NUREG/CR-3001
PNL-4342

Fuel Performance Annual Report for 1981

Prepared by W. J. Bailey/PNL
M. Tokar/NRC

Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory Commission

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.
   Washington, DC 20555

2. The NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission,
   Washington, DC 20555

3. The National Technical Information Service, Springfield, VA 22 61

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the NRC/GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

GPO Printed copy price: $5.00

NUREG/CR-3001
PNL-4342

Fuel Performance Annual Report for 1981

Manuscript Completed: November 1982
Date Published: December 1982

Prepared by
W. J. Bailey/PNL
M. Tokar/NRC

Pacific Northwest Laboratory
Richland, WA 99352

Prepared for
Division of Systems Integration
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NRC FIN B2501

ABSTRACT

This annual report, the fourth in a series, provides a brief description of fuel performance during 1981 in commercial nuclear power plants. Brief summaries of fuel operating experience, fuel problems, fuel design changes and fuel surveillance programs, and high-burnup fuel experience are provided. References to additional, more detailed information and related NRC evaluations are included.

CONTENTS

FIGURES

TABLES

1.0 INTRODUCTION

Monitoring the in-reactor performance of nuclear fuel in commercial light-water power reactors yields important feedback for safety considerations and licensing procedures. Members of the public, governing and advisory bodies, and the U.S. Nuclear Regulatory Commission (NRC) staff have expressed interest in a publicly available summary of in-reactor fuel performance. As a result, a series of annual reports, of which this is the fourth, has been implemented to provide such a summary. The first was NUREG-0633 (Ref. 1), which covered the period through calender year 1978. The second, NUREG/CR-1818 (PNL-3583) (Ref. 2), covered calendar year 1979. The third, NUREG/CR-2410 (PNL-3953) (Ref. 3), covered calendar year 1980.

As noted in the first report (Ref. 1) of this annual series, the Atomic Energy Commission (AEC) and then the NRC have requested operating nuclear reactor fuel performance details through the reporting requirements of Regulatory Guide 1.16 (Ref. 4). However, over the years the material covered in these reports has changed. The 1971 version of the guide requested that a summary of fuel performance characteristics be included in semiannual operating reports and that special topical reports be used for fuel inspection details. By 1975 though, only abnormal degradation of fuel cladding and an indication of failed fuel were reportable items.[a] Reporting requirements were further reduced in 1977: only abnormal degradation of fuel cladding was to be included and the requirement for an annual operating report was eliminated. Also, normal operation surveillance results, generic problems, and design trends are not addressed in the NUREG series of reports entitled "Nuclear Power Plant Operating Experience" (Refs. 5-10). Results of plant operating experience are also screened by the Nuclear Safety Analysis Center, which is operated by the Electric Power Research Institute (Ref. 11).

As a result, the primary intent of this report series is to summarize fuel operating experience, fuel system problems--especially generic type--that are of concern during the reporting period, fuel design changes, fuel surveillance programs, and high-burnup fuel experience. In preparing the reports, we attempt to provide a traceable path of references so that the reader can acquire a greater level of detail than is included in the annual summary.

(a) A report published in 1981, NUREG/CR-1380 (PNL-3325) (Ref. 12), elaborates on the reporting of abnormal degradation and fuel failures. The threshold for what constitutes abnormal degradation is not uniform and remains a matter of opinion. Therefore, the degree of reported degradation is not uniform. The definition of failed fuel is tied to the functional, legal, and detection requirements on the fuel. The designation of fuel as failed depends on which functional requirement is not met (safety, commercial, design), whether or not there is a legal contingency on that requirement (Technical Specification, fuel warranty, design basis), and which indicator is used (coolant or off-gas activity, sipping, strain, or deflecion). Thus, the definition can vary from outage to outage and from reload to reload for each utility as the considerations change.

This report, though focusing on fuel operating experience during calendar year 1981, includes some overlap with the previous year. For those problems first encountered prior to 1981, the pre-1981 information will be included for the sake of continuity. In addition, information received or action taken in early 1982 will be included if pertinent to the discussion of problem areas.

