated poison rod, and five having grid damage. Also, a previously suspected (Refs. 98-100) high rate of fuel assembly shoulder gap closure was confirmed (see 5.1.8,a). With regard to the rod perforations, one perforated fuel rod was associated with perimeter grid strip damage, but the cause(s) of the other rod perforations are undetermined. Though fretting due to lodged debris was again observed, it is not believed to have been a contributor to Cycle-2 rod failures

(Refs. 98 and 99). A total of 3 fuel assemblies that had been previously designated for use in Cycle 3 were reconstituted with 20 replacement rods and new fuel assembly caging. It should be noted that only 3 of the 20 replaced rods were perforated.

b. Calvert Cliffs-2

See 5.1.2,a for details.

c. Millstone-2

As noted in the previous annual report (Ref. 5, pp. 36-37), higher than normal concentrations of iodine-131 on June 9, 1982, indicated that fuel cladding defects were probably present. The estimate at that time was that there were 10 to 30 failed rods. Also, see 5.1.2,f; 5.1.2,g; 5.1.7,b; and 5.2.3,b. The plant was shut down on May 28, 1983, for refueling and maintenance (Ref. 103).

Cause of the Problem. A number of possible failure mechanisms were evaluated by the licensee (Ref. 103). It was concluded that the fuel rod failures apparently resulted from multiple sources (but none of these were indicative of a condition that may lead to continued serious fuel cladding degradation). Evidence of debris-induced fretting wear of cladding was found. Also, the licensee found one case of confirmed grid spring/fuel rod fretting, which was probably caused by damage to a cell that most likely occurred during fuel manufacturing or handling.

Corrective Action. Prior to the shutdown, power ramp rate and control rod movement restrictions were instituted to prevent additional defects (Refs. 104 and 105). Also, increases in power were limited to 3% per hour on a trial basis (Refs. 106 and 107).

A fuel rod failure investigation program was established by the licensee (Ref. 103). Subsequent to the plant shutdown, the entire core (217 fuel assemblies) was sipped and the results indicated the presence of 26 fuel assemblies that are believed to contain at least one leaking fuel rod per fuel assembly (Ref. 108). Of the 26 leaking assemblies, 5 are Combustion Engineering fuel assemblies and 21 are Westinghouse fuel assemblies (16 of the 21 had completed their first cycle of irradiation). Ultrasonic inspection results subsequently indicated that there were 32 failed fuel rods in the 26 assemblies (Ref. 103).

The licensee performed an extensive cleanup of the primary system to reduce the possibility of debris-induced wear (Ref. 103).

The problems described above necessitated a revision to the licensee's originally planned reload core for Cycle 6. A combination of new and previously discharged fuel assemblies were used to replace the leaking and two damaged fuel assemblies. Nine fuel assemblies, each with a single broken hold-down spring, were also used for Cycle 6 (the licensee decided that repair of these springs on the irradiated fuel assemblies would involve a high risk of damage to fuel pins).

NRC Action (Ref 103). The Cycle 6 core reload changes, together with operation without the thermal shield, were submitted to the NRC for approval. NRC approval was granted on December 30, 1983, and the plant achieved criticality on January 5, 1984. The NRC included this event in the May 1984 issue of Power Reactor Events, NUREG/BR-0051, Vol. 5, No. 6.

d. Point Beach-1

The recent Westinghouse report (Ref. 82) indicates that failed fuel assemblies were found in Point Beach-1 in 1983. As noted above, the failures were attributed to debris-induced fretting.

e. Surry-1

As described in the previous annual report (Ref. 5, pp. 39-40), the reactor coolant activity during Cycle 6 indicated the presence of leaking fuel assemblies. An intensive fuel examination program was initiated during the Cycle 6/7 refueling outage (Refs. 109 and 110). The licensee performed visual examination by remote video camera, fuel sipping operations, and ultrasonic measurements (to detect water in the fuel rods) on the fuel assemblies in the spent fuel pool following defueling. Some 36 of the 157 fuel assemblies were determined to be defective and were being evaluated. Blisters and through-wall holes were visually observed on several fuel rods. The fuel rod end cap or top plug was missing from three fuel rods, where the upper spring was visible. The abnormal degradation of the fuel cladding was reported in LER 83-014/01T-0 (Ref. 109), which will be updated when the cause of the failures has been determined. Also see 5.1.2, m and 5.1.4, b. The recent Westinghouse report (Ref. 82) indicates that, as noted above, Surry-1 was one of the plants where fuel failures due to debris-induced fretting have been found.

5.1.2 Iodine Spiking or Radioactive Gas Release

Iodine spiking (i.e., a temporary increase in coolant iodine concentration) is frequently observed at reactors where leaking fuel rods are present. These temporary increases in iodine concentrations have been observed to occur following shutdowns, start-ups, rapid power changes, and coolant depressurizations. An iodine spike is characterized by a rapid increase in the iodine concentration in the coolant by as much as three orders of magnitude, followed by a return to prespike concentration. 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 17) 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. It is important to note in Table 17 that there are

TABLE 17. Technical Specifications for Primary Coolant Activity^(a)

Plant	Type ^(b)	Limits on Coolant Activity		
		Dose Equivalent Iodine-131, ($\mu\text{Ci/g}$)	Other Isotopes, Value/E ^(c)	Standard ^(d)
Arkansas-1 ^(e)	PWR	3.5	72/E	No
Arkansas-2 ^(f)	PWR	1.0	100/E	Yes
Beaver Valley-1	PWR	1.0	100/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/E	Yes
Callaway-1	PWR			
Calvert Cliffs-1 and-2	PWR	1.0	100/E	Yes
Cook-1 and-2	PWR	1.0	100/E	Yes
Cooper Station	BWR	3.1	(g)	No
Crystal River-3	PWR	1.0	100/E	Yes
Davis-Besse-1	PWR	1.0	100/E	Yes
Diablo Canyon-1	PWR	1.0	100/E	Yes
Dresden-2	BWR	20.0 ^(h)	(g)	No
Dresden-3	BWR	20.0 ^(h)	(g)	No
Duane Arnold	BWR	1.2 ⁽ⁱ⁾	(g)	No
Farley-1 and-2	PWR	1.0	100/E	Yes
Fitzpatrick	BWR	3.1	(g)	No
Fort Calhoun-1	PWR	1.0	100/E	Yes
Fort St. Vrain	HTGR			
Ginna	PWR	3.0	84/E	No
Grand Gulf-1	BWR	0.2	100/E	Yes
Haddam Neck	PWR	(g)	68/E	No
Hatch-1	BWR	10	(g)	No
Hatch-2	BWR	0.2	100/E	Yes
Indian Point-2	PWR	(g)	60/E	No
Indian Point-3	PWR	1.0	100/E	Yes
Kewaunee	PWR	(g)	91/E	No
La Crosse	BWR	0.2	100/E	Yes
La Salle-1 and-2	BWR	0.2	100/E	Yes
Maine Yankee	PWR	1.0	100/E	Yes
McGuire-1 and-2	PWR	1.0	100/E	Yes
Millstone-1	BWR	0.2	100/E	Yes
Millstone-2	PWR	1.0	100/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/E	Yes
Oconee-1,-2, and-3	PWR	(g)	224/E	No
Oyster Creek-1	BWR	8.0 ^(h)	(g)	No
Palisades	PWR	1.0	100/E	Yes
Palo Verde	PWR			
Peach Bottom-2 and-3	BWR	2.0	(g)	No
Pilgrim-1	BWR	20.0 ^(h)	(g)	No

TABLE 17. (contd)

Plant	Type ^(b)	Limits on Coolant Activity		
		Dose Equivalent Iodine-131, ($\mu\text{Ci/g}$)	Other Isotopes, Value/ $\bar{E}$ ^(c)	Standard ^(d)
Point Beach-1 and-2	PWR	1.0	100/ $\bar{E}$	Yes
Prairie Island-1 and-2	PWR	1.0	100/ $\bar{E}$	Yes
Quad Cities-1 and-2	BWR	5.0	(g)	No
Rancho Seco-1	PWR	(g)	43/ $\bar{E}$	No
River Bend-1	BWR			
Robinson-2	PWR	1.0	100/ $\bar{E}$	Yes
Salem-1 and -2	PWR	1.0	100/ $\bar{E}$	Yes
San Onofre-1,-2, and-3	PWR	1.0	100/ $\bar{E}$	Yes
Sequoyah-1 and -2	PWR	1.0	100/ $\bar{E}$	Yes
St. Lucie-1 and-2	PWR	1.0	100/ $\bar{E}$	Yes
Summer-1	PWR	1.0	100/ $\bar{E}$	Yes
Surry-1 and-2	PWR	1.0	100/ $\bar{E}$	No ^(j)
Susquehanna-1, and-2	BWR	0.2	100/ $\bar{E}$	Yes
Three Mile Island-1	PWR	(g)	130/ $\bar{E}$	No
Trojan	PWR	1.0	100/ $\bar{E}$	Yes
Turkey Point-3 and-4	PWR	1.0 ^(k)	100/ $\bar{E}$	Yes
Washington Nuclear-2 ^(l)	BWR	0.2	100/ $\bar{E}$	Yes
Vermont Yankee-1	BWR	1.1	(g)	No
Watts Bar	PWR			
Yankee Rowe-1	PWR	1.0	100/ $\bar{E}$	Yes
Zion-1 and-2	PWR	1.0	100/ $\bar{E}$	Yes

(a) As of December 1981.

(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 Dose Equivalent Iodine-131.

(i) Increased sampling required when Dose Equivalent Iodine-131 is $>0.012 \mu\text{Ci/g}$.

(j) Other aspects are not standard.

(k) Not specified as either total iodine or Dose Equivalent Iodine-131

(l) Washington Public Power Supply System (WPPSS) Nuclear Project No. 2

differences in reporting requirements: a number of BWRs have allowable limits on coolant activity that are substantially higher than those for many of the PWRs.

As shown in Table 18, iodine spiking events occurred in 1983 at 10 PWRs: Calvert Cliffs-2, Davis Besse-1, Ginna, Millstone-2, North Anna-1, Palisades, Prairie Island-2, San Onofre-3, St. Lucie-1, and Surry-1. Iodine spiking events were reported in 1983, 1982, 1981, and 1980 by two reactors (Davis Besse-1 and St. Lucie-1); in 1983, 1982, and 1981 by two reactors (North Anna-1 and Palisades); and in 1983 and 1982 by one reactor (Millstone-2). One PWR (Ft. Calhoun-1) reported that an iodine spiking event occurred in early 1984. The 1983 and 1984 iodine spiking events and information on iodine spiking received and/or published by the NRC in 1983 and in early 1984 are discussed below:

a. Calvert Cliffs-2 (Refs. 111-122)

The dose equivalent iodine-131 in the reactor primary coolant system exceeded the limit of 1.0 $\mu\text{Ci/g}$ in Technical Specification 3.4.8 nine times in 1983: February 11 (Refs. 111 and 112), March 6 (Refs. 111 and 112), May 15 (Refs. 113 and 114), August 9 (Refs. 115 and 116), August 23 (Refs. 115 and 116), September 1 (Refs. 117 and 118), September 17 (Refs. 117 and 118), October 11 (Refs. 119 and 120), and December 17 (Refs. 121 and 122).

Cause of the Problem. For the first eight events, the licensee indicated that the iodine spiking was caused by a very small number of leaking fuel rods and a change in power level. For the event on December 17, 1983, the licensee indicated that the depressurization shock associated with the shutdown (reactor trip) resulted in an "iodine spike" from the fuel. This last event is similar to an earlier event in Calvert Cliffs-1 (Refs. 123 and 124), which is described in the fuel performance annual report for 1982 (Ref. 5).

