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Fuel Performance Annual Report for 1989

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Fuel Performance Annual Report for 1989

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ABSTRACT

This annual report, the twelfth in a series, provides a brief description of fuel performance during 1989 in commercial nuclear power plants and an indication of trends. 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 more detailed information and related U.S. Nuclear Regulatory Commission evaluations are included.

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1.0 EXECUTIVE SUMMARY

Interest in fuel performance during 1989 focussed primarily on extending burnup, with a goal of determining optimum fuel rod utilization without introducing increased leakage or other problems; a concomitant goal remained the determination and elimination of the causes of fuel rod failure.

In the sections that follow, the burnup levels attained in 1989 are discussed; the 1989 reliability of fuel rods along with the primary causes of fuel rod failure and the corrective actions being taken, are presented; an overview is provided of the major non-fuel core-related problems encountered during the year.

1.1 EXTENDING BURNUP

On the basis of the letter reports to the Nuclear Regulatory Commission from the nuclear fuel vendors[1-5] for calendar year 1989, an overview of the highest currently achieved burnups is given below. The data are from both fuel assemblies remaining in-core and those discharged during 1989.

Vendor	Plant or Test	Type	Burnup (GWd/MTU)	Comment
ANF	Tihange-1, Belgium	PWR	50.0	highest to date
	Big Rock Point	BWR	41.0	highest to date
	D.C. Cook, 17x17	PWR	44.0	discharged 1989
	Gundremmingen-3, FRG 9x9	BWR	40.0	discharged 1989
BWFC	Mark GdB, LTA	PWR	58.3	UO_2-Gd_2O_3
	Mark BZ, LTA, 15x15	PWR	58.3	Zirc -4 grids
C-E	ANO-2	PWR	43.0	discharged 1989
	St. Lucie-2	PWR	42.0	discharged 1989
	702 rods discharged		56-59.9	highest to date
GE		BWR	>45	bundle average
		BWR	60	peak pellet exp.
W	Zion-1 & -2	PWR	55	4 assemblies ave., 5 cycles
	North Anna-1	PWR	58.4	lead assembly ave., 4 18-mo cycles
	North Anna-1	PWR	>60.0	lead fuel rod ave.

Burnup goals are being extended to obtain basic information on fuel rod behavior as fuel rod lifetimes increase and to determine the feasibility of longer-term use of fuel, with the goals of minimizing spent fuel waste and ensuring safe extension of reactor cycles to a routine period of 24 months. About 90% of BWRs and over 80% of PWRs in the United States operate on nominal 18-month or 24-month refueling cycles; the remainder operate on 12-month cycles.[6]

Among the effects of longer cycles are an increase in cladding corrosion, fuel assembly bowing, fuel rod and assembly growth, and a possible degradation and increase in defect size because the failed fuel rod remains in the core for a longer time before removal.

Westinghouse[5] has found that neither extended burnup (assembly average of 55 GWd/MTU and beyond) nor extended residence times (seven cycles, to a discharge burnup of 40 GWd/MTU) has led to increased coolant activity, nor, therefore, to increased fuel failure/leakage.

Similarly, Advanced Nuclear Fuels[1] detected no fuel failures due to inherent manufacturing or design factors following irradiation to 50 GWd/MTU during 1989.

1.2 FUEL RELIABILITY

Along with the good fuel performance to extended burnups, the industry as a whole has experienced a steady decrease in reactor coolant Iodine-131 activity over the last several years. Average coolant activities quoted for calendar year 1989 for PWRs[1-3,5], normalized to standard coolant purification rate and corrected for tramp uranium, range from <0.001 μCi/g to 0.023 μCi/g. The industry median[1] coolant activity for BWR's in 1989 is 83 μCi/sec.

The decreasing Iodine-131 coolant activities indicate an increase in fuel reliability. Representative fuel reliability levels for 1989 were 99.997% or better throughout the industry for Zircaloy-4 clad fuel in PWRs, when fuel failure due to debris-caused fretting is not included;[1-3,5] approximately 99.986% for stainless-steel clad fuel, due to a vulnerability to debris-induced fretting in type 304 SS cladding;[2] and 99.98%[4] for the GE 8x8 BWR fuel.

Primary fuel failure causes during 1989 were debris-induced fretting for PWR fuel and pellet-cladding interaction (PCI) and crud-induced localized corrosion (CILC) for BWR fuel. Summaries of these failures during 1989, plus comments on other core components, are provided below.

1.2.1 Primary Cause of Fuel Rod Failure in PWRs - Debris Fretting

The major cause of the few fuel failures that do occur in PWRs is debris-induced fretting. Westinghouse[5] estimates that debris fretting represents the primary leakage mechanism for approximately 80% of the identified fuel rod leakers. Similarly, Combustion Engineering[3] estimated, on the basis of fuel examinations, that about 75% of the leaking fuel that occurred during 1989 was caused by debris fretting of the Zircaloy-4 cladding. This process occurs when bits of metallic debris in the primary coolant, which have fallen into the reactor primary coolant during maintenance operations or have broken loose from the reactor components, are swept through the system, getting caught at the orifices at the bottom of the fuel assembly spacer grids or other restricted areas. Vibrations induced by coolant flow cause the debris to rub against the fuel cladding until a breach develops. Following increased Iodine-131 activity in the coolant during reactor operation, the fuel failure is generally confirmed during reactor shutdown periods by ultrasonic testing (UT) and visual observation.

Utilities have taken aggressive action to halt debris-fretting. First, a major effort has been extended to prevent further introduction of debris into the system, with apparently good success. In addition, a number of design changes are being tested to minimize the effect of fretting from the already existing debris: ANF(7) and BWFC(2) have extended-length end fittings, most of the length of which is of solid stock. BWFC also has lowered the spacer grid to take advantage of the solid portion of the end cap. Combustion Engineering and Westinghouse have spacer grids (the GUARDIAN™ (8) and the Debris Filter Bottom Nozzle (DFBN)(5) respectively) with smaller holes to screen out more of the particles before they can reach regions of exposed fuel rods.

1.2.2 Primary Causes of Fuel Rod Failure in BWRs - PCI and CILC

General Electric has found pellet-cladding interaction (PCI) and crud-induced localized corrosion (CILC) to be the only two causes of cladding perforation in BWRs in recent periods. Although the effects of PCI can be alleviated by slow ascent to full reactor power, efficiency is lost by this tactic. GE is finding in lead use assembly (LUA) trials that Zr-lined (barrier-coated) cladding is effective in resisting PCI.

Crud-induced localized corrosion is now known to occur under certain conditions in the presence of Cu in the coolant system. The solution is to monitor the water chemistry carefully and to eliminate the sources of copper, originally from the condenser tubes and filter demineralizer condensate cleanup systems. Also, manufacturing processes have been developed which produce Zircaloy alloys that are more resistant to corrosion.

1.3 NON-FUEL CORE-RELATED PROBLEMS

In addition to the major problems encountered for fuel performance in reactor, as discussed above, there were several types of recurring problems that warrant mention, problems with other core components, with fuel and core component handling, and with procedures.

- Non-Fuel Core Components - There were five events in 1989 involving thinning of in-core instrumentation tubes. (See Section 5.1.3.)

- Fuel Handling - Fourteen fuel handling incidents occurred in 1989, including: four in which fuel assemblies were dropped or came loose and were out of line with their destination (Section 5.1.7 and 5.2.1); two in which assemblies were placed in an incorrect position (Section 5.1.7, 5.1.8); and one in which a fuel assembly was bent (through failure to follow procedure) (Sections 5.1.8 and 5.2.4.). Seven additional fuel handling events involved procedural violations, personnel error and administrative control deficiency (Sections 5.1.9, 5.1.10, 5.2.1, 5.2.5).

- Control Rod Malfunction, Failure and Maintenance/Installation Problems - Seventeen events (eight in foreign countries) involving the control rods or their controlling mechanism occurred during 1989, including rod failure, rod wear and cracking, and support pin problems (Sections 5.1.15 - 5.1.19, 5.2.6, 5.2.7).

- Personnel Error - Sixteen events in 1989 were due to personnel errors (two in foreign countries), one to personnel fatigue and one to miscommunication (Sections 5.1.36, 5.1.37, 5.2.22, 5.2.23); these are in addition to the fuel handling problems attributed to personnel errors.

- Failure to Follow Procedures, Defective Procedures - There were eleven events in 1989 in which there was non-compliance with procedures (Sections 5.1.38 & 5.2.18) and 18 events in which the procedures had failed to include adequate information to prevent the problems that occurred, training was inadequate, or there were management deficiencies (Sections 5.1.32 and 5.2.19).

From the numbers above, it is clear that, of non-fuel core-related problems, control rods require significant attention. By far the most frequent causes of problems, however, are people-related: failure to follow procedures, procedures with insufficient information and personnel errors (such as pushing the wrong button).

2.0 INTRODUCTION

This report is the twelfth[9-18] in a series which provides a compilation of the available information on nuclear reactor fuel performance, particularly new developments on the one hand and non-catastrophic off-normal behavior and problems on the other. A discussion of the evolution of the content of the current reports can be found in the "Fuel Performance Annual Report for 1988".[19]

The NRC regulation 10 CFR Part 50, Paragraph 50.73(a)(26)(ii)[20] requires reports on events in which the plant, including its principal safety barriers, was seriously degraded or was in an unanalyzed condition. Reporting on normal operation surveillance results, generic problems, and design trends is not required by NRC 10 CFR nor by the NUREG series entitled "Nuclear Power Plant Operating Experience"[21-27] and the Electric Power Research Institute (EPRI) reports.[28-30]

Thus the primary intent of this Annual Report series is to summarize fuel design changes and progress of the concomitant testing programs, progress toward high burnup goals and the problems that arise (whether due to conditions within the reactor or to operations and operators), fuel system problems (especially generic ones) that are of concern during the reporting period, and trends of general significance. References are provided for additional and background information. The main focus of the Annual Report for 1989 is on fuel operating performance during calendar year 1989, but there is some overlap with 1988 for continuity and with 1990 where the information has been received and is pertinent.

The sections in this Annual Report for 1989 are as follows:

1.0 Executive Summary

2.0 Introduction

3.0 Fuel Design Changes and Summary of Surveillance Programs

4.0 Fuel Operating Experience

5.0 Problem Areas Observed During 1989 (problems with control rods and other non-fuel components are in Appendix B)

6.0 Trends

7.0 Summary of High Burnup Fuel Experience

8.0 References

Appendix A--Historical Background on Fuel Reliability

Appendix B--Problem Areas Observed During 1989 with Non-Fuel Components.

As a basis for the design changes discussed in Section 3.0, typical fuel assembly[a] parameters and operating conditions for current light water reactor (LWR) fuel rod designs for use in pressurized water reactors (PWRs) and boiling water reactors (BWRs) are summarized in Table 1. Included in Table 1 and in the sections that follow is information on fuel from these five vendors:

1. Advanced Nuclear Fuels Corporation (ANF),[b] Richland, Washington
2. Babcock and Wilcox Fuel Company(BWFC),[c] Lynchburg, Virginia
3. Combustion Engineering, Inc. (C-E),[d] Windsor, Connecticut
4. General Electric Company (GE), San Jose, California
5. Westinghouse Electric Corporation (W), Pittsburgh, Pennsylvania.

(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) Previously known as Exxon Nuclear Company, Inc., (ENC); ANF is a Siemens Company.
(c) Previously known as Babcock & Wilcox Company (B&W); the B&W Fuel Company (BWFC) is a partnership between B&W and the American subsidiary of a French consortium of Cogema, Framatome, and Uranium Pechiney.
(d) Combustion Engineering, Inc. is now affiliated with Asea Brown Boveri (ABB).

TABLE 1. Typical Fuel Assembly Parameters[1]

Vendor	BWFC (B&W[14] Reactor System)		BWFC (W[14] Reactor System)			C-E	C-E	W	W	W	ANF	ANF	ANF	ANF	GE	GE	GE
Fuel Rod Array	15x15	17x17	15x15	15x15	17x17	14x14	16x16	14x14	15x15	17x17	15x15	17x17	8x8	9x9	7x7	8x8	8x8
Reactor Type	PWR	PWR	PWR	PWR	PWR	PWR	PWR	PWR	PWR	PWR	PWR	PWR	BWR	BWR	BWR	BWR	BWR
Assemblies per Core	177	205	157	157	193 (157)	217	217	121	193	193	193	193	560	724	764	560	560
Fuel Rods Per Assembly	208	264	204	204	264	176	236[a]	179	204	264	204	264	60	80	49	63	62
Empty Locations Per Assembly	17	25	21	21	25	20	20	17	21	25	21	25	4	1	None	1	2
Rod Pitch, mm (in.)	14.4 (0.568)	12.8 (0.502)	14.3 (0.563)	14.3 (0.563)	12.6 (0.496)	14.7 (0.580)	12.9 (0.5063)	14.1 (0.556)	14.3 (0.563)	12.6 (0.496)	14.3 (0.563)	12.6 (0.496)	16.3 (0.842)	14.5 (0.572)	18.7 (0.738)	16.3 (0.640)	16.3 (0.640)
System Pressure, MPa (psia)	15.2 (2200)	15.5 (2250)	13.9 (2015)	13.9 (2015)	15.5 (2250)	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	91.4	107.3	82.25	82.25	82.25	78.5	96.4	95.6	98.1	104.7	98.1	104.7	40.57	46	50.732	50.51	49.15
Average LHGR,[b] kW/m (kW/ft)	20.3 (6.20)	18.8 (5.73)	18.1 (5.53)	18.4 (5.60)	17.8 (5.43)	20.0 (6.09)	18.2 (5.54)	20.3 (6.20)	22.0 (6.70)	17.8 (5.44)	22.0 (6.70)	17.8 (5.44)	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)	24.4 (7.44)	22.6 (6.88)	25.1 (7.66)	25.5 (7.76)	27.6 (8.42)	24.00 (7.31)	21.00 (6.41)	24.36 (7.44)	26.40 (8.04)	21.36 (6.53)	26.40 (8.04)	21.4 (6.53)	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)	53.0 (16.16)	49.9 (15.20)	47.6 (14.5)	47.6 (14.5)	42.7 (13.0)	53.5 (16.3)	42.7 (13.0)	56.8 (17.3)	61.7 (18.8)	44.6 (13.6)	51.9 (15.83)	54.5 (16.6)	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)	2340 (4244)	2290 (4155)	2149 (3900)	2149 (3900)	1927 (3500)	2140 (3890)	1880 (3420)	2260 (4100)	2340 (4250)	1870 (3400)	2200 (3997)	1747 (3177)	2040 (3700)	2040 (3705)	2440 (4430)	1830 (3325)	1890 (3435)
Core Average Enrichment wt% ^{235}U	3.30[c]	3.15[c]	4.00[c]	3.41[c]	3.40[c]	3.89[c]	2.36	2.90	2.80	2.60	3.02	3.65	2.65	2.8	2.19	1.80	1.99
Max. Local Exposure MWd/MTU[d]	55,000	55,000	55,000	55,000	55,000	50,000	55,000	50,000	50,000	50,000	47,500	52,000	35,000	55,000	40,000	40,000	45,000
Cladding Material[e]	Zry-4	Zry-4	304SS	Zry-4	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	BWFC (B&W[14] Reactor System)		BWFC (W[14] Reactor System)			C-E	C-E	W	W	W	ANF	ANF	ANF	ANF	GE	GE	GE
Fuel Rod Length, m (in.)	3.904 (153.7)	3.878 (152.7)	3.218 (126.7)	3.197 (125.9)	3.848 (151.5)	3.71 (145.9)	4.09 (161.0)	3.87 (152.4)	3.80 (149.7)	3.85[f] (151.6)	3 86 (152.0)	3.86 (152.0)	3.99 (156.9)	3 99 (157.2)	4 09 (161.1)	4 09 (161.1)	4 20 (165 4)
Active Fuel Height, m(in.)	3.602 (141.8)	3.632 (143.0)	3.061 (120.5)	3.012 (118.6)	3.658 (144.0)	3.47 (136.7)	3.81 (150)	3.66 (144)	3.66 (144)	3.65[g] (143.7)	3.66 (144)	3.66 (144.00)	3.66 (144)	3.69 (145.24)	3.66 (144)	3.71 (146)	3.81 (150)
Plenum Length, m (in.)	0.298 (11.7)	0.242 (9.5)	0.122 (4.8)	0.159 (6.3)	0.164 (6.4)	0.22 (8.6)	0.25 (10.00)	0.18 (6.99)	0.21 (8.2)	0.16 (6.3)	0.17 (6.8)	0.18 (7.20)	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.)	10.92 (0.430)	9.63 (0.379)	10.72 (0.422)	10.72 (0.422)	9.50 (0.374)	11.18 (0.440)	9.70 (0.382)	10.72 (0.422)	10.72 (0.422)	9.50 (0.374)	10.77 (0.424)	9.14 (0.360)	12.74 (0.5015)	10.77 (0.424)	14.30 (0.563)	12.52 (0.493)	12.27 (0.483)
Cladding ID, mm (in.)	9.58 (0.377)	8.41 (0.331)	9.88 (0.389)	9.35 (0.368)	8.28 (0.326)	9.75 (0.384)	8.43 (0.332)	9.48 (0.3734)	9.48 (0.3734)	8.36 (0.329)	9.25 (0.364)	7.87 (0.310)	10.91 (0.4295)	9.25 (0.364)	12.68 (0.499)	10.80 (0.425)	10.64 (0.419)
Cladding Thickness, mm (in.)	0.673 (0.0265)	0.610 (0.024)	0.419 (0.0165)	0.686 (0.027)	0.610 (0.024)	0.711 (0.028)	0.635 (0.025)	0.617 (0.0243)	0.617 (0.0243)	0.572 (0.0225)	0.762 (0.030)	0.64 (0.025)	0.914 (0.036)	0.762 (0.030)	0.813 (0.032)	0.864 (0.034)	0.813 (0.032)
Diametral Gap[h], micron (mil)	213.4 (8.4)	198.1 (7.8)	165 (6.5)	178 (7.0)	165 (6.5)	190.5 (7.5)	178 (7.0)	190 (7.5)	190 (7.5	165 (6.5)	190 (7.5)	177.8 (7.0)	254 (10.0)	190 (7.5)	305 (12.0)	229 (9.0)	229 (9.0)
Fuel Pellet Diameter, mm (in.)	9.362 (0.3686)	8.209 (0.3232)	9.715 (0.3825)	9.17 (0.361)	8.115 (0.3195)	9.56 (0.3765)	8.26 (0.325)	9.29 (0.3659)	9.29 (0.3659)	8.19 (0.3225)	9.06 (0.3565)	7.70 (0.3030)	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.)	11.05 (0.435)	9.53 (0.375)	11.63 (0.458)	10.80 (0.425)	10.16 (0.400)	11.43 (0.450)	9.91 (0.390)	15.24 (0.600)	15.24 (0.600)	13.46 (0.530)	6.93 (0.273)	8.84 (0.348)	8.13 (0.320)	10.41 (0.410)	12.70 (0.500)	10.67 (0.420)	10.41 (0.410)
Fuel Pellet Density, %TD[i]	95	95	95	95	96/95[j]	95	95	94	95	95	94	94.0	95	94.5	95	95	95

(a) Unshimmed assemblies.
(b) LHGR = linear heat generation rate.
(c) Reload batch average enrichment.
(d) 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).
(e) Type 304 stainless steel (304SS), Zircaloy-4 (Zry-4), and Zircaloy-2 (Zry-2).
(f) The fuel rods in the Westinghouse PWR 17x17 fuel assemblies in South Texas -1 and -2 are 4.49 m (176.7 inches) long.
(g) The 17x17 fuel assemblies in South Texas -1 and -2 have an active height of 4.27 m (168 inches).
(h) Diametral gap = cladding ID - pellet diameter.
(i) Theoretical density (TD) of stoichiometric UO_2 is 10.96 g/cm^3.
(j) Design may use either density.

3.0 FUEL DESIGN CHANGES AND SUMMARY OF FUEL SURVEILLANCE PROGRAMS

Fuel System Design, Section 4.2 of the Standard Review Plan[31] requires that plans for testing, inspection, and surveillance of fuel be submitted and reviewed for each domestic nuclear power plant. The plans should include pre-irradiation 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 a 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). A supplementary fuel surveillance program appropriate for new fuel designs is noted in Reference 32.

Provided below is a summary of current design changes and fuel surveillance programs for each of the five fuel vendors, plus a summary of the surveillance programs being conducted by EPRI. Each section will address designs introduced in 1989, if any, improvements made in the past two or three years, and the surveillance programs under way to test these designs.

The information presented in these subsections is taken from the vendors' responses to the annual request from the Nuclear Regulatory Commission (NRC) for individual vendor input on fuel experience, design developments, etc. The responses vary in length from 1/2 page of text with 3 or 4 tables and a couple of figures to formal documents consisting of 15 pages of text and 10 pages of tables and figures, so that there is no uniformity in content or format. Because of this discrepancy in the provided information, no attempt has been made to present this information in a more uniform manner.

3.1 ADVANCED NUCLEAR FUELS CORPORATION (ANF) - (PWRs and BWRs)

The information which follows is taken from the "ANF Annual Fuel Performance Report."[1] Additional data and discussion are available in the 1991 Proceedings of the International Topical Meeting on Fuel Performance.[7]

3.1.1 Design Changes

No new design changes were specifically noted by ANF for 1989. Ongoing design evolution is discussed in a 1991 paper.[7] Some aspects of this evolution are: the introduction of the 9x9 array with several configurations of water rods, variable axial concentrations of gadolinia, beta-quenched cladding, use of fuel rod clips to prevent fuel failures due to baffle jetting, and the introduction of high thermal performance spacers and intermediate flow mixers.

The introduction of BWR 9x9 and PWR 17x17 fuel rod arrays has generally led to the reduction of rod linear heat generation rates. Additional benefits that result are lower fuel temperatures, less fission gas release, decreased pellet-clad interaction and lower clad stresses. In addition, the smaller diameter rods in BWR 9x9 arrays appear to provide greater resistance to failure, due to decreased pellet-cladding interaction (PCI).

3.1.2 Surveillance and Performance Programs

The status of ANF surveillance programs over several years in specific reactors is summarized, together with the surveillance information for other vendors, in Table 2. Additional general information is given in Reference 7.

3.2 B&W FUEL COMPANY (BWFC) [formerly Babcock & Wilcox Company (B&W)] - (PWRs)

The basis for this section is the "B&W Fuel Company 1989 Fuel Performance Report."[2] Additional, updated information is available in the 1991 Proceedings of the International Topical Meeting on LWR Fuel Performance.[48]

3.2.1 Design Changes

BWFC made no specific changes in 1989 to the design parameters of the Mark B, Mark C, and 15x15 stainless steel clad fuel rod array assemblies. Typical BWFC fuel assembly design parameters for the various current designs are given in Table 1. Developments over the past two years or so include the following:

For the Mark B8 design (1988):

- upper end fitting made easily removable, to facilitate field reconstitution
- increase in upper to lower end fitting distance, to provide more room for fuel rod growth
- increase in length of lower fuel rod end plug, largely solid metal, and lowering of position of lower spacer grid, to trap debris below the spacer grid at the solid portion of the end plug.

For the Mark-BW17 design, a debris resistant lower end fitting is in development. This design is compatible with Westinghouse Standard and Optimized Fuel Assemblies (OFAs), and has floating spacer grids, thicker fuel rod cladding, and a double fuel rod plenum.

3.2.2 Surveillance and Performance Programs

BWFC is cooperating in several fuel surveillance programs. Those in which there is progress of particular interest to report are:

TABLE 2. Major Fuel Surveillance Programs: Status Through 1989

Vendor	Fuel Type[a]	Power Plant	Planned No (Completed No) Operating Cycles	Scheduled Completion of Program	Interim Inspections to Date
Advanced Nuclear Fuels (formerly Exxon Nuclear)	15 x 15	Robinson-2	5(5)	Complete	3
	14 x 14	Prairie Island-2	3(3)	Complete	1
	8 x 8	Oyster Creek	5(5)	Complete	5
	11 x 11	Big Rock Point	4(4)	Complete	3
	14 x 14	Ginna	5(4)	1990	3
	17 x 17	Blayais-3	4(3)	1990	2
	8 x 8	WNP-2	4(2)	1991	2
	14 x 14	Calvert Cliffs	3(0)	1993	0
	15 x 15	Palisades	3(0)	1993	0
	9 x 9	Hatch-2	3(1)	1994	1
	9 x 9	Hatch-1	3(0)	1995	0
B&W Fuel Company partnership between Babcock & Wilcox and the American subsidiary of a French consortium)	15 x 15	Oconee-1	5[b]	Completed	3
	15 x 15[c]	Arkansas-1[d]	4	Completed	3
	15 x 15[e]	Rancho Seco	3	1990	2
	15 x 15[f]	Oconee-2	4	Completed	4
	15 x 15[g]	Oconee-2	3	Completed	3
	15 x 15[h]	Oconee-2	1	Completed	1
	15 x 15[h]	Oconee-1	3	Completed	3
	17 x 17[i]	Oconee-2	3	Completed	3
	15 x 15[j]	Oconee-1	4	1990	3
	15 x 15[k]	Oconee-2	3	Completed	2
	15 x 15[l]	Arkansas-1	4	Completed	3
	15 x 15[m]	Haddam Neck[n]	3	Completed	2
	17 x 17[o]	McGuire-1	3	1991	2
Combustion Engineering[p]	14 x 14[q]	Calvert Cliffs-1	5(5)	Completed	5
	14 x 14[q]	Fort Calhoun	6(6)	Completed	4
	14 x 14[r]	Calvert Cliffs-1	5(5), Part 1	Completed	5
	14 x 14[r]	Calvert Cliffs-1	5(5), Part 2	1991[s]	5
	14 x 14[t]	Calvert Cliffs-2	3(0)	1996	0
	16 x 16[t]	Arkansas-2[d]	3(3)	Completed	3
	16 x 16[u]	Arkansas-2	3(3)	Completed	3
	16 x 16[v]	Arkansas-2	5(5)	1991[s]	5
	16 x 16[t]	St Lucie-2	3(2)	Completed	1
	16 x 16[u]	Palo Verde-1	3(2)	1992	2
	16 x 16[w]	Palo Verde-1	3(1)	1994	1
	14 x 14[x]	Maine Yankee	12(12)	1991	3
General Electric	Barrier LTAs[y]	Quad Cities-1	(5)	--	--
	1983 LTAs[z]	Peach Bottom-3	(2)	--	--
	1984 LTAs[aa]	Duane Arnold	(2)	--	--
	1987 LTAs[bb]	Hatch-1	(1)	--	--
	Corrosion performance[cc]	Hatch-2	(1)	--	--
	1988 LTAs[dd]	Cooper	(1)	--	--
	Corrosion performance[ee]	Hatch-1	--	--	--
	1987 LUAs[ff]	Peach Bottom-2	--	--	--
Westinghouse	[gg]	North Anna-1	4(4)	--	--
	17 x 17 (OFA-Demo)[jj]	Farley-1	4(4)[hh] 5(5)[kk]	[ii]	4
	17 x 17 (OFA-Demo)[jj]	Salem-1	4(4)[hh]	--	3
	17 x 17 (OFA-Demo)[jj]	Beaver Valley-1	3(3)[ll]	--	3

TABLE 2. (contd)

Vendor	Fuel Type[a]	Power Plant	Planned No (Completed No) Operating Cycles	Scheduled Completion of Program	Interim Inspections to Date
Westinghouse (contd)	14 x 14 (OFA-Demo)[mm]	Point Beach-2	4(4)[nn]	(oo)	4
	17 x 17 (VANTAGE-5 Demo)	Summer-1	3(3)[pp]	(qq)	1
	IFBA Demo Fuel Rods[rr]	Turkey Point-3	(2)	--	--
	IFBA Demo Fuel Rods[ss]	Turkey Point-4	(2)	--	--
	IFM Demo Assembly[tt]	McGuire-1	(2)	--	--
	DFBN Assembly[uu]	3 Plants	--	--	--
	ZIRLO-Clad Fuel Rod Assembly[ww]	North Anna-1	3(1)[vv]		
	MO_2[xx]	R E Ginna	4(4)[yy]	--	--

(a) LTA = lead test assembly, MO_2 = mixed oxide (UO_2-PuO_2) fuel, R = retrofit fuel design, D = demonstration, OFA-Demo = Demonstration Optimized Fuel Assembly, IFBA = integral fuel burnable absorber, IFM = intermediate flow mixer, FPIP = Fuel Performance Improvement Program, DFBN = debris filter bottom nozzle, ZIRLO = an advanced zirconium alloy cladding that contains niobium
(b) For this entry, and the following entries for BWFC, scheduled completion means completion of irradiation
(c) LTAs of an advanced, extended-burnup design
(d) Arkansas Nuclear One-Unit 1 (also known as ANO-1)
(e) Current-design assemblies containing axially-blanketed fuel columns
(f) Current-design assemblies with special Zircaloy cladding materials and EPRI creep collapse specimen clusters[(33-40)]
(g) Current-design assemblies with lifted rods and cladding having a known spiral eccentricity in wall thickness[(41,42)]
(h) Current-design assemblies utilizing low-absorption spacer grid material (Zircaloy-4)[(43)]
(i) Two of these four LTAs are reconstitutable
(j) Gadolinia LTAs of an advanced, extended-burnup design
(k) Pathfinder LTA with 12 fuel rods with advanced Zircaloy cladding materials 6 rods have cladding with pure zirconium liners on the inside surface of the Zircaloy cladding and 6 rods have beta-quenched, Zircaloy-4 tubing[(44)]
(l) Same as (c), additional cycle of irradiation
(m) Four LTAs with Zircaloy-4 clad fuel rods to replace fuel assemblies with stainless steel-clad fuel rods
(n) Haddam Neck is also known as Connecticut Yankee
(o) Four 17 x 17 lead fuel assemblies (Mark-BW LA)
(p) For Combustion Engineering's major fuel research and development programs, the table entries show the status as of mid-1990
(q) Standard-design, high-burnup program
(r) Standard and advanced fuel design LTAs
(s) Hot cell examination of high burnup fuel yet to be performed
(t) Burnable poison irradiation program
(u) Standard surveillance program
(v) Standard and advanced fuel design, high-burnup program
(w) Advanced cladding designs
(x) Hot cell examination of high exposure control element assemblies
(y) Two bundles with barrier cladding involved
(z) Four bundles with improved design features involved

TABLE 2. (contd)

Footnotes for Table 2 continued below

(aa) Five bundles with improved design features involved
(bb) Four bundles Program objective lead use GE 8 x 8NB
(cc) Six fuel bundles Program objective cladding material process variables
(dd) Four fuel bundles Program objective lead use GE 8 x 8NB-1 features
(ee) Six fuel bundles Test objective Cladding material process variables effect
(ff) Four LUAs representing lead use GE 8 x 8NB production fuel
(gg) Eight fuel assemblies were irradiated as part of an EPRI program for their fourth consecutive 18-month operating cycle, four of the eight were in relatively high power positions and attained an assembly average burnup of about 58,100 MWd/MTU at discharge (May 1989), the lead fuel assembly average burnup was 58,417 MWd/MTU [45]
(hh) The two OFA-Demo assemblies in Farley-1 and the two assemblies in Salem-1 were discharged in 1984 after four cycles for examination Burnup achieved 39,170 MWd/MTU in Farley-1, and 34,400 MWd/MTU in Salem-1 [46]
(ii) Nondestructive postirradiation examinations were performed The assemblies were in good mechanical condition with no signs of deterioration See Reference 28 for the examination results
(jj) Two OFA-Demo assemblies
(kk) One of the two OFA-Demo assemblies was re-inserted for irradiation (fifth cycle) and achieved a burnup of 52,800 MWd/MTU [46] One standard fuel assembly (the symmetric partner to the OFA-Demo assembly in Cycle 7) was also irradiated for a fifth cycle and attained an average burnup of 52,080 MWd/MTU [46]
(ll) The two assemblies achieved a burnup of 35,500 MWd/MTU,[47] were discharged in 1984 after 3 cycles, and were examined
(mm) Two assemblies
(nn) The four assemblies completed their second cycle of irradiation in 1983 Subsequent examination showed one assembly had nine failed fuel rods (cause fretting wear at bottom Inconel spacer grid) The other three assemblies were in good condition, were returned to the core for a third and fourth cycle of irradiation, were discharged in 1985 and were examined [46] Average burnup achieved was 40,340 MWd/MTU [46]
(oo) Nondestructive examinations performed on the 4-cycle OFAs at the end of 1986 confirmed good performance through 4 cycles The assemblies were in good mechanical condition with no signs of deterioration See Reference 46 for the examination results
(pp) Four assemblies began power production in Cycle 2 in December 1984, completed two cycles of irradiation in March 1987, and were reinserted for a third cycle Each of the Summer demonstration assemblies contains 40 IFBA rods The assemblies also have IFMs
(qq) The four assemblies completed their third cycle of irradiation and were discharged in 1988 after attaining an accumulated average burnup of 46,050 MWd/MTU
(rr) The 4 IFBA rods were monitored during irradiation by in-core instrumentation
(ss) There were 28 IFBA rods in each of four demonstration assemblies, which allowed removal of some of the rods for postirradiation examination
(tt) One characterized IFM spacer grid demonstration assembly
(uu) Three fuel assemblies with DFBNs
(vv) The fuel rods attained a burnup of over 21,000 MWd/MTU in their first cycle, which was completed during February 1989 The rods are expected to surpass a burnup of 57,000 MWd/MTU at the completion of a third irradiation cycle
(ww) Two demonstration fuel assemblies with ZIRLO-clad fuel rods began irradiation in June 1987 ZIRLO is an advanced zirconium alloy that contains niobium ZIRLO is a trademark of Westinghouse Electric Corporation, Pittsburgh, Pennsylvania
(xx) Four assemblies with Westinghouse mixed oxide fuel rods were involved The mixed oxide (UO_2-PuO_2) fuel rods for Ginna were manufactured by Westinghouse but their irradiation was not part of a Westinghouse development program
(yy) The four assemblies were irradiated for the fourth cycle (i e , they were in the Cycle 11-14 cores) and were discharged Average burnup was 38,500 MWd/MTU

3.2.2.1 The DOE/Duke/AP&L/BWFC/Extended-Burnup Programs (Oconee 1 and ANO-1) [Joint effort among the U.S. Department of Energy, Duke Power Company (Duke), Arkansas Power & Light (AP&L), and BWFC.]