2.0 FUEL SURVEILLANCE REQUIREMENTS

Section 4.2, Fuel System Design, of the Standard Review Plan (Ref. 13) requires that plans for testing, inspection, and fuel surveillance be submitted and reviewed for each domestic nuclear power plant. The plans should include preirradiation verification of cladding integrity, fuel system dimensions, fuel enrichment, burnable poison concentration, and absorber composition. Postirradiation surveillance plans are dependent on whether the fuel design is an existing or new design, and if the fuel exhibited any unusual behavior or characteristics. These plans are then referenced and/or summarized in the plant's safety analysis report (SAR). Ref. 14 is an example of a required fuel surveillance program.

Typical fuel assembly parameters[a] and operating conditions for current commercial light-water reactor (LWR) fuel rod designs for use in pressurized water reactors (PWRs) and boiling water reactors (BWRs) are summarized in Table 1. The newer fuel rod designs are those with smaller fuel rod diameters and more fuel rods per assembly. In a fuel assembly that employs fuel rods of a newer design, the total assembly power is maintained while the individual fuel rods are operated at lower linear heat generation rates and temperatures. This design change is expected to aid in improving the irradiation behavior of commercial LWR fuel by reducing fission gas release from the fuel and by reducing the mechanical interaction between the fuel and cladding. For example, sample calculations indicate that shifting from a 7 x 7 to an 8 x 8 rod array in a BWR fuel assembly results in a reduction in fission gas release of about 15% at high burnups (e.g., the release value drops from 26% to 13% at 30,000 MWd/MTU and from 46% to 30% at 45,000 MWd/MTU).[b]

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

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

TABLE 1. Typical Fuel Assembly Parameters

VENDOR	B&W	B&W	C-E	C-E	W	W	W	ENC	ENC	GE	GE	GE
Fuel Rod Array	15x15	17x17	14x14	16x16	14x14	15x15	17x17	15x15	8x8	7x7	8x8	8x8 R
Reactor Type	PWR	PWR	PWR	PWR	PWR	PWR	PWR	PWR	BWR	BWR	BWR	BWR
Assemblies per Core	177	205	217	177	121	193	193	193	560	764	560	560
Fuel Rod Locations Per Assembly	225	289	196	256	196	225	289	225	64	49	64	64
Fuel Rods Per Assembly	208	264	176	236	179	204	264	204	60	49	63	62
Empty Locations Per Assembly	17	25	5	5	17	21	25	21	4	NONE	1	2
Rod Pitch, mm (in.)	14.4 (0.568)	12.8 (0.502)	14.7 (0.580)	12.9 (0.5063)	14.1 (0.556)	14.3 (0.563)	12.6 (0.496)	14.3 (0.563)	16.3 (0.842)	18.7 (0.738)	16.3 (0.640)	16.3 (0.640)
System Pressure, MPa (psia)	15.2 (2200)	15.5 (2250)	15.5 (2250)	15.5 (2250)	15.5 (2250)	15.5 (2250)	15.5 (2250)	15.5 (2250)	7.14 (1035)	7.14 (1035)	7.14 (1035)	7.14 (1035)
Core Average Power Density, kW/liter	91.4	107.3	78.5	96.4	95.6	98.1	104.7	98.1	40.57	50.732	50.51	49.15
Average LHGR,[a] kW/M (kW/ft)	20.3 (6.20)	18.8 (5.73)	20.0 (6.09)	17.5 (5.34)	20.3 (6.20)	22.0 (6.70)	17.8 (5.44)	22.0 (6.70)	15.2 (4.63)	23.1 (7.049)	17.9 (5.45)	17.7 (5.38)
Axial Peak LHGR, in an Average Rod, kW/M (kW/ft)	24.41 (7.44)	22.57 (6.88)	24.00 (7.31)	21.00 (6.41)	24.36 (7.44)	26.40 (8.04)	21.36 (6.53)	26.40 (8.04)	18.24 (6.02)	27.72 (9.16)	21.48 (7.09)	21.24 (6.99)
Max. Peak LHGR, kW/M (kW/ft)	53.0 (16.16)	49.9 (15.20)	53.5 (16.3)	42.7 (13.0)	56.8 (17.3)	61.7 (18.8)	44.6 (13.6)	51.9 (15.83)	47.6 (14.5)	60.2 (18.35)	44.0 (13.4)	44.0 (13.4)
Max. Fuel Temp., °C (°F)	2340 (4245)	2090 (4155)	2140 (3890)	1880 (3420)	2260 (4100)	2340 (4250)	1870 (3400)	2200 (3997)	2040 (3700)	2440 (4430)	1830 (3325)	1890 (3435)
Core Average Enrichment, wt% ^{235}U	3.10[b]	3.15[b]	2.35	2.36	2.90	2.80	2.60	3.02	2.65	2.19	1.80	1.99
Max. Local Burnup, MWd/MTU[c] GJ/kgU[d]	55,000 4752	55,000 4752	50,000 4320	55,000 4752	50,000 4320	50,000 4320	50,000 4320	47,500 4104	35,000 3024	40,000 3456	40,000 3456	45,000 3888
Cladding Material	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-2	Zry-2	Zry-2	Zry-2