Additional information provided by Combustion Engineering indicates that at the end of the cycle in 1984 all the fuel was sipped. Four assemblies were found to each contain a single leaking fuel rod. Visual observations indicated that three of the four rods were most likely failed by debris in the primary coolant. No cause was determined for the failure of the fourth rod.

Corrective Action. For the first eight events, the sampling frequency was increased, as stipulated in Technical Specification 3.4.8, until the levels decreased. No further action was deemed necessary by the licensee. Regarding the last event, no additional corrective action was deemed necessary by the licensee other than purification of the system utilizing an ion exchanger. The events in 1983 were terminated as shown in Table 19. As noted above, all fuel was sipped at the end of the cycle in 1984. The four failed rods were replaced with four stainless steel dummy rods and the assemblies returned to the core for the next cycle.

TABLE 18. Iodine Spiking Events

Reactor	Reactor Type		Nuclear Steam System Supplier (NSSS) ^(a)					Iodine Spiking or Radioactive Gas Release				
	BWR	PWR	AC	B&W	C-E	GE	<u>W</u>	1980	1981	1982	1983	1984
Arkansas-1		•		•					X			
Arkansas-2		•			•			X	X			
Brunswick-2	•					•		X	X	X		
Calvert Cliffs-1		•			•					X		
Calvert Cliffs-2		•			•						X	
Cook-2		•					•		X	X		
Crystal River-3		•		•				X	X			
Davis Besse-1		•		•				X	X	X	X	
Farley-1		•					•			X		
Ft. Calhoun-1		•			•							X
Ginna		•					•				X	
Hatch-2	•					•				X		
La Crosse	•		•						X			
Millstone-2		•			•					X	X	
North Anna-1		•					•		X	X	X	
Palisades		•			•				X	X	X	
Prairie Island-1		•					•			X		
Prairie Island-2		•					•				X	
San Onofre-3		•			•						X	
St. Lucie-1		•			•			X	X	X	X	
Surry-1		•					•		X		X	
Surry-2		•					•			X		
Trojan		•					•		X			
								<u>5</u>	<u>12</u>	<u>12</u>	<u>10</u>	<u>1</u>

(a) Allis-Chalmers (AC), Babcock & Wilcox (B&W), Combustion Engineering (C-E), General Electric (GE), and Westinghouse (W).

TABLE 19. Data On Iodine Spiking Events at Calvert Cliffs-2 in 1983

Peak Activity			Event Terminated (i.e., < 1.0 $\mu\text{Ci/g}$)	
Date	Time	$\mu\text{Ci/g}$	Date	Time
Feb. 11	2125	1.07	Feb. 12	0125
Mar. 6	1630	1.6	Mar. 6	2010
May 15	1900	1.38	May 16	0535
Aug. 9	2200	2.03	Aug. 11	1347
Aug. 23	*	1.08	Aug. 23	1545
Sept. 1	0115	1.05	Sept. 1	0450
Sept. 17	*	1.29	Sept. 17	1555
Oct. 11	1445	1.03	Oct. 11	1845
Dec. 17	0850	1.413	Dec. 17	1500

* Not stated.

b. Davis Besse-1 (Refs. 125-130)

On five days in 1983 (Jan. 15, Jan. 18, April 9, April 10, and May 10) the reactor coolant system dose equivalent iodine-131 exceeded the limit of 1.0 $\mu\text{Ci/g}$ in Technical Specification 3.4.8. The peak levels on the respective days were 2.67, 1.39, 1.39, 1.95, and 1.16 $\mu\text{Ci/g}$. At no time did the specific activity exceed the allowable limit of Technical Specification 3.4.8, which accommodates possible iodine spiking phenomena that may occur following a change in thermal power. Previous occurrences at this unit have been reported in LER 81-031 (Ref. 131), which is described in 5.1.2,c; LER 81-016; and LER 80-088 and discussed in the fuel performance annual reports for 1982 (Ref. 5) and 1981 (Ref. 4).

Cause of the Problem. The cause was a slight leakage of fission products through the fuel cladding. Some leakage is normal following a reactor trip due to reactor coolant system temperature and pressure changes.

Corrective Action. The iodine-131 level was monitored until it dropped below the Technical Specification limit. Table 20 (Ref. 126) lists the specific activity analysis, the time duration the specific activity exceeded the Technical Specification, and the maximum level reached for each occurrence.

c. Davis Besse-1 (Refs. 131-134)

The iodine spikes (LER 81-031) that occurred on May 12, July 30, Sept. 2, Oct. 16, Oct. 23, and Dec. 28, 1981, and on Dec. 4, 1982, are described in the fuel performance annual reports for 1982 (Ref. 5, p. 36) and 1981 (Ref. 4, p. 27). Revision 6 to that LER is included in a report (Ref. 131) published by the NRC in 1983.

TABLE 20. Data Associated with LER 83-002 at Davis Besse-1

Date of Occurrence	Specific Activity Analysis			Approximate Time Duration Tech. Spec. Level Exceeded	Maximum Level ($\mu\text{Ci/g}$)
	Date	Time	Activity ($\mu\text{Ci/g}$) Dose Equivalent I-131		
1/15/83	1/15/83	0810	0.17	21 hours	2.67
		2004	2.67		
		2155	2.54		
	1/16/83	0102	2.33		
		0400	2.40		
		0730	1.59		
		0930	1.58		
		1330	1.14		
		1730	0.73		
1/18/83	1/18/83	1354	0.14	8 hours	1.39
		2301	1.39		
	1/19/83	0256	1.11		
		0645	0.81		
4/9/83	4/8/83	1207	0.09	11 hours	1.39
		0420	1.06		
		0806	1.37		
		1145	1.39		
		1531	0.96		
4/10/83	4/10/83	0130	0.41	15 hours	1.95
		0620	1.95		
		1002	1.82		
		1341	1.31		
		1710	1.30		
		2110	0.89		
5/10/83	5/10/83	0811	0.06	8 hours	1.16
		1323	1.16		
		1600	0.95		
		2000	0.85		
	5/11/83	0820	0.33		

d. Ft. Calhoun-1 (Ref. 135)

During a normal plant shutdown for refueling, it was discovered on March 3, 1984, during a routine reactor coolant sample analysis that the reactor coolant radioactivity was in excess of the 1.0 $\mu\text{Ci/g}$ dose equivalent iodine-131. The event involves Technical Specifications 2.1.3(5) and 5.9.2.

Cause of the Problem. The cause was not stated in the report (Ref. 135) published by the NRC.

Corrective Action. The sample schedule (i.e., once every eight hours whenever the radioactivity exceeds 1.0 $\mu\text{Ci/g}$ dose equivalent iodine-131) of Table 3-4 of the Technical Specifications was initiated. The dose equivalent iodine-131 returned to below the limit at 0830 on March 4, 1984, at which time the every-eight-hour sampling was terminated.

e. Ginna (Refs. 136 and 137)

Following a unit trip on Jan. 17, 1983, from 100% power, an iodine-131 spike occurred in the reactor coolant system. Prior to the event, the iodine-131 concentration in the reactor coolant system was 0.0066 $\mu\text{Ci/g}$. Following the trip, the concentration rose to 0.127 $\mu\text{Ci/g}$ and peaked to 0.512 $\mu\text{Ci/g}$ during plant restart. This event is a limiting condition for operation (LC0), based on Technical Specification 3.1.4.3.a.

Cause of the Problem. The cause was not stated in LER 83-005 (Ref. 137) or in the report (Ref. 136) published by the NRC.

Corrective Action. The iodine-131 concentration in the reactor coolant system was returned to normal levels by normal letdown and no further actions were performed. After 21.7 hours above 0.2 $\mu\text{Ci/g}$, the iodine-131 concentration returned to 0.008 $\mu\text{Ci/g}$.

f. Millstone-2

The reactor coolant system iodine level exceeded the limit of 1.0 $\mu\text{Ci/g}$ in Technical Specification 3.4.8.a on May 28, 1983 (Refs. 138-141); Feb. 19, 1983 (Refs. 142 and 143); Dec. 31, 1982 (Refs. 144 and 145); Sept. 18, 1982 (Refs. 146 and 147); and June 9, 1982 (Refs. 108, 148, and 149). The duration of the event on May 28, 1983, was approximately 24 hours, as shown in Table 21. The duration of the event on Feb. 19, 1983, was approximately 20 hours, as shown in Table 22. The duration of the event on Dec. 31, 1982, was approximately 12 hours, as shown in Table 23.

Cause of the Problem. The events were believed to be caused by a small number of fuel rods with defects. See 5.1.2,g.

Corrective Action. Power increases were limited to 3% per hour (Refs. 142 and 144). The entire core was sipped during the refueling shutdown after Cycle 5; see 5.1.2,g for the results.

TABLE 21. Iodine Spiking Event on May 28, 1983, at Millstone-2

Time and Date of Sample		Dose Equivalent Iodine-131
Time	Date	($\mu\text{Ci/g}$)
0400	5/28/83	0.053
0615	5/28/83	1.569
0915	5/28/83	2.087
1115	5/28/83	1.923
1405	5/28/83	1.751
1715	5/28/83	1.594
2000	5/28/83	1.572
2300	5/28/83	1.386
0200	5/28/83	1.203
0515	5/28/83	1.006
0915	5/28/83	0.843

TABLE 22. Iodine Spiking Event on Feb. 19, 1983, at Millstone-2

Time and Date of Sample		Dose Equivalent Iodine-131
Time	Date	($\mu\text{Ci/g}$)
0540	2/19/83	1.132
0740	2/19/83	1.299
0935	2/19/83	1.318
1215	2/19/83	1.235
1550	2/19/83	1.144
1940	2/19/83	1.054
2340	2/19/83	1.018
0255	2/19/83	0.8764

TABLE 23. Iodine Spiking Event on Dec. 31, 1982, at Millstone-2

Time and Date of Sample		Dose Equivalent Iodine-131
Time	Date	($\mu\text{Ci/g}$)
0130	12/31/82	0.047
0450	12/31/82	0.022
1000	12/31/82	0.783
1345	12/31/82	1.216
1745	12/31/82	1.204
2147	12/31/82	1.073
1045	1/1/83	0.882
0545	1/1/83	0.755

g. Millstone-2 (Refs. 108, 138, 139, 148, and 149)

Higher than normal concentrations of iodine-131 in the reactor coolant system on June 9, 1982, during Cycle 5 operation indicated probable fuel cladding failures. (The estimate at that time was 10-30 rods.) This event was described in the fuel performance annual report for 1982 (Ref. 5, pp. 36-37) and at that time the cause of the event was unknown, but it had been speculated that a fuel pellet-cladding interaction (PCI) mechanism or fretting may have been involved. A revision (consists of an informational report) to the LER was submitted on Dec. 1, 1983, to the NRC (Refs. 148 and 149). Other iodine spiking events at this unit in 1982 and 1983 are described in 5.1.2,f.

Cause of the Problem. The informational report (Refs. 148 and 149) indicates that the identified fuel rod failures appear to be from multiple sources and do not indicate a situation that will lead to continued degradation of the fuel cladding.