Recent performance milestones include:

- Mark BEB 15x15 - Irradiated one of four advance lead test assemblies (LTAs) to a burnup of 57.3 GWd/MTU in 1988. These LTAs feature increased fuel rod plenum volume, decreased fuel rod initial fill-gas pressure, thicker fuel rod cladding, fully annealed Zircaloy-4 guide tubes, and several fuel rods containing annular pellets. Hot cell examination of one LTA, completed in 1989 after three cycles, confirms good performance at 47 GWd/MTU.[49-60]

- Mark-BW15 - LTAs completed their third cycle of irradiation in cycle 15 of Haddam Neck (Connecticut Yankee) in September 1989. An Echo 330 ultrasonic examination showed no leaking fuel rods even though debris had damaged many of the stainless steel clad fuel assemblies in cycle 15. Growth measurements after three cycles, with new upper end fittings after the second cycle, showed a fuel rod growth margin of 0.45 inch. Full batch implementation started with cycle 17 in 1991.

- Mark-BW17 - Four LTAs finished a second cycle in February, 1990 and poolside examination showed them to be in excellent condition after 27.7 GWd/MTU.

- UO_2-Gd_2O_3 extended burnup series - This design features urania-gadolinia fuel, annular fuel pellets, annealed guide tubes, Zircaloy-4 intermediate spacer grids, and a removable upper end fitting. Results of hot cell examination of 17 fuel rods from one LTA from the first cycle of irradiation, ended in 1984, have been completed and the results displayed the expected trends. One LTA has been irradiated through a fourth cycle in Oconee-1 to 58.3 GWd/MTU. Published progress reports are given in References 61-71. The urania-gadolinia program was scheduled for completion in 1990.

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

Full batch implementation of the low absorption grids of Zircaloy-4, for 15x15 fuel assemblies, began in 1984. As of December 31, 1989 a total of 1043 of these fuel assemblies had been irradiated, with a maximum assembly burnup of 58.3 GWd/MTU.

3.2.2.3 BWFC/Duke Advanced Cladding Pathfinder Program (Oconee 2)

The Pathfinder program completed its third cycle in February 1988. Echo 330 ultrasonic examination of the fuel rods showed no leakers. However, poolside examination showed that the beta quenched cladding had higher than expected oxidation.[72] This finding, coupled with results from similar projects, have led to abandoning further evaluation of beta-quench cladding application in PWRs.[2] (Note: This finding contrasts with the results of ANF's tests of B-quenched cladding in BWRs. See Section 4.1.2.)

3.3 COMBUSTION ENGINEERING INC. (C-E) (a division of Asea Brown Boveri Combustion Engineering Nuclear Power - ABB) - (PWRs)

The Combustion Engineering letter report from which the following sections are taken is in Reference 3. More recent information is found in Reference 8.

3.3.1 Design Changes

No specific design changes were noted by Combustion Engineering in their 1989 letter report.[3] A general discussion of C-E design evolution through 1990 can be found in Reference 8; particular attention is given to the performance of the following C-E designs:

- Zircaloy-4 cladding with lower nominal value and narrower range of allowable tin content and a high integrated annealing parameter.
- Erbia mixed with the UO_2 as a burnable absorber for PWR fuel management of high burnup, extended cycle operation.
- Debris resistant designs, in particular the GUARDIAN™ debris straining bottom grid.
- System 80[73] featuring an all-Zircaloy, reconstitutable fuel structure that has a lower core position in the reactor vessel, which reduces radiation fluence in the nozzle region and improves the expected small break loss-of-coolant accident (LOCA) performance.

3.3.2 Surveillance and Performance Programs

High burnup, extended cycle operation concerns are being addressed by a series of Lead Fuel Assembly (LFA) programs as discussed in the 1988 and 1991 International Topical Meetings on LWR Fuel Performance.[8,74] Both standard and advanced fuel designs are being evaluated. The performance programs currently in progress will provide hot cell evaluation of fuel and cladding with peak local burnup approaching 70 GWd/MTU. These programs include:

- Zircaloy-4 Fuel Rod and Assembly Guide Tube Growth
- Zircaloy-4 Fuel Rod Corrosion Behavior
- Erbia-Urania Fuel Behavior - Four Lead Fuel Assemblies containing 0.9 w/o Erbia in 3.4 w/o enriched UO_2 fuel pellets, fabricated in 1989, are operating in Calvert Cliffs II.

3.4 GENERAL ELECTRIC (GE) - (BWRs)

The General Electric design and surveillance information which follows is taken from Reference 4. More recent information through August of 1990 is given in Reference 75.

3.4.1 Design Changes

GE has made a variety of design modifications over the years to improve fuel corrosion resistance and overall fuel performance. Modified features include water rod configuration, spacer and upper tie plate, cladding surface treatment (involving material and heat treatment), axial zoning of gadolinia, fuel rod helium prepressurization, pellet dimensions, and pellet density. For PCI failures, the barrier concept for protecting the fuel cladding with Zr-lining has been tested since 1979, with periodic pool-side examination of representative bundles and fuel rods. Tests in Quad Cities-2 have included power increases for additional PCI resistance demonstration. No PCI-induced Zr-barrier fuel failures have been found in more than 680,000 barrier fuel rods exposed to at least one reactor cycle of operation.

CILC failures were discovered in 1979 in plants with copper alloy condenser tubes and filter demineralizer condensate cleanup systems, under certain specific conditions. Following the development of an out-of-reactor test of the susceptibility of Zircaloy to in-reactor nodular corrosion, manufacturing processes have been developed to improve the corrosion resistance of the Zircaloy starting material and to maintain that corrosion resistance throughout the fuel cladding fabrication.

3.4.2 Surveillance and Performance Programs

The fuel surveillance program adopted by GE and accepted by the NRC is described in four NRC reports.[76-79] A summary of the GE lead use assembly (LUA) surveillance program is contained in Table 3. Several of the Lead Use Assembly programs currently underway are discussed below:[4]

- 1983 LUAs - Four LUAs were loaded into Peach Bottom-3 in 1983 at the beginning of cycle 6 to test improved spacer and upper tie plate designs, axial zoning of gadolinia, and variations in cladding thickness, pellet dimensions, and fuel rod helium prepressurization. Poolside examination after one cycle in August 1985 and after two cycles in November 1987 showed characteristics of normal operation. Peach Bottom-3 returned to service in December 1989.

- 1984 LUAs - Five LUAs were loaded into Duane Arnold in 1985 at the beginning of cycle 8 to test water rod configuration, improved spacer and upper tie plate designs, cladding surface treatment, axial zoning of gadolinia, and variations in fuel rod helium prepressurization, pellet dimensions, and pellet density. Poolside examinations were made after one cycle in April 1987 and after two

TABLE 3. Summary of Ongoing Lead Use Assembly Surveillance Programs General Electric as of December 31, 1989[4]

Program	Reactor	No. of Bundles	No. of Completed Cycles of Operation	Bundle Average Exposure At Last Outage GWd/MTU	Objectives
Barrier LUA's	Quad Cities-1	1	5	43	Barrier cladding
1983 LUA's	Peach Bottom-3	4	2	24	Improved design features
1984 LUA's	Duane Arnold	5	2	28	Improved design features
1987 LUA's	Hatch-1	4	1	12	Lead use GE8 x 8NB
Corrosion Performance	Hatch-2	6	1	13	Cladding material process variables
GE8 x 8NB-1 Channel LUA's	Cooper	4	1	8	Lead use GE8 x 8NB-1 features
Corrosion Performance	Hatch-1	6	--	--	Cladding material process variables
1987 LUA's	Peach Bottom-2	4	--	--	Lead use GE8 x 8NB

cycles in October 1988, showing characteristics of normal operation. The next poolside examination was scheduled after the third cycle of operation in 1990.

- 1987 LUAs - Four LUAs representing GE8x8NB production use were loaded into Hatch-1 in 1987 (cycle 11). No evidence of CILC was found after one cycle in 1988. The next poolside examination was scheduled after the second cycle of operation in 1990.

- Cladding Corrosion Performance LUAs - Six LUAs were loaded into Hatch-2 in early 1988 (cycle 8) and six in Hatch -1 in late 1988 (cycle 12) to test cladding material, heat treatment, and surface conditioning. Both reactors have historically exhibited highly variable cladding corrosion performance. After one cycle and exposures up to 13,000 MWd/MTU, visual inspection revealed little or no visible nodular corrosion along the full length of the fuel rods. The next poolside examination of Hatch-2 bundles was scheduled in 1991, for Hatch-1 in 1990.

- GE8x8NB-1 Channel LUAs - Four LUAS representing GE8x8NB-1 production fuel bundle design features were loaded into Cooper in 1988 (cycle 12). Normal characteristics were found during poolside examination after one cycle in 1989. The second examination was scheduled for 1990.

- 1987 LUAs - Four LUAs representing production fuel were loaded into Peach Bottom-2 in 1989 (cycle 8). Poolside examination was scheduled for 1991.

- Zr barrier-coated cladding - Thirty two demonstration Zr barrier-coated bundles are currently operating in their fifth cycle in Quad Cities-2.

3.5 WESTINGHOUSE ELECTRIC CORPORATION (W) - (PWRs)

The Westinghouse report WCAP-8183, "Operational Experience with Westinghouse Cores"[5] is the basis for the following sections.

3.5.1 Design Changes

No new design changes were specifically noted during 1989. An overview of the Westinghouse design evolution is given in a 1991 paper by M.G. Balfour et al.[80] and in the letter report summary of 1989 fuel experience.[5] These ongoing developments include:

- The Debris Filter Bottom Nozzle (DFBN) - The DFBN has smaller flow holes than previous fuel assembly bottom nozzles in order to minimize passage of metallic debris large enough to cause fretting damage to fuel rods, while still providing a pressure drop equivalent to the previous fuel assembly bottom nozzle.

- Optimized Fuel Assemblies (OFAs) - To improve fuel utilization by enhancing neutron moderation and reducing parasitic capture, 14x14 and 17x17 OFAs employ a slightly-reduced fuel rod diameter compared to the non-OFA design, fuel rod, while retaining the same fuel rod pitch. While the top and bottom grids in the OFAs are made of Inconel, intermediate grids are of Zircaloy.

- VANTAGE 5 and 5H Fuel Assemblies - The VANTAGE 5 assembly has the same optimized fuel rod and Zircaloy grids as the OFA and has been improved further by incorporating features which reduce fuel cycle cost, increase core operating margins, and improve design and operating flexibility. These features include:

  - Integral Fuel Burnable Absorbers (IFBAs)

  - Intermediate Flow Mixer Grids (IFMs)

  - Axial blankets

  - A Reconstitutable Top Nozzle (RTN)

  - Increased discharge burnup.

- The VANTAGE 5H contains the VANTAGE 5 features but uses the non-OFA fuel rod of the Westinghouse Standard (LOPAR) fuel assembly and new low pressure drop Zircaloy grid design. Approximately 80% of the fuel pellets in an IFBA rod are coated with a thin zirconium boride coating which serves as a burnable absorber. IFM grids are small mixing vane grids that are located in the upper spans of the fuel assembly, between the Zircaloy structural grids, to provide more margin for departure from nucleate boiling.

The VANTAGE 5 assembly has features that reduce fuel cycle cost, increase core operating margins, and improve design and operating flexibility. These features include integral fuel burnable absorbers (IFBAs), intermediate flow mixer grids (IFMs), axial blankets, a reconstitutable top nozzle (RTN), and increased discharge burnup.

"The VANTAGE 5 assembly has the same optimized fuel rod and Zircaloy grids as the OFA" and has been improved further by incorporating "features which reduce fuel cycle cost, increase core operating margins, and improve design and operating flexibility." These features include: "1) Integral Fuel Burnable Absorbers (IFBAs), 2) Intermediate Flow Mixer Grids (IFMs), 3) Axial blankets," 4) "A Reconstitutable Top Nozzle (RTN)", and 5) "Increased discharge burnup." "Approximately 80% of the fuel pellets in an IFBA rod are coated with a thin zirconium diboride coating which serves as a burnable absorber. IFM grids are small mixing vane grids that are located in the upper spans of the fuel assembly," "between the Zircaloy structural grids," to provide more margin for departure from nucleate boiling.

- The VANTAGE 5H contains the VANTAGE 5 fuel features but uses the non-OFA fuel rod of the Westinghouse Standard (LOPAR) fuel assembly and a new low-pressure-drop Zircaloy grid design. Approximately 80% of the fuel pellets in an IFBA rod are coated with a thin zirconium boride coating that functions as a burnable absorber. IFM grids are small mixing vane grids that are located in the upper spans of the fuel assembly, between the Zircaloy structural grids, to provide more margin for departure from nucleate boiling.

- ZIRLO™ Cladding - This advanced cladding contains niobium, which provides additional resistance to corrosion, to permit fuel usage at higher burnups and/or higher temperatures.

3.5.2 Surveillance and Performance Programs

The fuel performance summary on a plant-by-plant is provided in Table 2. Summaries of several surveillance programs follow:

- Optimized Fuel Assemblies (OFAs) - From 1979 to 1986, ten demonstration OFAs (six 17x17 and four 14x14) were irradiated in four reactors (Point Beach Unit 2, Beaver Valley Unit 1, Salem Unit 1 and Farley Unit 1) to assembly average burnups in the range of 33,850 to 53,000 MWd/MTU. All assemblies were discharged in good condition except one that suffered fretting wear due to a nonstandard step in the manufacturing process.

In 1989, thirty two plants operated with at least one region of OFA fuel. Observations of OFA fuel at more than 20 plants during 1989 confirmed good overall performance. The following 1989 statistics are for at least one OFA region in-core:

No. of Plants	Cycles	Peak Region Ave. Burnup MWd/MTU
29	1st	22,000
27	2nd	38,300
25	3rd or 4th	45,400

- VANTAGE 5 and VANTAGE 5H Fuel Assemblies - Four VANTAGE 5 fuel demonstration assemblies (17x17) were loaded into the V.C. Summer Unit 1 cycle 2 core and began power production in December of 1984. After three cycles of irradiation they were discharged in September of 1988 with an average burnup of 46,050 MWd/MTU. All four demonstration assemblies exhibited no mechanical damage or wear and the IFM grids had no effect on the adjacent fuel assemblies. Individual VANTAGE 5 fuel features have been demonstrated at other nuclear plants: IFBA demonstration rods at Turkey Point Units 3 and 4 and the IFM grids at McGuire Unit 1. Full regions of reload fuel with at least one VANTAGE 5 fuel feature were in operation in 38 plants during 1989, including:

  - 13 plants with full regions containing axial blankets,
  - 36 plants with RTNs, 14 with IFBAs, 5 with IFM grids,
  - 21 plants began operating with the Debris Filter Bottom Nozzle (DFBN), and
  - 23 plants with assembly modifications for high burnups.

  In addition, two plants operated with an initial region of VANTAGE 5H fuel during 1989.

- ZIRLO™ Clad Fuel Rods - Two demonstration assemblies containing ZIRLO™ clad fuel rods began irradiation in the North Anna Unit 1 during June 1987. Their first cycle was completed in February 1989, with a burnup over 21,000 MWd/MTU. One of the assemblies was inserted for a second cycle and was expected to achieve a burnup of about 37,000 MWd/MTU in early 1991.

3.6 ELECTRIC POWER RESEARCH INSTITUTE (EPRI) PROGRAMS

The current status of the EPRI fuel performance surveillance program is the same as given in the 1988 Annual Report and is described in two reports.[82, 83] EPRI's research and development plan for 1987-89 is described in Reference 84. Additional information on the program is available in References 85-87. Two EPRI reports[88,89] contain information on design changes associated with BWR fuel. Use of the BWR Power Shape Monitoring System is described in Reference 90. Collection and formatting of data on reactor coolant activity and fuel rod failures is described in a 1986 report.[91] The lifetime of PWR silver-indium-cadmium control rods is described in a 1986 report.[92] Hydrogen water chemistry for BWRs is discussed in a 1987 report.[93] A 1987 paper[94] includes information on advances in LWR fuels. Zircaloy oxidation and hydriding under irradiation are discussed in a 1987 report.[95] Guidelines for improving fuel reliability are described in a 1989 report.[96]

4.0 FUEL OPERATING EXPERIENCE

As of the end of 1989, the total number of fuel assemblies that were in, or that had completed, operation in the United States was over 110,500[a] (over 65,700 in BWRs and about 44,800 in PWRs). The total number of fuel rods in fuel assemblies all over the world, as supplied and reported by the five major U.S. nuclear fuel vendors in their annual letter reports to the Nuclear Regulatory Commission, is over 15 million (about 4.2 million BWR type and 10.8 million PWR type); this compares with 14.4[19] million fuel rods throughout the world in 1988 (4.6 million BWR type and 9.8 million PWR type).

As of the end of 1989 there were 108 operable, licensed commercial reactors in the United States.[97] These plants generated 529.4 TWh[98] and achieved an average capacity factor of 61.7 % in 1989; the corresponding figures for 1988 are 525 TWh and 63.3% respectively, from the same source.[b]

A synopsis of domestic fuel performance is provided for each of the five domestic vendors in the sections which follow. The fuel integrity ratings of fuel from each vendor are also provided. These ratings are normally obtained from Iodine-131 activity levels initially, followed where possible by gas sipping or ultrasonic measurements; these methods are described in References 99-108. To assess the overall performance of fuel rods, the Institute of Nuclear Power Operations (INPO) Fuel Reliability Indicator (FRI)[109] has become a commonly used standard. The FRI for PWRs is the Iodine-131 coolant activity level normalized to a standard cleanup system flow rate (also referred to as the "uncorrected activity") and corrected for tramp uranium[c] and alternately referred to as the "corrected activity" or FRI value. For BWRs, the FRI value is determined from the rate of fission gas release measured at the steam jet air ejector. Lower FRI values are qualitatively indicative of fewer failed fuel rods in the core. The rule-of-thumb average Iodine-131 activity in reactor coolant is about 1.2×10^{-3} μCi of Iodine-131 per gram; in general, levels above this value signal the presence of leaking fuel rods. The specific coolant activity technical specification limit for each reactor depends on such factors as reactor power and coolant purification flow rate.

(a) This is lower than the number given in the Fuel Performance Annual Report for 1988 because it is from a different source. The total number of assemblies consists of a count of the in-core assemblies from Table 1 of a report from the Department of Energy [110] and the number discharged, from Table 5 of the same Reference.

(b) The figure for total electrical power generation given in Reference (19) is derived from a different source and is higher by about 5%.

(c) Tramp uranium is finely divided uranium oxide particles suspended in the coolant or deposited on core surfaces.

Historical information on fuel failure rates in BWRs and PWRs is provided in Appendix A. It should be noted that the definition of failed fuel is not uniformly applied;[a] in many cases the number of fuel failures is inferred from indirect evidence, while in other cases only directly observed failures are counted.[111]

Although overall commercial reactor operating experience continues to be excellent, there are sporadic events involving damage to or failure of fuel; those events are discussed in Section 5.0 and Appendix B.

4.1 ADVANCED NUCLEAR FUELS CORPORATION (ANF)

ANF fuel performance and fuel rod integrity through the end of 1989 are described in Reference 1.

4.1.1 Fuel Performance - Fuel Utilization and Burnup

As of the end of 1989, a total of 16,480 fuel assemblies containing 1,957,723 fuel rods had been irradiated, about 64% in BWRs and 36% in PWRs; and ANF fuel had been loaded into 47 commercial light water reactors (LWRs) (22 BWRs and 25 PWRs) in the United States, Europe, and Asia. The ANF fuel experience through December 31, 1989, is summarized in Table 4.

The exposure distribution of ANF fuel rods and assemblies, as of the end of 1989, is shown in Figure 1.[1] The highest exposure levels reached by ANF fuel to date are 41.1 GWd/MTU in 1985 for a group of BWR fuel rods irradiated at Big Rock Point in Michigan and 50.0 GWd/MTU in 1986 for PWR fuel irradiated at Tihange-1 in Belgium. ANF's BWR 9x9 fuel assemblies and PWR 17x17 fuel assemblies reached new high burnups in 1989: BWR 9x9 fuel at Gundremmingen-3 in Germany reached 40.0 GWd/MTU and PWR 17x17 fuel at Donald C. Cook-2 in Michigan reached 44.0 GWd/MTU.

(a) A two-volume report[112] published in 1980/1981, 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, or design), whether or not there is a legal contingency on that requirement (technical specification, fuel warranty, or design basis), and which indicator is used (coolant or off-gas activity, sipping, strain, or deflection). Definitions of fuel damage, failures, 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 Standard Review Plan (SRP).[31]

TABLE 4. Summary of Advanced Nuclear Fuels Corporation Fuel Experience through December 31, 1989[1]

A. Fuel Assemblies

Reactor Type	In Core: No. of Fuel Assemblies	In Core: Maximum Burnup, GWd/MTU	Discharged: No. of Fuel Assemblies	Discharged: Maximum Burnup, GWd/MTU	Total No. of Fuel Assemblies
BWR	7,674	34.4	2,847	41.1[a]	10,521
PWR	1,989	45.0	3,970	50.0	5,959
Total	9,663		6,817		16,480

B. Fuel Rods

Reactor Type	Number of Fuel Rods: In Core	Discharged	Total
BWR	525,862	176,686	702,548
PWR	447,902	807,273	1,255,175
Total	973,764	983,959	1,957,723

(a) Average of extended burnup rods transferred to a new host fuel assembly.

4.1.2 Fuel Rod Integrity

Historically, the overall ANF fuel rod integrity, based on failures that were judged to be from fuel related or unknown causes, has remained at better than 99.994%.[109] In 1989, the fuel rod reliability remained better than 99.997%. Failure statistics on all ANF fuel rods through December 31, 1989 are provided in Table 5.[1]

To assess the reliability of ANF fuel, ANF uses the INPO FRI described above. The FRI distribution for ANF PWR and BWR fuel is shown in Figure 2[1] and is derived from the 1989 yearly average for each reactor that operated with ANF fuel in the core. The median value for all PWR reactors containing ANF fuel is 1.26×10^{-3} μCi/ml, which compares well with the INPO PWR median value of 2.0×10^{-3} μCi/ml. The median value for BWR reactors containing ANF fuel in 1989 is $3.70 \times 10^{+2}$ μCi/ml, with the industry median being $8.30 \times 10^{+1}$ μCi/ml. ANF did not have any failures attributed to design or manufacturing in 1989. The five-year trend in the ANF FRI indicates a continued improvement in fuel performance.

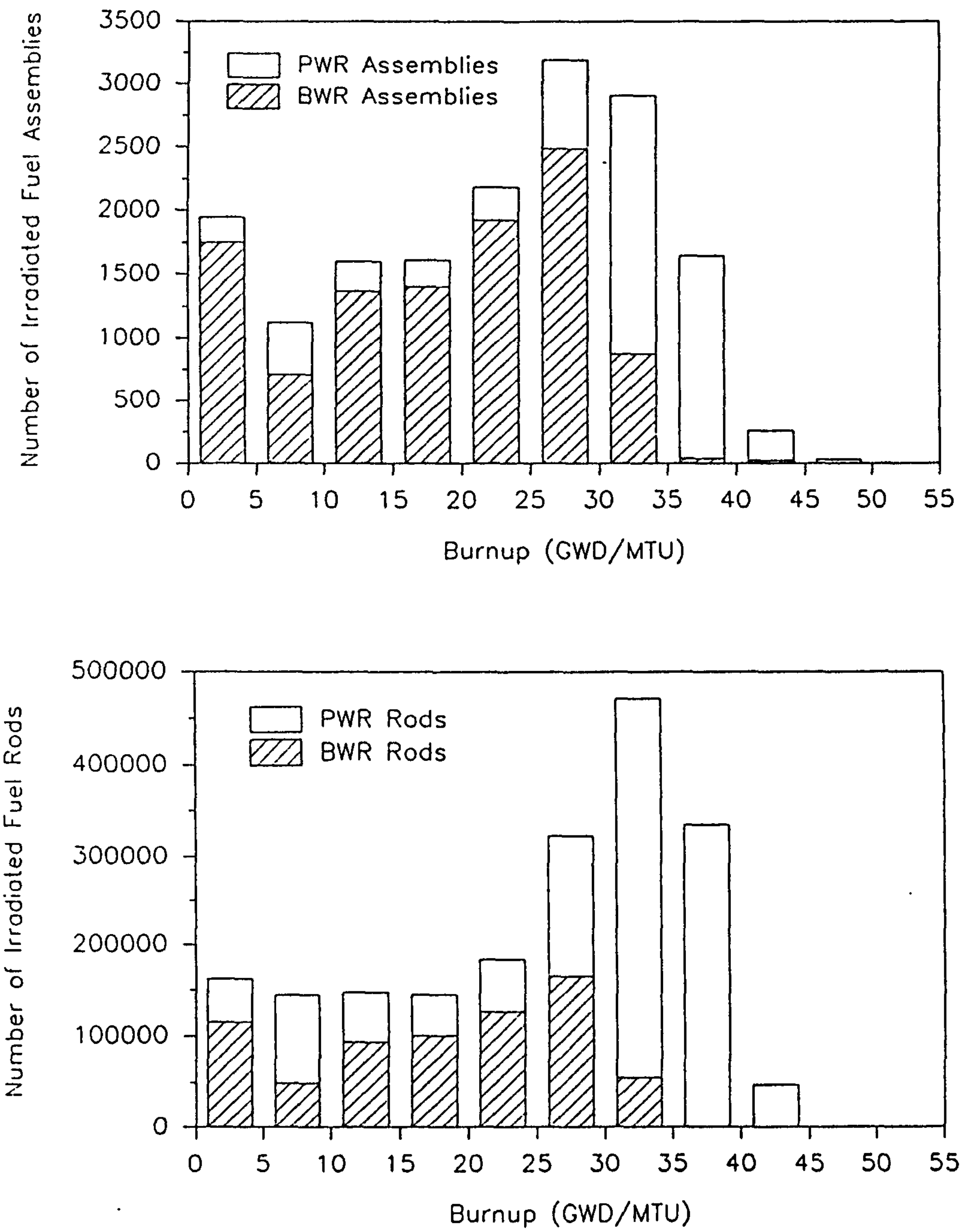

FIGURE 1. Distribution of Irradiated Advanced Nuclear Fuel Corporation's Fuel by Exposure Through the End of 1989[1]

TABLE 5. ANF Fuel Rod Failure Statistics as of December 31, 1989[1]

Reactor Type	No. of Irradiated Rods	Failed Rods Burnup Less Than Warranted, Fuel Related		Failed Rods Burnup Less Than Warranted, Core Related		All Other ANF Failures[a]		Total Failures	
		No.	Rate	No.	Rate	No.	Rate	No.	Rate
BWR	702,548	50	0.007%	98	0.014%	13	0.002%	161	0.023%
PWR	1,255,175	16	0.001%	114	0.009%	73	0.006%	203	0.016%
Total	1,957,723	66	0.003%	212	0.011%	86	0.004%	364	0.019%

(a) Failures not examined and/or above warranted burnup.

During 1989, leaks[a] in cladding attributable to causes other than fuel design or manufacturing were found by ANF to be from debris in the coolant stream trapped or lodged where it could cause fretting of the cladding.

ANF standard cladding continued to show good corrosion performance in all reactor environments, based on corrosion data collected during 1989. These data were obtained at three PWRs and four BWRs. Beta-quenched cladding reached exposures as high as 39.6 GWd/MTU, exhibiting resistance to corrosion particularly in those BWRs which are characterized as susceptible to crud-induced localized corrosion (CILC). (Note: This contrasts with BWFC's finding that in PWRs beta-quenching provides no particular advantage. See Section 3.2.2.3.)

4.2 B&W FUEL COMPANY (BWFC)

The performance and integrity of BWFC fuel throughout 1989 is provided in the letter report in Reference 2.

4.2.1 Fuel Performance - Fuel Utilization and Burnup

A summary of B&W Fuel Company's fuel rod experience, from the startup of their first reactor (Oconee-1) in April 1973 through December 1989, is provided in Table 6.[2] The operating status of BWFC-fueled reactors is shown in Table 7.[2]

Batch average burnups for the BWFC 15x15 Mark B fuel design have increased from 27.0 to 37.0 GWd/MTU. The peak burnup of a discharged fuel assembly (an LTA) in 1989 is 58.3 GWd/MTU following four cycles in core. The burnup experience for BWFC-supplied Zircaloy-clad fuel is summarized in Table 8[2] and for BWFC-supplied stainless steel-clad fuel in Table 9.[2]

(a) "Leaks" refer to the release of fission products to the primary coolant through a breach in the fuel rod cladding.

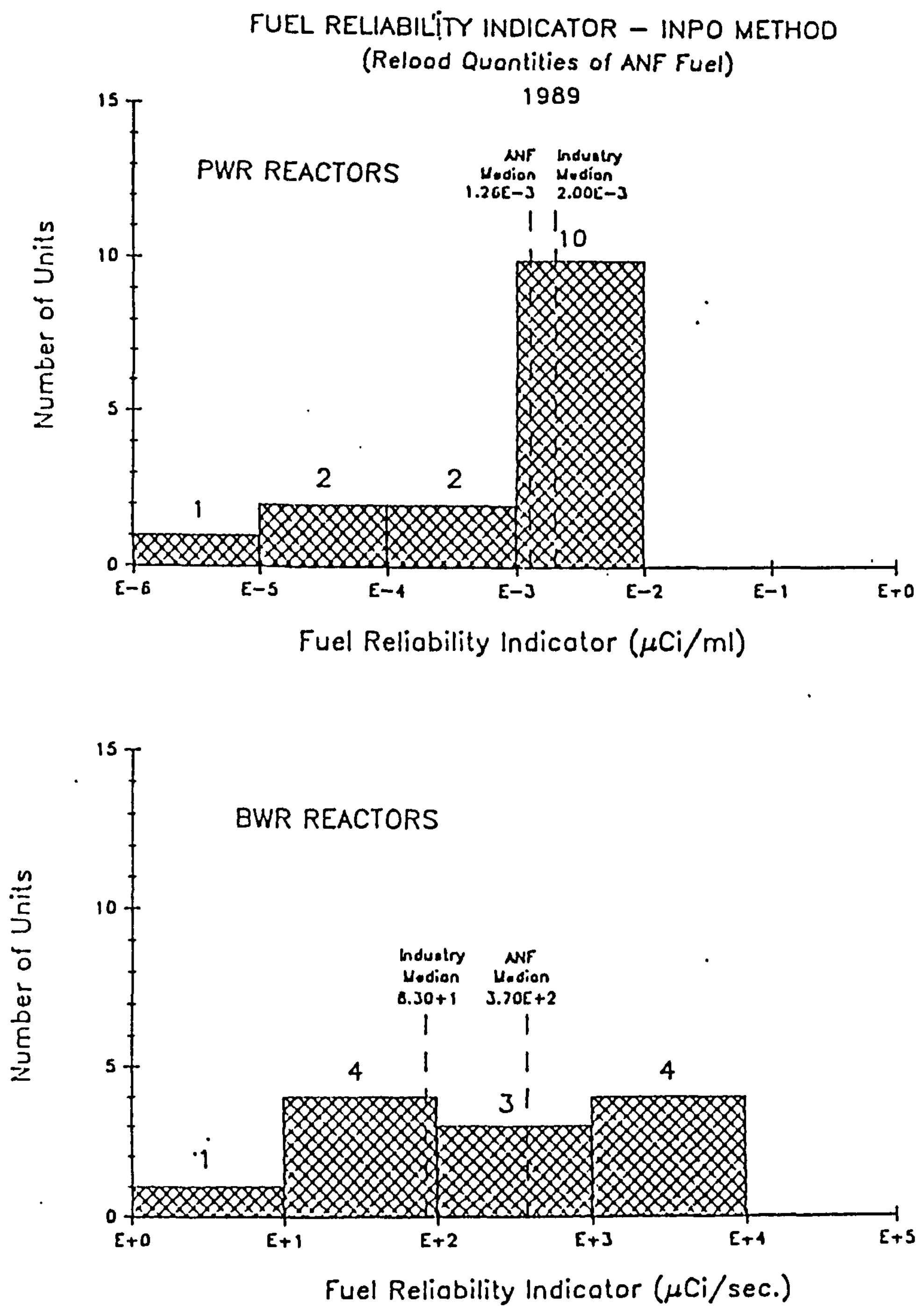

FIGURE 2. Advanced Nuclear Fuel Corporation PWR Fuel Reliability Indicator (FRI), Using the Institute of Nuclear Power Operations (INPO) Method: Annual Distribution of FRI for PWRs in 1989. (Note: 1.2E-04 is equivalent to 1.2×10^{-4})[1]

TABLE 6. 1989 Performance Summary for BWFC-Supplied Fuel Rods[a] (Cumulative to December 31, 1989)[2]

	Stainless Steel	Zircaloy-Clad Fuel: Fuel Assembly Rod Array Type 15 x 15	(Mark C) 17 x 17	(Mark BW) 17 x 17
1. Cumulative Number of Rods Irradiated Through Dec. 1989:	107,100	1,055,216	1,056	1,056
a. Maximum Rod-Average Burnup, GWd/MTU	39.2	60.8	36.4	15.5
b. Mean Rod-Average Burnup, GWd/MTU	27.8	27.7	30.1	15.3
2. Total Number of Rods Irradiated in 1989	32,028	247,712	--	1,056
3. Number of Irradiated Rods Incore on Dec. 31, 1989:	--	257,712	--	1,056
a. Maximum Rod-Average Burnup, GWd/MTU	--	40.6	--	15.5
b. Mean Rod-Average Burnup, GWd/MTU	--	21.6	--	15.3
4. Number of Rods Discharged in 1989:	32,028	70,304	--	--
a. Maximum Rod-Average Burnup, GWd/MTU	39.7	60.8	--	--
b. Mean Rod-Average Burnup, GWd/MTU	26.0	25.5	--	--
5. Estimated Number of Leaker Rods Generated in 1989	450[b]	8[c]	--	--

(a) Three Mile Island Unit 2 is excluded from this tabulation.
(b) Based on a combination of ultrasonic inspection and visual inspection during reconstitution. All failures examined had debris wear on cladding near bottom end cap.
(c) Estimated from equilibrium coolant radio-iodine behavior during full-power operation, or UT examination of fuel assemblies.

TABLE 7. Operating Status of BWFC-Fueled Reactors[2] (December 31, 1989)

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

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

4.2.2 Fuel Rod Integrity

In 1989 a total of eight leaking fuel rods occurred out of 257,712 BWFC Zircaloy-clad rods irradiated. Leaking from Zircaloy-clad rods was determined from coolant radio-iodine levels during full power operation and from ultrasonic inspection. No plant generated more than three leaking fuel rods; in only two plants were there more than one leaking fuel rod. The cause of these events has not yet been identified; in past years, poolside examination of leaking fuel rods showed debris in the core and spacer-grid fretting to be the primary causes of leaking fuel. The performance of BWFC Zircaloy-clad rods during 1989 represents a fuel integrity of 99.997%.