TABLE 1. (contd)

VENDOR	B&W	B&W	C-E	C-E	W	W	W	ENC	ENC	GE	GE	GE
Fuel Rod Length, m (in.)	3.90 (153.7)	3.88 (152.7)	3.71 (145.9)	4.09 (161.0)	3.87 (152.4)	3.80 (149.7)	3.85 (151.6)	3.86 (152.0)	3.99 (156.9)	4.09 (161.1)	4.09 (161.1)	4.20 (185.4)
Active Fuel Height, m (in.)	3.602 (141.8)	3.632 (143)	3.47 (136.7)	3.81 (150)	3.66 (144)	3.66 (144)	3.65 (143.7)	3.66 (144)	3.66 (144)	3.66 (144)	3.71 (146)	3.81 (150)
Plenum Length, m (in.)	0.298 (11.72)	0.242 (9.52)	0.22 (8.6)	0.25 (10.00)	0.18 (6.99)	0.21 (8.2)	0.16 (6.3)	0.17 (6.8)	0.27 (10.63)	0.41 (16.0)	0.36 (14.0)	0.25 (10.0)
Fuel Rod OD, mm (in.)	10.922 (0.430)	9.627 (0.379)	11.18 (0.440)	9.70 (0.382)	10.72 (0.422)	10.72 (0.422)	9.50 (0.374)	10.77 (0.424)	12.74 (0.5015)	14.30 (0.563)	12.52 (0.493)	12.27 (0.483)
Cladding ID, mm (in.)	9.576 (0.377)	8.407 (0.331)	9.86 (0.388)	8.43 (0.332)	9.48 (0.3734)	9.48 (0.3734)	8.36 (0.329)	9.25 (0.364)	10.91 (0.4295)	12.68 (0.499)	10.80 (0.425)	10.64 (0.419)
Cladding Thickness, mm (in.)	0.673 (0.0265)	0.610 (0.0240)	0.660 (0.026)	0.635 (0.025)	0.617 (0.0243)	0.617 (0.0243)	0.572 (0.0225)	0.762 (0.030)	0.914 (0.036)	0.813 (0.032)	0.864 (0.034)	0.813 (0.032)
Diametral Gap(e), micron (mil)	213.4 (8.4)	198.1 (7.8)	216 (8.5)	178 (7.0)	190 (7.5)	190 (7.5)	165 (6.5)	190 (7.5)	254 (10.0)	305 (12.0)	229 (9.0)	229 (9.0)
Fuel Pellet Diameter, mm (in.)	9.362 (0.3686)	8.209 (0.3232)	9.64 (0.3795)	8.26 (0.325)	9.29 (0.3659)	9.29 (0.3659)	8.19 (0.3225)	9.06 (0.3565)	10.66 (0.4195)	12.37 (0.487)	10.57 (0.416)	10.41 (0.410)
Fuel Pellet Length, mm (in.)	15.240 (0.600)	9.525 (0.375)	16.51 (0.650)	9.91 (0.390)	15.24 (0.600)	15.24 (0.600)	13.46 (0.530)	6.93 (0.273)	8.13 (0.320)	12.70 (0.500)	10.67 (0.420)	10.41 (0.410)
Fuel Pellet Density, %TD(f)	95	95	94.75	95	94	95	95	94	95	95	95	95

(a) LHGR = Linear heat generation rate.
(b) Reload batch average enrichment.
(c) MWd/MTU = number of megawatt days of thermal energy released by fuel containing one metric ton (10^3 kg) of heavy-metal atoms (e.g., U = uranium).
(d) GJ/kgU = gigajoule/kilogram of heavy metal (e.g., U = uranium).
(e) Diametral gap = cladding ID - pellet diameter.
(f) Theoretical density (TD) of stoichiometric UO_2 is 10.96 g/cm^3.