Corrective Action. At the end of Cycle 5, fuel sipping was performed on the entire core (217 assemblies) and revealed 26 fuel assemblies with failed cladding. Of the 26 leaking fuel assemblies, 5 were manufactured by Combustion Engineering and 21 by Westinghouse. In addition, visual and ultrasonic examinations were performed. The ultrasonic examinations revealed 32 failed fuel rods in the 26 leaking assemblies. Five of the 26 leaking assemblies were scheduled for discharge and the other 21 were scheduled for reinsertion (Refs. 138 and 139). All failed fuel assemblies, except one assembly that had one leaking fuel rod, were removed from the revised Cycle 6 loading pattern.

h. North Anna-1

The reactor coolant system dose equivalent iodine-131 specific activity exceeded the 1.0 $\mu\text{Ci/g}$ limit in Technical Specification 3.4.8.a on seven days in 1983: Nov. 18 (Refs. 150 and 151), Sept. 28 (Refs. 152 and 153), July 23 (Refs. 154 and 155), July 10 (Refs. 156 and 157), June 6 (Refs. 158 and 159), May 28 (Refs. 160 and 161), and May 9 (Refs. 162 and 163). Six iodine spiking events (two in 1983 and four in 1982) at this unit were discussed in the fuel performance annual report for 1982 (Ref. 5, pp. 37-38). Data for 1983 events are shown in Tables 24-30.

Cause of the Problem. The events were caused by known, but not specifically identified, minor fuel defects in the core. Iodine spiking occurs when rapid changes in reactor power are made if minor fuel defects are present. Iodine spiking with power changes is a well documented and widely accepted phenomenon (Refs. 150 and 151).

Corrective Action. The accelerated sampling frequency of Technical Specification 3.4.8 was implemented until the reactor coolant specific activity returned to less than the limit in Technical Specification 3.4.8.a.

TABLE 24. Iodine Spiking Event on Nov. 18, 1983, at North Anna-1

Time and Date of Sample		Dose Equivalent Iodine-131 ($\mu\text{Ci/g}$)
Time	Date	
0206	11/18/83	0.06
--Reactor trip at 0808 on 11/18/83--		
0850	11/18/83	0.089
1100	11/18/83	1.71
1320	11/18/83	1.84
1510	11/18/83	1.83
1700	11/18/83	1.76
1905	11/18/83	1.71
2300	11/18/83	1.23
0301	11/19/83	1.07
0700	11/19/83	0.49

TABLE 25. Iodine Spiking Event on Sept. 28, 1983, at North Anna-1

Time and Date of Sample		Dose Equivalent Iodine-131 ($\mu\text{Ci/g}$)
Time	Date	
1305	9/28/83	0.0638 (pre-trip sample)
1534	9/28/83	2.80
1730	9/28/83	2.42
2136	9/28/83	1.87
0151	9/29/83	1.58
0540	9/29/83	1.30
0927	9/29/83	1.10
1300	9/29/83	1.30
1747	9/29/83	1.19
2105	9/29/83	0.74

TABLE 26. Iodine Spiking Event on July 23, 1983, at North Anna-1

Time and Date of Sample		Dose Equivalent Iodine-131 ($\mu\text{Ci/g}$)
Time	Date	
0042	7/23/83	0.0603 (pre-shutdown sample)
1655	7/23/83	1.99
2045	7/23/83	2.62
0100	7/24/83	2.06
0305	7/24/83	1.73
0630	7/24/83	1.20
1120	7/24/83	0.728

TABLE 27. Iodine Spiking Event on July 10, 1983, at North Anna-1

Time and Date of Sample		Dose Equivalent Iodine-131 ($\mu\text{Ci/g}$)
Time	Date	
0054	7/9/83	0.09
0047	7/10/83	0.08
--Reactor trip at 1842 on 7/10/83--		
1924	7/10/83	1.90
2140	7/10/83	2.38
2340	7/10/83	2.19
0315	7/11/83	1.57
0800	7/11/83	0.95
1100	7/11/83	0.62

TABLE 28. Iodine Spiking Event on June 6, 1983, at North Anna-1

Time and Date of Sample		Dose Equivalent Iodine-131 ($\mu\text{Ci/g}$)
Time	Date	
2124	6/6/83	1.44
2329	6/6/83	1.34
0134	6/7/83	1.09
0400	6/7/83	0.88
0758	6/*/83	0.81

* Not stated.

TABLE 29. Iodine Spiking Event on May 28, 1983, at North Anna-1

Time and Date of Sample		Dose Equivalent Iodine-131 ($\mu\text{Ci/g}$)
Time	Date	
0121	5/28/83	0.104
0437	5/28/83	1.66
0630	5/28/83	1.97
0833	5/28/83	1.89
1233	5/28/83	1.27
1630	5/28/83	0.98
1935	5/28/83	0.88

TABLE 30. Iodine Spiking Event on May 9, 1983, at North Anna-1

Time and Date of Sample		Dose Equivalent Iodine-131 ($\mu\text{Ci/g}$)
Time	Date	
1800	5/9/83	1.92
2057	5/9/83	2.25
2200	5/9/83	1.89
0000	5/10/83	1.69
0203	5/10/83	1.66
0400	5/10/83	1.07
0730	5/10/83	1.69
0945	5/10/83	1.24
1300	5/10/83	0.89

i. Palisades (Refs. 164-167)

During plant startup, the primary coolant system sample revealed the dose equivalent iodine-131 to be $1.06 \mu\text{Ci/g}$, which exceeds the $1.0 \mu\text{Ci/g}$ limit in Technical Specification 3.1.4. The out-of-specification condition lasted approximately one hour. The iodine spiking events on October 17 and 30, 1982, at this plant (Refs. 166 and 167) were discussed in the fuel performance annual report for 1982 (Ref. 5, p. 38).

Cause of the Problem. The LER submitted in February 1983 (Refs. 164 and 165) states that there are no indications of abnormal cladding degradation and that iodine spiking following a change in thermal power is attributed to normal fuel leakage consistent with previously observed leakage during this cycle. The report (Ref. 168) issued by Exxon Nuclear in September 1983 indicates that the coolant activity increases in Palisades could be attributed to fuel failures.

Corrective Action. Procedures will be revised to increase letdown purification flow following a trip or major reduction in thermal power and when a return to power is anticipated.

j. Prairie Island-2

On three occasions in 1983, iodine spikes occurred following reactor shutdowns or trips. Spikes in dose equivalent iodine-131 values occurred in the reactor coolant system activity up to 3.18 Ci/g on June 9 (Ref. 169), up to 1.88 Ci/g on May 20 (Ref. 170), and up to 3.26 Ci/g on April 16 (Refs. 171 and 172). The limit in Technical Specification 3.1.D.1.a is 1.0 Ci/g dose equivalent iodine-131 (Ref. 172). The primary coolant specific activity prior to May 20, 1983, was at an equilibrium value of approximately $0.013 \mu\text{Ci/g}$ dose equivalent iodine-131 and prior to April 16, 1983, it was approximately 0.03 Ci/g . Data on the event that occurred on April 16, 1983, are shown in Table 31.

TABLE 31. Iodine Spiking Event on April 16, 1983, at Prairie Island-2

Time and Date of Sample		Dose Equivalent Iodine-131 ($\mu\text{Ci/g}$)
Time	Date	
1102	4/9/83	0.025
0110	4/10/83	0.0255
1758	4/10/83	0.0314
0729	4/11/83	0.0288
0730	4/12/83	0.0314
0729	4/13/83	0.0322
0801	4/14/83	0.0289
0808	4/15/83	0.0321
2045	4/15/83	0.0281
0153	4/16/83	3.26
0230	4/16/83	2.84
0810	4/16/83	1.74
1603	4/16/83	1.14
1958	4/16/83	1.10
0029	4/17/83	0.684
0500	4/17/83	0.637
0823	4/17/83	0.438
1312	4/17/83	0.315
1955	4/17/83	0.323
0400	4/18/83	0.142

Cause of the Problem. The activity is believed to have been caused by washout of fission products from the existing defects following reactor shut-down or trip. The reactor has been operating since mid-March 1983 with known fuel defects (estimated to be only one leaking fuel rod). There is no evidence of new defects being formed as a result of these transients.

Corrective Action. Cleanup flow was increased to 80 gpm until the activity had decreased below the limit (Ref. 172). Current planning is to limit load-follow operations on the unit to reduce iodine spiking and to inspect the fuel during the next outage to find and remove the leaking fuel rod (Ref. 172).

k. San Onofre-3 (Refs. 173-175)

The reactor coolant system specific activity exceeded the $1.0 \mu\text{Ci/g}$ dose equivalent iodine-131 limit in 1983 on Dec. 22, on Dec. 23 at 0100, and on Dec. 24 at 1145.

Cause of the Problem. These events were indications of iodine spiking following power changes associated with two reactor trips.

Corrective Action. The licensee invoked and met action statements "A" and "D" of LCO 3.4.7 in each event. As corrective action for each event,

purification flow was increased. The reactor coolant system specific activity was reduced to less than the limit at 1220 on Dec. 22, at 1230 on Dec. 23, and at 1945 on Dec. 24, 1983.

NRC Action. In August 1984, the NRC issued its evaluation (Ref. 175) of the basis and technical adequacy of the licensee's decision to continue to operate the unit in view of the high level of primary coolant activity experienced to date. The NRC review produced the following conclusions:

1. The licensee now estimates that 50 to 100 fuel rods have cladding defects.
2. The iodine ratio suggests that the defects are relatively large. Therefore, waterlogging should not be a problem.
3. The coolant activity has stabilized at about half the allowable Technical Specification limit, which suggests that no additional rods are failing.
4. The NSSS vendor, Combustion Engineering, has stated that potential for additional fuel damage in the current cycle will be reduced by keeping future power ascensions within previously used rates.

Based on the above, the NRC does not recommend any additional limitations on plant operation at this time. The NRC is continuing to monitor the coolant activity at San Onofre-3. Should the situation change in the future, the NRC will, of course, re-evaluate its conclusions, and take appropriate action as necessary.

1. St. Lucie-1

The reactor coolant system specific activity exceeded the Technical Specification 3.4.8.a limit of 1.0 $\mu\text{Ci/g}$ on February 26 (Refs. 176 and 177) and January 8, 1983 (Refs. 178 and 179). The latter event and the six events in 1982 were noted in the fuel performance annual report for 1982 (Ref. 5, p. 39). The event on February 26, 1983, is the 15th occurrence of this type at this unit (e.g., see Refs. 176-183). Data on the January 8, 1983, event are shown in Table 32.

TABLE 32. Iodine Spiking Event on Jan. 8, 1983, at St. Lucie-1

<u>Time and Date of Sample</u>		<u>Dose Equivalent Iodine-131 ($\mu\text{Ci/g}$)</u>
<u>Time</u>	<u>Date</u>	
0925	1/7/83	0.069
1145	1/8/83	1.023
1545	1/8/83	0.960
0640	1/9/83	0.920

Cause of the Problem. Iodine spiking is expected on power reductions based on past operating history (Refs. 176 and 177). After an extended period of power operation with a nominal level of fuel leakage, the reductions in power on January 8 and February 26 were sufficient transients to cause iodine buildup (iodine spiking phenomenon) in the primary coolant.

Corrective Action. The specific activity was first measured above its limit on Feb. 26, 1983, at 2235 and remained above the limit for approximately 28 hours. With the event on Jan. 8, 1983, the specific activity remained above the limit for four hours.

m. Surry-1

Three times in 1983 (Dec. 15, Nov. 22, and Feb. 7) the activity samples of the reactor coolant system indicated dose equivalent iodine-131 levels greater than the 1.0 $\mu\text{Ci/g}$ limit in Technical Specification 3.1.D.2. Durations of the events on Dec. 15, Nov. 22, and Feb. 7, 1983, were approximately 12, 22.5, and 36 hours, respectively. There were also 14 events in 1982 and 7 events in 1981; these are described in the fuel performance annual report for 1982 (Ref. 5, pp. 39-40). Data on the three events in 1983 are shown in Tables 33-35.