An investigation based on ultrasonic testing results determined that manufacturing variation contributed to the spacer grid fretting. Corrective action was taken to prevent those variations and a program is in place to follow both spacer grid manufacturing and fuel performance to prevent further occurrence of the problem.

TABLE 8. Summary of Burnup Experience for BWFC-Supplied Zircaloy-Clad Fuel[a] (December 31, 1989)[2]

Fuel Assembly Average Burnup, MWd/MTU	Assemblies Incore on December 31, 1989 No. of Assemblies	No. of Rods	Assemblies Discharged in 1989 No. of Assemblies	No. of Rods	Assemblies Discharged Through December 31, 1989 No. of Assemblies	No. of Rods
0 to 4,000	52	10,816	0	0	0	0
4,000 to 8,000	104	21,632	0	0	4	832
8,000 to 12,000	44	9,152	56	11,648	159	33,072
12,000 to 16,000	120	24,960	0	0	134	27,872
16,000 to 20,000	224	46,592	32	6,656	192	39,936
20,000 to 24,000	153	31,824	33	6,864	330	68,640
24,000 to 28,000	189	39,312[d]	16	3,328	1,154[b]	240,144
28,000 to 32,000	201	41,808	46	9,568	1,057	219,856
32,000 to 36,000	103	21,424	40	8,320	553[c]	115,136
36,000 to 40,000	44	9,152	72	14,976	312	64,896
40,000 to 44,000	5	1,040	41	8,528	80	16,648
44,000 to 48,000	0	0	0	0	10	2,080
48,000 to 52,000	0	0	0	0	1	208
52,000 to 56,000	0	0	1	208	1	208
56,000 to 60,000	0	0	1	208	2	416
TOTALS	1,239	257,712	338	70,304	3,989	829,728

(a) Three Mile Island Unit 2 is excluded from this tabulation.
(b) Includes two nonreconstitutable 17 x 17 LTA's (Mark C).
(c) Includes two reconstitutable 17 x 17 LTA's (Mark CR).
(d) Includes four 17 x 17 Lead Assemblies (Mark BW).

Of 32,028 stainless steel clad rods irradiated, an estimated total of 450 leaking fuel rods developed, due to extensive damage from debris fretting. All stainless steel clad rods were irradiated in the Haddem Neck (Connecticut Yankee) reactor; the fuel rod performance was determined by ultrasonic examination and visual inspections during an extensive reconstitution effort during the cycle 15 to 16 refueling. In spite of the large number of leaking rods in the one reactor, the average coolant iodine level for 1989 was slightly more than one fourth of the 1980 level, as is evident in Table 10. The performance for stainless steel clad rods represents a fuel integrity of 98.6% for 1989.

The number of leaking fuel rods shown in Table 6 was estimated by means of radio-iodine activity at full power operation and is therefore a best guess only. The BWFC ultrasonic testing method, the Echo 330 ultrasonic system, which uses a Lamb wave to detect the presence of water in the fuel-to-clad gap in individual fuel rods, permits a more precise determination of the number of

TABLE 9. Summary of Burnup Experience for BWFC-Supplied Stainless Steel Clad Fuel (December 31, 1989)[2]

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

(a) No fuel assemblies incore as of December 31, 1989 All fuel assemblies are offloaded for debris cleaning, inspection and reconstitution
(b) Includes four Lead Test Assemblies with Zircaloy clad fuel rods

leakers. To date, nine ultrasonic inspections have been performed on five BWFC-designed reactors (Arkansas Nuclear One Unit 1, Oconee-1, Oconee-2, Oconee-3, and Three Mile Island Unit 1) plus Westinghouse's Connecticut Yankee. The ultrasonic data revealed a large uncertainty in the radiochemistry projections. Investigations continue, combining visual inspection with the other methods for comparison and improved determinations.

4.2.3 Non-Fuel Core Components

Fuel Assembly Holddown Springs - In 1989, 19 broken holddown springs were found at two reactors - Oconee-2 and Oconee-3 - in fuel assemblies that were to be reinserted. The springs were replaced, inasmuch as the determination has been made that the broken springs do not pose a safety concern for continued reactor operation. Efforts to prevent broken springs include the use of increased wire diameter, the use of Inconel X-718 instead of X-750, and a change in the spring manufacturing process.

TABLE 10. Average Steady State Coolant Iodine Activity for B&W Designed Plants[2]

Date	I-131 Activity, μci/g
1980	0.086
1981	0.046
1982	0.031
1983	0.041
1984	0.051
1985	0.031
1986	0.014
1987	0.028
1988	0.035
1989	0.023

4.3 COMBUSTION ENGINEERING INC. (C-E)

A summary of the performance of Combustion Engineering fuel during 1989 is given in the C-E annual letter report.[3]

4.3.1 Fuel Performance - Fuel Utilization and Burnup

The highlights of the C-E fuel performance for 1989 are discussed below:

- A summary of Combustion Engineering Fuel Irradiated and/or Discharged and the batch averaged burnups achieved in 1989 is presented in Table 11. The highest batch averaged burnup in reactor at the end of 1989 was 43,000 MWd/MTU at Calvert Cliffs II. The highest batch averaged burnups at discharge during 1989 were 43,000 MWd/MTU at Arkansas-2 (ANO-2) and 42,000 MWd/MTU at St. Lucie-2. However, batch-averaged burnup at discharge of 56,800 and 48,200 MWd/MTU at Calvert Cliffs II and 51,600 MWd/MTU at Arkansas-2 were experienced in 1988[19] and are not included in the current table.

- The cumulative irradiation experience of active and discharged all-Zircaloy C-E assemblies through December 31, 1989 is shown in Table 12. The total number of C-E supplied rods, in reactor and discharged in 1989, was 1,556,158; the total number of C-E assemblies was 7,756.

- The status of the major C-E Fuel Research and Development Programs, as of mid-1990, has been incorporated into Table 2.

TABLE 11. Summary of Combustion Engineering Fuel Irradiated and/or Discharged in 1989[3]

Reactor/ (Fuel Cycle)	Fuel Batch	Number of Assemblies: In Reactor at End of Year	Number of Assemblies: Discharged During Year	Number of Fuel Rods: In Reactor at End of Year	Number of Fuel Rods: Discharged During Year	Batch-Averaged Burnup, MWd/MTU: On Dec. 31, 1989	Batch-Averaged Burnup, MWd/MTU: At Discharge
Arkansas-2/ (Cycles 7&8)	F	17	0	4,012	0	32,300	--
	G	0	49	0	11,324	--	43,000
	H	28	32	6,352	7,040	35,500	37,000
	J	68	0	15,312	0	19,000	--
	K	64	0	14,416	0	500	--
Calvert Cliffs-1/ (Cycle 10)	K	69	0	12,144	0	34,100	--
	L	52	0	9,152	0	23,000	--
	M	92	0	15,280	0	13,000	--
Calvert Cliffs-2/ (Cycle 8)	H	69	0	12,144	0	43,000	--
	J	60	0	10,560	0	34,000	--
	K	88	0	14,800	0	22,000	--
Fort Calhoun/ (Cycle 12)	M	44	0	7,552	0	25,000	--
	N	44	0	7,552	0	10,000	--
Maine Yankee/ (Cycle 11)	N	64	0	10,880	0	37,500	--
	P	72	0	12,400	0	27,000	--
	Q	72	0	12,464	0	13,000	--
Palo Verde-1/ (Cycle 2)	B	97	0	21,340	0	28,300	--
	C	64	0	14,720	0	25,200	--
	D	80	0	18,528	0	12,500	--
Palo Verde-2/ (Cycle 2)	B	69	0	15,180	0	29,500	--
	C	64	0	14,720	0	29,000	--
	D	108	0	24,400	0	16,500	--
Palo Verde-3 (Cycle 1)	A	69	0	16,284	0	15,300	--
	B	108	0	23,760	0	17,100	--
	C	64	0	14,720	0	12,500	--
St. Lucie-2/ (Cycles 4&5)	D	4	57	944	13,252	36,000	42,000
	E	57	27	13,212	6,068	35,000	33,000
	F	76	0	17,536	0	23,500	--
	G	80	0	18,448	0	9,000	--
San Onofre-2/ (Cycles 4&5)	A	1	5	236	1,180	15,000	30,000
	D	0	16	0	3,776	--	30,900
	E	0	88	0	20,320	--	36,000
	F	108	0	24,112	0	23,400	--
	G	108	0	24,112	0	500	--
San Onofre-3/ (Cycle 4)	A	5	0	1,180	0	28,200	--
	D	16	0	3,776	0	28,900	--
	E	88	0	20,320	0	32,000	--
	F	108	0	24,112	0	19,800	--

TABLE 11. (contd)

Reactor/ (Fuel Cycle)	Fuel Batch	Number of Assemblies: In Reactor at End of Year	Number of Assemblies: Discharged During Year	Number of Fuel Rods: In Reactor at End of Year	Number of Fuel Rods: Discharged During Year	Batch-Averaged Burnup, MWd/MTU: On Dec. 31, 1989	Batch-Averaged Burnup, MWd/MTU: At Discharge
Waterford-3/ (Cycles 3&4)	C	1	41	224	9,664	29,000	32,000
	D	48	44	11,232	10,080	33,500	34,000
	E	84	0	18,896	0	19,000	--
	F	84	0	18,896	0	1,500	--
Yankee Rowe (Cycles 19&20)	A	0	40	0	9,130	--	31,000
	B	36	0	8,222	0	26,100	--
	C	40	0	9,090	0	10,100	--

TABLE 12. Combustion Engineering Burnup Experience With All-Zircaloy Assemblies: Status as of December 31, 1989[3]

Fuel Assembly Batch-Average Burnup, MWd/MTU	In-Core Fuel Assemblies with Pressurized Fuel Rods: No. of Fuel Assemblies	In-Core Fuel Assemblies with Pressurized Fuel Rods: No. of Fuel Rods	Discharged Fuel Assemblies with Pressurized Fuel Rods: No of Fuel Assemblies	Discharged Fuel Assemblies with Pressurized Fuel Rods: No of Fuel Rods	Discharged Fuel Assemblies with Nonpressurized Fuel Rods: No of Fuel Assemblies	Discharged Fuel Assemblies with Nonpressurized Fuel Rods: No of Fuel Rods
0 to 3,999	256	57,424	0	0	0	0
4,000 to 7,999	0	0	6	1,048	0	0
8,000 to 11,999	320	67,218	25	4,400	208	40,500
12,000 to 15,999	261	53,500	444	97,172	190	35,351
16,000 to 19,999	437	98,364	389	77,252	24	3,840
20,000 to 23,999	248	48,064	265	50,836	0	0
24,000 to 27,999	284	62,750	942	185,586	0	0
28,000 to 31,999	347	74,196	1,003	183,518	0	0
32,000 to 35,999	178	34,160	892	176,504	0	0
36,000 to 39,999	49	11,308	432	87,304	0	0
40,000 to 43,999	190	36,236	358	68,170	0	0
44,000 to 47,999	0	0	0	0	0	0
48,000 to 51,999	0	0	3	579	0	0
52,000 to 55,999	0	0	1	176	0	0
56,000 to 59,999	0	0	4	702	0	0
	2,570	543,220	4,764	933,247	422	79,691

Total Assemblies Supplied = 7,756

Total Fuel Rods Supplied = 1,556,158

4.3.2 Fuel Rod Integrity

The corrected coolant iodine-131 activities reported for each reactor cycle at plants operating with C-E fuel are listed in Table 13. The corrected activities were obtained using the INPO (Institute of Nuclear Power Operators) standard FRI method described in paragraph 4.0.

- In Figure 3[3] the significant decrease in the average corrected coolant activity for all plants with C-E fuel over the 1987-1989 period is plotted. The average plant activity at the end of 1989 was 0.0096 μCi/g, the median 0.0023 μCi/g. These values compare reasonably well with the industry norm, as reported by INPO for the U.S. PWR industry in 1989.[3]
- It is estimated that 75% of the leaking fuel that operated during 1989 was caused by debris-induced fretting wear of the Zircaloy-4 fuel rod cladding. Many of these leaking fuel rods were removed and replaced with non-fueled rods during refueling outages, using C-E fuel assembly reconstitution methods.[3]
- The overall reliability of C-E fuel at the end of 1989, excluding failures due to debris-induced wear, is estimated to exceed 99.997%.[3]

4.4 GENERAL ELECTRIC (GE)

A summary of the GE fuel performance and fuel rod integrity is given in Reference 4.

4.4.1 Fuel Performance - Fuel Utilization and Burnup

As of December 31, 1989 over 3.8 million GE8x8 fuel type production Zircaloy-clad UO_2 rods were in or had completed operation in commercial BWRs. The cumulative number of fuel rods in GE8x8 bundles loaded, as a function of calendar year, is presented in Figure 4.[4] As of December 31, 1989 over 1.54 million GE fuel rods were in operation. The GE core loading by fuel type is shown in Figure 5.[4] As of December 31, 1989 GE had loaded approximately 1.17 million PCI-resistant barrier fuel rods in commercial BWRs.

In 1989, eighteen domestic and eight overseas GE BWR plants containing GE fuel had refueling outages, resulting in over 3800 new GE 8x8 fuel bundles being loaded. Over 50% (or 12 reloads) of this new fuel was the latest GE production fuel design (GE8x8EB and GE8x8NB).

The experience of GE production and developmental BWR Zircaloy-clad UO_2 fuel rods through December 31, 1989 included successful commercial reactor operation of fuel bundles to greater than 45,000 MWd/MTU bundle average exposure and approximately 60,000 MWd/MTU peak pellet exposure.

TABLE 13. Corrected Iodine-131[a] in the Primary Coolant of Reactors Containing Combustion Engineering Fuel in 1989[3]

Reactor	Fuel Cycle	Beginning of Cycle	End of Cycle	Corrected Iodine-131 μCi/g[b]
Arkansas-2[c]	7	05-18-88	09-25-89	0.0190
	8	11-18-89	02-15-91[d]	0.0003
Calvert Cliffs-1	10[e]	07-01-88	11-15-91[d]	0.0130
Calvert Cliffs-2	8	06-13-87	03-17-89	0.0520
Fort Calhoun	12[e]	01-29-89	02-17-90	0.0016
Maine Yankee	11[e]	12-13-88	04-07-90	0.0002
Palo Verde-1	2	03-01-88	04-08-89	0.0300
Palo Verde-2	2	06-18-88	02-14-90	0.0100
Palo Verde-3	1	10-25-87	03-05-89	0.0067
Saint Lucie-2	4	11-22-87	02-01-89	0.0081
	5	04-25-89	10-01-90	0.0009
San Onofre-2	4	12-09-87	09-08-89	0.0145
	5	11-17-89	08-15-91[d]	0.0003
San Onofre-3	4	08-16-88	04-15-90	0.0023
Waterford-3	3	05-29-88	09-22-89	0.0028
	4	11-19-89	03-15-91[d]	0.0050
Yankee Rowe	20	01-15-89	06-23-90	0.0020

(a) Corrected for tramp uranium and normalized to the same cleanup rate using the standard INPO method.
(b) End-of-cycle or end-of-1989 values.
(c) Arkansas Nuclear One-Unit 2 (also known as ANO-2).
(d) Projected end-of-cycle date.
(e) Contains fuel from Combustion Engineering and another supplier.

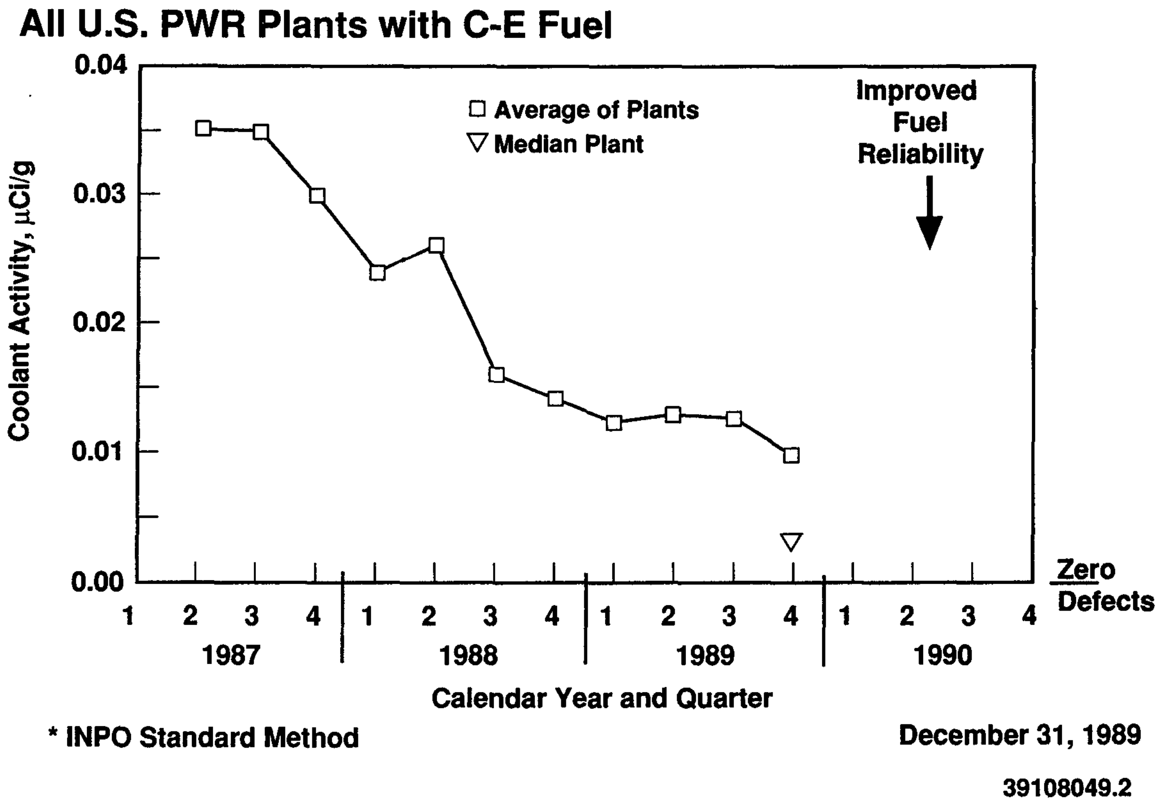

FIGURE 3. Corrected Coolant I-131 Activity* Versus Time[3]

4.4.2 Fuel Integrity

The GE8x8 fuel types have an overall fuel reliability rate from 1974 to the end of 1989 of greater than 99.98%.

4.5 WESTINGHOUSE ELECTRIC CORPORATION (W)

The summary of fuel performance and fuel rod integrity from Westinghouse for 1989 is found in Reference 5.

4.5.1 Fuel Performance - Fuel Utilization and Burnup

During 1989, 52 PWRs were refueled with Westinghouse fuel, and two plants started initial commercial power operation. A total of 73 commercial PWRs have used Westinghouse-supplied Zircaloy-clad fuel. These include 14x14, 15x15, and 17x17 fuel assemblies; 45 of these have operated with 17x17 assemblies through 1989.

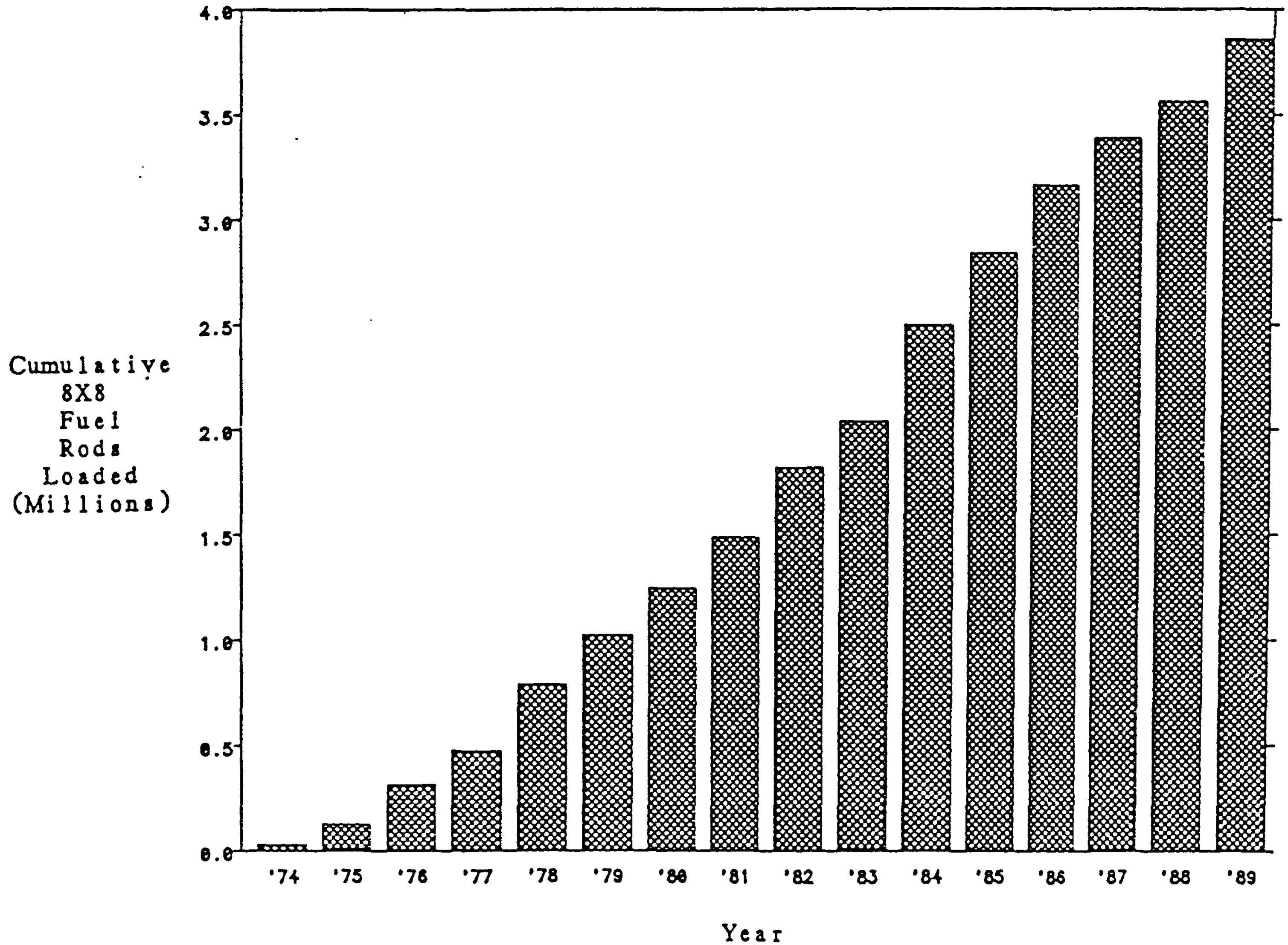

FIGURE 4. GE 8x8 BWR Fuel Rod Experience[4]

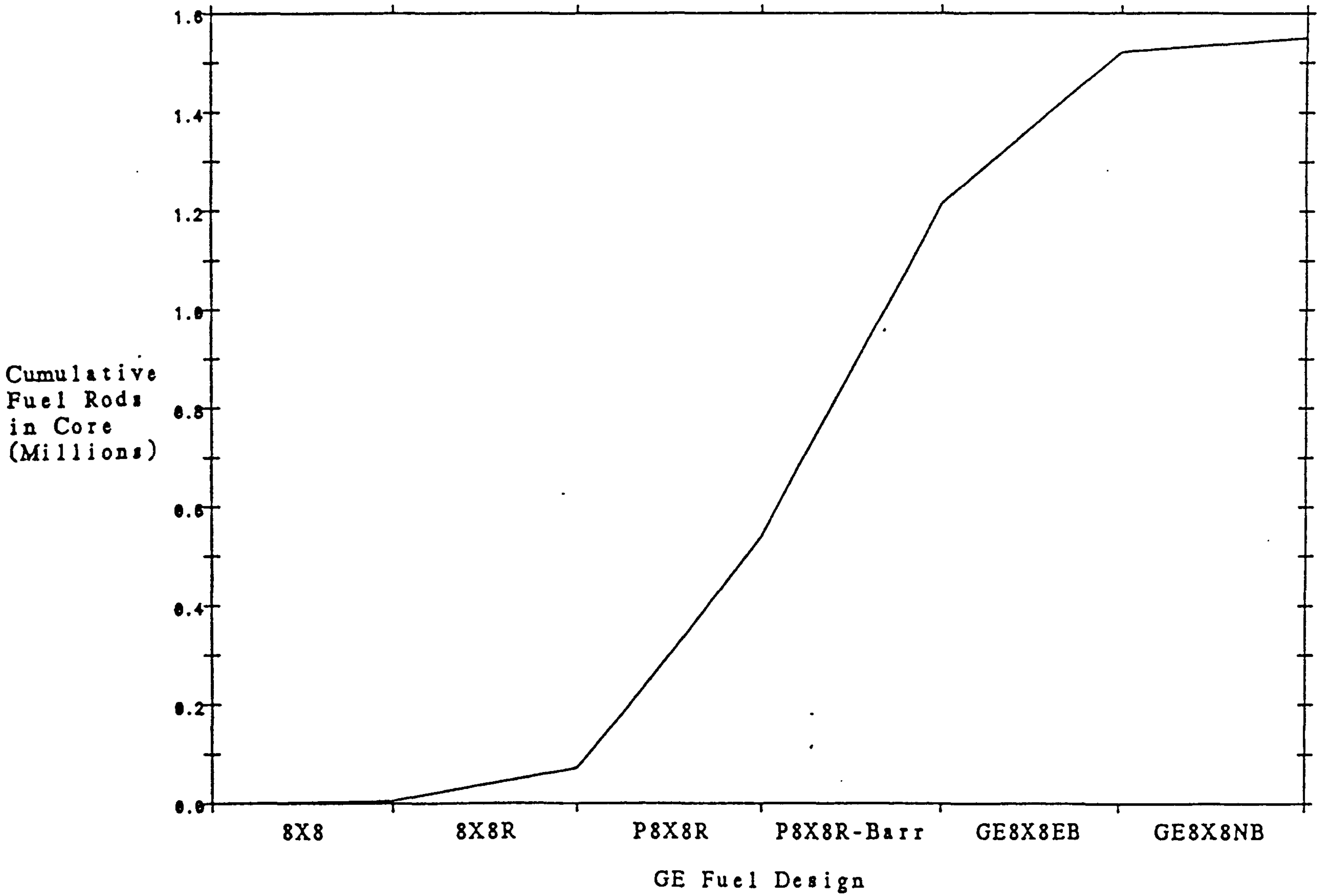

FIGURE 5. GE BWR Fuel Rods in Operation on 12/31/89[4]

At the end of 1989, a total of about 2.56 million W Zircaloy-clad fuel rods was in operation, representing 10,480 fuel assemblies. Including discharged fuel, the number of irradiated W Zircaloy-clad fuel rods totals 6.7 million, representing approximately 29,000 fuel assemblies.

The average burnup of all W discharged fuel is about 28 GWd/MTU, and the average burnup of all W fuel (in-core plus discharged) is about 25 GWd/MTU. A summary of burnup through the end of 1989 is presented in Table 14.[5] As shown, there are over 174,000 Westinghouse Zircaloy-clad rods (in-core and discharged) with assembly average burnups of 40 GWd/MTU or more. These represent 3365 fuel assemblies with assembly average burnups greater than 36 GWd/MTU, of which 772 had assembly average burnups greater than 40 GWd/MTU. A plant-by-plant status report showing peak region average burnup is given in Table 15.

4.5.2 Fuel Rod Integrity

The 1989 coolant activity level distribution for W-fueled plants, uncorrected and normalized, is shown in Table 16. In Figure 6 it can be seen that although the percentage of plants exhibiting uncorrected activity levels lower than 0.01 μCi/g peaked at 85% in 1987, it was still 80% in 1989, and there is an overall shift to lower activity levels over the years even as the number of plants increases; the number of plants exhibiting activity levels above 0.03 μCi/g decreased from 38% in 1982 to 2% in 1988 and 1989; and no plants using W fuel have exhibited Iodine-131 activity above 0.1 μCi/g since 1983.

During 1989 refueling outages, ultrasonic testing (UT) examinations were performed on ten reactors with the highest coolant activity to identify leaking rods. Thirty-two leaking fuel rods were found in nine reactors. Of the eleven rods examined to date, eight were leaking because of debris induced fretting, one because of a manufacturing-related problem, and for two there was no confirmed primary failure mechanism but tell-tale hydriding was evident.

Westinghouse reports no fuel failures due to corrosion or rod bow with any of its fuel, no failures due to primary hydriding since changes in manufacturing process and specifications were made in the early 1980s, and no evidence of fuel failure due to baffle jetting or cladding collapse during 1989.

4.5.3 Non-Fuel Core Components

- Rod Cluster Control Assemblies (RCCAs) - Over 2700 Westinghouse Full-length RCCAs are currently in service. Operational experience has determined that they are susceptible to fretting wear against upper internal guide cards while fully withdrawn and stationary, and to hairline cracks at the tips. In addition, the full-length hafnium absorber RCCAs exhibit localized hydriding in addition to

TABLE 14. Zircaloy-Clad Fuel Burnup Status Through 1989: Assemblywise Burnup Distribution of Westinghouse Zircaloy-Clad Fuel Rods[5]

A. ZIRC Active Rod Burnup Status as of 12/31/89--Combined Fuel Types:

Assemblywise Burnup, MWd/MTU	14 x 14 Rods	15 x 15 Rods	16 x 16 Rods	17 x 17 Rods	Totals Rods
0 - 3,999	7,160	28,560	9,400	143,088	188,208
4,000 - 7,999	19,260	50,592	3,760	158,928	232,540
8,000 - 11,999	24,976	16,320	9,165	284,592	335,053
12,000 - 15,999	14,272	16,932	34,075	271,128	336,407
16,000 - 19,999	15,859	49,164	3,760	284,064	352,847
20,000 - 23,999	27,804	42,432	3,760	165,528	239,524
24,000 - 27,999	25,931	36,108	8,460	194,568	265,067
28,000 - 31,999	22,458	37,128	7,520	189,552	256,658
32,000 - 35,999	24,030	32,436	3,055	164,736	224,257
36,000 - 39,999	17,005	11,628	1,880	73,392	103,905
40,000 - 43,999	4,296	5,916	0	14,520	24,732
44,000 - 47,999	0	0	0	3,168	3,168
48,000 - 51,999	0	0	0	0	0
52,000 - 55,999	0	0	0	0	0
56,000 - 59,999	0	0	0	0	0
TOTALS	203,051	327,216	84,835	1,947,264	2,562,366

B. ZIRC Discharged Rod Burnup Status as of 12/31/89 Combined Fuel Types:

Assemblywise Burnup, MWd/MTU	14 x 14 Rods	15 x 15 Rods	16 x 16 Rods	17 x 17 Rods	Totals Rods
0 - 3,999	0	0	0	0	0
4,000 - 7,999	4,293	0	0	4,488	8,781
8,000 - 11,999	23,249	6,528	6,815	54,384	90,976
12,000 - 15,999	18,067	50,388	9,635	172,920	251,010
16,000 - 19,999	61,755	92,616	6,110	364,584	525,065
20,000 - 23,999	53,321	109,344	7,755	163,152	333,572
24,000 - 27,999	128,480	123,216	20,680	328,439	600,815
28,000 - 31,999	135,616	282,336	23,735	395,736	837,423
32,000 - 35,999	215,597	238,884	20,210	409,396	884,087
36,000 - 39,999	75,144	151,572	8,695	265,975	501,386
40,000 - 43,999	20,406	36,516	1,880	66,000	124,802
44,000 - 47,999	2,148	13,056	0	3,696	18,900
48,000 - 51,999	0	0	0	264	264
52,000 - 55,999	0	816	0	528	1,344
56,000 - 59,999	0	0	0	1,056	1,056
TOTALS	738,076	1,105,272	105,515	2,230,618	4,179,481

TABLE 14. (contd)

C. ZIRC Total Rod Burnup Status as of 12/31/89 Combined Fuel Types:

Assemblywise Burnup, MWd/MTU	14 x 14 Rods	15 x 15 Rods	16 x 16 Rods	17 x 17 Rods	Totals Rods
0 - 3,999	7,160	28,560	9,400	143,088	188,208
4,000 - 7,999	23,553	50,592	3,760	163,416	241,321
8,000 - 11,999	48,225	22,848	15,980	338,976	426,029
12,000 - 15,999	32,339	67,320	43,710	444,048	587,417
16,000 - 19,999	77,614	141,780	9,870	648,648	877,912
20,000 - 23,999	81,125	151,776	11,515	328,680	573,096
24,000 - 27,999	154,411	149,324	29,140	523,007	865,882
28,000 - 31,999	158,074	319,464	31,255	585,288	1,094,081
32,000 - 35,999	239,627	271,320	23,265	574,132	1,108,344
36,000 - 39,999	92,149	163,200	10,575	339,367	605,291
40,000 - 43,999	24,702	42,432	1,880	80,520	149,534
44,000 - 47,999	2,148	13,056	0	6,864	22,068
48,000 - 51,999	0	0	0	264	264
52,000 - 55,999	0	816	0	528	1,344
56,000 - 59,999	0	0	0	1,056	1,056
TOTALS	941,127	1,432,488	190,350	4,177,882	6,741,847

design basis uniform hydriding. "Eddy current inspections of the hafnium RCCAs have shown that safe operation of the affected plants is not compromised, at least through the third 18-month cycle or a fourth annual cycle."

- Burnable Absorber Assemblies (BAs) - The Wet Annular Burnable Absorbers (WABAs) have been used routinely since 1983. No new incidents were reported with WABAs in 1989.

- Sources and Plugging Devices - No operational problems were reported with thimble plugging devices during 1989.