3.0 FUEL OPERATING EXPERIENCE

The following subsections provide synopses of domestic fuel operating experience. The six subsections include information from a)the five fuel vendors: Babcock & Wilcox Company, Combustion Engineering, Inc., Exxon Nuclear Company, Inc., General Electric Company, and Westinghouse Electric Cooperation, and b) the Electric Power Research Institute. While overall fuel operating experience continues to be excellent there are sporadic events involving damage to or failure of fuel, and those events are discussed in Section 4.0.

3.1 BABCOCK & WILCOX COMPANY (B&W)

B&W has issued the annual summary for 1981 (Ref. 15) of in-reactor fuel performance and ongoing development programs for B&W-designed commercial nuclear fuel. Additional information may be found in Ref. 16. B&W has irradiated a total of 647,728 Zircaloy-clad fuel rods during the nine-year operating history of B&W-designed reactors. In addition, stainless steel-clad fuel rods have been irradiated in Connecticut Yankee (Haddam Neck). A fuel integrity level of 99.992% was achieved with the 309,824 Zircaloy-clad fuel rods that were irradiated during 1981. See Section 4.0 for information on the fuel failures. A summary of B&W fuel rod experience, from the startup of their first reactor (Oconee-1) in April 1973 through December 1981, is provided in Table 2. The operating status of B&W-designed reactors is shown in Table 3. The burnup experience for B&W-supplied fuel is summarized in Table 4. Tables 2-4 are from Ref. 15.

B&W, in cooperation with Duke Power Company (Duke), Arkansas Power & Light (AP&L), Sacramento Municipal Utility District (SMUD), and the Department of Energy (DOE), is involved with programs to improve fuel utilization by extending the average burnup of LWR fuel assemblies to approximately 50,000 MWd/MTU and by developing and demonstrating advanced fuel designs (Ref. 15). The programs include the irradiation of B&W current-design and extended burnup-design 15 x 15 fuel assemblies to approximately 50,000 MWd/MTU. Two B&W papers (Ref. 17 and 18) concerning the development and performance of the extended burnup fuel were recently published. The programs also include demonstrating the in-reactor performance of a fuel-burnable poison mixture, UO_2-Gd_2O_3, (Ref. 19); axial blanket fuel; annular fuel pellets; and low absorption spacer grids (Ref. 20). See Section 5.0 for the status of the programs.

3.2 COMBUSTION ENGINEERING, INC. (C-E)

The 1981 performance of C-E fuel is described in a recent letter (Ref. 21) to the NRC. As of December 31, 1981, C-E has irradiated a total of 597,201 Zircaloy-clad fuel rods. Additional information on C-E fuel is provided in Refs. 22 and 23.

A summary of C-E fuel rods and assemblies irradiated and/or discharged and the batch-averaged burnups achieved in 1981 is presented in Table 5. The status of C-E burnup experience with all-Zircaloy assemblies is shown in

TABLE 2. 1981 Performance Summary for B&W-Supplied Fuel Rods[a]

Fuel Rod Type	Fuel Assembly Rod Array Type 15 x 15	17 x 17
1. Cumulative Number of Rods Irradiated Through Dec. 1981:	646,672	1,056
a. Maximum Rod-Average Burnup, MWd/MTU	41,600	35,300
b. Mean Rod-Average Burnup, MWd/MTU	9,990	29,500
2. Total Number of Rods Irradiated in 1981:	309,296	528
3. Number of Irradiated Rods In-Core on Dec. 31, 1981:	283,296	528
a. Maximum Rod-Average Burnup, MWd/MTU	41,600	35,300
b. Mean Rod-Average Burnup, MWd/MTU	5,030	33,900
4. Number of Rods Discharged in 1981:	52,000	0
a. Maximum Rod-Average Burnup, MWd/MTU	33,400	--
b. Mean Rod-Average Burnup, MWd/MTU	26,570	--
5. Estimated Number of Leaker Rods Generated in 1981[b]	24	