Cause of the Problem. The event was caused by known, but not specifically identified, fuel element defects in the reactor core. Post-rampdown/post-trip conditions in the core enhanced the release of fission products to the reactor coolant system, resulting in an iodine spike.

Corrective Action. An accelerated sampling frequency was implemented until the reactor coolant system specific activity returned to less than the Technical Specification 3.1.D.2 limit. Concerning the event on Dec. 15, 1983, no further corrective actions were to be taken at the time (Refs. 184 and 185). Following Cycle 6 operation, an intensive fuel examination program was initiated; some results of the examinations are described in 5.1.1,d and 5.1.4,b and also in the fuel performance annual report for 1982 (Ref. 5, pp. 39-40). Additional results (Ref. 109) will be described in the next annual report.

TABLE 33. Iodine Spiking Event on Dec. 15, 1983, at Surry-1

Time and Date of Sample		Dose Equivalent Iodine-131 ($\mu\text{Ci/g}$)
Time	Date	
1600	12/15/83	1.68
1800	12/15/83	1.52
1955	12/15/83	1.55
2230	12/15/83	1.37
0015	12/16/83	1.21
0400	12/16/83	0.685

TABLE 34. Iodine Spiking Event on Nov. 22, 1983, at Surry-1

Time and Date of Sample		Dose Equivalent Iodine-131 ($\mu\text{Ci/g}$)
Time	Date	
0620	11/22/83	2.17
0810	11/22/83	2.60
1010	11/22/83	2.59
1200	11/22/83	2.10
1600	11/22/83	2.08
2000	11/22/83	1.50
0015	11/23/83	1.31
0355	11/23/83	0.855

TABLE 35. Iodine Spiking Event on Feb. 7, 1983, at Surry-1

Time and Date of Sample		Dose Equivalent Iodine-131 ($\mu\text{Ci/g}$)
Time	Date	
2015	2/7/83	0.128 (pre-trip sample)
2305	2/7/83	7.20
0100	2/8/83	8.17
0300	2/8/83	8.08
0450	2/8/83	7.05
0738	2/8/83	5.78
0905	2/8/83	5.26
1305	2/8/83	3.46
1705	2/8/83	3.11
2000	2/8/83	2.14
2105	2/8/83	1.99
0105	2/9/83	1.75
0500	2/9/83	1.34
0730	2/9/83	1.28
0900	2/9/83	1.02
1110	2/9/83	0.827

5.1.3 PWR Fuel Assemblies Damaged by Baffle Jetting

a. Farley-1 (Refs. 186-189)

During unloading of the entire core for the fourth refueling and upflow modification work, fuel cladding damage was visually observed on 11 Cycle 4 baffle fuel assemblies on Jan. 30, 1983. The damaged assemblies were located at corner injection baffle joints, primarily at the top fuel span between grids 7 and 8. Some pellet and rod debris was found in and around the reactor vessel. Videotapes of the 11 assemblies exhibiting visual damage were forwarded to the NRC (Ref. 189). In addition to the 11 damaged fuel assemblies

visually observed, one assembly located at a center injection baffle joint in Cycle 4 and three assemblies in interior core positions were determined to be leaking based on results from sipping.

Cause of the Problem. The fuel cladding damage was due to vibrations induced by water jetting through the baffle joints. The baffle jetting is the result of a pressure differential across the baffle plate joints that exists due to reactor coolant flowing down on the outer surface and flowing up on the core side of the baffle plates.

Corrective Action. The damaged and leaking assemblies have been discharged, are stored in the spent fuel pool, and will not be utilized in the next reload core. A design modification was implemented to convert the coolant flow from down flow to up flow on the baffle exterior. Specifically, the modification was accomplished by plugging holes in the core barrel, by drilling holes in the upper former plate, and by closing all baffle gaps. The modification, performed under 10CFR50.59, was completed during the refueling outage that began on January 14, 1983 (Ref. 189). The licensee expects the modification to prevent recurrence of fuel damage due to baffle jetting.

5.1.4 Cladding Degradation

a. Salem-1 (Refs. 190-193)

On Nov. 21, 1982, during routine refueling operations, video inspection of removed fuel assemblies revealed cladding ruptures on one rod in fuel assembly D-20, which had been in the core for three cycles. The event possibly constitutes abnormal degradation of the cladding and is reportable in accordance with Technical Specification 6.9.1.8.C.

Cause of the Problem. The vendor (Westinghouse) performed a detailed inspection of the assembly with the failed rod and conducted an investigation of the failure. A review of the traceability data did not indicate any trend related to manufacturing that would indicate a tendency toward failure. A supplemental report (Refs. 190 and 191) submitted to the NRC in 1983 indicated that the rod appears to have failed due to secondary hydriding, without any apparent implications or correlations with respect to other fuel in Salem-1 or in the same batch. As noted in the fuel performance annual report for 1982 (Ref. 5, p. 48), a previous cladding failure was discovered (LER 82-005/01X-1) in fuel assembly C-04 on January 31, 1982; the failure in that event was attributed to excessive rod growth due to fuel lockup.

Corrective Action. Fuel assembly D-20 was transferred to the spent fuel pit. The assembly will not be used in future cycles. The present fuel examination program will continue surveillance for any abnormalities during refueling. The licensee assumed the failure to be of an isolated nature and deemed that no further action was necessary.

b. Surry-1 (Ref. 109)

Due to fuel failure indication during Cycle 6 operation (see Section 5.1.2,m), an intensive fuel examination program was initiated. Binocular inspection of all fuel unloaded from the core revealed abnormal degradation of 3 peripheral fuel rods on March 3, 1983. Fuel assembly OB3 was observed to have one small hole above grid 2 and one large hole (approximately $\frac{1}{4}$ inch diameter) above grid 4 in one of its fuel rods. Fuel assemblies J15 and 6A2 were observed to have a missing top end and plug in one fuel rod of each assembly. Subsequent high magnification TV inspection has disclosed one additional fuel assembly, 3A2, with a top end plug missing from one fuel rod. The separation of the plugs from the rods appeared to be at or near the circumferential (girth) weld. These events are reportable per Technical Specification 6.6.2.a(3). In addition, a small hole has been confirmed in one rod each of 8 other fuel assemblies. More fuel assemblies with defects were also found, as described below.

Cause of the Problem. The cause of the fuel failures will be discussed in the next annual report. See 5.1.1,d and 5.1.2,m.

Corrective Action. All leaking Cycle 6 fuel assemblies, intended for reuse in Cycle 7, will be eliminated from the Cycle 7 core design. A redesign of the Cycle 7 reload has been initiated.

All Cycle 6 fuel intended for Cycle 7 reuse has been sipped and visually inspected by binoculars. The results to date indicate 24 of a total of 76 Batch 8 fuel assemblies have defects. Inspection of all the remaining assemblies has not yet been completed. However defects have been identified in 12 of the remaining 81 Cycle 6 assemblies. Based on the results of the above inspections, selected fuel assemblies have been videotaped and are presently being reviewed.

Other inspections of the fuel are continuing. A high magnification fiber optics inspection tool is available to examine selected fuel rod end plug welds and the use of an ultrasonic inspection technique, the Brown Boveri Reaktor GmbH Failed Fuel Rod Detection System (FFRDS), should help specify failed fuel rods not specifically discovered by visual methods. Results of these examinations will be evaluated by Westinghouse Electric Corp. and the licensee (Virginia Electric and Power Co.).

An update report will be submitted to the NRC after more detailed evaluation and analysis have been performed.

5.1.5 BWR Fuel Assemblies Loaded Out of Proper Orientation

a. Hatch-2 (Refs. 194 and 195)

While performing "Fuel Movement Operation" Procedure HNP-2-9302, personnel discovered on May 30, 1983, that the 11 fuel assemblies inserted into the core had been loaded out of proper core orientation. The first eight assemblies

inserted were not loaded into their previous core locations around their respective Source Range Monitor (SRM) but were instead loaded on the opposite side of the core around another SRM, which is contrary to the requirements of Technical Specification 4.9.2.C.2. The plant condition was not affected by this event.

Cause of the Problem. The cause of the event was attributed to personnel error. Personnel used the core orientation program for Hatch-1 to compute the Hatch-2 data package (HNP-2-9302, Data Package 3) that is used to prepare fuel movement schedules for a full core reload. Because the orientations of the two cores are 180 degrees out of phase, the result was that 11 fuel assemblies were loaded into Hatch-2's core 180 degrees out of proper core orientation.

Corrective Action. The immediate corrective actions were as follows: core alterations were suspended until other corrective actions could be implemented; responsible personnel were counseled and retrained; and procedure HNP-2-9302 and its data sheets were revised, corrected and then verified as being correct before resuming refueling operations. Also, to prevent recurrence, the computer program will be revised to first ask the operator the unit the data is to be used on. It will then apply this data to the correct core orientation program for the respective unit.

5.1.6 Control Rod Guide Tube Support Pins

a. Beaver Valley-1 (Ref. 196)

In preparation for Cycle 4, the upper internals of the reactor vessel were reworked during the refueling outage to replace the control rod guide tube support pins. See 5.2.6,a.

b. Point Beach-1 (Ref. 197 and 198)

On February 25, 1984, a visual inspection of the control rod guide tube split pins revealed that three nuts were missing. The visual inspection was performed after an ultrasonic inspection of the pins identified cracks in 67 of 74 pins (2 pins per guide tube). This event will be discussed more fully in the next fuel performance annual report.

Cause of the Problem. The pins are made of Inconel X-750 and, because of the heat treatment temperature, are subject to stress corrosion cracking.

Corrective Action. The pin design and material are generic to Westinghouse PWRs, and the cracking has been addressed in a Westinghouse bulletin and an NRC information notice. See the fuel performance annual report for 1982 (Ref. 5, pp. 42 and 43).

c. Trojan

Broken control rod guide tube support pins were discovered in June 1984 at Trojan (Ref. 199). This event will be discussed more fully in the next fuel performance annual report.

5.1.7 PWR Hold-Down Springs

a. McGuire-1 (Refs. 200-204)

While the core was unloaded (the plant was shutdown to perform modifications on the steam generator), it was discovered on March 10, 1983, during inspection of the fuel assemblies that 21 of 94 nonfuel-bearing components (NFBC) equipped with the same spring design had broken hold-down springs. This event is reportable pursuant to Technical Specification 6.9.1.12 (i). Three of the damaged springs were confirmed to have double breaks but all pieces were retained about the hub of the handling portion of the core component.

Cause of the Problem. The event is due to a design deficiency, as the broken hold-down springs are theorized to have resulted from fatigue failure.

Corrective Action. No immediate corrective action was required since the core was unloaded. No missing portions of the removed hold-down assembly springs were identified; hence, no loose parts were introduced into the reactor coolant system by these spring failures. The core was reloaded but the NFBCs that used hold-down springs of the old design were removed. A reload type safety evaluation was performed by Westinghouse to justify the modified core design.

b. Millstone-2 (Refs. 103, 108, and 205-208)

Underwater camera inspections of all 144 irradiated Westinghouse-manufactured fuel assemblies on August 11, 1983, identified 15 broken hold-down springs. At that time, the entire core was in the spent fuel pool while work was being done on the thermal shield in containment. Fifteen fuel assemblies each had one broken spring: eight of the broken springs were in once-burned "G" fuel assemblies and seven were in twice-burned "F" fuel assemblies. Each of the springs exhibited a brittle-type fracture.