TABLE 15. Westinghouse Fuel Performance Status Report - 1989[5] (through December 31, 1989)

Reactor	Location	Owner	Date of First Electrical Power	Current Cycle No	Peak Region Average Burnup (MWd/MTU)
Jose de' Cabrera	Spain	Union Electrica S A	9/68	--	30,770
Beznau-1	Switzerland	Nordostschweizerische Kraftwerke AG	8/69	19	35,420
R E Ginna	U S A	Rochester Gas & Electric	12/69	19	39,960
Point Beach-1	U S A	Wisconsin Electric Power	12/70	17	41,010
Point Beach-2	U S A	Wisconsin Electric Power	8/72	16	40,650
Surry-1	U S A	Virginia Electric Power	8/72	10	35,260
Turkey Point-3	U S A	Florida Power & Light	11/72	11	34,280
Surry-2	U S A	Virginia Electric Power	2/73	10	35,920
Indian Point-2	U S A	Consolidated Edison	7/73	10	37,470
Turkey Point-4	U S A	Florida Power & Light	6/73	12	35,130
Zion-1	U S A	Commonwealth Edison	8/73	11	37,070
Prairie Island-1	U S A	Northern States Power	12/73	13	36,750
Zion-2	U S A	Commonwealth Edison	12/73	11	36,710
Kewaunee	U S A	Wisconsin Public Service	3/74	--	34,050
Prairie Island-2	U S A	Northern States Power	12/74	13	36,130
D C Cook-1	U S A	Indiana & Michigan Elec	2/75	11	36,800
Trojan	U S A	Portland General Electric	12/75	12	37,740
Millstone-2 (non-Westinghouse plant with Westinghouse fuel)	U S A	Northeast Utilities	11/75	--	33,450
Indian Point-3	U S A	Power Authority of the State of New York	5/76	7	36,510
Beaver Valley-1	U S A	Duquesne Light	5/76	7	33,470
Salem-1	U S A	Public Service Electric & Gas	12/76	9	34,880
KORI-1	Korea	Korea Electric Power	7/77	10	32,090
Farley-1	U S A	Alabama Power Company	8/77	10	36,870
D C Cook-2	U S A	Indiana & Michigan Elec	3/78	--	35,320
North Anna-1	U S A	Virginia Electric Power	4/78	8	36,970
North Anna-2	U S A	Virginia Electric Power	8/80	7	40,100
Sequoyah-1	U S A	Tennessee Valley Authority	10/80	4	35,480
Salem-2	U S A	Public Service Elec & Gas	5/81	5	36,350
Farley-2	U S A	Alabama Power Company	5/81	7	37,200
McGuire-1	U S A	Duke Power Company	9/81	6	38,290

TABLE 15. (contd)

Reactor	Location	Owner	Date of First Electrical Power	Current Cycle No.	Peak Region Average Burnup (MWd/MTU)
Krsko	Yugoslavia	Savske Elektrarne, Ljubljana and Elektroprivreda, Zagreb	10/81	8	32,090
Sequoyah-2	U.S.A.	Tennessee Valley Authority	12/81	4	35,340
V. C. Summer-1	U.S.A.	South Carolina Elec. & Gas	11/82	5	34,760
KORI-2	Korea	Korea Electric Power	4/83	6	33,050
McGuire-2	U.S.A.	Duke Power Company	5/83	6	36,110
Diablo Canyon-1	U.S.A.	Pacific Gas & Electric	11/84	4	39,320
Maanshan-1	Taiwan	Taiwan Power Company	5/84	4	31,020
Callaway-1	U.S.A.	Union Electric Company	10/84	4	38,180
KORI-3 (was KNU-5)	Korea	Korea Electric Power	1/85	4	34,820
Catawba-1	U.S.A.	Duke Power Company	1/85	4	35,220
Byron-1	U.S.A.	Commonwealth Edison	2/85	3	33,910
Maanshan-2	Taiwan	Taiwan Power Company	2/85	3	34,730
Wolf Creek	U.S.A.	Kansas Gas & Electric	6/85	4	33,950
Diablo Canyon-2	U.S.A.	Pacific Gas & Electric	10/85	3	29,530
Millstone-3	U.S.A.	Northeast Utilities	2/86	3	30,450
KORI-4 (was KNU-6)	Korea	Korea Electric Power	4/86	4	29,890
Catawba-2	U.S.A.	Duke Power Company	5/86	3	31,460
Younggwang-1 (was KNU-7)	Korea	Korea Electric Power	8/86	4	33,060
Younggwang-2 (was KNU-8)	Korea	Korea Electric Power	11/86	3	30,850
Shearon Harris	U.S.A.	Carolina Power/Light	1/87	3	26,290
Byron-2	U.S.A.	Commonwealth Edison	2/87	2	23,880
Vogtle-1	U.S.A.	Georgia Power	3/87	2	25,660
Braidwood-1	U.S.A.	Commonwealth Edison	7/87	2	19,020
Beaver Valley-2	U.S.A.	Duquesne Light	8/87	2	24,470
South Texas-1	U.S.A.	Houston Lighting	4/88	2	14,510
Braidwood-2	U.S.A.	Commonwealth Edison	5/88	1	--
Vogtle-2	U.S.A.	Georgia Power	4/89	1	9,570
South Texas-2	U.S.A.	Houston Lighting	5/89	1	5,100

TABLE 16. Summary of Coolant Activity in 1989[a] for Westinghouse Corporation Reactors[5]

Activity Range I-131[d] μCi/g	Uncorrected[b] I-131 No. of Plants	Uncorrected[b] I-131 Percentage in Range	Corrected[c] I-131 No. of Plants	Corrected[c] I-131 Percentage in Range
0.030 to 0.100	1	2%	1	2%
0.010 to 0.030	11	18%	7	12%
0.003 to 0.010	17	28%	10	16%
0.001 to 0.003	17	28%	13	21%
Below 0.001	15	24%	30	49%

(a) I-131 values are given as of the end of 1989 (December basis).
(b) Normalized Measured data.
(c) Normalized Measured data corrected for tramp uranium.
(d) All data have been normalized to 100% power and the same cleanup rate.

TABLE 17. Comparison of Coolant Activity[a] from 1982 to 1989 in Westinghouse Corporation Reactors[5]

Activity Range (I-131, μCi/g)	1989 No. of Plants	1989 % in Range	1982 No. of Plants	1982 % in Range
0.10 to 0.30	--	--	1	4
0.030 to 0.10	1	2	10	38
0.010 to 0.03	11	18	3	11
0.003 to 0.010	17	28	8	31
0.001 to 0.003	17	28	2	8
Below 0.001	15	24	2	8

(a) I-131 uncorrected values are for the end of each year (December basis). All data have been normalized to 100% power and the same clean-up rate. Uncorrected - Uses normalized measured data with no adjustments for tramp uranium.

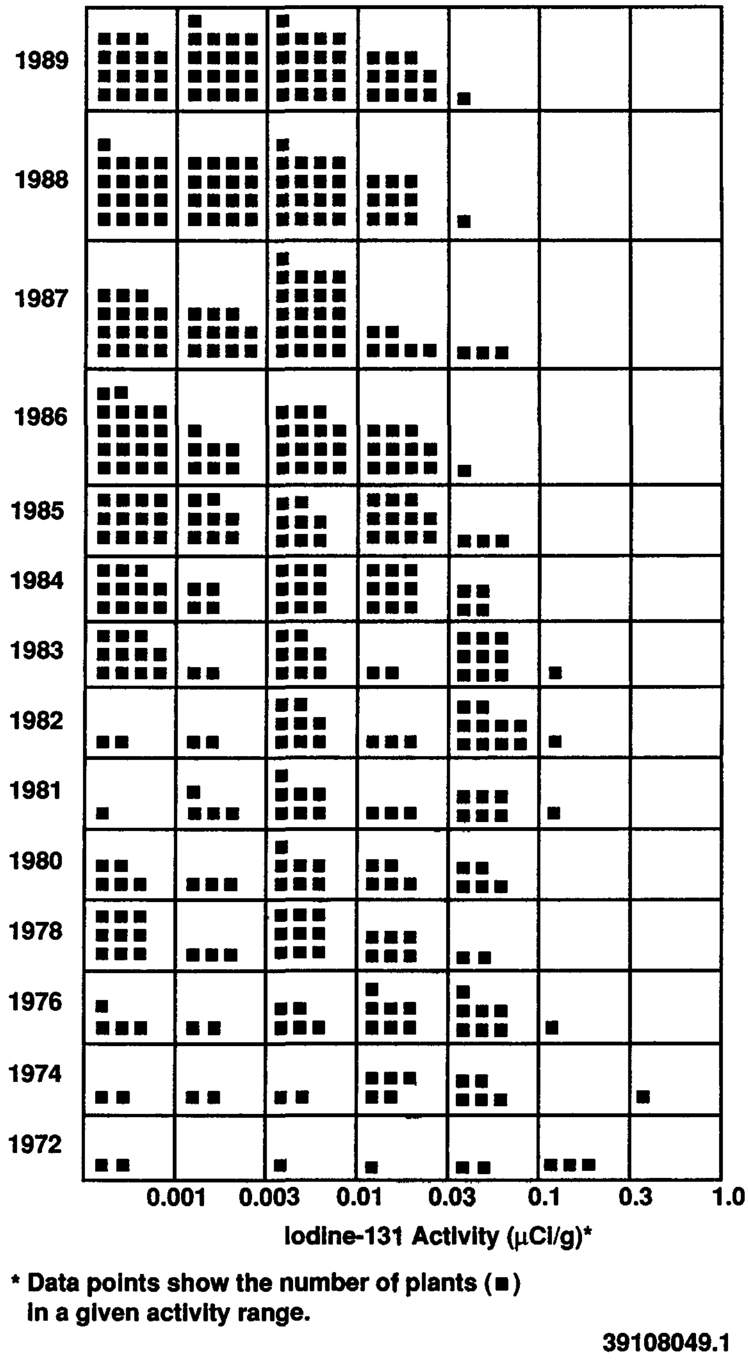

FIGURE 6. Uncorrected Reactor Coolant Activity Distributions for Westinghouse-Fueled Plants[45,113-117]

5.0 PROBLEMS OBSERVED DURING 1989

This section contains information on events or items that involve actual or potential fuel failure or damage or are of concern or interest to the fuel systems. Because of the length of the section and the amount of detailed information provided, part of Section 5.0 is placed in Appendix B. In general, fuel related events are placed in Section 5.0 and nonfuel related events are placed in Appendix B. There are a few exceptions to the events placed in Appendix B. To aid the reader, an index for Section 5.0 is provided here and in Appendix B.

INDEX

5.1 PROBLEMS IN 1989 THAT ARE SIMILAR TO THOSE IN 1988

5.1.1 Fuel Systems: Failures, Damage, or Potential for Damage

There were 23 events (or items of interest) concerning fuel systems that involved failures, damage, or the potential for damage. Of the 23, 9 (8 in 1989, 4 in 1988) involved U.S. plants and 12 (7 in 1989, 5 in 1988) involved plants in other countries. Fuel handling events are described in Sections 5.7 through 5.11. Events involving handling of irradiated fuel are listed in Table 18. Events involving handling of new (unirradiated) fuel are listed in Table 19. The events (or items of interest) pertaining to fuel failures, fuel damage, or the potential for fuel failure/damage are discussed below.

5.1.1.1 Oconee-2

A 1989 EPRI report[118] indicates that Duke Power Company "discovered that most leaking fuel rods at Oconee-2 occurred in two locations: directly adjacent to the center instrument tube and in or adjacent to corners. Arkansas Power and Light observed defects at the same locations at Arkansas-1."

5.1.1.2 Quad Cities-1

At Quad Cities-1 on September 21, 1989, during transfer of new fuel from the new fuel storage vault to the fuel pool, one fuel assembly (LYT191) was released from the refueling grapple and fell upon the spent fuel storage racks.[119] The cause of the event was a combination of personnel error and procedural deficiency. The grapple control switch was inadvertently left in the "release" position after attempting to unlatch. The unlatching was due to the adjacent fuel assembly not being fully seated.

5.1.1.3 San Onofre-2

Souther California Edison believes more than five fuel rods in a core have failed, they elect to conduct an inspection. Therefore, during the last few outages, fuel at San Onofre-2 have been inspected by ultrasonic testing.[120]

5.1.1.4 Sequoyah-1

The NRC is proposing to fine the Tennessee Valley Authority (TVA) for a significant failure to comply with NRC regulatory requirements at Sequoyah-1. During reactor trips on May 19 and 23 and June 6, 1988, the average temperature of the reactor coolant system dropped below the analyzed value. Had the fuel in the reactor core been approaching the end of its useful life, such a condition could have increased the probability and consequences of an accident. Information on this 1988 event was published in 1989.[121]

TABLE 18. Fuel Handling Events Involving Irradiated Fuel

Year	Plant	Fuel Assembly Fell Dropped, Bent, or Bumped	Fuel Rod Broken	Fuel Rod Damaged	Fuel Rod Dropped	Fuel Assembly Improperly Grappled	Fuel Assembly Inadvertently Lifted	Spacer Grids Damaged	Fuel Assembly Contacted By Another Fuel Assembly
1990	Bruce-4								X(l)
1989	Limerick-1				X				
	Limerick-2	(a)		(a)					
	North Anna-1 and -2(b)								X(b)
	Sequoyah-1 and -2	X(c)							
	St. Laurent				X(n)				
	Vogtle-1(k)	X(k)							
	Kruemmel (Germany)	X(m)							
1988	Palisades						X		
	Vogtle-1	X(d)							
1987	Oyster Creek			X(e)				X(e)	
	North Anna	(f)							
1986	Diablo Canyon-1	X							
	Haddam Neck	X(g)					X(g)		
1985	Cooper					X			
	Fitzpatrick						X		
	Summer-1							X	
1984									
1983	Beaver Valley-1	X(h)							
	Turkey Point-4	X,X(i)							
1982	Browns Ferry-2		X						
	Hatch-1		X						

TABLE 18. (contd)

Year	Plant	Fuel Assembly Fell Dropped, Bent, or Bumped	Fuel Rod Broken	Fuel Rod Damaged	Fuel Rod Dropped	Fuel Assembly Improperly Grappled	Fuel Assembly Inadvertently Lifted	Spacer Grids Damaged	Fuel Assembly Contacted By Another Fuel Assembly
1981	Cook-1	X							
	Millstone-1	X							
	Prairie Island-1	X							
1980									
1979									
1978	Dresden-1	X							

(a) Center stringer assembly containing seven startup neutron source pins was dropped 35 ft through water to cask pit floor. One source pin damaged (amounted to >$2K); pin was determined to be unacceptable for use.
(b) A fresh fuel assembly was set on top of an irradiated fuel assembly that was in a spent fuel storage rack in the spent fuel storage pool; visual inspection revealed no damage to either fuel assembly.
(c) One fuel assembly bent; however, examination later revealed no damage to the fuel assembly.
(d) Visual inspection revealed no apparent damage to the fuel assembly.
(e) Upper tie plate and the eight tie rods separated from the remainder of the previously damaged spent fuel bundle during movement in the spent fuel pool.
(f) Fuel assembly undamaged, but there was potential for damage (the fuel assembly was hooked to the crane but was still in the storage rack when the crane moved laterally).
(g) One fuel assembly was lifted and dropped.
(h) A rod cluster control assembly was also damaged.
(i) Two events occurred at this plant.
(j) Series of dropping events - no release of radioactivity.
(k) Spend fuel bundle halted over core, tilted.
(l) Fueling machine cylinder driven downward carrying a fuel channel.
(m) Assembly dropped from crane.
(n) Fuel element broke loose from handling system.

TABLE 19. Fuel Handling Events Involving Fresh (Unirradiated) Fuel

Year	Plant	Fuel Assembly Fell or Dropped	Fuel Assembly Damaged	Fuel Assembly Contacted By Another
1989	North Anna-1 and -2[(a)]			X[(b)]
	Quad Cities-1	X		
	Dungeness-B[(h)]	X[(h)]		
1988	Fitzpatrick	X		
	Washington Nuclear-2	X[(c)]	X[(d)]	
1987	Callaway	X[(e)]		
	Grand Gulf-1	X[(f)]	X[(g)]	
1986				
1985	Farley-2		X	
1984				
1983				
1982				
1981				
1980				
1979	Pilgrim	X		
1978				
1977	Arkansas-1		X	
1976				
1975	Dresden-1		X	
	Crystal River-3	X		
1974				
1973	Maine Yankee		X	
	Turkey Point-4	X		

(a) North Anna-1 and -2 share one spent fuel storage pool.
(b) A fresh fuel assembly was set on top of an irradiated fuel assembly that was in a spent fuel storage rack in the spent fuel storage pool; visual inspection revealed no damage to either fuel assembly.
(c) Two fuel assemblies fell from shipping container.
(d) Both fuel assemblies are to be shipped to vendor for repair (estimated cost: $100,000).
(e) Truck carrying 14 fuel assemblies overturned.
(f) Two fuel bundles fell about 2 to 2.5 ft to turbine deck.
(g) Both bundles damaged and are to be replaced.
(h) String of new fuel assemblies dropped onto top of reactor.

5.1.1.5 Surry-1

Following the shutdown of Surry-1 on September 14, 1988, an iodine spiking event occurred. The iodine spike was suspected to have been caused by fuel element defects in the reactor core. The fuel assemblies were to be inspected during the outage and fuel assemblies replaced as appropriate. Information on this 1988 event was published in 1989.[122]

5.1.1.6 Surry-1

In mid-September 1988, water chemistry indicated that one of the new fuel assemblies was leaking at Surry-1. According to the article,[123] the leak could not be found by ultrasonic testing (UT) because the leak was so new.[a] The plant had to resort to the time-consuming sipping process to find the leaking fuel assembly. Information on the 1988 event was published in 1989.[123]

5.1.1.7 BWR Channel Bowing

On September 29, 1989, the NRC issued an information notice cautioning BWR operators that failure to account for bowing of fuel channel boxes in computer core modeling can lead to fuel overheating and dryout.[124] The warning stems from the discovery in August 1988 of four dried out and damaged fuel rods in separate fuel assemblies at Sweden's Oskarshamn-2.[125]

5.1.1.8 Cycle Length and Its Effects on Fuel

About 90% of BWRs and over 80% of PWRs in the U.S. operate on nominal 18-month or 24-month refueling cycles; the remainder operate on 12-month cycles.[126] Long cycles modify the irradiation history experienced by the fuel assembly.[127] Remaining longer in the same location in the reactor core tends to increase fuel assembly bow, which can affect handling. Also, allowances for the effects of burnup gradients on rod growth may require adjustment. The consequences of fuel failure are amplified by longer cycles. On the average, there is a longer time until removal of the failed fuel and more time for propagation of the defect and associated effects.

5.1.1.9 Westinghouse Fuel

In a February 1989 article,[128] Westinghouse indicated they have 2.5 million fuel rods in operation and that a) they have "...somewhere between 80 and 96 leaking rods," b) "debris is our largest single actor right now..." as a cause for leaking fuel rods, and c) after examining a large number of failures over the years, there are about eight that they couldn't characterize.

(a) Another reason could be that there was insufficient water (typically need 0.5 gram or more) in the failed rod(s) for UT to detect it.

5.1.1.10 Argentina

It is indicated in a 1989 article[129] that at Argentina's Atucha-1 on August 22, 1988, three damaged fuel-bearing pressure tubes allowed some fuel pellets and other debris to fall into the vessel of the prototype pressurized heavy water reactor (PHWR). Extracting the fuel pellets and metal shards (many tiny in size) from the radioactive vessel has been a logistical and engineering nightmare. Scores of other pressure tubes are being inspected by the Comision Nacional de Energia Atomica (CNEA), for possible damage from metal fatigue or corrosion, which could be possible causes of the mishap.

5.1.1.11 Belgium

Out of nearly 41,000 Belgonucleaire mixed oxide (UO_2-PuO_2) fuel rods irradiated to date, no failures have occurred since 1980.[130] Of the 41,000 rods, over 38,000 were made by the MIMAS process. The few failures experienced before 1980 were due to causes that, at the time, also affected commercial UO_2 fuel.

5.1.1.12 France

A French paper[131] presented at a symposium in August 1989 includes a discussion of degradation problems noted in their PWRs. Rubbing has been encountered on the surface of Zircaloy-4 fuel cladding (due to cross flow through baffle joints, loose parts, grid contact or manufacturing defects), on AISI 304 stainless steel control rod tubing (due to hydraulic disturbance and contact with guide tubes), and on AISI 316 stainless steel in-core instrumentation tubes (due to movement against guide tubes). There have been numerous failures in service of Alloy X750 guide tube pins due to stress corrosion cracking (a phenomenon well documented in the literature). Defects have also been encountered in core baffle assembly fixing bolts, which have been attributed to a variety of possible mechanisms (fatigue, corrosion, creep, etc.) combined with severe radiation conditions.

In another paper (by P. L. Andersen et al.) at the same symposium, it was indicated that irradiation-assisted stress corrosion cracking, a problem observed in the early 1960s in fuel elements and associated with high cladding stresses due to fuel swelling, has been detected more recently in lower-stress components such as instrument dry tubes and control blade handles and sheaths.

5.1.1.13 France

It is stated in a 1989 article[132] that Electricite de France (EDF) has indicated that over half of the fuel failures in their system so far are due to events external to the fuel itself.

5.1.1.14 India

A recent article[133] provides information on atomic power stations in India. The Tarapur station had as many as 100 fuel bundles develop pinholes in both BWR units in the first cycle. One of the incidents at the Madras station involved a pair of fuel bundles getting stuck in the pressurized heavy water reactor's (PHWR's) fuel transfer port.

5.1.1.15 Sweden

It is indicated in a 1989 article[134] that sipping (i.e., leak testing) at a Swedish BWR in August 1988 identified failure of four, first-cycle, 64-rod, water-cross (SVEA) fuel assemblies in a mixed 8x8 and SVEA core. The cause of the failures was dryout. The principal reason for the fuel rod overpower was, in each case, the combined influence of excessive bow of channels of two high burnup 8x8 fuel bundles adjacent to the failed assembly. Large-bow channels were being recycled and had exposures of 54,000 to 68,000 MWd/MTU; bow measurements showed a marked acceleration of bow in the range of 40,000 to 50,000 MWd/MTU. ABB Atom indicates that great caution must be used in re-using BWR fuel channels. The fuel failure mechanism is not inherently related to the SVEA design.

5.1.1.16 Sweden

Additional information on the channel box bowing problem noted at Sweden's Oskarshamn-2 (BWR) in August 1988 is provided in a 1989 article.[135]

5.1.1.17 Sweden

It is stated in a 1989 article[136] that Sweden's Oskarshamn-3 was shut down for two weeks in December 1988 because of a fuel failure. The failed fuel assembly was found by sipping; the activity was coming from one "severely damaged" rod. The failure is unusual because they "have had very good experience" otherwise with ABB-supplied fuel.

5.1.1.18 Sweden

An article[137] published in February 1989 indicates that fuel failures have occurred during the past five years in seven Swedish plants (four BWRs and three PWRs). The fuel rod failure rates are about 0.003% per year. The fuel rod failures are listed on the next page.

Cause of Failure	Number of Failed Rods During 1982-1987 BWR	PWR
Fabrication defect	1	1
Debris-induced fretting	3	4
Pellet-cladding interaction	11	0
Spacer grid/rod wear	0	6
Baffle jetting	--	3
Not determined	2	5
Total per plant per year	1.0	1.3

5.1.1.19 USSR

A recent article[138] indicates that fuel failure has been a recurrent Soviet problem.

5.1.1.20 Vermont Yankee

The end of a fuel rod broke into the 18-year-old BWR, scattering microscopic particles of fuel in the reactor cooling system. The particles were so small, in such relatively small quantity, and so finely dispersed that they did not trigger the alarm or cause any real health effects, but the employees have taken issue on safety grounds.[139]

5.1.1.21 Connecticut Yankee

Connecticut Yankee, a Westinghouse 616-MW PWR, has 343 fuel rods in 88 fuel assemblies with through-wall cracks. The cause is metal flakes left over from thermal shield maintenance during the last outage. The outer two rows of the assemblies contain 75% of the damaged fuel pins.[140]

5.1.1.22 France

St. Laurent, a magnox reactor, scrammed on November 2, 1988 because of a leaking fuel rod, one of 45,000 in the graphite core.[141]

5.1.1.23 USSR

A LOCA and partial core melt may have taken place on a Soviet submarine which caught fire off the Norwegian coast on June 26, 1989, judging from radioactive water samples and steam seen escaping from the vessel. Another accident is believed to have taken place on July 16, the third in four months. Details were not forthcoming from the Soviets.[142]

5.1.2 Issues/Concerns with Generic Implications

There were eight events or items of interest (six in 1989, two in 1988) in the U.S. and four events or items of interest (one in 1989, three in 1988) in foreign countries that involved issues or concerns with generic implications. Those events or items of interest are described below.

5.1.2.1 Brunswick-2

Brunswick-2 lost all off-site power (cause: repair crew error) for 10 hours on June 17, 1989.[143] The operators had to trip the unit manually, to avoid possible core power oscillations, even though they knew the trip meant loss of off-site power to emergency systems, because NRC changed BWR operating procedures late last year to require a trip in the circumstances Brunswick was in. NRC considers the Brunswick incident to have safety significance because the unit was dependent on its emergency diesel generators to power emergency buses for 10 hours, precursor to a station blackout and an identified main contributor to the risk of core melt.

5.1.2.2 South Texas-1

Information published in 1989 indicates that Westinghouse notified South Texas-1 on September 1, 1988, of the existence of a flow anomaly similar to that identified in other Westinghouse four-loop plants.[144] The flow anomaly is a thermal-hydraulic instability in the reactor vessel that results in a slight decrease in coolant flow to certain areas of the reactor core. The departure from nucleate boiling (DNB) penalty resulting from the anomaly exceeds the available generic margin. At that time, Westinghouse recommended maintaining reactor coolant system flow above 400,000 gpm until further analysis can be completed.

An updated report was issued by the licensee in December 1988 and information from that report was published by the NRC in 1989.[145] Recently completed safety analyses for South Texas-1 support operation at the Technical Specification reactor coolant system flow rate of 395,000 gpm. Implementation of the new safety analyses requires a revision to the design basis as discussed in the Final Safety Analysis Report (FSAR) and Technical Specifications. Until the changes are approved by the NRC, the utility will maintain a coolant flow at or above 400,000 gpm when operating at 100% power.

5.1.2.3 Wolf Creek-1

A 1989 article[146] indicated that the potential for swelling in the cladding of thin hafnium control "rodlets," first noticed in 1988 at Wolf Creek,[147] was a "full-fledged safety concern." The concern centered on the possibility that a swollen control rod could slow drop time or occlude the inner diameter of its guide tube.[148] Subsequent safety analysis showed that the maximum increase in individual rod drop times in the worst-case scenario would be less than one-tenth of a second.[146] Of the fourteen U.S. PWRs (11 operating units and 3 under construction) equipped with hafnium control rods, several are replacing the hafnium control rods with silver-indium-cadmium, but others have not experienced swelling and will keep them under scrutiny. Westinghouse will now make these rods only by special request and considers the matter now to be a minor licensing issue. "Swelling is thought to occur when hydrogen, under PWR operating pressures, diffuses through the stainless steel cladding and pierces the protective film on the hafnium.[149] The subsequent hydriding of the hafnium causes expansion" (20 to 25 mils were measured at Wolf Creek-1). See Section 5.1.17.10 TAIWAN. One control rod could not be fully reinserted into Taiwan's Maanshan-1 in September 1988--a broken tip on one of the rods was found and the hydriding phenomenon was identified as the cause of the failure.[150]

5.1.2.4 BWR Channel Bowing

See Section 5.1.1.7 for details on this 1988/1989 event and item of interest.

5.1.2.5 BWR Thermal-Hydraulic Stability

A 1989 paper[151] by NRC provides a status report on the regulatory review of BWR thermal-hydraulic stability. It is expected that implementation of BWR owner's group recommendations, possibly involving hardware modifications, for long-term resolution of the stability issue will commence in 1990.

5.1.2.6 PWR Primary Water Stress Corrosion Cracking

"Mounting evidence of primary water stress corrosion cracking (PWSCC) in Inconel-600 primary system pressure boundary penetrations" (includes in-core instrumentation and control rod drive penetrations) "has caught the attention of U.S. regulators and prompted industry fears that PWR owners could be facing a generic problem equivalent to the large-diameter pipe cracking that plagued U.S. BWRs in the early 1980s.[152] The possibility of circumferential cracking and sudden failure of Inconel-600 penetrations raises safety concerns because the resulting leaks from the primary system would be unisolable."

5.1.2.7 Cycle Length and Its Effects on Fuel

See Section 5.1.1.8 for details on this 1989 item of interest.

5.1.2.8 Generic Issue 82: Beyond Design Basis Accident in Spent Fuel Pools

It is concluded by the NRC in a report, NUREG-1353,[153] that no new regulatory requirements are warranted concerning the use of high-density spent fuel storage racks. The conditional probability of a Zircaloy cladding fire in the event of a complete loss of water was found to be 0.25 for BWRs and 1.0 or PWRs. The value/impact and cost-benefit evaluations for the proposed alternatives for Generic Issue 82 do not indicate that cost-effective options are available to mitigate the risk of beyond design basis accidents in spent fuel pools.

5.1.2.9 France

"In France on April 1, 1989, a control rod failed to drop into the core of Gravelines-4 (PWR)."[154] "Video inspection revealed that the control rod had broken off and fallen to the bottom of a fuel assembly and its spring was stuck within the guide tube, causing the control rod cluster to stick at an intermediate position." "Electricite de France (EDF) had previously thought that a broken rod could not prevent control rod cluster drop--an assumption disproved at Gravelines." "Analysis showed that the local wear on the control rod casing was far more severe than had been predicted by studies." "The French nuclear regulatory agency (SCSIN) said the incident showed that "everything must be done to inspect the control rod clusters and have no more clusters break," but also that "we have to reconsider the criteria" for control rod wear." "The more severe criteria for control rod wear dictate replacing about 30 of the 53 clusters on each reactor." "The control rod cluster problem is projected to cost at least 100-million francs ($15-million U.S. at current rates) this year."[154]

5.1.2.10 Sweden

See Section 5.1.1.15 for details on this 1988 event.

5.1.2.11 Sweden

See Section 5.1.1.16 for details on this 1988 item of interest.

5.1.3 Thinning of In-Core Instrumentation Tubes

In 1989, there were five events or items of interest involving the thinning of in-core instrumentation tubes. Of the five, three involved U.S. plants and two involved plants in other countries. These events or items of interest are discussed below.

5.1.3.1 Beaver Valley-1

On October 10, 1989, at Beaver Valley-1, eddy current testing identified nine in-core instrumentation guide thimble tubes with degradation in excess of specified limits (based on ASME Code allowable stress limits, tube degradation of up to 60% is acceptable).[155] An analysis also projected degradation in excess of 60% of the wall thickness for an additional nine tubes by the end of the next fuel cycle (these nine tubes were repositioned to prevent unacceptable wall thinning). The degradation is apparently due to mechanical wear of the thimbles against reactor vessel internals induced by the coolant flow characteristics through the vessel. Failure of a guide thimble tube is bounded by the analysis in Section 14.3, "Loss of Coolant Accident," in the plant's updated Final Safety Analysis Report.

5.1.3.2 Diablo Canyon-1

Eddy current inspection of the in-core neutron monitoring system thimble tubes was performed at Diablo Canyon-1 on October 20, 1989.[156] It was determined that the degradation in 28 thimble tubes exceeds 50% of the wall thickness. During the current refueling outage, 33 tubes will be replaced and 12 tubes will be repositioned. Thirteen other tubes showed degradation of less than 35% of the wall thickness.

5.1.3.3 South Texas-1

One of the tasks scheduled to be performed during the refueling outage that is to start on August 4, 1989, at South Texas-1 is to install thicker-walled in-core instrumentation thimble tubes.[157]

5.1.3.4 Belgium

A January 1989 article[158] includes a discussion of the wear on in-core instrumentation guide tube thimbles and the striking variations in the wear exhibited by three groups of Belgian plants: Tihange-1; Doel-4 and Tihange-3; and Doel-3 and Tihange-2. Those results emphasize the importance of subtle design differences. It was also noted in the article that it was not widely known that through-wall cracking of three in-core thimbles occurred at a U.S. plant (Salem-1) in March 1981.

5.1.3.5 France

See Section 5.1.1.12 for details on this 1989 item of interest.

5.1.4 Crud-Induced Localized Corrosion (CILC)

There were three events or items of interest regarding crud-induced localized corrosion (CILC) that involved U.S. plants. Items of interest were published in 1989 and 1988; events occurred starting as early as 1980. The events and items of interest are described below.

5.1.4.1 Hatch-1 and -2

Hatch-1 and -2 were designed with Admiralty brass condenser tubes and filter-demineralized cleanup systems; hence, they both have been susceptible to CILC failures because the condenser tubes contain a small amount of copper.[159,160] Since 1980, about 108 fuel assemblies experienced cladding failures during 13 of the 15 cycles during the decade; most of the failures were caused by CILC.[159] There have been three barrier rod failures (these were the result of manufacturing defects).[159] During the past year, the utility has had the tubing in the main condensers replaced with titanium tubing; however, it will take some time--perhaps over a year--for the reactor water copper levels to decrease far enough to make CILC failures very unlikely.[159,160]

5.1.4.2 Limerick-1

Articles published in 1989 and 1988 provide additional information on the indications of damage to fuel cladding by crud-induced localized corrosion (CILC) that were discovered at Limerick-1 during a refueling outage that ended in August 1987. Pinhole-sized leaks in as many as 30 fuel rods have forced the utility to operate at reduced power (lowered to 57%) during the last four months of its operating cycle.[161,162] A high copper concentration in the feedwater, caused by poor water chemistry, was originally targeted as the cause of the problem but the utility is now rethinking that scenario.[163] Fuel loaded during that outage was specially heat treated, but that treatment failed to do the trick, and the utility has found more cladding failures in various fuel bundles, including some in the specially treated fuel. What is puzzling to the utility and the fuel manufacturer (General Electric Company) is why CILC did not affect some of the older fuel rods but did attack new rods, some of which were heat-treated and some of which were not.

5.1.4.3 Nine Mile Point-2

Niagara Mohawk Power Corp.'s Nine Mile Point-2 reactor remained down, after a forced outage for other matters, in order to correct a circulation water pH/copper dissolution problem.[164]

5.1.5 Flow-Induced Fretting

There was one item of interest in 1989 involving flow-induced fretting at U.S. plants. That item is described below.

5.1.5.1 Braidwood-1 and -2

Westinghouse Electric Corporation is to inspect the control rods at Braidwood-1 and -2 for wear or fretting caused by flow-induced vibrations.[165]

5.1.6 Iodine Spiking

There were two iodine spiking events, one in 1989 and one in 1988, at U.S. plants. Those events are described below and are included in Table 20 and Table 21.

5.1.6.1 Limerick-1

On January 11, 1989, an iodine spike occurred at Limerick-1.[166] The reactor coolant dose equivalent iodine-131 specific activity exceeded the Technical Specification limit of 0.2 microcuries/gram and remained above that level for 14 hours and 30 minutes.

5.1.6.2 Surry-1

See Section 5.1.1.5 for details on this 1988 event.