(a) Connecticut Yankee (Haddam Neck) and Three Mile Island-2 are excluded from this tabulation.
(b) Estimated from equilibrium coolant radioiodine behavior during full-power operation (iodine-131, iodine-133, and iodine-135 activity levels). The relatively small percentage of 17 x 17 type fuel (lead test assemblies) operating with 15 x 15 type fuel precludes differentiation of probable leakers among the two fuel types.

TABLE 3. Operating Status of B&W-Designed Reactors (as of December 31, 1981)

Reactor	Cycle	Maximum Assembly Burnup, MWd/MTU: In-Core Fuel	Maximum Assembly Burnup, MWd/MTU: Fuel Discharged to Date
Oconee-1	7	40,000	40,000
Oconee-2	5	36,200	33,700
Oconee-3	6	33,100	31,950
Three Mile Island-1[a]	5	25,200	32,200
Arkansas-1[b]	5	29,800	33,250
Rancho Seco	5	31,200	37,750
Crystal River-3	4	17,900	29,900
Davis Besse-1	2	23,050	13,100

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

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

TABLE 4. Summary of Burnup Experience Through December 31, 1981, for B&W-Supplied Fuel[a]

Fuel Assembly Batch Average Burnup, MWd/MTU	In-Core Assemblies on December 31, 1981: No. of Assy's	In-Core Assemblies on December 31, 1981: No. of Rods	Assemblies Discharged in 1981: No. of Assy's	Assemblies Discharged in 1981: No. of Rods	Assemblies Discharged Through December 31, 1981: No. of Assy's	Assemblies Discharged Through December 31, 1981: No. of Rods
0 to 3,900	188	39,104	0	0	0	0
4,000 to 7,900	132	27,456	0	0	4	832
8,000 to 11,900	192	39,936	0	0	103	21,424
12,000 to 15,900	284	59,072	4	832	177	36,816
16,000 to 19,900	350	72,800	3	624	121	25,168
20,000 to 23,900	61	12,688	21	4,368	202	42,016
24,000 to 27,900	110	22,880	141	29,328	690[b]	143,632
28,000 to 31,900	28	5,824	81	16,848	383	79,664
32,000 to 35,900	70[c]	14,672	0	0	65	13,520
36,000 to 39,900	0	0	0	0	1	208
40,000 to 43,900	1	208	0	0	3	624
Totals	1,416	294,640	250	52,000	1,749	363,904

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

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

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

TABLE 5. Summary of Combustion Engineering Fuel Irradiated and/or Discharged in 1981

Reactor (Fuel Cycles)	Fuel Batch	No. of Assemblies Irradiated	No. of Assemblies Discharged	No. of Fuel Rods Irradiated	No. of Fuel Rods Discharged	Batch-Averaged Burnup, MWd/MTU as of December 1981	Batch-Averaged Burnup, MWd/MTU at Discharge
Arkansas-2[a] (Cycles 1 and 2)	A	61	60	14,396	14,160	16,500	13,400
	B	60	---	13,440	---	18,700	---
	C	56	---	12,808	---	15,200	---
	D	60	---	14,160	---	5,100	---
Calvery Cliffs-1 (Cycle 5)	D	1	---	176	---	39,100	---
	E	52	---	9,152	---	30,100	---
	F	72	---	12,672	---	21,200	---
	G	92	---	16,192	---	9,600	---
Calvert Cliffs-2 (Cycles 3 and 4)	B	1	1	164	164	---	34,500
	C	68	68	11,632	11,632	---	34,100
	D	84	59	14,784	10,384	28,400	22,300
	E	64	---	11,264	---	18,900	---
	F	128	---	21,824	---	8,300	---
Fort Calhoun (Cycles 6 and 7)	D	1	---	176	---	44,900	---
	E	12	12	2,112	2,112	---	34,300
	F	36	28	6,336	4,928	31,800	30,200
	G	44	---	7,744	---	21,800	---
Maine Yankee (Cycles 5 and 6)	E	2	1	160	160	22,100	28,100
	G	32	32	5,568	5,569	---	32,700
	H	40	40	7,040	7,040	---	31,500
	I	72	---	12,576	---	26,200	---
Millstone-2 (Cycle 4)	B	1	---	164	---	25,700	---
	D	72	72	12,672	12,672	---	31,400
	E	72	---	12,672	---	23,300	---
Palisades (Cycle 4)	D	60	60	12,960	12,960	---	30,600
St. Lucie-1 (Cycles 4 and 5)	C	1	1	176	176	---	35,700
	D	60	60	10,560	10,560	---	29,600
	E	68	3	11,968	528	23,200	23,300
	F	88	---	14,912	---	14,100	---
	G	64	---	11,024	---	600	---