A recent Westinghouse report (Ref. 82) indicates that there were 16 hold-down spring failures, each on a different fuel assembly.

In addition to the broken springs, the stainless steel top nozzle posts of Westinghouse fuel assemblies exhibited a varying degree of fretting due to interaction with the Inconel springs. This fretting has been found in both assemblies with and without broken springs. Fuel assemblies with broken springs do not necessarily exhibit fretting. The fretting was severe on some top nozzle posts.

Fourteen of the 15 broken springs show a correlation with location in the reactor core. Except for fuel assembly F-73, all posts with a broken spring

were located adjacent to the reactor core shroud in either Cycle 4 or Cycle 5. No correlation has been found with the top nozzle post fretting.

Examination of 51 of 72 discharged Batch "E" Combustion Engineering-manufactured fuel assemblies has shown no broken springs and very little fretting. These assemblies have been burned three times (i.e., the batch average burnup is approximately 33,000 MWd/MT).

There is no detectable correlation between the broken springs or fretted fuel assemblies with the failed fuel assemblies described in Millstone-2 LER 83-019/03L-1 (see 5.1.2,g).

This event is reportable under Technical Specification 6.9.1.8.i.

Cause of the Problems. The probable cause of the failures is system flow-induced spring vibration near the core periphery, leading to fatigue failure (Refs. 82, 205, and 206). The reason for the fretting of the end posts is not known (Refs. 207 and 208).

Corrective Action. Starting in Cycle 6, all new fuel will have a new spring design to reduce spring vibration and resist fatigue failure (Ref. 205 and 206). The top nozzle post fretting problem is still under investigation (Refs. 207 and 208). Further information will be provided by the licensee as it becomes available (Refs. 205 and 206).

c. Oconee-1 (Refs. 209-212)

During the refueling outage at the end of Cycle 7, all 177 fuel assemblies were examined and the video inspection revealed three, and a possible fourth, broken hold-down springs on the Oconee-1 fuel assemblies. The fourth broken spring was positively identified on July 1, 1983. This event is reportable pursuant to Technical Specification 6.6.2.1a(9). Analysis of potential loss of hold-down force, loose parts, or interference with control rod assembly (CRA) movement indicates that operation with broken hold-down springs is within the technical specifications.

Cause of the Problem. This occurrence was caused by component failure. The three Mark B6 (Inconel X-750) spring failures were caused by fatigue-induced cracking at an existing surface flaw, which then propagated by fatigue. The three springs were part of the same batch that experienced similar failures in Oconee-1 (LER 80-015), Oconee-2 (LER 82-003), and Oconee-3 (LER 82-007). The cause of the failure of the fourth spring, a Mark B7 (Inconel 718) type, is not yet known.

Correction Action. The Fuel Assembly Inspection Program will continue until future analysis of hold-down spring results indicates that such a program is no longer necessary. The broken fuel assembly hold-down springs will be replaced with mark B10 springs prior to reloading them into the core. The Mark B7 broken spring will be examined by B&W to determine the failure mode. This information will be provided when available.

d. Sequoyah-2 (Ref. 213)

There were 94 nonfuel-bearing components (NFBCs) with a total of 94 barrel-shaped UHI springs in the core during Cycle 1. The 94 NFBCs consisted of 92 burnable poison assemblies and 2 primary source assemblies. Postirradiation examination of the NFBCs during the refueling outage following Cycle 1 revealed that there were 17 broken springs with no double breaks and for which all parts were accounted. The hold-down springs that failed on the McGuire-1 NFBCs (see 5.1.7,a) and the hold-down springs on the Sequoyah-2 NFBCs are of the same design.

Cause of the Problem. Failure of these springs was due to their susceptibility to the frequency of the main coolant pump.

Corrective Action. The assemblies will be discharged at the end of Cycle 1. The springs have been redesigned to eliminate this problem.

5.1.8 Closing of PWR Rod Shoulder Gap

a. Arkansas-2 (Refs. 98-101 and 214-218)

This event at Arkansas-2 in 1982 was described in the fuel performance annual report for 1982 (Ref. 5, pp. 50-51). Refs. 98-101 and 214-217 were issued in 1983. It was reported (Refs. 98 and 99) that the previously suspected (Ref. 100) high rate of fuel assembly shoulder gap closure was confirmed.

b. St. Lucie-2 (Ref. 219)

The licensee (Florida Power and Light Co.) and Combustion Engineering (C-E) discussed the axial growth concern with the NRC at a meeting on March 3, 1983.

Cause of the Problem. At Arkansas-2 (Ref. 5, pp. 50-51), it was indicated that a combination of two factors were said to have led to this circumstance. First, the design models used by C-E to predict fuel rod growth underpredicted the actual rod growth. Second, the annealed guide tubes grew at a slower rate than predicted.

Corrective Action. Batch C fuel assemblies and 16 Batch B fuel assemblies at St. Lucie-2 were modified. The modifications involved installing shims on the guide tubes to increase the shoulder gap and machining the holddown and flow plates of the upper end fitting to a reduced thickness for compensation to maintain the overall assembly length. The licensee stated that the modifications will provide assurance that the observations at Arkansas-2 (see 5.1.8,a) have been factored into the St. Lucie-2 design.

NRC Action. The NRC issued a summary (Ref. 219) of the meeting with the licensee and the fuel vendor to discuss fuel assembly modifications to resolve the axial growth concern.

5.1.9 Deformation of PWR Spent Fuel Storage Rack

a. Maine Yankee (Refs. 220-223)

During fuel handling on November 21, 1982, the inner surface of about 10% of the spent fuel storage cells in one type of spent fuel storage rack (designated Phase 1) was observed to be deformed and bulging into the fuel assembly storage space. The deformation is associated with the aluminum-clad neutron absorber that forms the storage cell walls. This event was reportable under Technical Specification 5.9.1.6.1. An update on this spent fuel storage cell damage was provided on Jan. 21, 1983 (Refs. 220 and 221).

Cause of the Problem. The cause of the bulge was determined to be hydrogen gas at low pressure that resulted from an aluminum and water chemical reaction within the cell wall. Entrapped water and local flaws, most likely, combined to cause the bulging.

Corrective Action. One cell exhibiting inward bulging was removed and examined. The remaining bulged areas will be vented to relax the affected cell wall deformation. In the long term, these racks are scheduled to be replaced in accordance with Maine Yankee Proposed Change No. 70, which is now under NRC licensing review. The replacement racks, some of which are now nearly completely fabricated (Ref. 221), are designed to reduce the likelihood of the observed bulging (i.e., by use of stainless steel cladding and vented neutron absorber enclosures).

5.2 NEW PROBLEMS IN 1983

5.2.1 Fuel Rods in a PWR Optimized Fuel Assembly Damaged by Fretting

a. Point Beach-2 (Refs. 224 and 225)

During the shutdown for the spring 1983 refueling outage, higher than normal levels of iodine-131 in the reactor coolant prompted investigation of the fuel assemblies. Sipping of all reload fuel assemblies yielded positive results in only one of four demonstration optimized fuel assemblies (OFAs). The OFAs had attained a burnup of 20,000 MWd/MTU. Breakaway force measurements and remote visual (TV camera) inspections on May 23, 1983, located 11 rods with fretting damage in ZD3, the leaking OFA: 9 rods with through-wall penetrations and 2 rods without through-wall penetrations. The abnormal cladding degradation was reported as required by Technical Specification 15.6.9.2.A.3.

The other 3 OFAs were visually inspected and, along with 69 reload standard assemblies, were wet and dry sipped; no anomalies were noted. Also, a sampling of rods was measured for breakaway forces (no abnormal readings recorded) and visually inspected (no anomalies noted).

Cause of the problem. Fretting at the bottom spring clip grid elevation in ZD3 probably caused the through-wall penetrations of the rods. The spring clips are made of Inconel 718.

Corrective Action. Assembly ZD3 will not be reloaded into the core. A replacement candidate has been tentatively selected. Possible causes being investigated by the fuel vendor (Westinghouse) and the licensee include: the grid manufacturing process, possible problems in the fabrication practices of the OFA, and the design of the grid and assembly. An update report is to be submitted to the NRC within 30 days of the startup of the unit (planned for early July 1983). Corrective actions are to be included in that report.

5.2.2 Cladding Wear on PWR Rod Cluster Control Assemblies

a. Kewaunee (Refs. 226 and 227)

During the Cycle 9-10 refueling shutdown, an underwater visual inspection on April 1, 1984, of three rod cluster control assemblies (RCCAs) revealed apparent wear marks on the cladding of the RCCA rodlets. The wear marks were found to occur at a position that correlates to the location of the guide cards used to position the rodlets in the guide housing when the RCCAs are parked in their normally full-out position. Although this event does not meet the reporting criteria of 10CFR50.73(A), it has been reported as an LER that may be of generic interest.

Cause of the Problem. The cladding wear is attributable to the design of the guide cards and is a result of vibratory interaction between the rodlets and the guide cards during long periods of steady-state power operation.

Corrective Action. Because none of the rodlets exhibited wear that exceeds the Westinghouse criteria for RCCA wear depth, the RCCAs are acceptable for operation in Cycle 10. Visual inspections of the remaining RCCAs are presently planned for the Cycle 10-11 refueling shutdown.

5.2.3 Binding of Components and Top Nozzle Fretting in PWR Fuel Assemblies

a. Millstone-2 (Refs. 108 and 228-231)

During spent fuel visual inspections performed at the end of Cycle 5, evidence of binding of the top nozzle flower to the top nozzle posts was observed on Sept. 2, 1983, in fuel that was manufactured by Westinghouse. Binding of the flower to the top nozzle posts in fuel assembly F-73 has resulted in damage to the flower. In addition, restricted flower movement relative to one top nozzle post in fuel assembly F-37 has resulted in the deformation of a guide tube. This event involves Technical Specification 6.9.1.8.e and was briefly mentioned in the fuel performance annual report for 1982 (Ref. 5, p. 52).

Cause of the Problem. Restricted motion of the flower when it was contacted by the fuel alignment plate resulted in bending of the flower in fuel assembly F-73 in a CEA location and deformation of the guide tube in fuel assembly F-37 in a non-CEA location.

Corrective Action. Tests were performed to verify free travel of the flowers in all irradiated Westinghouse Cycle 6 fuel assemblies. Fuel assemblies F-73 and F-37 were eliminated from Cycle 6.

b. Millstone-2 (Refs. 205-208)

Severe fretting was identified on the top nozzle posts of a number of irradiated Westinghouse-manufactured fuel assemblies. See 5.1.7,b for details.

5.2.4 Fuel Investigations at Two B&W Reactors

The only anomalies related to the performance of B&W fuel occurring in 1983 included the detection by radiochemistry analysis of fuel failures at two plants. An investigation is currently being conducted to identify probable cause(s) of the failures. Thus far this investigation indicates no correlation between the incidents. The following paragraphs describe the events associated with the occurrence of the failures and the status of the investigations currently underway.

a. Arkansas-1 (Ref. 40)

Initial criticality for Cycle 6 of Arkansas-1 (i.e., Arkansas Nuclear One, Unit 1) was achieved on April 7, 1983. The startup period was typified by frequent trips from 100% full power and included two operational abnormalities that subjected the fuel to high risk situations.