TABLE 20. Iodine Spiking or Radioactive Gas Release Events

Reactor	Type		Supplier (NSSS)[a]					Iodine Spiking or Radioactive Gas Release (No. of Events)									
	BWR	PWR	AC	B&W	C-E	GE	W	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989
Arkansas-1		•		•					X								
Arkansas-2		•			•			X	X								
Big Rock Point	•					•						X[b]					
Brunswick-2	•					•		X	X	X							
Calvert Cliffs-1		•			•					X							
Calvert Cliffs-2		•			•						X(9)						
Catawba-1		•					•						X(1)				
Cook-1		•					•						X(2)				
Cook-2		•					•		X	X							
Crystal River-3		•		•				X	X					X(1)			
Davis Besse-1		•		•				X	X	X	X(5)						
Farley-1		•					•			X							
Ft. Calhoun-1		•			•							X(1)					
Ginna		•					•				X(1)						
Hatch-2	•					•				X				(c)			
La Crosse	•		•						X								
Limerick-1	•					•											X(1)
Millstone-2		•			•					X(3)	X(2)						
North Anna-1		•					•		X	X(4)	X(7)				X(1)		
Palisades		•			•				X	X(2)	X(1)						
Prairie Island-1		•					•			X							
Prairie Island-2		•					•				X(3)						
San Onofre-2														X(1)			
San Onofre-3		•			•						X(3)	X(10)	X(5)	X(1)			
St. Lucie-1		•			•			X	X	X(6)	X(2)						
Surry-1		•					•		X(7)	X(13)	X(3)	X(4)	X(3)	X(4)	X(1)	X(1)	
Surry-2		•					•			X							
Trojan		•					•		X								
Yankee Rowe		•					•		X					X(1)			
No. of Reactors								**5**	**13**	**13**	**10**	**4**	**4**	**5**	**2**	**1**	**1**

(a) Allis-Chalmers (AC), Babcock & Wilcox (B&W), Combustion Engineering (C-E), General Electric (GE), and Westinghouse (W).
(b) Plant output voluntarily restricted because of high off-gas release rate (however, it is only about 5% of Technical Specification limit).
(c) Plant has been run at 85% power since January 1986 to decrease off-gas activity caused by leaking fuel.[167,168]

TABLE 21. Iodine Spiking or Radioactive Gas Release Events at Domestic Plants

Year	Number of Plants	Total Number of Events
1989	1	1
1988	1	1
1987	2	2
1986	5	$\geq$8
1985	4	11
1984	4	16
1983	10	36
1982	13	$\geq$36
1981	13	$\geq$19
1980	5	$\geq$5

5.1.7 Fuel Handling: Fuel Dropped/Broken/Damaged or Potential for Damage Existed

There were five fuel handling events (two in 1989, two in 1988) at U.S. plants and one in 1989 in France in which fuel was dropped or the potential for damage existed. Those events are described below.

5.1.7.1 Limerick-1

On February 28, 1989, a fuel rod was dropped at Limerick-1 during fuel reconstitution activities.(169) There was no release of radioactivity. The fuel rod was retrieved, inspected, found to be undamaged, and returned to its proper location. Dropping of the fuel rod was due to inexperience and fatigue of the fuel handler.

5.1.7.2 North Anna-1 and -2

An event occurred on April 27, 1989, at the spent fuel storage pool that is shared by North Anna-1 and -2.(170) The licensee's operators attempted to insert a new (i.e., nonirradiated) fuel assembly into a spent fuel storage rack position already containing a spent fuel assembly. Both fuel assemblies were visually inspected; no damage was noted.

5.1.7.3 Palisades

As the upper guide structure was being removed at Palisades on September 3, 1988, it was observed that a fuel bundle had been removed from the core and was hanging from the upper guide structure. The fuel bundle (K-28) was separated from the structure and set in a restrained configuration atop the core. The cause of the event was attributed to the bundle adhering to the bundle guide pins on the upper guide structure. A gauge obtained from the fuel vendor was used to verify tie plate locating hole center-to-center spacing and inner diameter bore. No significant deviations were noted. Inspection of the guide pins indicated that no bending had occurred, nor was any physical damage induced. Root cause for the event is indeterminate. Information on this 1988 event was published in 1989.[171,172]

5.1.7.4 Vogtle-1

In 1989, the NRC published information on an event that involved a power supply disturbance on October 20, 1988, at Vogtle-1 that led to a computer memory loss in the refueling machine.[173] The refueling machine halted with spent fuel bundle 5C42 suspended directly over its previous core location. The bundle was manually lowered and core alterations were temporarily stopped. When core alterations were resumed, fuel bundle 5C42 was unlatched in order to withdraw the refueling machine mast. However, the fuel bundle was not fully inserted and was apparently resting on its guide pins. When unlatched, the fuel bundle leaned sideways and came to rest against the core baffle. The fuel bundle was removed from the core and transferred to the fuel handling building. Visual examinations revealed no apparent damage to the fuel bundle. Full insertion of fuel bundles is confirmed by the computer circuitry while the refueling machine is under computer control. However, less precise methods are employed during manual operation. Specific measures to enhance full insertion confirmation of fuel bundles during manual operations are being evaluated and are expected to be implemented by February 1, 1989.

5.1.7.5 France

Several events occurred in rapid succession on September 1, 1989 at St. Laurent-AI, a gas-cooled reactor. A fuel element broke loose from the handling machine and fell seven meters into its original position, with no apparent consequences. While attempting to inspect the channel containing the fallen fuel element, two more fuel elements were dropped, one from five meters and the other from one meter, also with no apparent consequences. Finally, normal ventilation of the fuel handling machine was interrupted by a separate accident, and for 40 minutes an emergency ventilation system was used to ensure cooling of the fuel elements in the fuel handling machine.[174]

5.1.8 Fuel Handling: Fuel in Incorrect Position

There were two fuel handling events in 1989 at U.S. plants that involved placing of a fuel assembly in an incorrect position. These events are described below.

5.1.8.1 North Anna-1 and -2

See Section 5.1.7.2 for details of this 1989 event.

5.1.8.2 Quad Cities-1

In October 1989, a refueling crew inserted a BWR fuel assembly in the wrong location in the core. The foreman then attempted to cover up the error by ordering two unauthorized movements of fuel assemblies. Also, communication with the control room was not maintained, which is another failure to follow procedures. The foreman's license was suspended.[175]

5.1.9 Fuel Handling: Crane Operation

There were two fuel handling events (one in 1989, one in 1988) involving crane operation at U.S. plants. Those events are described below.

5.1.9.1 Calvert Cliffs-1

On December 30, 1988, it was discovered that one of the administrative controls at Calvert Cliffs-1 was not being properly maintained and that a heavy load (the spent fuel cask crane load block) had been moved over the spent fuel pool. Information on this 1988 event was published in 1989.[176]

5.1.9.2 Oyster Creek

On January 14-15, 1989, a fuel pool gate (FPG) may have been moved over irradiated fuel stored in the fuel pool at Oyster Creek.[177] The Technical Specifications state that no object in excess of the weight of one fuel assembly may be moved over stored irradiated fuel. Although no one could state conclusively that this did occur, an observation and an analysis indicate that this may have occurred. The cause of the incident was personnel error. This was a voluntary report.

5.1.10 Fuel Handling: Procedural Violations

There were seven fuel handling events (two in 1989, five in 1988) at U.S. plants that involved procedural violations. Those events are described below.

5.1.10.1 Arkansas-2

On February 29, 1988, at Arkansas-2, NRC inspectors observed housekeeping discrepancies on the fuel handling bridge during fuel handling activities over the reactor vessel. The discrepancies included loose tools, loose debris, and excessive dirt on the floor of the fuel handling bridge. Information on this 1988 event was published in 1989.[178]

5.1.10.2 Browns Ferry-2

On January 5, 1988, with 74 fuel assemblies loaded, fuel loading at Browns Ferry-2 was halted by plant management to evaluate NRC concerns with the reload procedures. NRC was concerned that fuel loading was being performed without adequate neutron monitoring due to inadequate safety review of Technical Specification amendments. Information on this 1988 event was published in 1989.[179]

5.1.10.3 Brunswick-1

An NRC resident inspector discovered that the standby gas treatment system, which would be used to process radioactive gas in the event of a mishap, was inoperable during the period December 11-14, 1988, at which time irradiated fuel assemblies were being moved inside the secondary containment building at Brunswick-1. Information on this 1988 event was published in 1989.[180]

5.1.10.4 Byron-1

The NRC cited Byron-1 for violation of NRC safety regulations on October 12, 1988, during lowering of the water level in the refueling area for maintenance. Water was removed by a pump from the reactor vessel faster than it was draining into the vessel from the refueling area. The water level in the reactor remained well above the top of the fuel and adequate cooling capability was available. Since the suction point of the pump is above the top of the fuel, drawing water at the point could not result in the fuel being uncovered. The incident occurred because plant personnel relief on visual observation of the water level in the refueling area, which proved to be misleading, and on a temporary water level device that did not accurately indicate the level. Information on this 1988 event was published in 1989.[181]

5.1.10.5 Clinton-1

On January 22, 1989, it was determined that reactor core alterations were being performed at Clinton-1 in one quadrant of the core without an operable source range monitor (SRM) in the adjacent quadrant, which is a violation of Technical Specification 3.9.2[182] The cause of this event is attributed to utility licensed operator error.

5.1.10.6 Harris-1

Plant personnel at Harris-1 were in the process of transferring spent fuel from the shipping cask to the spent fuel storage pool on August 27, 1989, when it was discovered that the fuel building operating floor equipment hatch was in the storage location on the operating deck and not installed as required.[183] Fuel movement was immediately stopped; fuel movement resumed after the hatch was installed. The event was caused by procedural inadequacies.

5.1.10.7 Point Beach-1

At Point Beach-1 on April 20, 1988, a warning device (flashing red light) was not in use during a fuel assembly transfer, which is a violation of the plant's procedure (HP-3). The radiation field at the containment wall exceeded 1000 mrem/hr. During the time the red light was inoperative, fewer than 10 irradiated fuel assemblies were sent through the fuel transfer system. When fuel is moved through the transfer canal, radiation emanates from a gap between the containment wall and the containment floor and transient dose rates of up to 6,000 mrem/hr are observed for approximately 10 seconds. Information on this 1988 event was published in 1989.(184)

5.1.11 Fuel Handling: Other Events

There were two other fuel handling events (one in 1989, one in 1988) at U.S. plants and one item of interest on fuel handling at foreign plants. Those events and the item of interest are described below.

5.1.11.1 Davis Besse-1

Fuel loading at Davis Besse-1 was delayed when loose parts were discovered in the reactor vessel on July 2, 1988. The debris consisted of two pieces of high pressure injection/make-up nozzle thermal sleeve, an apparent paint chip, and a rag. Information on this 1988 event was published in 1989.(185)

5.1.11.2 San Onofre-2

On September 20, 1989, core alterations (removal of in-core nuclear instruments) were performed at San Onofre-2 without complete containment closure.(186) The event was attributed to deficient administrative controls.

5.1.11.3 USSR

The Soviet Union plans to operate a closed fuel cycle; hence, spent fuel has to be transported to reprocessing plants.(187) Thus, transport of spent fuel from VVER-1000 reactors is one of the main challenges facing the Soviet nuclear power industry. According to the safety regulations at VVER-1000s, it is only possible to unload spent fuel from on-site storage when the reactor is shut down.

5.1.12 Debris in Reactor Vessel/Potential For It

There were three events in 1988 at U.S. plants and one item of interest pertaining to plants in the U.S. and other countries that involve debris in the reactor vessel or the potential for it. These events and the item of interest are described below.

5.1.12.1 Davis Besse-1

Fuel loading at Davis Besse-1 was delayed when loose parts were discovered in the reactor vessel on July 2, 1988. The debris consisted of two pieces of high pressure injection/make-up nozzle thermal sleeve, an apparent paint chip, and a rag. Information on this 1988 event was published in 1989.[185]

5.1.12.2 Diablo Canyon-1

Diablo Canyon-1 responded to items of violation cited by the NRC. One of the violations involved the reactor vessel head cable tray area--loose tools were found on April 14, 1988, that were not entered on the provided log. A procedure will be developed to prevent introduction of foreign materials into the reactor coolant system. Information on this 1988 event was published in 1989.[188]

5.1.12.3 Diablo Canyon-1

Incidents of loss of cleanliness control were identified on April 9, 12, 21, 22, and May 10, 1988, at Diablo Canyon-1 by NRC and licensee personnel, including the discovery on April 22, 1988, of foreign material on the reactor vessel upper internals. One of the activities underway from April 6 to 9, 1988, was control rod drive mechanism weld repair. Information on this 1988 event was published in 1989.[189]

5.1.12.4 Westinghouse Fuel

See Section 5.1.1.9 for details on this item of interest.

5.1.12.5 Vermont Yankee

See item under 5.1.1.20.

5.1.13 Failure of Inflatable Seal

There was one event in 1988 at a U.S. plant that involved the failure of an inflatable seal. That event is described below.

5.1.13.1 Surry

There was a Severity Level-III violation (i.e., under the NRC's policy, such a violation is one that is cause for significant concern) at the Surry Power Station on May 17, 1988. The event involved the sudden failure of an inflatable seal in the reactor refueling cavity and the subsequent loss of 30,000 gallons of water from the cavity. Information on this 1988 event was published in 1989.[190,191]

5.2 NEW PROBLEMS IN 1989 (AND A FEW IN 1990)

Described below in 24 sections are various new problems with fuel systems that occurred in 1989 (plus a few in 1990).

5.2.1 Fuel Systems: Failures, Damage, or Potential for Damage

There were 7 events (1 in 1990, 6 in 1989) at U.S. plants and 12 events (1 in 1990, 11 in 1989) at plants in other countries concerning fuel systems that involved failures, damage, or the potential for damage. Those events are described below.

5.2.1.1 Haddam Neck[a]

The occurrence of an unusually large number of defective fuel rods (213 leaking fuel rods in 67 fuel assemblies) at Haddam Neck has made the utility extend their outage (originally scheduled for September 2 to November 2, 1989) by two months to repair the fuel assemblies.[264] The fuel rod failures are being attributed to tiny debris left in the primary system from the previous outage.

The licensee for the reactor (one of the last using stainless steel-clad fuel rods) is in the process of obtaining a license to use Zircaloy-clad fuel rods instead of stainless steel-clad fuel rods.

5.2.1.2 Haddam Neck

Ultrasonic testing on November 17, 1989, revealed a significant number of failed fuel rods at Haddam Neck.[265] Approximately 233 failed fuel rods were identified in 88 of 109 fuel assemblies scheduled for reinsertion. The failures were caused by debris-induced fretting. The debris lodged between the lower fuel assembly nozzle and the first spacer grid. Although the source of the debris has not been confirmed, it appears to be a machining by-product from the thermal shield support system repairs that were performed during the last refueling outage. The affected fuel assemblies that are to be reinserted in the core will be reconstituted to remove the failed rods. A root-cause evaluation of the event is to be conducted.

5.2.1.3 Haddam Neck

A recent article[266] indicates that Haddam Neck has extended its outage to repair 286 degraded fuel rods in 88 fuel assemblies, most of which the licensee suspects are leakers. The cause of the problem is being attributed to tiny debris left in the primary coolant system from previous outage work. The licensee stated that "...It's rare for Connecticut Yankee to have any

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(a) The following four entries (Sections 5.2.1.1, 5.2.1.2, 5.2.1.3, and 5.2.1.4) pertain to the same event. Although debris-induced fretting is an old problem, it is new at Haddam Neck.

leaking fuel pins." The licensee is in the process of getting approval to replace stainless steel-clad fuel rods with ones clad with Zircaloy.

5.2.1.4 Haddam Neck

Haddam Neck has extended its outage until at least April 1990 because fuel damage is more extensive than originally thought.[267] The reactor has 343 fuel rods in 88 fuel assemblies with through-wall cracks caused by metal flakes left over from a job done on the thermal shield during the last outage. The flakes, the largest of which is the size of a fingernail, apparently caused nicks and some cracks. Seventy-five percent of the damaged fuel rods have been in the outer two rows of rods of the fuel assemblies. The fuel assemblies are being cleaned and reconstituted.

5.2.1.5 Limerick-2

On June 13, 1989, at Limerick-2, the center stringer assembly containing seven startup neutron source pins unthreaded from the top of the assembly and dropped approximately 35 feet through the water to the cask pit floor.[268] Visual inspection (by underwater camera) indicated that one source pin was damaged (amounted to >$2,000). The pin was determined to be unacceptable for use. The event was the result of a personnel error and a procedural deficiency.

5.2.1.6 Oyster Creek

A review of the containment spray system logic on March 8, 1989, determined that the system at Oyster Creek would not perform as expected during a design basis loss-of-coolant accident (LOCA) due to the design of the system logic.[269] This occurrence is considered to have potential safety significance in that the loss of net positive suction heat (NPSH) to the core spray pumps could lead to core damage during a LOCA. Any core damage would be minimized by the fact that other sources of water external to the primary containment would be used to provide cooling to the core if the torus was unavailable.

5.2.1.7 Point Beach-1

On April 21, 1989, it was confirmed at Point Beach-1 that the estimated time to core uncovery, assuming a single train of the Emergency Core Cooling System (ECCS) during the time of transfer from the refueling water storage tank to the containment sump, was probably considerably shorter than previously assumed.[270] Previous analyses had not considered entrainment and steam voiding in the core. The licensee issued an order to provide guidance to operating crews for an event of this type.

5.2.1.8 San Onofre-1

An evaluation at San Onofre-1 on February 27, 1989, determined that design provisions intended to trip the reactor in the event of a reactor coolant pump (RCP) locked rotor did not satisfy the single failure criteria.[271] Specifically, with a concurrent RCP locked rotor and a failure of a reactor coolant low measured flow reactor protection system (RPS) trip, an RCP overcurrent trip signal to the RPS would not have actuated in sufficient time (six seconds) to preclude exceeding core design limits. The cause of the condition was related to an absence of available design basis documentation in combination with an inadequate interdisciplinary review.

5.2.1.9 Sequoyah-1 and -2

The licensee for Sequoyah-1 and -2 submitted a response to an item of violation cited by NRC in Inspection Report 50327 and 50238/8907.[272] The event occurred on February 11, 1989, during an attempt to raise the upender without first having the fuel transfer card fully inserted. The result of this action was a bent irradiated fuel assembly (this is a repeat of Violation 50328/84-36-01 that occurred at Sequoyah-2). However, it is stated later in the article that an examination revealed no damage to the fuel assembly and that the damage to the fuel transfer cart was caused by personnel error in failure to follow the procedure.

5.2.1.10 Vogtle-1

On March 20, 1990, a construction truck knocked down a power line outside the Alven W. Vogtle plant in Waynesboro, Georgia.[273] The event caused Vogtle-1, which was down for refueling, to lose power to its core cooling system; Vogtle-2 tripped off-line for unknown reasons when Vogtle-1 lost power. Vogtle-1 was without power for 36 minutes. NRC and Georgia Power investigators visited the plant in an effort to determine why the backup generator failed and why the power interruption of Vogtle-1 caused an automatic shutdown of Vogtle-2.

5.2.1.11 Canada

In January 1990, an accident occurred during routine on-power fueling of one of the 480 fuel channels at Canada's Bruce-4.[274,275] After the fueling machine had been positioned and locked on to the designated channel, the 20-ton bridge that carries the fueling machine cylinder was driven 16 inches downward towards a different channel position on the reactor face. This displacement apparently damaged an end fitting or ruptured the fuel channel, releasing heavy water coolant onto the reactor structure floor. Personnel are expecting to remove 8 irradiated fuel bundles from the tilted fueling machine and 13 bundles from the affected channel.

5.2.1.12 Federal Republic of Germany[a]

On July 24, 1989, the refueling machine telescope mast manipulator, which was suspended above the reactor vessel, snapped off at Isar-1 in West

(a) The following three entries (Sections 5.2.1.12, 5.2.1.13 and 5.2.1.14) pertain to the same event.

Germany.[276] The manipulator's plunge was broken by a refueling platform, but the collision spilled 67 ball bearings from a rotation device on the platform into the open vessel. Thus far, 45 bearings have been recovered with magnets. Bavarian regulators have stated that the remaining 13 bearings must be found before restart of the BWR will be allowed. Utility officials acknowledge that damage to fuel cladding from a loose ball bearing cannot be completely excluded. In the meantime, 235 of the 594 fuel assemblies in the Isar-1 core have been replaced and inspected.

5.2.1.13 Federal Republic of Germany

Isar-1 (BWR) was restarted in September 1989 after West German licensing officials ruled that the nine ball bearings assumed to be left in the reactor after the July 24, 1989, refueling accident posed no danger to reactor safety.[277] In the accident, the refueling machine telescope mast manipulator, suspended above the reactor vessel, snapped off and collided with the refueling platform, causing 67 ball bearings to be dumped into the core. About 80,000 individual reactor parts and components were inspected. Experts determined that the nine ball bearings are not likely to be dislodged by currents generated by recirculation pumps. It would have been possible to find the bearings but since they are presumed to be in inaccessible corners of the core, a licensing official said "the costs would have been prohibitive with respect to the benefits accrued" in enhanced reactor safety.

5.2.1.14 Federal Republic of Germany

Fuel assemblies were inspected in September 1989 at Isar-1 (BWR) in the Federal Republic of Germany after a slight increase in radioactivity was detected in the primary coolant circuit.[278] The release was due to hairline cracks in a defective fuel element cladding. It was feared initially that the cause might be traced to the ball bearings that were left in the primary coolant circuit after the recent maintenance outage. The failed fuel was in an area that the missing balls could not have reached, and it was decided that the cause of the event was probably a reaction between the cladding and the fuel pellets.

5.2.1.15 Federal Republic of Germany

West Germany's Kruemmel (BWR) has been put back into operation after nine weeks of interruption caused by a fuel assembly that was dropped from the crane during reloading into the water-filled storage pool.[279]

5.2.1.16 France

On September 1, 1989, at France's Saint Laurent-A1 (a gas-cooled reactor), a fuel element broke loose from the fuel handling system and fell seven meters into its original place, with no apparent consequence for cladding integrity.[280] During attempts to inspect the channel with the fallen fuel element, two more fuel elements were dropped, one from five meters and the other from one meter, also with no apparent consequences.

5.2.1.17 Japan[a]

At Japan's Fukushima-II-3 on January 6, 1989, one of the recirculation pumps registered "wild vibration" and led operators to completely shut down the reactor on January 7.[281] Inspection revealed foreign materials (turbine blade, bolts, metallic pieces, etc.) in pumps and the reactor vessel. The lower part of the reactor vessel is being emptied to permit flushing out all the foreign materials inside the vessel as well as the fuel bundles. The fuel bundles are to be cleaned in April or May 1989. Every fuel rod (more than 48,000 total) will undergo thorough inspection.[282]

5.2.1.18 Japan

As indicated above, a pump at Japan's Fukushima-II-3 (BWR) failed on January 6, 1989, and parts of the pump were swept through the piping toward the reactor vessel. Small pieces have been found in the reactor vessel and in the core itself. To examine the core, fuel bundles are moved to the spent fuel pool and are flushed with air and water to dislodge the metal particles. The fuel bundles are examined via fiber optics to assess any damage to the surface of the fuel bundle.[283]

5.2.1.19 Japan

In the incident at Japan's Fukushima-II-3 noted above, about 31.3 kg of metal were lost to abrasion. Powdered metal, "up to several kilograms," is thought to have found its way to the fuel assemblies. This incident is attributed to insufficient welding penetration of the reactor recirculation pump bearing ring.[284]

5.2.1.20 Japan

Rare stress corrosion cracking of the traversing in-core probe (TIP) system was discovered on September 18, 1988, at Japan's Hamaoka-1 (BWR).[285] A General Electric spokesman knew of no instance of similar cracking in U.S. BWRs. The minor leak in the in-core probe housing came from a 13-mm crack in the upper part of the tube.

5.2.1.21 Japan

On October 4, 1989, a leaking fuel assembly was found at Japan's Ohi-1 (PWR).[286]

5.2.1.22 Japan

Fuel was loaded in Mutsu, Japan's nuclear-powered ship, in 1972.[287] The 36-MW PWR core was operated until 1974 and then, because of a radiation leak, the reactor was shut down and the reactor vessel sealed. After 14 years, the

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(a) The following three sections (Sections 5.2.1.17, 5.2.1.18, and 5.2.1.9) pertain to the same event.

reactor was opened and some components (included 12 fuel rods and 12 control rods) were examined. Corrosion was detected on one fuel rod, one control rod, and other reactor components. It is believed by the Japanese that the corrosion may have been caused by local and limited water quality deficiency, a result of water level changes made during some shield reinforcement work. A March 1990 article[288] indicates that after 16 years of nonoperation the reactor is to start up again for a six-phase power test; the reactor is scheduled to achieve criticality on March 29, 1990. Part of the power test is to be performed in port and part at sea. If fuel problems arise during the power test, it will be interesting to see if they are traceable to the water quality problem.

5.2.1.23 Switzerland

In May 1989, Asea Brown Boveri (ABB) performed a complete fuel cladding inspection at Switzerland's Beznau-1.[224] This is the first time a Swiss reactor has undergone ultrasound fuel cladding inspection. ABB has used ultrasound techniques to inspect the cladding on some fuel rods in Europe, but this work at Beznau-1 was the first time ultrasound was used to inspect cladding of an entire core in a European reactor. The plant operator (NOK) suspected that some rods in the core of Beznau-1 were flawed; following the inspection, several single rods were replaced.

5.2.1.24 Switzerland

During the shutdown for repairs on July 13, 1989, an increase in reactor coolant activity led the Beznau-1 plant operator (NOK) to suspect a fuel element was defective.[289] All 120-odd fuel elements are being inspected to pinpoint the problem and to replace any leaking element.

5.2.1.25 United Kingdom

An incident in February 1989 at Dungeness-B, a British advanced gas-cooled reactor, involved dropping onto the top of the reactor a string of new fuel assemblies that were being prepared for loading into the reactor.[290] Some fragments from shattered fuel assemblies, which consist of graphite sleeves around clusters of stainless steel-clad fuel pins, entered the reactor.

5.2.2 Issues/Concerns with Generic Implications

There were two 1990 items of interest pertaining to issues/concerns with generic implications. These items are discussed below.

5.2.2.1 Krypton-85 From Decayed Spent Fuel

The NRC recently issued an information notice to all holders of operating licenses or construction permits alerting them to potential problems resulting from the accidental release (e.g., at the spent fuel pool working floor) of krypton-85 from decayed spent fuel.[291] Direct exposure to this gas would result in a dose to the skin approximately 100 times the whole-body dose.

5.2.2.2 Fort St. Vrain

Public Service Co. of Colorado (PSC) has ceased operation of its ailing Fort St. Vrain nuclear power plant after hairline cracks were found in tubes supplying steam to the turbine generator. The reactor had been shut down since August 18, 1989, when a routine test indicated a control rod problem. While the control rod was being replaced, a separate examination of the steam generators revealed the cracks. Although the cracks are not a public safety concern, the financial burden of correcting them makes early closure more feasible.[368].

5.2.3 Unexpected Power Loss/Increase

There were two events (one in 1990, one in 1989) at U.S. plants involving unexpected power loss or power increase. Those events are described below.

5.2.3.1 Dresden-2

On March 14, 1989, there was an unexpected power increase upon entering the remote load following mode at Dresden-2.[292] The event was due to a procedural deficiency. Although core flow increased to slightly above the 100% limit in Technical Specification 3.3.G during this event, safety significance was minimal as maximum core thermal power and other nuclear fuel limits were not exceeded. This was the first occurrence in which 100% core flow was exceeded while operating in the economic generation control (EGC) load following mode.

5.2.3.2 Vogtle-1

See Section 5.2.1.10 for details on this 1990 event.

5.2.4 Fuel Handling: Fuel Dropped/Broken/Damaged or Potential for Damage Existed

There were two events in 1989 at U.S. plants and one event in 1990 at a plant in another country involving fuel handling and cases where fuel was dropped, broken, or damaged or the potential for damage existed. Those events are described below.

5.2.4.1 Limerick-2

See Section 5.2.1.5 for details on this 1989 event.

5.2.4.2 Sequoyah-1 and -2

See Section 5.2.1.9 for details on this 1989 event.

5.2.4.3 Canada

See Section 5.2.1.11 for details on this 1990 event.

5.2.5 Fuel Handling: Procedural Violation

There was one fuel handling event in 1989 at a U.S. plant involving a procedural violation. That event is discussed below.

5.2.5.1 Three Mile Island-2

At Three Mile Island-2 on January 20, 1989, core alterations (i.e., movement of fuel within the reactor vessel) were performed without the supervision of a fuel handling senior reactor operator (FHSRO).[318] Table 6.2-1 of the Technical Specifications requires a core alteration to be directly supervised by a senior reactor operator or an FHSRO. The root cause of this event was personnel error by the duty task supervisor and FHSRO in that they jointly failed to adequately communicate.

5.3 OLD PROBLEMS THAT DID NOT RECUR OR THAT WERE SOLVED

There were no fuel failures reported in the U.S. in 1989 that were due to baffle jetting (see Table 22), pellet-cladding interaction (PCI), or primary hydriding. Also, no events involving hold-down springs (see Table 23) were reported in 1989 by U.S. plants.

TABLE 22. PWR Fuel Assemblies with Damaged or Failed Fuel Rods Due to Baffle Jetting

Year	Plant(s)	Number of Fuel Assemblies
1989		
1988		
1987		
1986[a]	Beaver Valley-1	1
	McGuire-1	2
	Point Beach-1	1
1985	Beaver Valley-1	
	McGuire-1	
	North Anna-1	1
	Point Beach-1	4[b]
	Point Beach-2	2
	Yankee Rowe	3
1984	Yankee Rowe	1
1983	Farley-1	11
1982	Trojan	17
1981	Farley-1	
	Trojan	1
	Yankee Rowe	1[c]
1980	Trojan	2
1979		
1978	Trojan	
1977		
1976	Point Beach-1	1
1975		

TABLE 22. (contd)

Year	Plant(s)	Number of Fuel Assemblies
1974		
1973	Point Beach-1	

(a) Two defective fuel assemblies also found at Goesgen (Switzerland).
(b) Only a slight indication of damage to fuel rods was noted on two of these assemblies.
(c) Only had one fuel rod that was bowed due to baffle jetting.

TABLE 23. Events Involving Hold-Down Springs

		Cause of Event(s)		
Year	Plant	Broken Springs(s)	Loose Spring Clamp	Broken Spring Clamp Screw(s)
1989				
1988				
1987				
1986				
1985				
1984	Point Beach-2			X
	Surry-1		X	X
1983	McGuire-1	X		
	McGuire-2	X		
	Oconee-1	X		
	Sequoyah-2	X		
1982	Davis Besse-1	X		
	Oconee-1	X		
	Oconee-2	X		
	Oconee-3	X		
1981	Arkansas-1	X		
1980	Crystal River-3	X		
	Davis Besse-1	X		
	Oconee-1	X		

6.0 TRENDS

No major new problems surfaced during 1989. What follows is one possible way of sorting the items in Section 5.0 "Problem Areas Observed in 1989" - by major reactor system. This provides a system-focussed perspective to supplement the problem-oriented perspective found in Section 5.0, in an effort to uncover possible new trends in problems.

The systems to be discussed are listed below; for easy reference, each entry in the following text will also include its Section 5 number. Some pertinent entries were documented in 1988 or 1990, they are so identified.

- 6.1 CONTROL ROD SYSTEMS
  - 6.1.1 Flow-Induced Fretting
  - 6.1.2 Hafnium Rod Swelling
  - 6.1.3 Insertion without Appropriate Control
  - 6.1.4 Procedural Deficiency or Violation/Personnel Error
  - 6.1.5 Sticking
  - 6.1.6 Stress Corrosion Cracking
  - 6.1.7 Valves
  - 6.1.8 Wear
- 6.2 EMERGENCY/SAFETY SYSTEMS
- 6.3 FUEL SYSTEMS
  - 6.3.1 Bowing
  - 6.3.2 Crud-Induced Localized Corrosion (CILC)
  - 6.3.3 Fuel Alignment Pins
  - 6.3.4 Fuel Handling
  - 6.3.5 Iodine Spiking
  - 6.3.6 Leaks
  - 6.3.7 Procedural Deficiency or Violation/Personnel Error
- 6.4 IN-CORE INSTRUMENTATION
- 6.5 POWER - REACTOR POWER RATING
  - 6.5.1 Axial Shape Index
  - 6.5.2 Power Exceeded
  - 6.5.3 Thermal-Hydraulic Instability
- 6.6 POWER - REACTOR POWER SUPPLY SYSTEM
- 6.7 PRIMARY COOLING SYSTEM
  - 6.7.1 Debris
  - 6.7.2 Core Coolant Flow/Lowering of Water Level
  - 6.7.3 Stress Corrosion Cracking
  - 6.7.4 Impurities
  - 6.7.5 Unborated Water
  - 6.7.6 Unanalyzed Condition
- 6.8 OTHER REACTOR SYSTEMS
  - 6.8.1 Containment
  - 6.8.2 "Hot Particles"
  - 6.8.3 Inflatable Seal
  - 6.8.4 New Fuel Radiation Monitor
  - 6.8.5 Core Operating Limit Supervisory System (COLSS)
- 6.9 SPENT FUEL POOLS

6.10 PERSONNEL ERROR, PROCEDURE VIOLATIONS, TRAINING
- 6.10.1 Personnel Error
- 6.10.2 Procedure Violation/Noncompliance
- 6.10.3 Deficiency in Training
- 6.10.4 Lack of Administrative Control
- 6.10.5 Defective Procedure

6.11 UNKNOWN ROOT CAUSE

In cases in which an item could be classed under more than one category, the additional categories are underlined in the paragraph where the item first appears; the item is not listed in more than one category. This effort to cross-reference the items is incomplete, however, and there may be duplication in Section 6.10 on Personnel Error. The longest lists of problem items involve Personnel Errors, some of which appear also in the section on Fuel Systems. The fuel systems and the control systems, two of the major systems, exhibit the most events.

6.1 CONTROL ROD SYSTEMS

6.1.1 Flow Induced Fretting

5.1.5.1 - Braidwood-1 & -2 - A 1989 article[319] indicates that the control rods are to be inspected for fretting and wear.

6.1.2 Hafnium Rod Swelling

5.1.2.3 - Wolf Creek-1 and Taiwan, Maanshamn (1988) - Swelling of hafnium control rods has become a full-fledged safety concern.

6.1.3 Insertion without Appropriate Control

5.1.15.2 - Fort St. Vrain - A control rod pair failed to scram during a scram surveillance test.

5.1.15.3 - McGuire-2 - During a routine rod cluster control assembly test, a reactor trip occurred because control rods dropped into the core. The cause was marked as Unknown.