(a) Arkansas Nuclear One-Unit 2.
(b) Batch E Assembly in Cycle 6 is different from the Batch E Assembly in Cycle 5.

Table 6. A statement about the fuel integrity level was not included in Ref. 21; however, the typical equilibrium iodine-131 activity concentrations in the primary coolants of C-E reactors during 1981 are listed in Table 7. See Section 4.0 for information on the fuel failures. Tables 5-7 are from Ref. 21.

A paper describing C-E's experience with fuel at high burnups was recently published (Ref. 24). Fuel performance data supporting the design and operation of C-E fuel are being acquired under surveillance programs sponsored by EPRI and DOE (Refs. 24-28). See Section 5.0.

3.3 EXXON NUCLEAR COMPANY, INC. (ENC)

A report (Refs. 29 and 30) summarizing the ENC fuel performance through December 31, 1981 has been published. Additional information is provided in Refs. 31-33. ENC has irradiated a total of 527,856 fuel rods in 4024 fuel assemblies and the fuel integrity levels achieved have been 99.987% and 99.43%, respectively. See Section 4.0 for information on the fuel failures. ENC fuel has been loaded into 23 domestic and foreign reactors (includes 9 BWRs and 14 PWRs). During 1981, 11 ENC fuel assemblies (3 in domestic plants and 8 in foreign plants) were found by sipping or by visual inspection to contain leaking fuel rods. Table 8 (Ref. 29) summarizes the ENC fuel assembly distribution and fuel rod performance as of the end of 1981. The maximum assembly average burnups attained were 39,100 MWd/MTU in BWRs and 46,800 MWd/MTU in PWRs. The exposure range of ENC fuel is shown in Figure 1. ENC fuel failures observed in 1981 are listed in Table 9. Tables 8 and 9 and Figure 1 are from Ref. 29.

ENC, in cooperation with customer utilities, DOE, EPRI, and PNL, is conducting a number of test fuel programs that are aimed at evaluating the performance of several proposed design modifications (Refs. 25, 29, 31, and 34-38). See Section 5.0.

3.4 GENERAL ELECTRIC COMPANY (GE)

Table 10 (Ref. 39) provides a summary of GE's BWR/2-5 experience with 8 x 8 fuel as of January 1982 and shows that GE has irradiated a total of 1,489,246 fuel rods and achieved a fuel integrity level of greater than 99.98%. See Section 4.0 for information on the fuel failures. One GE BWR 6 with Mark III containment (Kuosheng-1) has begun operation (Ref. 40-42). GE is conducting fuel surveillance programs that involve lead test assemblies and, in some cases, demonstration reloads (Refs. 25, 39, and 43-47). A recent paper (Ref. 48) describes the impact of extended burnup on the BWR fuel cycle. A summary of GE's lead test assembly and extended burnup bundles is shown in Table 11 (Ref. 39). See Section 5.0.