The first operational abnormality occurred on September 26, 1983, when it was discovered that a group 7 control rod assembly (CRA) had been inserted to 60% beyond the group average position. The malpositioned CRA was then withdrawn to the group average position in less than one hour while the reactor was operating at 90% full power. The quadrant tilt exceeded Technical Specification limits after the incident.

During the same time period, attempts were made to initiate lithium control procedures recommended by B&W. While trying to raise lithium to approximately 1.5 ppm, an excess amount of lithium was accidentally injected to the primary coolant. Lithium levels rose from 0.5 to 2.5 ppm, a level above the recommended band for minimum crud solubility.

Defect estimates following this period indicated approximately ten leakers were present in core.

Cause of the Problem. Analysis of the cause of the increased activity at Arkansas-1 can only be speculative at best, since no fuel sipping or examinations have thus far been made. However, the type of local transient experienced by the fuel rods in the vicinity of the group 7 CRA may have induced higher than normal local strain rates in the cladding, putting the fuel at higher risk for failure.

b. Oconee-1 (Ref. 40)

Initial criticality for Cycle 7 at Oconee-2 was reached on December 7, 1983. With the exception of one reactor trip from 40% full power, a smooth startup period followed. Full power was achieved on December 17, 1983, and

maintained through February except for a brief decrease to 85% full power on January 17, 1984. At approximately five effective full power days (EFPD's) into the cycle, as full power equilibrium conditions were being established, isotopic iodine activity in the primary coolant increased rapidly. By the end of December, iodine activity equilibrated to a level indicating approximately twenty leaking rods in core. Iodine ratios indicated average-size defects. Defect estimates based on radiochemistry data from previous cycles indicate that as many as twelve leaking rods could have been leakers from previous cycles.

Cause of the Problem. An initial review of operational records and manufacturing records has failed to identify any clean-cut reason for the leaking rods. A more in-depth review is planned to be completed before the end of the cycle.

5.2.5 Misloaded Pellets in a PWR Fuel Assembly

a. Sequoyah-2 (Ref. 232)

Nineteen fuel rods containing pellets of an improper diameter and enrichment were found to be loaded into fuel assembly P39 that had been delivered to the Tennessee Valley Authority (TVA) for use in Sequoyah-2. Specifically, it amounted to approximately a 24-inch length of pellets beginning 12 inches from the bottom in each of the 19 fuel rods. The errant pellets were of a smaller diameter and a lower enrichment than the proper pellets. The incident was discovered on August 24, 1983, through a routine overcheck evaluation of inspection records and documentation by a quality control engineer.

The Westinghouse Water Reactor Divisions Safety Review Committee reviewed the factors relating to this incident and concluded that although this incident does not represent a substantial hazard, it does represent a failure to comply with 10CFR50.46, relating to substantial safety hazards. This conclusion was based upon analysis of the impact on the safety margins and licensing bases as contained in the Sequoyah-2 FSAR assuming this condition was undetected and fuel assembly P39 operated as originally delivered.

Cause of the Problem. The controls employed by Westinghouse Nuclear Fuel Division (NFD) manufacturing to prevent such an occurrence include (1) physical barriers between production lines to separate pellets with different enrichments; (2) gamma scanning of fuel rods for final acceptance; and (3) application of administrative procedures and documentation. However, departure from procedures resulted in the release of the assembly to TVA.

Corrective Action. Although the fuel assembly (rods) was shipped to the reactor site, it was not placed in the reactor. The incident was discovered and corrected prior to the loading of the fuel in the reactor. An analysis was conducted to assess the impact on reactor performance had the assembly been loaded into the core. The analysis showed that the center temperature melting point of 5080°F would have been exceeded based on present evaluation methods.

Similarly, the peak cladding temperature as contained in the current ECCS analysis for Sequoyah-2 would also exceed the Appendix K limit of 2200°F. However, coolability of the assembly and the overall core would not be affected.

Additionally, an investigation was performed to determine whether or not this event represented an isolated incident. A detailed review was made of manufacturing records for the time period that included the manufacture of the Sequoyah-2 reload and involved several thousand lots (25 rods per lot) of fuel rods. No evidence of misloaded pellets owing to pellet tray mix-up was noted. This information coupled with an assessment of the manufacturing process led to the conclusion that this misloaded assembly was an isolated event. Westinghouse NFD has taken corrective actions that result in additional controls to prevent a recurrence. The following series of actions has been identified and implemented at the Westinghouse Columbia Fuel Fabrication Plant:

1. Quality control instructions concerning operations of the gamma scan unit have been modified to clarify requirements for reporting and recording defect indications. Where previously procedure language provided certain latitude regarding when defect indications were entered into the data collection system, the requirement has been clarified to require defect input at the time a defect is indicated by the scanner.
2. A generic review of critical manufacturing and quality control procedures has been initiated to identify and clarify any ambiguity in procedure wording. In conjunction with this, meetings have been conducted by manufacturing and quality control management with operating and inspection personnel. Responsibilities for compliance to area procedures were discussed.
3. Quality Control has implemented a daily overcheck of pellet tray tags to verify correct loading by contract, enrichment and production line.
4. Procedures have been issued defining use and control of enrichment barriers on the manufacturing floor. In addition an area has been established for enrichment clean out and inspection of pellet tray carts.

In addition to the above implemented corrective actions, further actions are being explored. These include a device to be installed at rod loading stations that would prevent undersized pellets from entering the tube, administrative controls for delivery of pellet tray carts to rod loading lines, reinstitution of color coding for pellet tray tags, appropriate program changes to the computerized data collection system to permit computer verification of blend, enrichment and pellet tray numbers at rod loading, and modifications of data input required to release the fuel rods.

The NRC staff was informed of the return and replacement of the 19 defective rods. The circumstances resulting in the return and modification of assembly P39 were discussed via a telecon with the members of the NRC's Core

Performance Branch on September 27, 1983. Westinghouse has also informed TVA that the incident does not represent a substantial safety hazard but does represent a failure to comply with 10CFR50.46.

5.2.6 PWR Fuel Assembly and RCCA Damaged During Placement of Upper Internals in the Reactor Vessel

a. Beaver Valley-1 (Ref. 196)

A fuel assembly and Rod Cluster Control Assembly (RCCA) were damaged during placement of the upper internals in the reactor vessel following fuel shuffling in preparation for Cycle 4. The upper internals had been reworked during the refueling outage to replace the control rod guide tube support pins, which are used to align the bottom of the control guide tube assembly with the top of the upper core plate. This incident was briefly described in the fuel performance annual report for 1982 (Ref. 5, p. 54).

Cause of the Problem. When installing the upper internals into the reactor vessel, a control rod guide tube support misalignment in the K-2 position caused a downward deflection of the fuel assembly.

Corrective Action. The fuel assembly and RCCA in the K-2 position will be removed from the core. A new core loading pattern was developed and was included in a letter (Ref. 196), dated Sept. 1, 1983, to the NRC. The new pattern involves removal of the fuel assembly at the K-2 position and replacing it with a fuel assembly from the R-8 position; a new RCCA will also be used in core location K-2. A new Region 6A fuel assembly will be placed in the R-8 position; it will be the only fuel assembly of that type in Cycle 4. This position was chosen for the new fuel assembly since this will minimize the effects on power distribution and core peaking factors.

5.2.7 PWR Fuel Assemblies Fall or are Dropped

a. Turkey Point-4 (Refs. 233 and 234)

While lifting spent fuel assembly X-13 out of its designated rack in the Turkey Point-4 spent fuel pool on April 5, 1983, the hoisting cable on the fuel handling crane parted and fuel assembly X-13 dropped back into its rack from the fully lifted position. This event is reportable in accordance with Technical Specification 6.9.2.A.2. There is no indication of fuel cladding damage, as radiation monitor readings and survey results are unchanged from pre-event values. LERs similar to the one for this unit were 73-002 and 75-007.

Cause of the Problem. A malfunction of the two limit switches on the hoisting crane failed to stop the upward movement of the fuel assembly. This caused the cable to be overstressed and the cable subsequently parted.

Corrective Action. Repairs have been made to the limit switches. Additional inspections of fuel assembly X-13 were conducted in the spent fuel pool and no signs of cladding damage were revealed.

b. Turkey Point-4 (Refs. 235-238)

During loading of the core, it was observed on April 17, 1983, that fuel assembly X-04 had leaned across the empty core center and was resting against fuel assembly W-51. Fuel assembly X-17 was also observed to be leaning approximately 12 inches from upright. Refueling operations had been proceeding normally before the event: 3 assemblies were loaded, assembly X-04 was loaded, and then 4 more assemblies were loaded. A reconstruction of the event indicates that assembly X-04 initially struck assembly Z-06, skirted along it and displaced assembly X-17 and came to rest against assembly W-51. This event is reportable in accordance with Technical Specification 6.9.2.a.9. A similar event in this unit was LER 83-002 (Refs. 233 and 234), which is described in 5.2.7,a.

Cause of the Problem. According to all indications, assembly X-04 was set down properly. It has been concluded that assembly X-04 was on the bottom, but not properly located on the guide pins. Since assembly X-04 was used in a previous fuel cycle, bowing or twisting of the assembly might have given the appearance that the top was aligned even though the bottom was not.

Corrective Action. Assemblies X-04, W-51, X-17, Z-06 and four other potentially contacted assemblies (Z-35, W-11, Z-30, and Z-26) were removed to the spent fuel pool for inspection. No breach of cladding was found on any assembly. Results of additional tests and inspections performed on X-04 will be described in the next annual report. Assembly Z-06 had 13 damaged rods and was returned to the manufacturing plant for repairs. Assembly W-51 had 8 damaged rods and was rejected for further use (it had been used for three cycles). Assemblies X-17 (had 2 scratched rods), W-11 (had 6 scratched rods), Z-35, Z-30, and Z-26 were placed back in the core.

Refueling procedures have been modified, adding new requirements for increased lighting and for TV camera and visual verification of fuel assembly location.

5.2.8 Heat Flux Spike at a BWR

a. Browns Ferry-2 (Refs. 239 and 240)

An event that occurred on Feb. 22, 1984, is briefly described below and will be covered more fully in the fuel performance report for 1984. During startup, the reactor scrammed due to a high flux spike on channels "C" and "F" of the intermediate range monitors (IRMs) of the neutron monitoring system. The IRMs function to prevent fuel damage that can result from abnormal transients that occur while operating in the intermediate power range.

Cause of the Problem. The short period (less than 5 seconds) and subsequent reactor scram were the result of the continuous withdrawal of a high-worth control rod.

Corrective Action. Administrative controls to prevent recurrence of fast periods were inadequate to ensure that high-worth rods were properly identified prior to use. The plant's written procedures and the nuclear engineer qualification program will both be revised to include formal emphasis on the nuclear engineer's duties and proper control rod sequencing with respect to high-worth rods.

5.2.9 Control Rods Inserted in Reverse Order at a BWR

a. Quad Cities-1 (Refs. 241 and 242)

On March 10, 1983, while preparing for an outage, the reactor operator inadvertently inserted the control rods in the reverse order as specified by the approved sequence procedure. Control rods were inserted in this manner until the reactor was manually scrammed as per normal shutdown procedure at approximately nine percent of rated core thermal power. Due to the routine surveillances, which verify rod coupling integrity, and the relatively short amount of time that the reactor was operated under 20 percent of core thermal power, the probability of a design bases reactivity insertion event was minimal. This LER was submitted in accordance with the requirements of Technical Specification 6.6.B.1.f.

Cause of the Problem. The cause of the occurrence is personnel error. It is apparent that an attempt was made by those involved to implement a properly prepared control rod sequence to be used for shutting the unit down. However, a personnel error led to the misapplication of a rod worth minimizer (RWM) computer printout as an approved sequence.