5.1.15.5 - Prairie Island-2 - In scrams in December 1989, the electrically energized latches that secure the reactor's array of control rods in their raised position lost power, allowing the rods to drop and disrupting neutron production in one sector of the reactor. The root cause is essentially Unknown.

5.1.15.6 - Turkey Point-4 - The control rod system began to insert rods automatically but stopped after only four steps of insertion. The operators could not insert the control rods manually with the control system, so the reactor was tripped.

5.1.15.7 - Yankee Rowe - During operation at 100% power, the operators discovered that Group C control rods could not be moved due to an inoperable control rod cam motor.

5.1.15.8 - France, Blayais-4 (1988) - Two of the 53 control rod clusters were blocked due to damaged guide cards on three tubes.

5.1.14.4 - Switzerland, Leibstadt (1988) - A loose control rod coupling was discovered - the only Class A incident at a power reactor in Switzerland in 1988.

5.1.18.1 - Braidwood-1 - Lightning-induced voltage transients removed power to various rod drive control cards and allowed numerous control rods to drop.

5.1.18.3 - Catawba-2 (1988) - A control rod dropped into the core during testing due to the failure of a fuse.

6.1.4 Procedural Deficiency or Violation/Personnel Error/Miscommunication

5.1.15.1 - Dresden-2 - Some control rod drive (CRD) hydraulic control unit charging header ball check valves were inoperative or missing, due to Procedural Deficiency.

5.1.15.4 - Perry-1 - Operation Condition 2, startup, was completed with a control rod inoperable, due to Procedural Deficiency.

5.1.18.2 - Brunswick-2 (1988) - With reactor in Operational Condition 5, a control rod was in the fully withdrawn position but the shorting links had not been removed from the circuitry.

5.1.18.5 - Fort St. Vrain (1988) - Licensee personnel handled control rods with bare hands.

5.1.18.6 - Limerick-1 - Two control rods had their uncoupling rods misaligned during the plant's second cycle, following misalignment during installation (Design Deficiency).

5.1.18.7 - Millstone-1 - Four restraining metal straps on control rod drive system hydraulic control units (HCUs) were found missing in August 1989, due to personnel error.

5.1.18.9 - Turkey Point-3 (1988) - Maintenance was performed on the control rod system without documented instructions or appropriate drawings.

5.2.8.2 - Trojan - Miscommunication between the licensee and the nuclear steam supply system vendor in 1976 caused a nonconservative assumption to be used, with the result that the 100% power reference temperature used in the rod control system was different from that in the safety analysis. This was reported in 1989.

6.1.5 Sticking

5.1.2.9 - France, Gravelines-4 - A control rod broke off and fell to the bottom of a fuel assembly and its spring was stuck in the guide tube, causing the control rod cluster to stick at an intermediate position.

5.2.6.1 - Fort St. Vrain - One of 37 control rod pairs could not be inserted more than a third of the way and became stuck outside the reactor; the head of one of the control rod's Inconel clevis pin bolts had developed a crack, broken off, and become wedged between the control rod and the guide tube.

5.2.6.2-5.2.6.5 - Finland, Olkiluoto-1 - A metallic powder normally used in sandblasting was found in the control rod drives. The material appears to correspond to old stainless steel BWR oxide layers.

5.2.6.6 - Japan, JAERI's nuclear safety research reactor - The reactor failed to reach criticality when the bottom of one rod had become disconnected and was still in the core.

6.1.6 Stress Corrosion Cracking

5.1.17.1 - Arnold - In November 1988, transgranular stress corrosion cracking of control rod drive piping was observed.

5.1.17.2 - Diablo Canyon-1 (1988) - Canopy seal welds on four control rod drive mechanism head adapter plugs were leaking, because of transgranular stress corrosion cracking.

5.1.17.3 - Palisades (1988) - A total of 14 control rod drive seal housings were found to be cracked, due to contaminant-induced transgranular stress corrosion. Studies indicate that it would take approximately five years for a 0.030-inch initial depth crack to propagate through the entire housing wall.

5.1.17.4 - Pilgrim (1984-1989) - Control rod drive collet retainer tube weld defects were determined to be due to intergranular stress corrosion cracking (IGSCC).

5.1.19.6 - France, Gravelines-1 - Guide tube support pins have been replaced for the second time. A new support pin more resistant to cracking is under development.

5.2.1.22 - Japan, Mutsu (1990) - On this reactor which has been shut down for 16 years and has recently been opened, corrosion has been found on one fuel rod, one control rod and other reactor components, possibly due to local and limited water quality deficiency.

6.1.7 Valves

5.1.14.3 - River Bend (1987) - A valve supplying cooling water to one of the control rod drives was mispositioned and 18 lock wires on the hydraulic control units attached to several of the valves were missing.

6.1.8 Wear

5.1.16.2 - Monticello - Refueling activities in November 1989 included removing 9 core blades because of wear and moving 30 other to maximize their useful lives.

5.1.17.8 - France - New (1989) criteria for replacement of worn control rods: cladding pierced through or worn over 20% of its circumference next to the seventh guide plate. All control rod clusters on 900-MW PWRs are to be changed by the end of 1990.

5.1.19.1 - Federal Republic of Germany, Biblis and Unterweser - Central pins, the counterpart of Westinghouse-design control rod guide tube split pins, will be replaced in the years 1989 through 1991 if any crack indications are found.

6.2 EMERGENCY/SAFETY SYSTEMS

5.1.20.3 - Haddam Neck - A discrepancy was discovered in the Design Basis Large-Break Loss-of-Coolant Accident (LBLOCA) analysis. A nonconservative reactor vessel lower plenum volume was used in the Interim Acceptance Criteria (IAC) model.

5.1.20.6 - North Anna-1 - An input error for the LBLOCA analysis for the 18% steam generator tube plugging licensing case was discovered.

5.2.1.6 - Oyster Creek - The containment spray system logic would not perform as expected during a design basis LOCA, due to the design of the system logic.

5.2.1.7 - Point Beach-1 - It was discovered that the estimated time to core uncovery, assuming a single train of the Emergency Core Cooling System (ECCS) during the time of transfer from the refueling water storage tank to the containment sump, was probably considerably shorter than previously assumed.

5.2.1.8 - San Onofre-1 - It was determined that the design provisions intended to trip the reactor in the event of a reactor coolant pump (RCP) locked rotor did not satisfy the single failure criteria.

6.3 FUEL SYSTEMS

6.3.1 Bowing

5.1.1.7 - There is a need to account for channel bowing in computer modeling of fuel performance.

5.1.1.15 - Sweden, Oskarshamn-2 - Channel bowing lead to failure of four first-cycle fuel assemblies, due to dry-out.

6.3.2 Crud-Induced Localized Corrosion (CILC)

5.1.4.1 - Hatch-1 &-2 (1980-1989) - Most of the 108 fuel assembly failures during 13 of the last 15 cycles were due to CILC.

5.1.4.2 - Limerick-1 (1987-1989) - Pinhole-sized leaks in as many as 30 fuel rods have forced the reactor to run at reduced power.

6.3.3 Fuel Alignment Pins

5.1.19.2 - Federal Republic of Germany, Biblis-A - Twenty two Inconel X-750 fuel alignment pins were replaced with new austenitic steel pins. Pins at Obrigheim and Grafenrheinfeld have been replaced. Pins at Grohnde, Phillipsburg-2, and Neckarwestheim-1 have been or are to be inspected.

5.1.19.3 - Federal Republic of Germany, Biblis-B - Sixty seven of 386 fuel alignment pins inspected were replaced prior to January 1990.

5.1.19.4 - Federal Republic of Germany, Unterweser (1988-1989) - Thirty four fuel alignment pins of Inconel X-750 were replaced with stainless steel. Inspection of all fuel alignment pins was required at all West German PWRs following a refueling accident at Brokdorf in 1988.

6.3.4 Fuel Handling

5.1.1.2 - Quad Cities-1 - Fuel assembly fell from refuelling grapple.

5.1.7.1 - Limerick-1 - A fuel rod dropped during fuel reconstitution activities but was not damaged. Personnel Fatigue was involved also.

5.1.7.2 - North Anna-1 &-2 - An attempt was made to insert a new fuel assembly into a spent fuel storage rack position. Categorized also as Personnel Error.

5.1.7.3 - Palisades (1988) - A fuel bundle was left hanging from the upper guide structure due to the bundle adhering to the bundle guide pins.

5.1.7.4 - Vogtle-1 (1988) - Power loss caused suspension of fuel in refueling machine and subsequent manual operation of the handling machine.

5.1.9.1 - Calvert Cliffs-1 (1988) - A heavy load was moved over the spent fuel pool, contrary to proper Administrative Control.

5.2.1.5 - Limerick-2 - The center stringer assembly containing seven startup neutron source pins unthreaded from the top of the assembly and dropped approximately 35 feet through the water to the cask put floor. (Personnel Error and Procedure Deficiency.)

5.2.1.9 - Sequoyah-1 & -2 - During an attempt to raise the upender without first having the fuel transfer card fully inserted, an irradiated fuel assembly was bent.

5.2.1.11 - Canada, Bruce-4 (1990) - During routine on-power refueling of one of the 480 fuel channels and after the fueling machine had been positioned and locked on the designated channel, the 20-ton bridge that carries the fueling machine cylinder was driven 16 inches downward toward a different channel position of the reactor face, apparently damaging an end fitting or rupturing the fuel channel and releasing heavy water onto the reactor structure floor.

5.2.1.12 - Federal Republic of Germany, Isar-1 - The refueling machine telescope mast manipulator, which was suspended above the reactor vessel, snapped off, hitting the refueling platform and causing 67 ball bearings from a rotation device on the platform to spill into the open vessel. All but nine were recovered before restart of the BWR was allowed.

5.2.1.15 - Federal Republic of Germany, Kruemmel - Nine weeks of interrupted operation were caused by dropping a fuel assembly into the water-filled storage pool during reloading.

5.2.1.16 - France, Saint Laurent-A1 - A fuel element broke loose from the fuel handling system and fell seven meters into its original place, with no apparent consequence for cladding integrity.

5.2.1.25 - United Kingdom, Dungeness-B - A string of new fuel assemblies that were being loaded into the reactor were dropped onto the top of the reactor.

6.3.5 Iodine Spiking

5.1.1.5/6 - Surry-1 - On September 1, 1988 an iodine spike occurred; sipping confirmed that a fuel leak existed.

5.1.6.1 - Limerick-1 - An iodine spike occurred on January 11, 1989 exceeding the Technical Specification limit of 0.2 microcuries/gram.

6.3.6 Leaks

5.1.1.1 - Oconee - Most leaking in fuel rods occurs directly adjacent to the center instrument tube and in or adjacent to corners.

5.1.1.14 - India, Tarpur - As many as 100 fuel bundles developed pinholes in the first cycle.

5.1.1.18 -Sweden (1984-1989) - Fuel failures have occurred in four BWRs and three PWRs in the past five years.

5.1.1.19 - USSR - Fuel failures are a recurrent problem in the USSR.

5.2.1.1 - 5.2.1.4 - Haddam Neck - Approximately 283 failed fuel rods were identified in 88 of 109 fuel assemblies scheduled for reinsertion. These were all attributed to debris-induced fretting.

5.2.1.14 - Federal Republic of Germany, Isar-1 - A release of radioactivity found in the primary coolant was found to be due to hairline cracks in fuel element cladding, probably due to pellet-cladding interaction (PCI).

5.2.1.21 - Japan, Ohi-1 - A leaking fuel assembly was found in October.

5.2.1.23 - Switzerland, Beznau-1 - Asea Brown Boveri performed a complete ultrasonic fuel cladding inspection, the first time a Swiss reactor has undergone ultrasound fuel cladding inspection.

6.3.7 Procedural Deficiency or Violation/Personnel Error Related to Fuel

5.1.10.1 - Arkansas-2 (1988) - Housekeeping discrepancies were observed on the fuel handling bridge during fuel handling activities.

5.1.10.2 - Browns Ferry-2 (1988) - Fuel loading may have been performed without adequate neutron monitoring, due to inadequate safety review.

5.1.10.3 - Brunswick-1 (1988) - The standby gas treatment system was inoperable during a period when irradiated fuel assemblies were being moved.

5.1.10.4 - Byron-1 (1988) - Water was removed from the reactor vessel faster than it was being from the refueling area.

5.1.10.5 - Clinton-1 - Reactor core alterations were performed in one quadrant without an operable source range monitor (SRM) in the adjacent quadrant.

5.1.10.6 - Harris-1 - The fuel building operating floor equipment hatch was not installed as required during the process of transferring spent fuel from the shipping cask to the storage pool.

5.1.10.7 - Point Beach-1 (1988) - A warning device was not in use during a fuel assembly transfer.

5.2.5.1 - Three Mile Island-2 - Movement of fuel within the reactor vessel was performed without the supervision of the fuel handling senior reactor operator, a Technical Specification violation.

5.2.11.1 - France, Tricastin-2 - The fuel assembly cooling system had not been switched on during refueling, a Level 1 event.

5.2.12.1 - France, Dampierre - Some fuel rods were discovered to contain undersized pellets. This was classed as a Level 1 event because the discovery had not been made prior to fuel loading.

6.4 IN-CORE INSTRUMENTATION

5.1.3.1 - Beaver Valley-1 - Nine in-core instrumentation guide thimble tubes had degraded due to mechanical wear.

5.1.3.2 - Diablo Canyon-1 - Twenty-eight in-core neutron monitoring system thimble tubes had degraded beyond 50% of wall thickness.

5.1.3.3 - South Texas-1 - Thicker-walled in-core instrumentation thimble tubes were to be installed during the August 1989 refueling.

5.1.3.4 - Belgium, Tihange-1, 2, and 3 and Doel-3 and 4 (1981-1989) - This item includes a discussion of wear on in-core instrumentation in these reactors, as a function of subtle design differences.[158]

5.2.1.20 - Japan, Hamaoka-1 (1988) - Rare stress corrosion cracking of the traversing in-core probe (TIP) system caused a minor leak in the in-core housing from a 13-mm crack in the upper part of the tube.

6.5 POWER - REACTOR POWER RATING

6.5.1 Axial Shape Index

5.1.26.1 - San Onofre-2 - A manual trip was initiated in September 1989, because of the approach of the axial shape index (ASI) to the core protection calculator auxiliary trip setpoint.

6.5.2 Power Exceeded

5.1.22.4 - LaSalle-2 (1988) - The power level increased to 118% of rated power when two pumps that recirculate water through the reactor vessel automatically shut down when the reactor was at 85% of full power.

5.1.22.6 & 7 - San Onofre-2 (1988) - It was determined in 1988 that the reactor ran in excess of 102% of rated power in 1984 due to a manufacturing defect in a feedwater flow venturi. And during some of the time between August and October, 1988 it may have run slightly in excess of 100% of indicated power due to several factors which caused a decrease in indicated power relative to actual plant power.

5.1.20.4 - Hope Creek (1988) - The reactor was operated at 101.2% of rated power, nominally, due to nonconservative calculational errors for feedwater flow transmitters.

5.1.20.5 - McGuire-1 - The reactor operated at greater than 100% thermal power due to Procedural Deficiency.

5.1.24.1 - Limerick-1 (1988) - Prior to a controlled scram, the power began increasing due to the positive reactivity effect of decreasing moderator temperature, a Personnel Error.

5.2.3.1 - Dresden-2 - An unexpected power increase occurred upon entering the remote load following mode, due to a Procedural Deficiency.

5.1.2.8 - Susquehanna-2 (1988) - Reactor power increased to 101% due to a pressure transient, the result of mispositioning of an isolation valve, a Personnel Error.

5.1.23.1 - Fort St. Vrain - Reactor power was found to be 83.4% (when maximum authorized power was 82%), because a reheat steam at temperature flow had not been accounted for in the secondary heat balance calculation of reactor power, a Personnel Error.

6.5.3 Thermal-Hydraulic Instability

5.1.2.2 - South Texas-1 (1988) - A flow anomaly occurred that was similar to those in Westinghouse four-loop plants.

5.1.2.5 - A 1989 paper[151] by the NRC discusses long-term resolution of BWR thermal-hydraulic instability.

5.1.20.1&.2 - Cook-2 - The rated thermal power was exceeded because of an incorrect change in blowdown constants in a thermal output computer program.

6.6 POWER--REACTOR POWER SUPPLY SYSTEM

5.1.2.1 - Brunswick-2 - A manual reactor trip was specified by new regulations under the conditions that prevailed in this loss-of-off-site power incident, even though the emergency systems under which the reactor would be shut down would have to run off the emergency diesel generators.

6.7 PRIMARY COOLING SYSTEM

6.7.1 Debris

5.1.1.10 - Argentina, Atucha-1 (1988) - Three damaged fuel-bearing pressure tubes allowed some fuel to fall into the vessel; extracting the shards has been a problem.

5.1.11.1 - Davis Besse-1 (1988) - Two pieces of high pressure injection/ make-up nozzle thermal sleeve, an apparent paint chip and a rag were discovered in the reactor vessel.

5.2.1.17,18 & 19 - Japan, Fukushima-II-3 - Vibration in one of the recirculation pumps was found to be due to foreign materials (turbine blade, bolts, metallic pieces, etc.) in pumps and the reactor vessel. All 48,000 fuel rods were to undergo inspection. Part of this problem was attributed to insufficient welding penetration of the reactor recirculation pump bearing ring.

5.2.10.1 - 5.2.10.7, 5.2.1.1 - 5.2.1.4, and 5.2.1.17 - 5.2.1.19 - Two events of debris, one in a U.S. plant (Haddam Neck) and one in Japan (Fukushima II-3) occurred in 1989.

6.7.2 Core Coolant Flow/Lowering of Water Level

5.1.21.3 - Nine Mile Point-2 (1988) - The reactor was inadvertently operated at about 104.5% of rated core flow due to an "unanalyzed condition".

5.1.30.2&3 - Clinton-1 - Three separate incidents of lowering the water level below the required 23 feet above the top of the reactor pressure vessel flange occurred, due in one case to deficient surveillance test procedure and in the other two to insufficient training.

5.1.30.4 - Nine Mile Point-2 - (1988) - Power was lost to the feed water pumps, allowing the water to boil off from 183 to 108 inches above the core, at which point the high pressure core spray and reactor core isolation cooling systems were activated.

6.7.3 Stress Corrosion Cracking

5.1.2.6 - A generic problem may be surfacing as evidence mounts for primary water Stress Corrosion Cracking (PWSCC) in Inconel-600 primary system pressure boundary penetrations.[152]

6.7.4 Impurities

5.2.14.1 - Calvert Cliffs-1 - An abnormal sulfate concentration existed in the primary coolant system at the time of startup.

6.7.5 Unborated Water

5.2.15.1 -San Onofre-1 - Approximately 440 gallons of unborated water was added to the reactor refueling cavity water during decontamination activities and resulted in a positive reactivity addition, but the 5% shutdown margin was not approached.

6.7.6 Unanalyzed Condition

5.1.21.3 - Nine Mile Point-2 - The reactor was inadvertently operated at about 104.5% of rated core flow due to an "unanalyzed condition".

6.8 OTHER REACTOR SYSTEMS

6.8.1 Containment

5.1.27.1 - Cook-2 (1988) - For periods of two to five minutes an open pathway existed from the containment atmosphere to the auxiliary building during core alteration and fuel movement.

5.1.27.2 - Farley-1 - Unlatching of control rod drive mechanisms was performed without having established containment integrity (Personnel Error).

5.1.27.3 - Farley (1990) - Containment integrity was breached during replacement of fuel within the reactor containment.

5.1.27.4 - McGuire (1988) - During fuel unloading operations, containment was breached when three temporary penetrations were found to be leaking.

5.1.27.5 - Millstone-3 - Fuel building integrity was lost during fuel movement when a door was left open, by Personnel Error.

5.1.27.7 - Zion-1 - While the core was being off-loaded, there was a loss of containment closure.

6.8.2 "Hot Particles"

5.1.28.1 - Several papers[219,249-251] provide information on the problem of hot-particle contamination in an estimated 70% of the nuclear power plants surveyed by EPRI.

6.8.3 Inflatable Seal

5.1.13.1 - Surry (1988) - An inflatable seal in the reactor refueling cavity suddenly failed, causing loss of 30,000 gallons of water from the cavity (a Level-III violation, causing significant concern).

6.8.4 New Fuel Radiation Monitor

5.1.34.1 - Palo Verde-2 (1988) - A new fuel radiation monitor failure was attributed to a malfunction of the unit's clock.

6.8.5 Core Operating Limit Supervisory System (COLSS)

5.1.34.3 - San Onofre-3 - The COLSS was found to be inoperable, so that the plant was operating with a departure from nucleate boiling ratio (DNBR).

6.9 SPENT FUEL POOLS

5.1.2.8 - NUREG-1353[153] No new regulatory requirement are warranted concerning the use of high-density spent fuel storage racks.

5.1.31.1-Rancho Seco - Before refueling could begin in 1989, a leak in the 5-mm thick stainless steel lining of the spent fuel pools had to be repaired.

5.2.13.1 - St. Lucie-2 - The water level in the spent fuel pool was raised high enough to flood the intake ventilation ducts that line the perimeter of the pool, a Personnel Error.

5.2.16.1 - Fitzpatrick - Surveys provided in the spent fuel pool to support ongoing work were inadequate to detect the presence of an object emitting up to 1000 R/hr on contact, which appeared in the work area.

5.2.16.2 - McGuire-1 - The neutron absorber panels originally installed in the racks are shorter than the active fuel length of the stored fuel assemblies; this, combined with the shrinkage of the Boraflex neutron absorber, could potentially have greater effects on reactivity than allowable.

6.10 PERSONNEL ERROR, PROCEDURES VIOLATIONS, TRAINING

6.10.1 Personnel Error

5.1.14.1 - Davis Besse-1 - A personnel error during maintenance caused a group of control rods to drop into the reactor core, stopping the nuclear chain reaction. Control room personnel erroneously believed that the nuclear reactor was continuing and resumed startup procedures, withdrawing the rods that had just dropped.

5.1.36.1 - 5.1.36.13 - Nine events in 1989 and four in 1988 involved personnel errors.

5.2.22.1 - 5.2.22.5 - There were five additional events in 1989 involving personnel errors.

6.10.2 Procedure Violation/Noncompliance

5.1.1.4 - Sequoyah-1 - The average temperature of the reactor dropped below the analyzed value, due to failure to comply with NRC regulatory requirements.

5.1.12.2 - Diablo Canyon-1 - Loose tools were found in the reactor vessel head cable tray area, where they could fall into the coolant system.

5.1.14.2 - Palo Verde-1 - Due to deletion of action requirements in the Technical Specifications, a control element assembly (CEA) slipped, during the performance of a surveillance, approximately 4 inches farther below the other CEAs in its group than is allowed by the Technical Specifications.

5.1.33.3 - Sequoyah-2 - Inadequate work control resulted in two emergency core cooling system pumps to be inoperable while the reactor was in hot standby.

5.1.31.2 - San Onofre-2 (1988) - The spent fuel handling machine (SFHM) may have been operated over the fuel storage pool while postaccident cleanup units were not operable, a violation of Technical Specification 3.9.12.

5.1.38.1 - 5.1.38.10 - Four events involving procedural noncompliance occurred in 1989 and 6 in 1988.

6.10.3 Deficiency in Training

5.1.32.1-5.1.32.17 - Thirteen cases of defective procedures or training deficiencies are noted in these sections for 1989 and four for 1988.

6.10.4 Lack of Administrative Control

5.1.11.2 - San Onofre-2 - Core alterations were performed without complete containment closure, due to lack of adequate Administrative controls.

6.10.5 Defective Procedure

5.2.19.1 - 5.2.19.4 - There were four events in 1989, three in the United States and one in France, involving defective procedures.

6.11 UNKNOWN ROOT CAUSE

5.1.39.1 - 5.1.39.7 - Four events in 1989 and two in 1988 could not be traced to a specific root cause.

5.2.24.1 - See 5.2.1.10 Vogtle-1 (1990) - A new event caused by miscommunication.

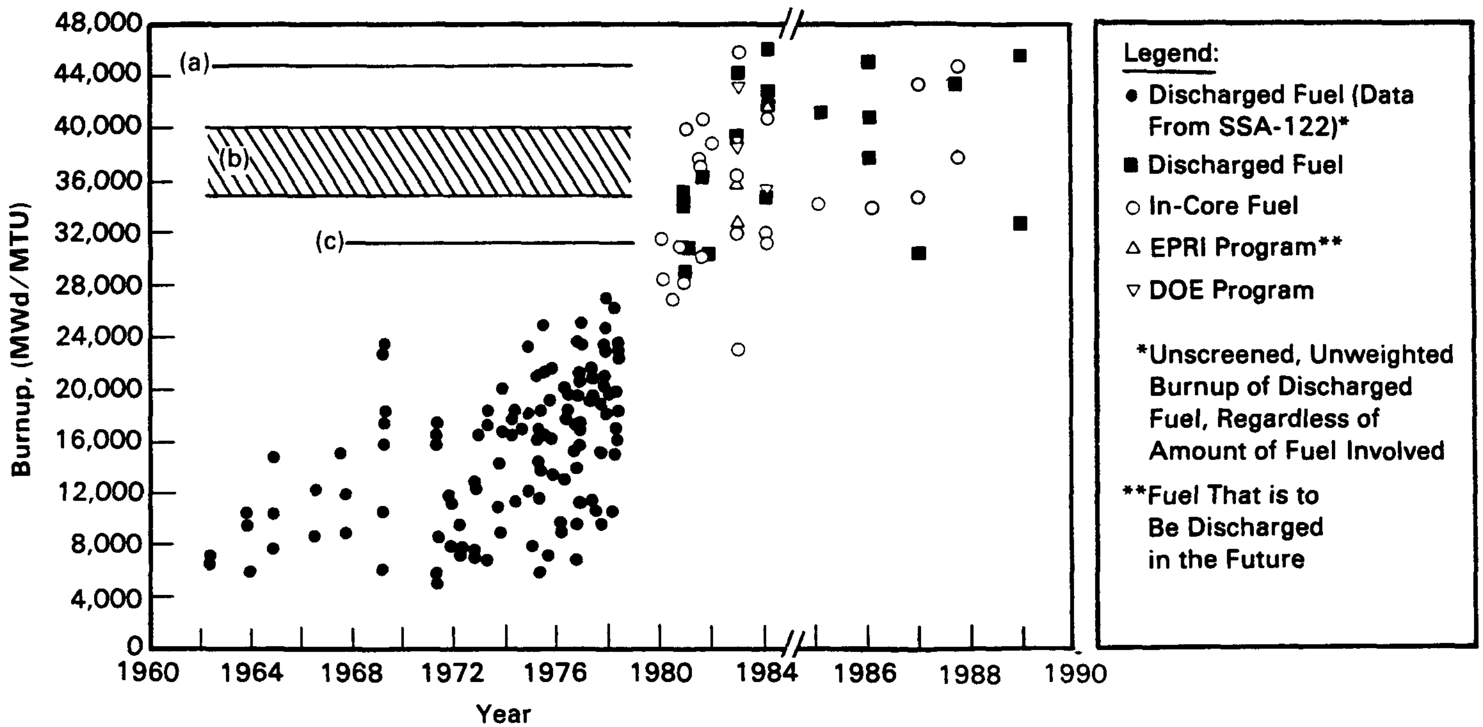
7.0 SUMMARY OF HIGH-BURNUP FUEL EXPERIENCE

An historic perspective of burnup experience in the United States for BWR fuel is given in Figure 7 and for PWR fuel in Figure 8. Most of the early data for Figures 7 and 8 were obtained from Reference 320.

The goals for burnup have been reflected in the fuel reload order burnup warranties; in 1984 the warranty for average batch burnup for BWR fuel was 31 GWd/MTU and for PWR fuel was 36 GWd/MTU.[321] Burnup goals have increased from the DOE goals of 45 GWd/MTU for BWRs and 50 GWd/MTU for PWRs initiated in 1978[322,323] to the EPRI goal of 60 GWd/MTU by 1997.[84] The peak rod-average burnup is generally 5 to 10 GWd/MTU higher than the batch-average burnup levels. Burnup increases are being spurred by the trend toward 18- and 24-month reactor operating cycles in place of annual cycles, so that earlier predictions of stepped burnup increases are being outpaced.[86,322,324-336]

A summary of the 1989 high burnup achievements of the five domestic fuel vendors is given in the table below.[1-5]

Vendor	Plant or Test	Type	Burnup (GWd/MTU)	Comment
ANF	Tihange-1, Belgium	PWR	50.0	highest to date
	Big Rock Point	BWR	41.0	highest to date
	D.C. Cook, 17x17	PWR	44.0	discharged 1989
	Gundremmingen-3, FRG 9x9	BWR	40.0	discharged 1989
BWFC	Mark GdB, LTA	PWR	58.3	UO_2-Gd_2O_3
	Mark BZ, LTA, 15x15	PWR	58.3	Zirc -4 grids
C-E	ANO-1	PWR	43.0	discharged 1989
	St. Lucie-2	PWR	42.0	discharged 1989
	702 rods discharged		56-59.9	highest to date
GE		BWR	>45	bundle average
		BWR	60	peak pellet exp.
W	Zion-1 & -2	PWR	55	4 assemblies ave., 5 cycles
	North Anna-1	PWR	58.4	lead assembly ave. 4 18-mo cycles
	North Anna-1	PWR	<60.0	lead fuel rod ave.

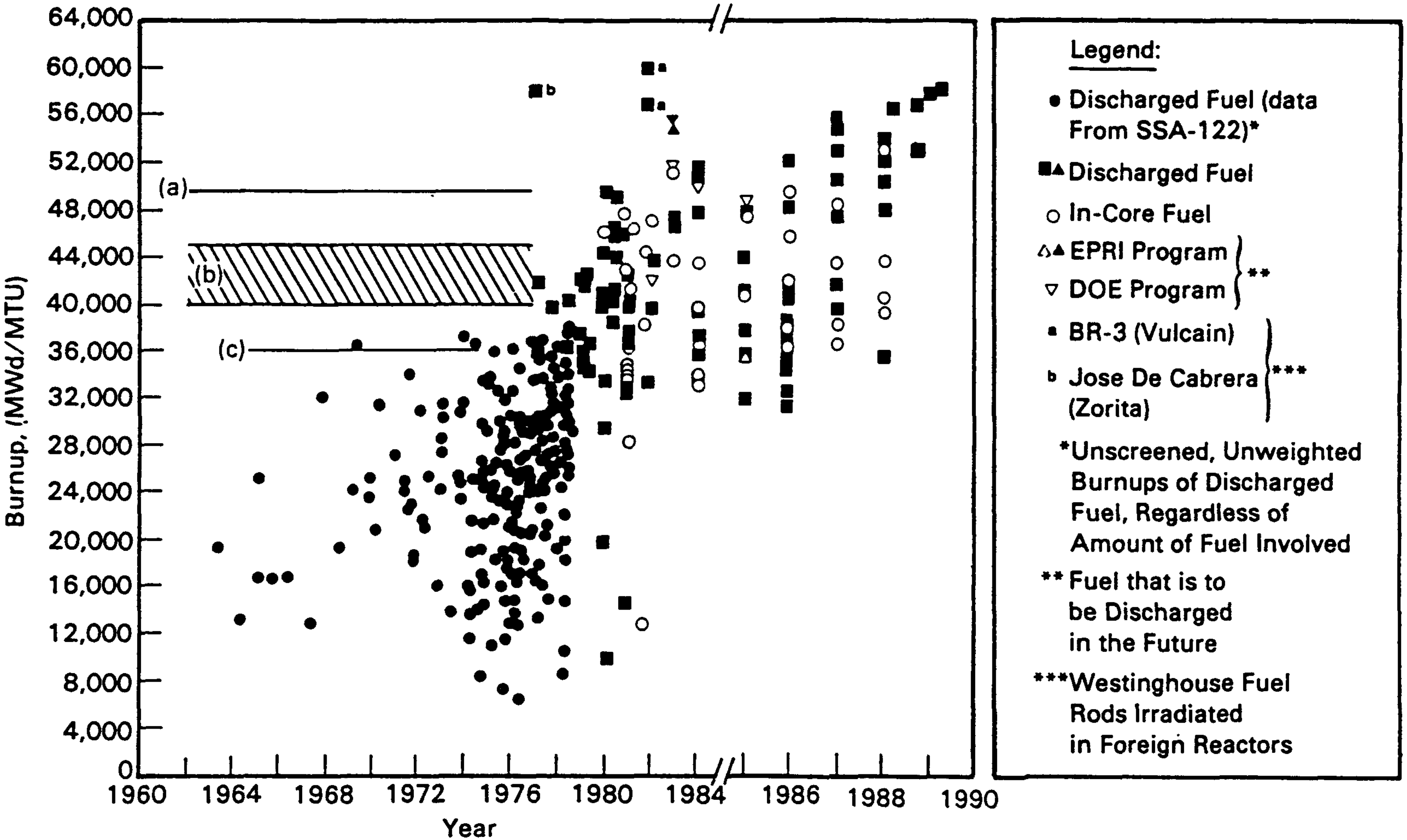


(a) EXTENDED BURNUP GOAL (BATCH AVERAGE BURNUP)

(b) EXTENDED BURNUP (BATCH AVERAGE) GENERIC APPROVALS BY NRC, 1985-86

(c) 1984 PERFORMANCE SPECIFICATION

FIGURE 7. Domestic BWR Fuel Burnup Experience



(a) EXTENDED BURNUP GOAL (BATCH AVERAGE BURNUP)

(b) EXTENDED BURNUP (BATCH AVERAGE) GENERIC APPROVALS BY NRC, 1985-86

(c) 1984 PERFORMANCE SPECIFICATION

FIGURE 8. Domestic PWR Fuel Burnup Experience

Burnup statistics for fuel assemblies discharged from U.S. BWRs and PWRs are summarized below.[327]

BWRs

In 1989 alone:

Number of assemblies discharged during 1989:		4,101
# of assemblies discharged with burnup	>25 GWd/MTU	1,761
	>30 GWd/MTU	498
	>35 GWd/MTU	20
Average burnup in GWd/MTU		21.5

Cumulative since 1968:

Total assemblies discharged since 1968		41,681
# of assemblies discharged with burnup	>25 GWd/MTU	15,249
	>30 GWd/MTU	2,253
	>35 GWd/MTU	38
	>40 GWd/MTU	7
Average burnup in GWd/MTU		21.0

PWRs

In 1989 alone:

Number of assemblies discharged during 1989:		2,869
# of assemblies discharged with burnup	>25 GWd/MTU	2,360
	>35 GWd/MTU	1,434
	>45 GWd/MTU	38
	50-60 GWd/MTU	1
Average burnup in GWd/MTU		32.3

Cumulative since 1968:

Total assemblies discharged since 1968		28,691
# of assemblies discharged with burnup	>25 GWd/MTU	22,072
	>35 GWd/MTU	6,444
	>45 GWd/MTU	69
Average burnup in GWd/MTU		29.1

Statistics which include assemblies from other countries, with references to various sources through 1988, are as follows (taken from the 1988 Fuel Performance Report[19]):

BWRs

Cumulative average burnup through 1988:

Assemblies attaining burnups	>25 GWd/MTU	>12,200
Fuel rods attaining burnups	>36.0 GWd/MTU	>6,000
	40-42 GWd/MTU	>350

PWRs

Cumulative average burnup through 1988:

		Assemblies	Rods (99% Zr-clad)
Attaining burnups	>36 GWd/MTU	~5,250	~1.08 million
	>40 GWd/MTU	~1,050	~172,270
	>48 GWd/MTU	~21	4,745
	>52 GWd/MTU	>15	3,694
	>55 GWd/MTU		1,056
	>56 GWd/MTU	10	
	>58 GWd/MTU	4	

Individual PWR rod irradiations have attained rod average burnups as high as 61.5 GWd/MTU.[323] Concerns regarding the possible effects of extended burnup are discussed in the following paragraphs.