Corrective Action. The reactor was manually scrammed. Further corrective actions will be implemented as specified in a supplemental report, pending further investigations and recommendations from the Commonwealth Edison Company's Quad Cities Nuclear Power Station and Corporate management.

5.2.10 Overflowing or Partial Draining of Water from Refueling Cavity and/or Fuel Storage Pool

Described below are the events at Connecticut Yankee (Haddam Neck) in 1984 and at Peach Bottom-3 in 1983. In 1982, partial draining of water from spent fuel pools occurred at Davis Besse-1 (Refs. 243 and 244) and Trojan (Ref. 245) and overflowing of water from the spent fuel pool (volume estimated to be 23,000 gallons) occurred at Salem-1 (Refs. 246 and 247). Such events have occurred earlier at these plants: Arkansas-2 (LER 81-019), Brunswick-2 (LER 76-153), Davis Besse-1 (LER 80-046), St. Lucie-1 (LER 76-001), Surry-1 (LER 81-015), Trojan (LER 80-009), and Turkey Point-4 (LER 70-014).

a. Connecticut Yankee (Refs. 248-250)

On August 21, 1984, during a refueling outage but prior to the beginning of fuel movement, about 200,000 gallons of slightly radioactive water drained from the refueling pool onto the floor of the containment building (Refs. 248

and 249). None of the water was released to the environment. Though neither occurred, two scenarios were possible in which irradiated fuel could have been exposed to air and workers could have been exposed to high levels of radiation. This event will be described more fully in the next annual report.

Cause of the Problem. The rubber gasket on the refueling cavity seal failed (Ref. 248). The original all-metal seal had been replaced by the licensee with a seal that consisted in part of flexible rubber components (Ref. 249).

Corrective Action. The licensee immediately notified the NRC and suspended refueling (Ref. 248). Design modifications were made by the licensee to prevent recurrence of the seal failure and to ensure that neither of the scenarios could occur.

NRC Action. The NRC concurred (Ref. 248) with the licensee that the safety issues had been resolved and permitted refueling to resume. The NRC issued an Inspection and Enforcement bulletin on August 24, 1984 (Ref. 250). The NRC issued an order and a citation on December 13, 1984 (Ref. 249).

b. Peach Bottom-3 (Ref. 251)

On March 3, 1983, while shut down for refueling, a small quantity of slightly contaminated water was released from the reactor building into the storm drain system. The water overflowed the open reactor cavity and fuel pool and drained to the storm drain system under railroad doors at elevation 135 ft. The dilution of the small amount of radioactive water discharged by additional water in the storm drain system resulted in no detectable release to the river. The event involves Technical Specification 3.8. A previous similar occurrence at this unit was LER 81-013/01T.

Cause of the Problem. The release was caused by inadvertant initiation of the A and B residual heat removal (RHR) pumps.

Corrective Action. The release path was isolated, sampling begun and the small amount of water in the catch basin was pumped to radwaste for processing.

5.2.11 Other Problems

a. Core Unloading/Loading

(1.) Browns Ferry-3 (Refs. 252 and 253)

In preparation to resume unloading of the core on Oct. 10, 1983, the operator observed that more than one control rod could be selected without the use of the one rod permissive bypass, which was in violation of Technical Specification 3.10.A.1.

Cause of the Problem. It was suspected that a reed relay on a relay logic card was stuck.

Corrective Action. The problem cleared and would not repeat. The refueling interlocks functional test was completed and core unloading was resumed. This was a random event and no recurrence control is required.

(2.) Brunswick-1 (Refs. 254 and 255)

While performing routine surveillance (Technical Specification 4.9.10.2.1e) on January 23, 1983, during an extended refueling outage, the control operator noticed that one fuel bundle was located around each of the following withdrawn control rods: 22-15, 34-23, 34-27, and 26-43. This fuel shuffle error involves Technical Specifications 3.9.10.2 and 6.9.1.9b.

Cause of the Problem. A procedural inadequacy allowed the four fuel bundles to be placed in separate fuel cells with withdrawn control rods. The procedure (FH-11) did not define responsibility for assuring that fuel was placed in cells with inserted control rods.

Corrective Action. The control rods were inserted within one hour. The procedure was revised to require all control rods to be inserted prior to fuel movement.

(3.) Cooper (Refs. 256 and 257)

During the process of reloading fuel and performing a control rod shuffle, abnormal difficulty was experienced in setting one of the fuel support pieces in the core. In order to lower a TV camera into the core and observe whether any physical obstruction was preventing installation of the fuel support piece, one adjacent fuel bundle was removed from the core. Subsequent to successfully installing the fuel support piece, the fuel bundle was relocated in the core in the same position that it occupied prior to experiencing difficulty with the adjacent fuel support piece. It was discovered on July 21, 1983, that loading the bundle back into the core at this point conflicted with Technical Specification 3.10.A.2, which requires all control rods to be inserted in the core for loading fuel into the core. Removal and reinstallation of the fuel bundle in the proper location did not change the previously established subcritical reactivity margin.

Cause of the Problem. The licensed senior reactor operators and reactor engineering representative failed to clearly understand Technical Specification 3.10.A.2 in that although the fuel movement sequence could be altered to permit removal of fuel already installed in the core, it was not permissible to reload this fuel back into the core unless all control rods were inserted.

Corrective Action. This occurrence was discussed with the refueling floor supervisor and the reactor engineering representative to prevent recurrence of this event. Procedures have been reviewed by reactor engineers and found to be adequate when implemented correctly. Additional training has been scheduled on Technical Specification requirements for Operations personnel. The Operations Supervisor is to brief the crews. The Reactor Engineering Supervisor is to brief reactor engineering personnel.

(4.) Dresden-2 (Refs. 258 and 259)

During core unloading operations on January 13, 1983, the grapple control rod interlock would intermittently clear while the grapple was over the core. This event was reportable under Technical Specification 6.6.B.2.(b).

Cause of the Problem. The grapple-over-core limit switch was not maintaining full contact with the rail. This is the first reported event for this switch (General Electric Model CR 115 G 307).

Corrective Action. Core unloading was immediately stopped and no control rod movement occurred. The limit switch was adjusted and DOS 800-1 completed to verify operability.

(5.) McGuire-2 (Refs. 260 and 261)

While moving an in-core detector during initial fuel loading, the rope used to hold the detector cable fell into the reactor vessel on March 7, 1983, and was drawn into the "C" hot leg. The rope was found wrapped around the Residual Heat Removal (RHR) impeller hub. This event violates Technical Specification 3.9.8.2 and is reportable under Technical Specification 6.9.1.11(b).

Cause of the Problem. This incident is a result of personnel error in allowing loose material to be in the vicinity of the open vessel.

Corrective Action. RHR pump 2A was secured from service for 3 hours (with pump 2B also secured) to allow rope retrieval efforts. Since the fuel being placed in the core was new, there was no residual heat present and stopping the pumps did not affect the core temperature. Both pumps were available if needed. Fuel loading was suspended and no decrease in boron concentration was made with the pumps secured. Following unsuccessful retrieval efforts, pump 2A was restarted and fuel loading was completed. The rope was removed on March 15, 1983. Personnel will be counseled.

b. Refueling Platform/Bridge Interlocks

(1.) Browns Ferry-1 (Refs. 262 and 263)

This event on Nov. 12, 1983, involved the failure of the refueling platform position interlock, which prevents travel over the core with one rod withdrawn and the fuel grapple extended. Technical Specification 3.10.A.1 is applicable.

(2.) Peach Bottom-3 (Refs. 264 and 265)

The refueling bridge interlock was inoperable on Mar. 12, 1983, during movement of four fuel bundles from the reactor to the spent fuel pool. Technical Specification 3.10.A.1 is applicable.

c. High Chloride Concentration in Reactor Coolant System

Five plants (one BWR, four PWRs) reported high chloride concentrations in their reactor coolant systems in 1983: Browns Ferry-1 (Refs. 266 and 267), Davis Besse-1 (Refs. 268 and 269), Farley-2 (Refs. 270 and 271), Ginna (Refs. 272 and 273), and Indian Point-2 (Refs. 274 and 275). The 1982 event in Cook-1 (Ref. 276) is included in a report (Ref. 277) published by the NRC in 1983. The 1982 event in La Crosse (Ref. 278) is included in a report (Ref. 279) published by the NRC in 1983.

d. Reactor Coolant System Has High Conductivity

Two BWRs submitted LERs in 1983: Quad Cities-2 (Refs. 280 and 281) reported that the condensate conductivity exceeded the limit, resulting in high reactor coolant system conductivity, and Susquehanna-1 (Refs. 282 and 283) reported that on Jan. 26, 1983, excessive reactor water conductivity and pH occurred twice (cause: organic substance entered condenser). Susquehanna-1 (Refs. 284-286) reported that high reactor coolant system conductivity was noted on Jan. 8, 1983 (cause: reactor water cleanup system was inadvertently isolated). The 1982 events at Brunswick-2 (Ref. 287), La Crosse (Ref. 278), La Salle-1, and Pilgrim (Ref. 158) are included in reports (Refs. 288, 279, 289, and 290, respectively) published by the NRC in 1983.

e. Reactor Coolant System Has High Oxygen Concentration

Two PWRs submitted LERs in 1983: Calvert Cliffs-2 (Refs. 291 and 292) and Crystal River-3 (Refs. 293 and 294).

f. Heavy Load Passes Over Irradiated Fuel in Spent Fuel Pool

(1.) Millstone-2 (Refs. 295 and 296)

While at 0% power and during routine refueling operations, the spent fuel gate, a load of approximately 4500 pounds, traveled over irradiated fuel assemblies in the spent fuel pool on Nov. 29, 1983. The gate was being moved from the transfer canal to the cask laydown pit when the discovery was made. The load was then returned over irradiated fuel to a safe position by the transfer canal. The event had no impact on refueling operations. The LER was submitted in accordance with Technical Specification 6.9.1.9.b.

Cause of the Problem. Due to physical limitations on the travel of the crane, the load could not be positioned over the designated safe load path.

Corrective Action. When it was discovered that the load was over irradiated fuel assemblies, it was moved to a safe position. To prevent recurrence, special rigging procedures have been incorporated into the crane travel procedure to allow the gate to travel over the designated safe load paths. No further action is required.

g. Spent Fuel Pool Racks

(1.) Browns Ferry-1 (Refs. 297 and 298)

During fuel unloading on April 24, 1983, at the beginning of a refueling outage, 130 fuel bundles were unloaded from the core into a high-density spent fuel storage rack (HDSFR) that had not had the required postinstallation test for the presence of boron. Technical Specification 5.5.B requires the effectiveness of the spent fuel storage pool to be less than 0.95.

Cause of the Problem. Personnel failed to verify acceptability of the HDSFRs prior to preparation of fuel assembly transfer forms. Additionally, plant procedures were inadequate in that neither they nor the refueling floor tagboard identified the untested HDSFRs.

Corrective Action. Fuel handling operations were halted and the problem investigated. Plant procedures will be modified to provide tracking of HDSFR testing.

(2.) San Onofre-1 (Refs. 299-301)

While inspecting the spent fuel pit (SFP) for debris on May 5, 1983, two upper lateral braces of the SFP storage racks were found bent and not in contact with the east SFP wall. This follow-up report was submitted in accordance with Technical Specification 6.9.2.a. A formal structural evaluation has shown that the storage racks met the original design criteria in the "as-loaded" condition but not in the "full-loaded" condition.