An International Topical Meeting on LWR Fuel Performance was held April 21-24, 1991 in Avignon France; this meeting was jointly sponsored by the American Nuclear Society (ANS) and the European Nuclear Society (ENS). Presented at this meeting were the most recent high burnup fuel experience in the U.S. and several foreign countries. Some of the high burnup issues discussed during this meeting were cladding growth, fission gas release, fuel rod and assembly growth, channel box bow, decrease in fuel thermal conductivity, and the fuel rim effect. The papers presented during this meeting will be discussed further in future fuel operating experience reports.

A number of vendor publications reporting on inspection of the irradiated fuel provide valuable information on the effects and possible ramifications of the higher burnup levels. Summaries of several of these reports and articles, taken from the Fuel Performance Report for 1988,[19] are presented in the paragraphs which follow. The reports and articles are discussed in reverse chronological order because extended burnup experience, goals and issues have changed with time.

A 1989 article[330] indicates that high burnup fuel from two C-E reactors, Arkansas-2 and Calvert Cliffs-1, is to be examined in hot cells at Atomic Energy of Canada Limited's (AECL) Chalk River Nuclear Laboratories. The

19 fuel rods from Arkansas-2 have burnups in the range of 28,100 to 58,100 MWd/MTU; the 12 fuel rods from Calvert Cliffs-1 have burnup values in the range of 47,900 to 59,000 MWd/MTU. After the examinations, AECL will dispose of the fuel by burying it at Chalk River.

A 1988 Westinghouse paper[46] includes information on the variation of fuel rod growth with fluence, the variation of fuel assembly growth with burnup, and a comparison of the distribution of cladding corrosion after four and five cycles of operation.

A 1988 article[347] indicates that the current wave of high burnup fuel assemblies being discharged from reactors will virtually end the in-reactor testing of high burnup fuel by vendors and utilities. Funding by the U.S. for extended burnup research has dropped off dramatically in the past few years. DOE's annual funding fell to zero several years ago; it had reached a high of $16 million in 1979.

Licensing of fuel for extended-burnup operation is discussed by the NRC in a 1987 paper;[337] in that paper it is indicated that irradiation experience to date with extended-burnup fuel has revealed no evidence of degradation of fuel safety or performance for burnups to the NRC-approved levels. The NRC has generically approved batch-average burnups of 35,000 to 40,000 MWd/MTU and 40,000 to 45,000 MWd/MTU for BWRs and PWRs, respectively.[337] The regulatory perspective on extended burnup fuel is discussed in a 1982 paper.[345] The NRC has reviewed vendor topical reports that address extended burnup experience, methods and test data.[17]

A 1987 Westinghouse report[328] documents the results of post-irradiation examinations of two PWR fuel assemblies that were irradiated for five cycles. One was an optimized fuel demonstration assembly that attained a burnup value of 52,774 MWD/MTU. The other was a standard fuel assembly that achieved a burnup value of 52,100 MWd/MTU. Visual inspections showed that the fuel assemblies were in good mechanical condition with no evidence of deterioration. High oxide thicknesses, which did not impair the cladding, were noted on several "white" rods, but that corrosion behavior is not considered by Westinghouse to be representative of typical behavior. The anomalous behavior is believed by Westinghouse to be plant- or region-related.

Projected benefits to the LWR fuel cycle from extended burnup are discussed by DOE in papers[322,340-343] and an article.[344] The effects on LWR fuel cycles of DOE-sponsored development in extending fuel burnup are discussed in a 1986 paper.[323] Improvements in fuel utilization and performance are described by DOE in a 1987 paper.[322]

Some concerns with extended-burnup fuel are noted in three 1987 papers.[337-339] Two 1986 papers[335,336] and Table A.2 (in Appendix A) indicate that extending fuel burnup has not had an obvious detrimental effect on fuel behavior.

A Babcock & Wilcox report[331] issued in October 1986 includes the examination results for 16 fuel rods from a PWR fuel assembly irradiated in

Oconee-1 to an assembly average burnup level of 50,160 MWd/MTU. In general, B&W concludes that the test program confirmed the soundness of the Mark B fuel assembly design. B&W indicated that analysis of the postirradiation examinations through 50,000 MWd/MTU identified two areas of concern for routinely irradiating batches of fuel assemblies to this burnup level. These concerns are a) the pellet-cladding reaction zone (and its effect on fuel rod growth and performance) and b) cladding waterside oxidation and hydriding conditions (and their effects on cladding ductility). B&W concludes that additional data in these areas are needed to model and predict confidently their extended burnup behavior. To alleviate these concerns, B&W suggested design changes (e.g., increase fuel assembly length or decrease fuel rod lengths, use thicker cladding) that could be made. B&W indicated that advanced cladding may also reduce these concerns and that development and investigations in this area should be continued.

A Combustion Engineering report[329] issued in September 1986 contains examination results for 12 fuel rods from a PWR fuel assembly irradiated in Fort Calhoun. The 12 rods have rod-average burnup values ranging from 49,700 MWd/MTU to 55,700 MWd/MTU. Combustion Engineering drew the following conclusions from the examinations: 1) the fission gas release remains below 2% for burnup values up to 56,000 MWd/MTU and does not exhibit a pronounced dependence on burnup up to that level, 2) an interaction layer has formed at the pellet-cladding interface as a result of long residence time with interfacial pressure between the pellets and the cladding, 3) the pellet-cladding interaction has enhanced fuel rod growth at high burnup, and 4) the ductility retained by the cladding shows a significant decrease when local fuel burnups are greater than 55,000 MWd/MTU.

A good data base exists for most fuel performance parameters for burnups up to 34,000 MWd/MTU and 47,000 MWd/MTU for BWR fuel and PWR fuel, respectively.[346] An exception is the irradiation-induced growth of Zircaloy-clad fuel rods and the associated fuel assemblies.

The fuel rod failure rate is not expected to increase as burnup is extended.[333,334] Results from DOE programs aimed at increasing fuel burnup indicate that no significant unexpected phenomena or trends that would limit burnups to lower values have been encountered in the burnup ranges studied with fuel of both traditional and advanced designs.[333] The programs have benefitted the nuclear industry by producing good design and licensing data on fuel that has been irradiated to high burnup levels. In the United States, the only known case of failure of fuel operating in the extended burnup range occurred in a core that had many debris-induced failures of fuel of traditional design.[333]

Babcock & Wilcox stated in an October 1985 article[332] that they observed a drastic loss of cladding ductility in one Oconee (PWR) fuel assembly that had attained a burnup value of 50,600 MWd/MTU; the loss occurred during the last (fifth) cycle of irradiation. However, the ductility of the cladding was still found to be acceptable at this burnup level. Application of high-burnup experience to rod consolidation is described by B&W in a 1986 paper.[333] It is indicated in another B&W paper[334] that the oxide thickness buildup is projected to be very sensitive to the temperature of the cladding surface. Hence, operation late in life of fuel rods at high linear heat generation rates may be restrictive.

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APPENDIX A

HISTORICAL BACKGROUND ON FUEL ROD RELIABILITY

APPENDIX A

HISTORICAL BACKGROUND ON FUEL RELIABILITY

This appendix consists of a table detailing historical information on the reliability of LWR Zircaloy-clad and stainless steel-clad fuel.

Fuel Vendor:

- Advanced Nuclear Fuels Corp. (ANF) [previously Exxon Nuclear Company, Inc. (ENC)]

Year	Annual Fuel Rod Reliability, %[a]	Comment	Ref. No.
1989	99.997		1
1988	>99.994		349
1987	≥99.995[b,c]	See Table 5.	350
1986	99.995[b,c]		351
1985	99.994[b,c]		16
1984	99.995[b,c]		15
1983	99.998,[b,c] 99.87[b,d]		14
1982	99.998[b,c]		13
1981	99.998,[b,c] 99.987[b,d]		12
1980	100		11
1979	(e)		10
1978	(f)		9

- B&W Fuel Company (BWFC) [Babcock & Wilcox (B&W)]

Year	Annual Fuel Rod Reliability, %	Fuel Failure Index	Ref. No.
1989	99.997 (98.6[g])		2
1988	99.99 (99.98[g])		353
1987	99.98 (99.98[g])		354
1986	99.999 (99.998[g])		355
1985	99.995 (99.997[g])	See Figure 1 in Ref. 335	16
1984	99.990 (100[g])	"	15
1983	99.991 (100[g])	"	14
1982	99.994	"	13
1981	99.992	"	12
1980	99.997	"	11
1979	~99.97	"	10
1978	99.9 to 99.99	"	9

- B&W Fuel Company (BWFC) [Babcock & Wilcox (B&W)] (contd)

Year	Annual Fuel Rod Reliability, %	Fuel Failure Index	Ref. No.
1977	--	"	
1976	--	"	
1975	--	"	
1974	--	"	

- Combustion Engineering, Inc. (C-E)

Year	Annual Fuel Reliability, %	Defect Level, %	Ref. No.
1989	99.997[w]		3
1988	(h)		356
1987	(h)		357
1986	(h)	--	358
1985	(h)	--	16
1984	99.98		15
1983	99.98	--	14
1982	99.99	--	13
1981	(h)	--	12
1980	(h)	--	11
1979	>99.99	<0.01[i,j]	10, 359
1978	99.99	0.01[j]	9, 359
1977	99.98	0.02[k]	359, 360
1976	99.98	0.02[k]	359, 360
1975	99.97	0.03[k]	359, 360
1974	>99.75	<0.25[k]	359, 360
1973	99.96	0.04[k]	359, 360
1972	99.99	0.01[k]	359, 360
1971	99.99	0.01[k]	359, 360

- General Electric Company (GE)

Year	Annual Fuel Rod Reliability for All 8 x 8 Fuel Types, %	Ref. No.
1989	99.98 for all 8 x 8 fuel types, 1974 to end of 1989	4
1988	>99.97 for all 8 x 8 fuel types	361
1987	>99.99 for all 8 x 8 fuel types	362
1986	>99.99 for all 8 x 8 fuel types	363
	99.994 for nonbarrier designs	364
	>99.999 for barrier designs	364
1985	>99.99 for all 8 x 8 fuel types	16
1984	>99.99 for all 8 x 8 fuel types	15
	100.000[l] for barrier 8 x 8 fuel	15
1983	99.993 and 99.998[m] for all 8 x 8 fuel types	14
1982	>99.98 for all 8 x 8 fuel types	13

- General Electric Company (GE) (contd)

Year	Annual Fuel Rod Reliability for All 8 x 8 Fuel Types, %	Ref. No.
1981	>99.98 for all 8 x 8 fuel types	12
1980	>99.98 for all 8 x 8 fuel types	11
1979	99.984 for 8 x 8 fuel	10
	99.998 for 8 x 8R plus 8 x 8 R(PP) fuel types[n]	10
1971-1978	See Table A.1 in Appendix of Reference 13	9, 13

- Westinghouse Electric Corporation (W)

Year	Annual Fuel Rod Reliability %	Cladding Defect Level, % of Rods	Average Coolant Activity Level,(p) µCi/g	Fourth Quarter % of Design Basis Activity Release Rate(p,q) in W-Fueled Reactors	Range of Maximum Iodine-131 Activity In Primary Coolant in W-Fueled Reactors, µCi/g	Ref. No.
1989	99.994	0.006	0.0047	--	See Table 16 and Figure 6 in this report	5
1988	99.994	0.006	0.0049	--	See Table 17 and Fig. 6 in this 1988 report	45
1987	99.994[d]	0.006	0.0045	--	See Table 17 and Figure 6 in Reference 113	113
1986	(o)	--	0.0060,[r] 0.0070[s]	--	<0.001 to ≤0.100	114
1985	(o)	--	0.0086,[r] 0.0092[s]	--	<0.001 to 0.100[t]	16
1984	(o)	--	0.008	--	0.008 to 0.121, <0.001 to 0.1[u]	15
1983	(o)	--	0.030	--	0.0001 to 0.102, <0.001 to 0.3[u]	14
1982	(o)	--	0.0296,[r] 0.041;[s]	--	0.0005 to 0.105, <0.001 to 0.3[u]	13
1981	(o)	--	--	<0.001 to 6.38	<0.001 to 0.3[u]	12
1980	(o)	--	--	<0.01 to 4.2	<0.001 to 0.1[u]	11
1979	~99.983	~0.017	--	--	--	10
1978	(o)	--	--	--	<0.001 to 0.1[u]	9
1977	99.938 to 99.9999	0.0001 to 0.062	--	--	--	9
1976	(o)	--	--	0.05 to 5.2[v]	<0.001 to 0.3[u]	
1975	99.75 to 100.00[v]	0.00 to 0.25[v]	--	0.0 to 15[v]	--	

- Westinghouse Electric Corporation (W) (contd)

Year	Annual Fuel Rod Reliability %	Cladding Defect Level, % of Rods	Average Coolant Activity Level,(p) μCi/g	Fourth Quarter % of Design Basis Activity Release Rate(p,q) in W-Fueled Reactors	Range of Maximum Iodine-131 Activity In Primary Coolant in W-Fueled Reactors, μCi/g	Ref. No.
1974	99.790 to 99.999[v]	0.001 to 0.210[v]	--	0.1 to 21[v]	<0.001 to 1.0[u]	
1973	99.91 to 99.999[v]	0.001 to 0.09[v]	--	0.1 to 2.8[v]	--	
1972	99.74 to 100[v]	0 to 0.26[v]	--	0.1 to 6.0[v]	<0.001 to 0.3[u]	
1971	99.23 to 100[v]	0 to 0.77[v]	--	0.1 to 22[v]	--	
1970	99.24 to 99.999[v]	0.001 to 0.76[v]	--	0.1 to 76[v]	--	
1969	99.64 to 100[v]	0 to 0.36[v]	--	0.0 to 36[v]	--	
1968	--	--	--	--	--	

(a) See references for reliability of BWR and PWR fuel rods, respectively.
(b) On a cumulative basis.
(c) The fuel reliability value is based on fuel failures that were judged to be from fuel-related or unknown causes and were 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).
(d) The fuel reliability value is based on fuel failures from all causes.
(e) Annual fuel reliability not stated (9 BWR fuel rods and 4 PWR fuel rods were reported as failed). As of December 1979, ANF (previously ENC) had 2190 fuel assemblies in domestic and foreign plants.
(f) Annual fuel reliability not stated (7 BWR fuel rods failed and 1 or 2 PWR fuel rods may have failed). As of November 1978, ANF (previously ENC) had 1342 fuel assemblies in domestic plants.
(g) Reliability of stainless steel-clad fuel.
(h) Annual fuel rod reliability of fuel rods not stated by C-E, but they provided data on coolant activity. In their input[358] for the annual report for 1988, C-E indicates that the overall fuel rod reliability of their fuel fabricated since 1984 is estimated to be 99.997%, excluding failures caused by debris-induced fretting wear and by baffle jetting in the Yankee Rowe plant (an older Westinghouse plant).
(i) As of February 1, 1979.
(j) See Figure 1 in Reference 359.
(k) See Figure 1 in Reference 359 and Figure 1 in Reference 360.
(l) Based on 1983 data.[15]
(m) Reliability of 8 x 8 fuel if fuel failures involving crud-induced localized corrosion (CILC) are excluded.
(n) R = retrofit design, PP = prepressurized.
(o) Westinghouse did not state a fuel rod reliability (integrity) value. Westinghouse continues to evaluate fuel performance in terms of coolant activity level.
(p) In Revision 5 of WCAP-8183,[365] Westinghouse reported that, starting June 30, 1976, they were reporting fuel performance in terms of coolant activity level. Westinghouse indicated that the prior concept of a "cladding defect level" implies that all defects introduce activity into the coolant at the same rate; however, leak rates of defected rods can decrease (or increase) as a function of time. Hence, Westinghouse decided to abandon reporting of reactor core condition in terms of a number of defects and started reporting activity of iodine-131 in the coolant as a percentage of the coolant design basis activity. In Revision 9 of WCAP-8183,[366] Westinghouse states that "the coolant design basis activity varies somewhat from plant to plant depending upon such factors as reactor power and coolant purification flow rate; however, a value of approximately 2 μCi of iodine-131 per gram of coolant water can be used for purposes of comparison. Since the coolant design basis activity was based on an inferred 1-percent defect level, the new basis of reporting (activity) produces a number approximately 100 times larger than the previous basis (inferred defects). That is, 1 percent of design basis activity would previously have been reported as 0.01 percent defected rods." Starting in 1982, Westinghouse provided data on average coolant activity level (also maximum iodine-131 activity in the primary coolant for each Westinghouse-fueled reactor) in terms of μCi/g.

- Westinghouse Electric Corporation (W) (contd)

Year	Annual Fuel Rod Reliability %	Cladding Defect Level, % of Rods	Average Coolant Activity Level,(p) µCi/g	Fourth Quarter % of Design Basis Activity Release Rate(p,q) in W-Fueled Reactors	Range of Maximum Iodine-131 Activity In Primary Coolant in W-Fueled Reactors, µCi/g	Ref. No.

Footnotes continued

(q) Activity release rate calculated from coolant activity averaged over the quarter and presented as percent of that iodine-131 release rate which establishes the basis for design of plant shielding and coolant cleanup system equipment.
(r) Excludes fuel failures due to baffle jetting.
(s) Includes fuel failures due to baffle jetting.
(t) See Figures 8 and 9 in Reference 16, Figure A.3 in Reference 13, and Reference 367.
(u) See Figure 9 in Reference 16.
(v) The range of values noted in WCAP-8183, Revisions 1-6, for individual plants in all four quarters of the given year is shown. For an idea of the average annual fuel reliability (or defect level), see Figure 9 in Reference 16.
(w) Excluding failures caused by debris induced fretting wear.

APPENDIX B

5.0 PROBLEMS OBSERVED DURING 1989

APPENDIX B

5.0 PROBLEMS OBSERVED DURING 1989

INDEX FOR APPENDIX B

5.1.14 Control Rod Operation

There were three events (two in 1988, one in 1987) at U.S. plants and one event in 1988 at a plant in a foreign country that involved control rod operation. The events are described below.

5.1.14.1 Davis Besse-1

During reactor startup (plant at 2% power) at Davis Besse on December 18, 1988, a personnel error during maintenance troubleshooting caused a group of control rods to drop into the reactor core, which stopped the nuclear chain reaction. Control room personnel erroneously believed that the nuclear reactor was continuing and resumed startup procedures by withdrawing control rods that had previously dropped into the core. Technical aspects of the incident were not of high safety significance, but the violations of NRC requirements were indicative of a significant breakdown in control of NRC-licensed activities in the control room. Information on this 1988 event was published in 1989.[192]

5.1.14.2 Palo Verde-1

At Palo Verde-1 on November 5, 1988, a control element assembly (CEA) slipped approximately 10 inches below the other CEAs in its group during performance of a surveillance test. Under the conditions existing at the time of the event, Action 6 of Technical Specification 3.3.1 requires that each CEA be aligned within 6.6 inches of all other CEAs in its group. Therefore, the unit was in a condition outside the action statement when the CEA slipped (cause: apparent intermittent ground on lower gripper coil). The cause of the unit being in a condition outside the action statement was an inappropriate deletion of action requirements made during a revision of the Technical Specifications. Information on this 1988 event was published in 1989.[193]

5.1.14.3 River Bend

Information was published in 1989[194] concerning an event on June 15, 1987, at River Bend in which it was discovered that a valve that supplies cooling water to one of the control rod drives was mispositioned. The utility also discovered that 18 lock wires on hydraulic control units attached to several of the valves were missing. The reactor was cleared for restart after the utility tested and fully inserted each control rod and also did a complete review to verify that all safety-related valves were in proper position.

5.1.14.4 Switzerland

A loose control rod coupling was discovered in 1988 at the Leibstadt BWR in Switzerland.[195] It was the only Class A incident at a power reactor in Switzerland in 1988. The federal nuclear safety inspectorate, HSK, said this defect, if not detected and repaired, could have led to a control rod drop, but added that the reactor was designed to handle such an event.

5.1.15 Control Rod System Failure/Malfunction

There were eight events at U.S. plants and one in a foreign plant in 1989, and one event in 1988 at a plant in a foreign country that involved failure or malfunction of a control rod system. The events are described below.

5.1.15.1 Dresden-2

On January 30, 1989, it was found during an outage at Dresden-2 that some control rod drive (CRD) hydraulic control unit charging header ball check valves were inoperative (some check valve balls were missing).[196] Due to a concern that the degraded ball check valves could have resulted in a failure to insert or a slower scram insertion time for the affected CRDs at low reactor pressures, a 10 CFR 50.72 notification was made. Safety significance was mitigated by the fact that scram insertion times would have been unaffected under normal operation conditions. The cause of the missing check valve balls was determined to be a procedural deficiency.

5.1.15.2 Fort St. Vrain

At Fort St. Vrain on April 27, 1989, the control rod pair in Region 3 failed to scram during a scram surveillance test.[197] The event was caused by excessive shims (a result of a measurement error during refurbishment) in the control rod drive gear train, which resulted in excessive loading of the first stage bearing.

5.1.15.3 McGuire-2

At McGuire-2 on March 3, 1989, during a routine rod cluster control assembly test, a reactor trip occurred because control rods dropped into the core.[198] This event was assigned a cause of unknown because it could not be determined during the course of the investigation what caused the control rods to drop into the core.

5.1.15.4 Perry-1

Operational Condition 2, startup, was completed at Perry-1 on July 23, 1989, with a control rod inoperable (it was untrippable as a result of an improper valve lineup).[199] Causes of the event were procedural deficiency and personnel error.

5.1.15.5 Prairie Island-2

In scrams on December 21 and 26, 1989, at Prairie Island-2, the electrically energized latches that secure the reactor's array of control rods in their raised position lost power, allowing the rods to drop and disturbing the reactor geometry by disrupting neutron production in one sector.[200] The cause of the event essentially remains a mystery--so far without generic implications. The NRC's findings so far are "not totally clear" on the root cause of the reactor trips; the NRC has stated that "It was some sort of intermittent short in the control rod circuitry."

5.1.15.6 Turkey Point-4

On September 15, 1989, at Turkey Point-4, the rod control system, which is designed to lessen primary system temperature by lowering control rods a step at a time, began to insert the rods automatically but stopped after only four steps of insertion.[201] The operators tried to manually insert the control rods with the control system, but were unsuccessful and manually tripped the reactor.

5.1.15.7 Yankee Rowe

During normal operation at 100% power at Yankee Rowe, the control room operator observed on April 23, 1989, that the Group C control rods could not be moved (cause: inoperable control rod cam motor).[202] After troubleshooting, the control rods dropped into the core. The cause of the inoperable cam motor was a broken compression connector on the motor's brake solenoid circuit. No abnormalities were found in the control rod's circuits and components. A root cause for the event could not be positively determined.

5.1.15.8 France

It is indicated in a 1989 article[203] that on December 24, 1988, 2 of the 53 control rod clusters at France's Blayais-4 were blocked. It was found that guide cards (thin metal fins that help guide the control rods within the guide tube) were damaged on three tubes, probably due to their being hit during introduction of control rod stems. The three tubes were replaced.

5.1.15.9 France

On April 1, 1989, one of 53 control rod clusters at Gravelines-4 failed to drop completely into the core. The unit was restarted on May 13 after repair and inspection.[204]

5.1.15.10 Perry-1

The Cleveland Electric Illuminating Company failed to follow proper procedure when they neglected to determine the cause of the failure of two control rods to pass the insertion time-test in July and again in November 1989. Instead, in July they retested both rods, which passed the second time and remained in use. In November, one rod passed after repeated trials and remained in use; the other failed to pass and was declared inoperable.[205]

5.1.16 Control Blade/Rod Wear

There was one event and one item of interest in 1989 pertaining to control blade or rod wear at U.S. plants. The event and item of interest are discussed below.

5.1.16.1 Braidwood-1 and -2

See Section 5.1.5.1 for details on this item of interest.

5.1.16.2 Monticello

A longer-than-usual outage for this 580-MW BWR, which ended November 9, 1989 after 83 days, was needed because of heavier than normal work load, due in turn to additional wear on equipment during the longer operating cycle. The refueling activities included removing nine core blades because of wear and moving 30 others to maximize their useful lives.[215]

5.1.17 Control Rod System Swelling/Wear/Corrosion/Cracking

There were six events (one in 1989, three in 1988, one in 1986, and one in 1984) at U.S. plants and two events (one in 1989, one in 1988) at plants in foreign countries that involved swelling, wear, corrosion, or cracking of control rod systems. There were also three items of interest on those subjects in 1989. The events and items of interest are described below.

5.1.17.1 Arnold

On November 20, 1988, transgranular stress corrosion cracking of control rod drive piping was observed at Arnold. Analysis indicated that the cracking was the result of a high concentration of chloride contamination on the external surface of the piping. The source of the chlorides was traced to apparent leaching from electrical cable insulation jackets in a conduit directly above the affected control rod drive piping tube bundle. Information on this 1988 event was published in 1989.[207]

5.1.17.2 Diablo Canyon-1

On February 25, 1988, an unexplained increase in containment airborne radiation was observed at Diablo Canyon-1. Examination revealed that canopy seal welds on four control rod drive mechanism head adapter plugs were leaking. The leaks were initiated at the inside diameter of the canopy and were caused by transgranular stress corrosion cracking. Information on this 1988 event was published in 1989.[208]

5.1.17.3 Palisades

An event occurred at Palisades on December 17, 1986.[209,210] Updated information was issued by the licensee in December 1988 and published by the NRC in 1989.[211] Initially, three control rod drive seal housings were found to be cracked. In September 1988, 11 more cracked housings were found. It is postulated that the crack indications are the result of contaminant-induced transgranular stress corrosion cracking. Studies indicate that it would take approximately five years for a 0.030-inch initial depth crack to propagate through the entire housing wall.

5.1.17.4 Pilgrim

In 1989, the NRC published information from an updated report[212] that was submitted by the licensee regarding control rod drive collet retainer tube weld defects that were found at Pilgrim on August 22, 1984. A control rod drive collet retainer tube had a longitudinal, through-wall crack, which was determined to have been caused by intergranular stress corrosion cracking (IGSCC) in the cold worked and weld-sensitized stainless steel. It was determined that the cracking could not lead to mechanical failure of the control rod drive.

5.1.17.5 Wolf Creek-1

See Section 5.1.2.3 for details on this 1988 event.

5.1.17.6 PWR Primary Water Stress Corrosion Cracking

See Section 5.1.2.6 for details on this 1989 item of interest.

5.1.17.7 France

See Section 5.1.1.12 for details on this 1989 item of interest.

5.1.17.8 France

Electricite de France (EDF) and French safety authorities have agreed on new criteria for replacement of worn control rods, according to a 1989 article.[213] The new criteria, which are more complex than the previous ones, basically require EDF to replace any control rod whose cladding is either pierced through or worn over 20% of its circumference next to the seventh guide plate. EDF estimates that this will lead to replacement of 30 to 35 control rod clusters out of the 53 on each 900-MW PWR. All control rod clusters on 900-MW PWRs are to be changed by the end of 1990. Vibration makes the control rods fret against the guide plates. The origin of the vibration is linked to coolant flow around the rods, but the phenomenon is still not completely understood. According to the French, similar problems have apparently not cropped up at comparable PWRs in other countries, notably the U.S. and Japan, and the French are trying to understand why.

5.1.17.9 France

See Section 5.1.2.9 for details on this 1989 event.

5.1.17.10 Taiwan

It is indicated in a 1989 article[214] and a 1988 memorandum[147] that one control rod could not be fully re-inserted into Taiwan's Maanshan-1 in September 1988. A broken tip on one of the rods was found and the hydriding phenomenon was identified as the cause of the failure. See "1.2.3 Wolf Creek-1."

5.1.17.11 Monticello

See Section 5.1.16.2 for details on this 1989 event.

5.1.18 Control Rod System Installation/Maintenance Error

There were nine events (four in 1989, four in 1988, and one in 1987) at U.S. plants that involved control rod system installation or maintenance errors. Those events are discussed below.

5.1.18.1 Braidwood-1

Lightning-induced voltage transients at Braidwood-1 on July 18, 1989, removed power to various rod drive control cards and allowed numerous control rods to drop.[216] The root cause of the event was inadequate protection and isolation of the rod control system from lightning-induced voltage transients.

5.1.18.2 Brunswick-2

The licensee for Brunswick-2 was cited for a violation (Severity Level III) of the plant's technical specification. On March 8, 1988, with the reactor in Operational Condition 5, a control rod (10-39) was in the fully withdrawn position but the shorting links had not been removed from the circuitry. Information on this 1988 event was published in 1989.[217]

5.1.18.3 Catawba-2

On June 6, 1988, at Catawba-2, a control rod dropped into the core during testing. Cause of the event was failure of a fuse. The fuse that failed in the rod control system circuitry was not the type fuse specified by Westinghouse. The activity that installed the improper fuse could not be identified. Information on this 1988 event was published in 1989.[218]

5.1.18.4 Fort St. Vrain

See Section 5.1.15.2 for details on this 1989 event.

5.1.18.5 Fort St. Vrain

At Fort St. Vrain on March 23, 1988, licensee personnel handled control rods with bare hands. This violation involving mishandling of control rods was due to a combination of errors. Workmen were inattentive to detail and also failed to sufficiently review the work plan. The information on this 1988 event was published in 1989.[219]

5.1.18.6 Limerick-1

It was identified at Limerick-1 on May 6, 1989, that two control rods had their uncoupling rods misaligned.[220] These control rods had been improperly verified as coupled to their drives during the plant's second cycle. During the first refueling outage, the uncoupling rods for the two control rods were misaligned during installation due to personnel error (a procedural inadequacy and design deficiency also contributed to the event).

5.1.18.7 Millstone-1

Four restraining metal straps on control rod drive system hydraulic control units (HCUs) were found missing at Millstone-1 on August 22, 1989.[221] The event is attributed to personnel error. A specific site analysis is not available to determine the capability of the HCUs to remain functional during a seismic event without the straps and thus it is an unanalyzed condition. Straps were reinstalled on the affected HCUs.

5.1.18.8 River Bend

See Section 5.1.14.3 for details on this 1987 event.

5.1.18.9 Turkey Point-3

Turkey Point-3 was cited by the NRC for a violation of Section 5.1.6.1 of ANSI N18.7-1972. The event occurred on January 13, 1988, when maintenance was performed on the control rod system without documented instructions or drawings appropriate to the circumstances. Fuses were removed from the system without a complete understanding of what circuitry the fuses supplied. Consequently, portions of the rod control circuitry for three rods were de-energized while only one rod was thought to be affected. During a plant shutdown, this unexpectedly resulted in multiple rods dropping into the core, requiring a manual reactor trip. Information on this 1988 event was published in 1989.[222]

5.1.19 Control Rod Guide Tube Support Pins

There were six items of interest (including three events) in 1989 at plants in two foreign countries. The items and events are described below.

5.1.19.1 Federal Republic of Germany

Replacement of central pins, the Kraftwerk Union (KWU) counterparts of the Westinghouse-design control rod guide tube split pins, at West Germany's Biblis and Unterweser is to occur over the next two years, according to a 1989 article.[223] The pins must be examined and replaced if any crack indications are found this year.

5.1.19.2 Federal Republic of Germany

All fuel alignment pins at West Germany's Biblis-A were inspected with ultrasound devices, and 22 pins made of Inconel X-750 were replaced with new pins made of austenitic steel.[224] The alignment pins at Grohnde, Grafenrheinfeld, Phillipsburg-2, Obrigheim, and Neckarwestheim-1 in West Germany have been or are to be inspected by Kraftwerk Union (KWU). Pins have been replaced at Obrigheim and Grafenrheinfeld. KWU hopes to be awarded a contract to inspect pins at West Germany's Biblis-B this fall; a contract is also pending for pin inspection at the Borssele PWR in the Netherlands.

5.1.19.3 Federal Republic of Germany

A recent article[225] indicates that 386 fuel alignment pins were ultrasonically examined at Biblis B (PWR) in the Federal Republic of Germany and 67 of the pins were replaced.

5.1.19.4 Federal Republic of Germany

After the recent inspection of all fuel alignment pins at Unterweser (West Germany), 34 pins made of Inconel X-750 were replaced with stainless steel pins.[226] Inspection of fuel alignment pins was required at all West German PWRs following the refueling accident at Brokdorf in 1988.

5.1.19.5 France

See Section 5.1.1.12 for details on this 1989 item of interest.

5.1.19.6 France

A June 1989 article[227] indicates that the guide tube support pins were replaced for the second time at Gravelines-1, a French PWR. A new support pin more resistant to cracking is under development.

5.1.20 Nonconservative Assumptions/Incorrect Data

There were six events (five in 1989, one in 1988) at U.S. plants that involved use of nonconservative assumptions or incorrect data. Those events are discussed below.

5.1.20.1 Cook-2

It was discovered on November 8, 1989, that the rated thermal power was exceeded at Cook-2 because of an incorrect change in blowdown constants in a thermal output computer program.[228]

5.1.20.2 Cook-2

On March 21, 1989, with Cook-2 operating at 100% power (Cook-1 was refueling), it was suspected that a discrepancy existed in the Westinghouse P-250 computer-calculated thermal power value, which, if true, would lead to exceeding the rated thermal power.[229] Investigation revealed that the blowdown-mass enthalpy term was not included in the thermal output program. Worst case analysis indicated a potential 1.8% difference between the P-250 calculated and actual thermal power. It is expected that detailed evaluation of Cook-1, following startup, will similarly reduce the actual power deviation. Changes have been or will be made to ensure that the plants operate conservatively.