Cause of the Problem. The lateral braces were apparently bent during previous pit evolutions by items suspended from the SFP bridge crane.

Corrective Actions. Fuel assemblies adjacent to bent braces and the SFP liner will be visually inspected for damage. No additional fuel assemblies will be stored in the racks or will any fuel assemblies already stored in the racks be moved pending further evaluation or corrective action. Any long-term corrective action to restore the SFP storage racks to their original structural integrity will be held in abeyance pending resolution with the NRC regarding the "Plan for Managing Plant Retrofit" (Ref. 301).

(3.) Rancho Seco (Refs. 302 and 303)

An event that occurred on Feb. 29, 1984, at the spent fuel pool is briefly described below and will be covered more fully in the fuel performance annual report for 1984. While moving a spent fuel rack as part of the spent fuel reracking project, a rack became bound against an adjacent rack. As a result, one of the two slings being used to lift the rack failed. The second sling remained intact and the rack was safely lowered without dropping.

Cause of the Problem. The rack had been improperly rigged and the load cell for the lifting crane was set too high.

Corrective Action. All work was stopped until the investigation of the event was completed. Governing procedures have been revised to clarify proper rigging and load cell usage and to require sign-offs by the cognizant SMUD engineers and quality control inspectors at appropriate steps in the procedures.

h. Design/Calculational Errors

Eight plants (two BWRs, six PWRs) submitted LERS during 1983. Browns Ferry-3 (Refs. 304 and 305) provided an update on a linear heat generation rate calculational error. Cook-1 (Refs. 306 and 307) reported that an in-core detector computer code had wrong values. La Salle-1 (Refs. 308 and 309) indicated that incorrect nodal exposure data were entered into a process computer. Maine Yankee (Refs. 310 and 311) provided an update on a computer program error, which affected coefficients relating maximum one pin power to assembly power. Oconee-3 (Refs. 312 and 313) reported that axial power shaping rod (APSR) limits were found to be not conservative (design error involved) and that EPRI was notified of the possible generic problem. Prairie Island-1 (Ref. 314) reports that reanalysis indicates possible high cladding temperature following a LOCA (an error in the input to one of the codes used in the ECCS analysis was discovered). Surry-1 (Refs. 315 and 316) indicates that it was notified by Westinghouse that the rod drop accident analysis, as presented in the FSAR, may not represent the limiting case. Zion-1 (Refs. 317 and 318) reported that an error in a depletion calculation for the core was found, which could affect core peaking factors during load-follow operations and that an evaluation revealed that the original analysis was not still bounding.

i. Hot Channel Factors

Three PWRs submitted LERs in 1983. McGuire-1 (Refs. 319 and 320) and McGuire-2 (Refs. 320 and 321) were notified by Westinghouse about a potential violation of hot channel factor limits. Salem-1 (Refs. 322-325) provided an update on a potential unreviewed safety question involving the hot channel factor (licensee no longer has a safety question).

j. Erratic Behavior of Power Range Instrument

One PWR reported such an event in 1983. Summer-1 (Refs. 326 and 327) experienced erratic behavior from a power range instrument during three events. Cause of erratic behavior is unknown. The licensee is still evaluating the cause of the problem.

k. Fuel Transfer Tubes

On April 4, 1984, it was determined (Ref. 328) that adequate local leak rate testing of the Farley-1 and -2 fuel transfer tubes had not been performed since May 29, 1979, and Sept. 8, 1980, respectively.

1. Containment Integrity Violated

One PWR submitted an LER in 1983. St. Lucie-1 (Refs. 329 and 330) reported that while in Mode 6 refueling during core alterations, containment integrity was violated twice (inner and outer personnel hatch doors left open). The events in 1982 at Oconee-3 (Ref. 331) and Salem-2 (Ref. 332) were included in reports (Refs. 333 and 334, respectively) published by the NRC in 1983.

m. Shield Plugs Not Removed

Pilgrim-1 (Refs. 335 and 336) reported that during refueling operations the pool shield plugs separating the dryer/separator pool from the refuel cavity were not removed in accordance with Technical Specification 3.5.F.5. Fuel movement was immediately stopped and the plugs removed.

n. Unknown Cause for Control Rods That Dropped, Slipped or Overtraveled

Four PWRs submitted LERS in 1983 and in each event the cause was unknown at that time. Millstone-2 reported that Control Element Assembly (CEA)-62 (Refs. 337 and 338) and CEA-16 (Refs. 339 and 340) dropped into the core on April 21, 1983, and Feb. 18, 1983, respectively. A similar case (Refs. 341 and 342) occurred at Millstone-2 in 1982. Quad Cities-2 (Refs. 343 and 344) reported that a control rod overtraveled on Sept. 1, 1983. St. Lucie-1 reported that two control rods dropped in 1983: these are the eighteenth (Refs. 345 and 346) and nineteenth (Refs. 347 and 348) occurrences of dropped rods at this plant due to an unknown cause. Earlier occurrences are described in other LERs (e.g., Refs. 349-351). St. Lucie-2 reported that four control rods dropped or slipped in 1983 (Refs. 352-359).

o. Unknown Cause for Inoperable Control Rods

(1.) Calvert Cliffs-2 (Refs. 360 and 361)

An update on an event involving an inoperable control rod assembly was submitted on Jan. 27, 1983. Following a reactor trip on Feb. 12, 1982, it was discovered that Control Element Assembly (CEA)-19 had stuck at the approximately 8-inch withdrawn position. Attempts to move the CEA failed and it was declared inoperable (involves Technical Specifications 3.1.3.1 and 3.1.3.4).

Cause of the Problem. There was no evidence of a causative agent for the event or of any sustained damage.

Corrective Action. CEA-19 was freed and tested satisfactorily on Feb. 13, 1982. All other CEAs remained operable and adequate shutdown margin was maintained throughout this event. No similar events have been reported. During refueling, the CEA (and the CEA extension shaft) and its associated fuel assembly (2F123) were visually inspected with a TV camera. Eddy current examinations were performed on the fuel assembly guide tubes and the CEA and there were no anomalous indications. The center CEA finger was profiled (by eddy current testing) and the results showed no indications that would explain the CEA becoming lodged.

(2.) Farley-1 (Refs. 362 and 363)

The rod control system was declared inoperable on July 18, 1983. No cause for the event could be determined.

p. Control Rod Drive Coupling Potentially Overloaded

One BWR submitted an LER in 1983. Susquehanna-1 (Refs. 364 and 365) provided an update on clamps on control rod drive fittings that were potentially overloaded.

q. Control Rod Uncoupled or Coupling Failure

Four BWRs submitted LERs in 1983. Brunswick-1 (Refs. 366 and 367) provided an update on a control rod drive coupling failure. Fitzpatrick (Refs. 368 and 369) reported finding a control rod uncoupled (bent uncoupling rod prevented coupling). La Salle-1 (Refs. 370 and 371) reported that a control rod uncoupled during startup due to spring damage and crud buildup. Pilgrim (Refs. 372 and 373) reported that an indication was received that a control rod was uncoupled (no root cause for the event could be determined at that time).

r. Rod Control Assemblies Not On Surveillance Sheet

Salem-2 (Refs. 374 and 375) reported that it was discovered on April 1, 1984, during routine surveillance of control rod assemblies that four rods were missing from the surveillance check-off sheet. The event was caused by a typographical error. Since July 1983, surveillance was not performed on these four rods on six different occasions.

5.3 OLD PROBLEMS (1982) THAT DID NOT RECUR OR THAT WERE SOLVED

5.3.1 BWR Fuel Rod Breakage During Visual Inspection

No such events were reported by domestic BWRs in 1983.

The event at Hatch-1 in 1982 (Refs. 376 and 377) was described in the fuel performance annual report for 1982 (Ref. 5). Ref. 376 was published by the NRC in April 1983.

5.3.2 Hydriding of PWR Fuel Rods

There were no such events reported by domestic PWRs in 1983.

The event at Indian Point-2 in 1982 (Refs. 378-381) was described in the fuel performance annual report for 1982 (Ref. 5). Refs. 378 and 380 were published by the NRC in 1983. An updated report (Ref. 379) on the five fuel assemblies that experienced cladding degradation (hydriding) was submitted by the licensee to the NRC in 1983.

6.0 SUMMARY OF HIGH-BURNUP FUEL EXPERIENCE

Domestic BWR fuel burnup experience is summarized in Figure 5. Domestic PWR fuel experience is summarized in Figure 6. Most of the data on fuel discharged prior to 1979 in the two figures was obtained from Ref. 382. As indicated in Section 3.0, typical fuel reload orders today for a BWR and a PWR have warranties for average batch burnups of about 31,000 MWd/MTU and 36,000 MWd/MTU, respectively (Ref. 24).

Present optimum extended burnups (discharge batch averages) are typically about 50,000 MWd/MTU and 45,000 MWd/MTU for PWRs and BWRs on 12-month cycles, respectively, and about 10% higher for 18-month cycles (Ref. 383). EPRI anticipates that fuel assemblies designed for improved high burnup performance will not be available for full reloads until the late 1980s and that there will be a gradual increase in burnup of 1000-2000 MWd/MTU per year (Ref. 92).

In testimony given at hearings conducted on March 16 and 17, 1982 by the subcommittee on Energy Research and Production of the House Committee on Science and Technology, DOE (Ref. 384) indicated that fuel from each of the five U.S. fuel vendors, equivalent to 22 fuel assemblies, has been irradiated in seven commercial LWRs and that selected assemblies have achieved burnups up to 47,000 MWd/MTU. As noted in Section 4.0, BWR and PWR fuel assembly burnups of >45,000 MWd/MTU and 55,400 MWd/MTU, respectively, have been attained. Projected benefits to the LWR fuel cycle from extended burnup are discussed by DOE in a recent paper (Ref. 385) and in a recent article (Ref. 386). A final report on a DOE-sponsored study on extended burnup in PWRs has been issued (Ref. 91).

The regulatory perspective on extended burnup fuel is discussed in Ref. 387. The NRC is currently reviewing vendor topical reports that address extended burnup experience, methods and test data.

FIGURE 5. Domestic BWR Fuel Burnup Experience

28/6L

FIGURE 6. Domestic PWR Fuel Burnup Experience

7.0 ITEMS OF GENERAL SIGNIFICANCE

In notices published on May 6, 1982 and July 26 and August 29, 1983 in the Federal Register (Refs. 8, 9, and 11), the NRC indicated that it was amending its regulations to improve the information received in Licensee Event Reports from nuclear power plant licensees (Refs. 7 and 10). Paragraph 50.73(a)(2)(ii) of the new LER rule (which became effective January 1, 1984) requires events to be reported where the plant, including its principal safety barriers (i.e., fuel cladding, reactor coolant system boundary, or the containment), was seriously degraded or in an unanalyzed condition. Examples of situations addressed by this paragraph are "fuel cladding failures in the reactor or the storage pool that exceed expected values, that are unique or widespread, or that resulted from unexpected factors."

The NRC evaluated the Exxon Nuclear 17 x 17 PWR fuel design and concluded on December 27, 1982, that the associated Exxon Nuclear topical report (as revised in responses to NRC staff questions and subject to plant-specific provisions) will serve as an acceptable generic reference for that fuel design (Ref. 388).

In July 1983 and in connection with SRP-4.2 Appendix A requirements, the NRC transmitted an evaluation report, "Fuel Grid Testing Review," to all four U.S. PWR fuel vendors and concluded that the fuel grid testing methods of all four are appropriate and acceptable (Refs. 389-392).

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