5.1.20.3 Haddam Neck

At Haddam Neck (also known as Connecticut Yankee) on April 25, 1989, a discrepancy was discovered in the Design Basis Large-Break Loss-of-Coolant Accident (LBLOCA) analysis.[230] A nonconservative reactor vessel lower plenum volume was used in the Interim Acceptance Criteria (IAC) model. The result of the error would be a peak cladding temperature above the 2300°F IAC limit. The immediate corrective action included implementation of administrative controls to reduce the plant's Technical Specification limit on linear heat generation rate and a commensurate reduction in the axial offset operating window. The event was reportable per 10 CFR 50.73(A)(2)(V)(D) since a condition existed that alone could have prevented the fulfillment of a safety system to mitigate the consequences of an accident.

5.1.20.4 Hope Creek

Hope Creek was cited for violating its facility operating license by operating the reactor on September 21, 1988, at 101.2% of rated power, nominally, with a worst case of 102.2% of rated power. The higher power resulted from nonconservative calculational errors for feedwater flow transmitters, but the higher power was within the margin assumed in the design basis. Information on this 1988 event was published in 1989.[231]

5.1.20.5 McGuire-1

On June 30, 1989, it was discovered that McGuire-1 operated at greater than 100% thermal power. The event was assigned a cause of inappropriate action (for calibration data, the figure from the Operator Aid Computer was used, but the figure was wrong) with a contributory cause of procedural deficiency.[232]

5.1.20.6 North Anna-1

An input error in the large-break loss-of-coolant (LBLOCA) analysis for the 18% steam generator tube plugging licensing case was discovered at North Anna-1 on August 8, 1989.[233] Results of the reanalysis determined that correction of the error resulted in peak cladding temperatures (PCTs) that exceeded the current licensing basis and the 2200°F limit specified in 10 CFR 50.46. As a corrective action, administrative limits were placed on North Anna-1 operating parameters to reduce the PCT below the 10 CFR 50.46 limit during a large-break LOCA.

5.1.21 Unanalyzed Condition

There were three events (one in 1989, two in 1988) at U.S. plants that involved unanalyzed conditions. Those events are described below.

5.1.21.1 Grand Gulf-1

On September 23, 1988, System Energy Resources, Inc. (SERI) determined that there existed situations during cold shutdown and refueling at Grand

Gulf-1 where certain loads manually handled over irradiated fuel may not have been bounded by analyzed events and may not have been restricted by administrative controls. Further evaluations and administrative controls will be required prior to the next outage. Information on this 1988 event was published in 1989.[234]

5.1.21.2 <u>Haddam Neck</u>

See Section 5.1.20.3 for details on this 1989 event.

5.1.21.3 <u>Nine Mile Point-2</u>

An updated report was submitted by the licensee regarding an event at Nine Mile Point-2 on April 19, 1988; the updated report was published by the NRC in 1989.[235] The reactor was inadvertently operated with greater than 100% of rated reactor core flow (it was later determined that an "indicated" core flow of 100% of rated was an actual core flow of 104.5% of rated). The event was due to a poor electrical connection and resulted in plant operation in an unanalyzed condition.

5.1.22 <u>100% Power Exceeded</u>

There were eight events (three in 1989, five in 1988) at U.S. plants in which 100% of rated power was exceeded. Those events are discussed below.

5.1.22.1 <u>Cook-2</u>

See Section 5.1.20.1 for details on this 1989 event.

5.1.22.2 <u>Cook-2</u>

See Section 5.1.20.2 for details on this 1989 event.

5.1.22.3 <u>Hope Creek</u>

See Section 5.1.20.4 for details on this 1988 event.

5.1.22.4 <u>La Salle-2</u>

On March 9, 1988, two pumps that recirculate water through La Salle-2's reactor vessel automatically shut down when the plant was being operated at about 85% power. As a result, the power level increased to a peak of 118% of the reactor's rated power before the reactor automatically shut down. Information on this 1988 event was published in 1989.[236]

5.1.22.5 <u>McGuire-1</u>

See Section 5.1.20.5 for details on this 1989 event.

5.1.22.6 San Onofre-2

It was determined on December 16, 1988, that San Onofre-2 operated in excess of 102% of rated power from December 23, 1983, to January 4, 1984, because of a manufacturing defect in a feedwater flow venturi. Information on this event was published in 1989.[237]

5.1.22.7 San Onofre-2

On October 31, 1988, it was determined that San Onofre-2 had been operated at an estimated actual power slightly in excess of 100% for a portion of the time between August 27 and October 21, 1988. During that period, however, the plant was never continually operated at greater than 100% indicated power nor was it operated at an estimated actual power in excess of 102%, thus preserving the initial power assumption utilized in the safety analyses. Cause of the event was attributed to a decrease (several factors involved) in indicated plant power relative to actual plant power. Information on this 1988 event was published in 1989.[238]

5.1.22.8 Susquehanna-2

Inadvertent reactor core isolation cooling initiation and injection occurred at Susquehanna-2 on December 15, 1988. Reactor power increased to 101% during the event. No degradation of fuel was evident. The event was the result of a pressure transient due to mispositioning of an isolation valve by a technician. Information on this 1988 event was published in 1989.[239]

5.1.23 Other Power Limit Exceeded

There was one event in 1989 at a U.S. plant in which an intermediate power limit was exceeded. That event is described below.

5.1.23.1 Fort St. Vrain

On June 23, 1989, it was discovered that the reheat steam attemperation flow had not been accounted for in the secondary heat balance calculation of reactor power.[240] The licensee took immediate action and updated the secondary heat balance calculation. Reactor power was found to be 83.4%, which exceeded Fort St. Vrain's maximum authorized operating limit of 82%. It was also found that the reactor power was in excess of 82% for approximately four hours. The root cause for this event was identified to be inadequate procedures; appropriate procedural changes have been implemented.

5.1.24 Unexpected Power Increase

There was one event in 1988 at a U.S. plant that involved an unexpected power increase. That event is described below.

5.1.24.1 Limerick-1

On April 9, 1988, a controlled shutdown was in progress at Limerick-1; however, approximately three minutes prior to the scram, power began increasing due to the positive reactivity effect of decreasing moderator temperature. The reactor was shut down; there was no release of radioactive material to the environment as a result of the event. The cause of the event was cognitive personnel error. The licensed reactor operator did not adequately anticipate and observe the effects of decreasing moderator temperature on core reactivity. Information on this 1988 event was published in 1989.[241]

5.1.25 100% Core Coolant Flow Exceeded

There was one event in 1988 at a U.S. plant in which 100% core coolant flow was exceeded. That event is described below.

5.1.25.1 Nine Mile Point-2

See Section 5.1.21.3 for details on this 1988 event.

5.1.26 Axial Shape Index

There was one event in 1989 at a U.S. plant involving the axial shape index. That event is discussed below.

5.1.26.1 San Onofre-2

A manual trip was initiated on September 2, 1989, at San Onofre-2 because of the approach of the axial shape index (ASI) to the core protection calculator auxiliary trip setpoint.[242] The ASI describes the axial power distribution of the reactor core. At the end of a fuel cycle, the effect of a decrease in plant power on ASI is greater than at any other time in the cycle. As a result, strict controls must be employed to maintain the ASI within limits and prevent a trip. Although action was taken to control the ASI, it was not sufficient to maintain the ASI within its limits.

5.1.27 Containment Integrity

There were seven events (one in 1990, four in 1989, and one in 1988) at U.S. plants involving breaching of containment integrity. Those events are described below.

5.1.27.1 Cook-2

At Cook-2 on August 29, 1988, it was found that for short periods (typically two to five minutes) an open pathway from the containment atmosphere to the auxiliary building existed during core alteration and fuel movement. The cause of the event was an inadequate procedure. Information on this 1988 event was published in 1989.[243]

5.1.27.2 Farley-1

Core alterations (unlatching of control rod drive mechanisms) were performed on September 30 and October 1, 1989, at Farley-1 without having established containment refueling integrity.[244] This event was caused by cognitive personnel error.

5.1.27.3 Farley

Containment integrity was breached on April 19, 1990, at the Farley plant during the replacement/placement of fuel within the reactor containment.[245] Removal of the bonnet to a valve and some handhole covers created an air path from the containment atmosphere to the outside atmosphere.

5.1.27.4 McGuire-1

An updated report[246] indicates that during fuel unloading operations at McGuire-1 on October 25, 1988, containment integrity was breached when three temporary penetrations were found to be leaking. Fuel movement was suspended until the penetrations were resealed and leak tested.

5.1.27.5 Millstone-3

Fuel building integrity was lost during fuel movement at Millstone-3 on May 24, 1989, due to an open door caused by personnel error.[247] The immediate corrective action was to stop fuel handling until the door could be closed.

5.1.27.6 San Onofre-2

See Section 5.1.11.2 for details on this 1989 event.

5.1.27.7 Zion-1

At Zion-1 on September 21, 1989, it was discovered during refueling operations (the core was being off-loaded) that there was a loss of containment closure.[248] No radioactive release occurred during the time the vent path existed. The event was caused by a combination of procedural deficiency and improper planning.

5.1.28 Containment Airborne Contamination

There was one event in 1988 at a U.S. plant involving an unexplained increase in containment airborne contamination. That event is described below.

5.1.28.1 Diablo Canyon-1

See Section 5.1.17.2 for details on this 1988 event.

5.1.29 "Hot" Particles

There were several items of interest in 1989 pertaining to "hot" particles. Those items are described below.

5.1.29.1 "Hot" Particles

Several papers[219,249-251] presented in 1989 provide information on "hot" particles (i.e., irradiated fuel fragments). Hot-particle issues have been in current focus since the Three Mile Island-2 accident dosimetry highlighted the basic problem.[249] A survey conducted by EPRI[252] indicated that ~70% of the nuclear power plants have had some problem with hot particles. Both BWRs and PWRs have found hot-particle contamination in the plant environment and, occasionally, on radiation workers. A hot particle on the skin produces a very steep dose gradient; the dose drops off very rapidly as the distance from the particle increases. The local absorbed dose produced from a hot particle on the skin may exceed the administrative limit established by the utility and on occasion exceeds the regulatory limit, which results in an overexposure reportable to the NRC. The particles are sometimes called "fleas" because they seem to jump or hop from place to place.

5.1.30 Lowering of Water Level

There were four events (two in 1989, two in 1988) at U.S. plants involving lowering of water levels. Those events are discussed below.

5.1.30.1 Byron-1

See Section 5.1.10.4 for details on this 1988 event.

5.1.30.2 Clinton-1

Unexpected isolation of shutdown cooling and lowering of the water level (by five inches) of the upper containment fuel pools occurred at Clinton-1 on February 3, 1989.[253] Water level in the upper pools dropped below the required level of 23 feet above the reactor pressure vessel flange. The cause of the event was apparently a deficient surveillance test procedure.

5.1.30.3 Clinton-1

During two separate incidents on February 26, 1989, at Clinton-1, the water level in the upper containment fuel pools dropped below the normal level of 23 feet above the top of the reactor pressure vessel flange.[254] This event was attributed to insufficient training of operators on what constitutes "other specified conditions" of Technical Specification 3.0.4.

5.1.30.4 Nine Mile Point-2

On April 13, 1988, power was lost to the feed water pumps at Nine Mile Point-2. The water level above the core boiled off from 183 inches to 108 inches, at which point the high pressure core spray and the reactor core isolation cooling systems were activated. Information on this 1988 event was published in 1989.[255]

5.1.31 Spent Fuel Pool

There were two items of interest in 1989 and one event in 1988 that involved spent fuel pools. The items and the event are described below.

5.1.31.1 Rancho Seco

Rancho Seco is scheduled for defueling, starting November 18, 1989; however, before defueling can commence, crews at the plant are repairing a leak in the spent fuel pool's 5-mm thick stainless steel lining.[256]

5.1.31.2 San Onofre-2

On February 3, 1988, it was noted at San Onofre-2 that the spent fuel handling machine (SFHM) may have been operated over the fuel storage pool while the postaccident cleanup units (PACU) were not operable. This is contrary to the Technical Specification 3.9.12 action statement. Information on this 1988 event was published in 1989.[257]

5.1.31.3 Generic Issue 82: Beyond Design Basis Accident in Spent Fuel Pools

See Section 5.1.2.8 for details on this 1989 item of interest.

5.1.32 Defective Procedure/Training or Management Deficiency

There were 18 events (14 in 1989, 4 in 1988) at U.S. plants involving defective procedures or training or management deficiencies. Those events are discussed below.

5.1.32.1 Clinton-1

See Section 5.1.30.2 for details on this 1989 event.

5.1.32.2 Clinton-1

See Section 5.1.30.3 for details on this 1989 event.

5.1.32.3 Cook-2

See Section 5.1.27.1 for details on this 1988 event.

5.1.32.4 Davis Besse-1

See Section 5.1.11.1 for details on this 1988 event.

5.1.32.5 Dresden-2

See Section 5.1.15.1 for details on this 1989 event.

5.1.32.6 Fort St. Vrain

See Section 5.1.23.1 for details on this 1989 event.

5.1.32.7 Grand Gulf-1

See Section 5.1.21.1 for details on this 1988 event.

5.1.32.8 Harris-1

See Section 5.1.10.6 for details on this 1989 event.

5.1.32.9 Limerick-1

See Section 5.1.7.1 for details on this 1989 event.

5.1.32.10 Limerick-1

See Section 5.1.18.6 for details on this 1989 event.

5.1.32.11 McGuire-1

See Section 5.1.20.5 for details on this 1989 event.

5.1.32.12 McGuire-2

A 15% power reduction was made at McGuire-2 on October 24, 1989, and the required iodine-131 dose equivalent sample was analyzed, but the results were invalid because of insufficient purge time prior to obtaining the sample.[258] This event was attributed to management deficiency and inappropriate action.

5.1.32.13 Palo Verde-1

See Section 1.14.2 for details on this 1988 event.

5.1.32.14 Perry-1

See Section 5.1.15.4 for details on this 1989 event.

5.1.32.15 Quad Cities-1

See Section 5.1.1.2 for details on this 1989 event.

5.1.32.16 San Onofre-2

See Section 5.1.11.2 for details on this 1989 event.

5.1.32.17 Zion-1

See Section 5.1.27.7 for details on this 1989 event.

5.1.32.18 Perry-1

See item under 5.1.15.10 for details on this 1989 event.

5.1.33 Design/Installation/Maintenance Deficiency

There were three events (one in 1989, two in 1988) at U.S. plants involving deficiencies in design, installation, or maintenance. Those events are discussed below.

5.1.33.1 Limerick-1

See Section 5.1.18.6 for details on this 1989 event.

5.1.33.2 Nine Mile Point-2

See Section 5.1.21.3 for details on this 1988 event.

5.1.33.3 Sequoyah-2

In 1989, the NRC published information from an updated report[259] that was submitted by the licensee on an event at Sequoyah-2 on April 7, 1988. The event involved inadequate work control resulting in two emergency core cooling system pumps being inoperable while the reactor was in Mode 3 (hot standby).

5.1.34 Equipment Inoperable/Malfunction

There were five events (two in 1989, three in 1988) at U.S. plants that involved inoperable or malfunctioning equipment. Those events are described below.

5.1.34.1 Palo Verde-2

In 1989, the NRC published information from an updated report[260] that was submitted by the licensee concerning an event on December 7, 1988, at Palo Verde-2 in which a new fuel radiation monitor was found to be inoperable (it indicated zero millirem/hour instead of the actual radiation level adjacent to the new fuel storage racks). The last accurate reading was on December 4, 1988. The event was attributed to a malfunction of the monitor's clock; however, the cause of the failure could not be confirmed.

5.1.34.2 San Onofre-2

See Section 5.1.31.2 for details on this 1988 event.

5.1.34.3 San Onofre-3

While San Onofre-3 was operating at 100% power, with the core operating limit supervisory system (COLSS) having been removed from service for quarterly preventive maintenance, it was discovered on May 2, 1989, that the COLSS backup computer system (CBCS) had failed.[261] No alarms had been received to alert the operators of this condition. With both COLSS and CBCS inoperable, the plant was operating with a departure from nucleate boiling ratio (DNBR) below that allowed by Technical Specification 3.2.4. Investigation determined that the CBCS had failed as a result of a memory error due to an indeterminate cause. The computer failure was not detected due to the absence of a CBCS failure alarm. Design changes have been implemented.

5.1.34.4 Sequoyah-2

See Section 5.1.33.3 for details on this 1988 event.

5.1.34.5 Yankee Rowe

See Section 5.1.15.7 for details on this 1989 event.

5.1.35 Manufacturing Defect

There were three events, one in 1988 and one in 1990 at U.S. plants, and one in 1989 in Canada. These events are described below.

5.1.35.1 San Onofre-2

See Section 5.1.22.6 for details on this 1988 event.

5.1.35.2 Vermont Yankee

See item under 5.1.1.20 for details of this 1990 event.

5.1.35.3 Canada

In Bruce-3, twelve of the unit's 480 fuel-carrying pressure tubes were found to be leaking. These tubes, and all tubes built subsequent to Bruce-3, are of zirconium-niobium alloy, so hydriding was not suspected. They may have had a manufacturing defect or a structural weakness at the rolled joint connection.[307]

5.1.36 Personnel Error

There were 13 events (9 in 1989, 4 in 1988) at U.S. plants that involved personnel errors. Those events are described below.

5.1.36.1 Clinton-1

See Section 5.1.10.5 for details on this 1989 event.

5.1.36.2 Davis Besse-1

See Section 1.14.1 for details on this 1988 event.

5.1.36.3 Farley-1

See Section 5.1.27.2 for details on this 1989 event.

5.1.36.4 Fort St. Vrain

See Section 5.1.18.5 for details on this 1988 event.

5.1.36.5 Limerick-1

See Section 5.1.24.1 for details on this 1988 event.

5.1.36.6 Millstone-1

See Section 5.1.18.7 for details on this 1989 event.

5.1.36.7 Millstone-3

See Section 5.1.27.5 for details on this 1989 event.

5.1.36.8 North Anna-1 and -2

See Section 5.1.7.2 for details on this 1989 event.

5.1.36.9 Oyster Creek

See Section 5.1.9.2 for details on this 1989 event.

5.1.36.10 Oyster Creek

At Oyster Creek on May 15, 1989, a reactor-coolant sample was not taken and analyzed for dose equivalent iodine-131 activity (such activity can be an indicator that failed fuel is present in the core) following a reactor power change of more than 15% as required by the plant's Technical Specifications.[262] The root cause of the event was personnel error. Two contributing factors were also evident.

5.1.36.11 Perry-1

See Section 5.1.15.4 for details on this 1989 event.

5.1.36.12 Quad Cities-1

See Section 5.1.1.2 for details on this 1989 event.

5.1.36.13 Susquehanna-2

See Section 5.1.22.8 for details on this 1988 event.

5.1.37 Personnel Fatigue

There was one event in 1989 at a U.S. plant involving personnel fatigue. That event is discussed below.

5.1.37.1 Limerick-1

See Section 5.1.7.1 for details on this 1989 event.

5.1.38 Procedural Noncompliance

There were 10 events (4 in 1989, 6 in 1988) at U.S. plants involving procedural noncompliance. Those events are described below.

5.1.38.1 Catawba-2

On August 25, 1989, power was reduced more than 15% at Catawba-2 within a one-hour period but the required sample for analysis for iodine (the detection of iodine-131 may indicate the presence of failed fuel) was not taken, which is a violation of the plant's technical specifications.[263]

5.1.38.2 Davis Besse-1

See Section 5.1.12.1 for details on this 1988 event.

5.1.38.3 Diablo Canyon-1

See Section 5.1.12.2 for details on this 1988 event.

5.1.38.4 Diablo Canyon-1

See Section 5.1.12.3 for details on this 1988 event.

5.1.38.5 McGuire-2

See Section 5.1.32.12 for details on this 1989 event.

5.1.38.6 Oyster Creek

See Section 5.1.36.10 for details on this 1989 event.

5.1.38.7 San Onofre-2

See Section 5.1.31.2 for details on this 1988 event.

5.1.38.8 San Onofre-3

See Section 5.1.34.3 for details on this 1989 event.

5.1.38.9 Sequoyah-1

See Section 5.1.1.4 for details on this 1988 event.

5.1.38.10 Turkey Point-3

See Section 5.1.18.9 for details on this 1988 event.

5.1.39 Unknown Cause for Event

There were six events (four in 1989, two in 1988) at U.S. plants and one 1989 item of interest pertaining to events with unknown causes. The events and the item of interest are described below.

5.1.39.1 McGuire-2

See Section 5.1.15.3 for details on this 1989 event.

5.1.39.2 Palisades

See Section 5.1.7.3 for details on this 1988 event.

5.1.39.3 Palo Verde-2

See Section 5.1.34.1 for details on this 1988 event.

5.1.39.4 Prairie Island-2

See Section 5.1.15.5 for details on this 1989 event.

5.1.39.5 San Onofre-3

See Section 5.1.34.3 for details on this 1989 event.

5.1.39.6 Yankee Rowe

See Section 5.1.15.7 for details on this 1989 event.

5.1.39.7 Westinghouse Fuel

See Section 5.1.1.9 for details on this 1989 item of interest.

5.2.6 Control Rod System Failure/Malfunction

There were four events in 1989, one event at a U.S. plant and four events at plants in another country, that involved failure or malfunction of control rod systems. Those events are described below.

5.2.6.1 Fort St. Vrain

Fort St. Vrain (HTGR) was idled for about two years starting in 1984, when 6 of the 37 control rod pairs failed to insert automatically on scram signal.[293,294] On August 17, 1989, during a weekly surveillance test, operators received a "slack cable" alarm for one of the 37 control rod pairs. Although the pair could be withdrawn, the pair could only be reinserted about one-third of the normal rod travel. Then, the pair became stuck outside the reactor. The utility discovered that the head of one of the control rod's Inconel clevis pin bolts had developed a crack, broken off, and wedged itself between the control rod and the guide tube. The bolts had been replaced in 1985.

5.2.6.2 Finland[a]

Finnish officials are checking the possibility that a metallic powder, normally used for sandblasting but found on September 10, 1989, in Olkiluoto-1's control rod drives, was put there by a saboteur during a refueling outage that ended three months ago.[295] Some control rods had jammed on September 7. Cleanup of the 120 control rod drives will be slow going, since only four drives can be effectively cleaned per day.

5.2.6.3 Finland

Additional information has been published on the recent control rod drive failure at Finland's Olkiluoto-1 (BWR) that was caused by metal powder, which may have been introduced deliberately.[296]

5.2.6.4 Finland

Finnish law enforcement officials are continuing to investigate how almost 20 kilograms of a metallic granular powder (0.1-0.5 mm in diameter) used in sandblasting got into the control rod drives at Olkiluoto-1 (BWR).[297] Sabotage has not been ruled out though officials are placing greater emphasis now on non-criminal activities. A comprehensive cleanup of the reactor, rods,

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(a) There are four entries (Sections 2.6.2, 2.6.3, 2.6.4 and 2.6.5) that pertain to the same event.

and rod drive mechanisms was started immediately. Fuel was removed from the core while the core support structure and control rod guide tubes were vacuumed.

5.2.6.5 Finland

The metal powder that fouled control rod drives in September 1989 at Finland's Olkiluoto-1 (BWR) may have been lying in a low-flow area of piping since the plant was started up in 1978.(298) Earlier, the possibility of sabotage was suggested, but investigations have failed to find any evidence to support that theory. Analysis of the powder samples, which appear to correspond to old stainless steel BWR oxide layers, indicates that the oxide formation is many years old. Pressure tests in May 1989 may have loosened the powder deposition. Following cleanup, only traces of the powder remain and tests show that these small amounts will not affect the control rod performance. See also References 303-305.

5.2.6.6 Japan

On July 26, 1989, Japan Atomic Energy Research Institute's (JAERI's) nuclear safety research reactor (NSRR) failed to achieve criticality when all three control rods were withdrawn.(299) Inspection revealed that the bottom portion of one rod (~20 cm long and containing boron carbide and air) had become disconnected and was still in the core. Attachment screws were found to be loosened.

5.2.6.7 France

See item under 5.1.15.9 for this 1989 event.

5.2.6.8 France

"Chinon-B1 was down for refueling July 25, 1989 when a planned test showed that one of the reactor's 53 control rod clusters did not drop in the prescribed time."(300)

5.2.7 Control Rod System Corrosion/Cracking

There were two events in 1989 at plants in another country involving control rod system corrosion or cracking. Those events are described below.

5.2.7.1 Japan

See Section 5.2.1.20 for details on this 1989 event.

5.2.7.2 Japan

See Section 5.2.1.22 for details on this 1989 event.

5.2.8 Nonconservative Assumptions/Incorrect Data

There were two events in 1989 at U.S. plants involving use of nonconservative assumptions or incorrect data. Those events are discussed below.

5.2.8.1 Point Beach-1

See Section 5.2.1.7 for details on this 1989 event.

5.2.8.2 Trojan

On September 9, 1989, it was found at Trojan that the 100% power reference temperature used in the rod control system was different than the value in the safety analysis.[301] The nonconservatism was due to a miscommunication between the licensee and the nuclear steam supply system (NSSS) vendor in 1976.

5.2.9 Oxide Thickness in Excess of Design Limits

There was an item of interest in 1989 regarding oxide thicknesses in excess of design limits. The item is discussed below.

5.2.9.1 Oxide Thickness

Oxide thicknesses in excess of design limits have been observed on ANF fuel in several PWRs.[302] The cladding involved was fabricated by a specific supplier. There were no fuel failures in spite of cladding thickness losses of up to 100 microns due to higher than normal waterside corrosion. Assembly burnups close to 42,000 MWd/MTU were achieved without fuel failure, even with a measured oxide thickness as high as 165 microns.

5.2.10 Debris in Reactor Vessel

There were three events in 1989, one at a U.S. plant and two at plants in other countries, that involved debris in the reactor vessel. Those events are described below.

5.2.10.1 Haddam Neck

See Section 5.2.1.1 for details on this 1989 event.[a]

5.2.10.2 Haddam Neck

See Section 5.2.1.2 for details on this 1989 event.

(a) Entries in Sections 2.10.1, 2.10.2, 2.10.3, and 2.10.4 pertain to the same event.

5.2.10.3 Haddam Neck

See Section 5.2.1.3 for details on this 1989 event.

5.2.10.4 Haddam Neck

See Section 5.2.1.4 for details on this 1989 event.

5.2.10.5 Japan

See Section 5.2.1.17 for details on this 1989 event.[a]

5.2.10.6 Japan

See Section 5.2.1.18 for details on this 1989 event.

5.2.10.7 Japan

See Section 5.2.1.19 for details on this 1989 event.

5.2.10.8 Finland

See Sections 5.2.6.2 - 5.2.6.5 for this 1989 event.

5.2.11 Fuel Assembly Cooling System

There was one event in 1989 at a plant in another country involving the fuel assembly cooling system. That event is described below.

5.2.11.1 France

During refueling of France's Tricastin-2, it was discovered on January 15, 1989, that the fuel assembly cooling system had not been switched on.[306] As only three assemblies had been inserted at the point, there was no heatup of the coolant; however, due to the violation of technical specifications as well as the lessons to be learned from the incident (apparently due to an error in operating documents), the incident was classified at Level 1 on the French nuclear severity scale.

5.2.12 Undersize Fuel Pellets

There was one event in 1989 at a plant in another country that involved undersize fuel pellets. That event is discussed below.

5.2.12.1 Undersize Pellets

On February 10, 1989, the French limited power at the Dampierre PWR station after it was discovered that some fuel rods contained a few undersized

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(a) Entries in Sections 2.10.5, 2.10.6, and 2.10.7 pertain to the same event.

pellets.[308] The power limitation was lifted on February 13 after it was verified that the smaller fuel did not adversely affect safety. The incident was classified at Level 1 because of EDF's failure to discover the fuel anomaly before fuel loading. Only a small number of pellets were concerned (40 out of 37 million) but the same anomaly could have been repeated in pellets already loaded into Dampierre-1 and -4 and Cruas-4, which also use this type of fuel.[309]

5.2.13 Raising of Water Level

There was one event in 1989 at a U.S. plant involving raising the water level. That event is described below.

5.2.13.1 St. Lucie-2

At St. Lucie-2 on February 22, 1989, the water level in the spent fuel pool (SFP) was raised high enough to flood the intake ventilation ducts that line the perimeter of the pool.[310] The overflow of water into the SFP ventilation ducting rendered the safety-related portion of the fuel handling building (FHB) ventilation system inoperable. The root cause of the event was operator errors by utility personnel.

5.2.14 Impurities in Primary Coolant

There was one event in 1989 at a U.S. plant involving an impurity in the primary coolant. That event is discussed below.

5.2.14.1 Calvert Cliffs-1

An NRC inspection report indicates that abnormal sulfate concentrations existed in the primary coolant system at Calvert Cliffs-1 at the time of startup in March 1989.[311] The response from the licensee indicates that a peak value of 1.9 ppm (within the limits of Chemistry Procedure CP-204) before the plant startup was commenced on March 26, 1989.

5.2.15 Addition of Unborated Water

There was one event in 1989 at a U.S. plant (PWR) involving the addition of unborated water. That event is described below.

5.2.15.1 San Onofre-1

On January 23, 1989, approximately 440 gallons of unborated water was added to the reactor refueling cavity water during decontamination activities at San Onofre-1 and resulted in a positive reactivity addition.[312] There was no safety significance to the event since the minimum required shutdown margin of 5% was not approached. The event was caused by cognitive personnel error.

5.2.16 Spent Fuel Pool

There were three events in 1989, two at U.S. plants and one at a plant in another country, and a 1990 item of interest pertaining to spent fuel pools. The events and the item of interest are discussed below.

5.2.16.1 Fitzpatrick

At Fitzpatrick on June 12, 1989, the surveys provided to support ongoing work in the spent fuel pool were inadequate to identify the presence of an object emitting up to 1000 R/hr on contact, which appeared in the work area.[313] The object was a highly radioactive particle that floated to the surface of the spent fuel pool. The doses of the two workers in close proximity to the source were calculated to be 780 mrem (whole body) and 960 mrem (extremity).

5.2.16.2 McGuire-1

In September 1989, analysis indicated that the formation of gaps in the Boraflex neutron-absorbing material in high density spent fuel storage racks (the subject of NRC Information Notice No. 87-43) was unlikely to occur in the racks at McGuire-1.[314] However, the potential existed for a different problem to develop at McGuire-1. The neutron absorber panels originally installed in the racks are shorter than the active fuel length of the stored fuel assemblies. This combined with shrinkage of the Boraflex neutron absorber could potentially have greater effects on reactivity than the condition referred to in the NRC Information Notice. This event was attributed to design deficiency because of the unanticipated environmental interaction.

5.2.16.3 Krypton-85 From Decayed Spent Fuel

See Section 5.2.2.1 for details on this 1990 item of interest.

5.2.16.4 Federal Republic of Germany

See Section 5.2.1.15 for details on this 1989 event.

5.2.17 Lack of Design Basis Documentation/Inadequate Review

There was one event in 1989 at a U.S. plant involving the lack of design basis documentation and an inadequate review. That event is described below.

5.2.17.1 San Onofre-1

See Section 5.2.1.8 for details on this 1989 event.

5.2.18 Procedural Noncompliance

There was one event in 1989 at a plant in another country involving procedural noncompliance. That event is described below.

5.2.18.1 France

See Section 5.2.11.1 for details on this 1989 event.

5.2.19 Defective Procedure

There were four events in 1989, three at U.S. plants and one at a plant in another country, involving defective procedures. Those events are described below.

5.2.19.1 Dresden-2

See Section 5.2.3.1 for details on this 1989 event.

5.2.19.2 Fitzpatrick

See Section 5.2.16.1 for details on this 1989 event.

5.2.19.3 Limerick-2

See Section 5.2.1.5 for details on this 1989 event.

5.2.19.4 France

See Section 5.2.11.1 for details on this 1989 event.

5.2.19.5 France

An error at Gravelines-1 that left pressurizer relief valves unable to open at their nominal setpoints has been acknowledged as a serious defect in the maintenance quality control program. The mistake was due to the insertion of some solid screws that had been used erroneously for over a year and had recently been replaced. The solid screws looked like the correct hollow screws and had been left unmarked in the tool box.[315]

5.2.20 Design Deficiency

There were two events in 1989 at U.S. plants involving design deficiencies. Those events are described below.

5.2.20.1 McGuire-1

See Section 5.2.16.2 for details on this 1989 event.

5.2.20.2 Oyster Creek

Section 5.2.1.6 for details on this 1989 event.

5.2.21 Equipment Failure

There was one event in 1989 at a plant in another country involving equipment failure. That event is described below.

5.2.21.1 Federal Republic of Germany

See Section 5.2.1.12 for details on this 1989 event.

5.2.21.2 Federal Republic of Germany

See Section 5.2.1.13 for details on this 1989 event.[a]

5.2.22 Personnel Error

There were five events in 1989 at U.S. plants and two in foreign plants involving personnel errors. Those events are described below.

5.2.22.1 Limerick-2

See Section 5.2.1.5 for details on this 1989 event.

5.2.22.2 San Onofre-1

See Section 5.2.15.1 for details on this 1989 event.

5.2.22.3 Sequoyah-1 and -2

See Section 5.2.1.9 for details on this 1989 event.

5.2.22.4 St. Lucie-2

See Section 5.2.13.1 for details on this 1989 event.

5.2.22.5 Three Mile Island-2

See Section 5.2.5.1 for details on this 1989 event.

5.2.22.6 Germany

Technicians preparing for a test on Fessenheim-1 mistakenly closed off feedwater to a reactor cooling system of unit 2, which was operating at full power, instead of to the system of unit 1. The error was quickly detected in the control room and corrected.[316]

(a) Pertains to same event noted in Sections 2.2.1 and 2.1.12.

5.2.22.7 France

Blayais-3 was down for refueling on July 11, 1989, when an in-core measurement thermocouple tube was damaged during closing of the reactor vessel head. After repair of the thermocouple, the reactor restarted July 27.[317]

5.2.23 Miscommunication

There was one event in 1989 at a U.S. plant involving miscommunication. That event is described below.

5.2.23.1 Trojan

See Section 5.2.8.2 for details on this 1989 event.

5.2.24 Unknown Cause for Event

There was one event in 1990 at a U.S. plant in which the cause is unknown. That event is described below.

5.2.24.1 Vogtle-1

See Section 5.2.1.10 for details on this 1990 event.

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