TABLE 6. Combustion Engineering Burnup Experience with All-Zircaloy Assemblies: Status as of December 31, 1981

Fuel Assembly Batch-Averaged Burnup, MWd/MTU	In-Core Fuel Assemblies with Pressurized Fuel Rods		Discharged Fuel Assemblies with Pressurized Fuel Rods		Discharged Fuel Assemblies with Non-Pressurized Fuel Rods	
	No. of Fuel Assemblies	No. of Fuel Rods	No. of Fuel Assemblies	No. of Fuel Rods	No. of Fuel Assemblies	No. of Fuel Rods
0 to 3,900	64	11,024	0	0	0	0
4,000 to 7,900	60	14,160	6	1,048	0	0
8,000 to 11,900	220	38,016	25	3,696	208	40,500
12,000 to 15,900	144	28,296	217	41,792	190	35,351
16,000 to 19,900	125	24,940	248	43,128	24	3,840
20,000 to 23,900	254	44,688	226	39,796	0	0
24,000 to 27,900	73	12,740	372	62,916	0	0
28,000 to 31,900	85	14,960	646	112,652	0	0
32,000 to 35,900	0	0	135	23,144	0	0
36,000 to 39,900	1	176[a]	0	0	0	0
40,000 to 43,900	0	0	1	162	0	0
44,000 to 47,900	1	176	0	0	0	0
Totals	1,027	189,176	1,876	328,334	422	79,691

(a) This fourth-cycle Calvert Cliffs-1 assembly includes some fuel rods in their fifth cycle.

TABLE 7. Typical Equilibrium Iodine-131 Activity Concentrations in the Primary Coolants of Combustion Engineering Reactors During 1981

Reactor	Fuel Cycle	Beginning of Cycle	End of Cycle	1981 Equilibrium Conditions Power, %	Iodine-131, µCi/g
Arkansas-2[a]	1	12/06/78	03/28/81	100	0.10 to 0.15
	2	07/02/81	08/82[b]	100	0.05 to 0.07
Calvert Cliffs-1	5	12/21/80	04/16/82	100	0.003 to 0.015
Calvert Cliffs-2	3	12/06/79	01/17/81	NA[c]	NA
	4	03/13/81	10/82[b]	100	0.006 to 0.015
Fort Calhoun	6	06/08/80	09/18/81	95	0.07
	7	12/17/81	01/83[b]	NA	NA[d]
Maine Yankee	5	03/09/80	05/08/81	97	0.005
	6	07/12/81	10/8^[b]	97	0.002[d]
Millstone-2	4	10/20/80	12/05/81	100	0.006 to 0.01
Palisades	4	05/24/80	08/29/81	100	0.03 to 0.09[d]
St. Lucie-1	4	05/07/80	09/08/81	100	0.01 to 0.06
	5	11/29/81	05/83[b]	NA	NA

(a) Arkansas Nuclear One-Unit 2.
(b) Target end-of-cycle date.
(c) Not available (NA).
(d) Predominately non-Combustion Engineering supplied fuel.

3.5 WESTINGHOUSE ELECTRIC CORPORATION (W)

The operational experience in Westinghouse cores with Zircaloy-clad fuel up to December 31, 1981, is summarized in a recent report (Ref. 49). Westinghouse-supplied fuel has been employed in 40 commercial PWRs. During 1981, 23 plants were refueled and 4 plants commenced initial commercial power operation. Through 1981, Westinghouse has irradiated a total of 2,421,800 Zircaloy-clad fuel rods. Table 12 (Ref. 49) provides a burnup summary for Westinghouse Zircaloy-clad fuel discharged and being irradiated through 1981. The burnup performance from 1974 through 1981 is illustrated in Figure 2 (Ref. 49). Westinghouse continues to evaluate fuel performance in terms of coolant activity level (i.e., instead of fuel integrity level). Table 13 (Ref. 49) presents a performance summary for Westinghouse fuel on a plant-by-plant basis and includes data on coolant activity level. See Section 4.0 for information on the fuel failures. Westinghouse is conducting a number of fuel surveillance programs (Refs. 50-52). See Section 5.0.

TABLE 8. Summary of Exxon Nuclear Fuel Assembly Distribution and Fuel Rod Performance as of December 31, 1981

A. Fuel Assemblies

Reactor Type	Number of Fuel Assemblies: In Core as of December 31, 1981	Discharged in 1981	Discharged Prior to 1981	Total	Total Failures: Number	Rate, %
BWR	1455	32	498	1985	12	0.60
PWR	1570	128	341	2039	11	0.54
Total	3025	160	839	4024	23	0.57

B. Fuel Rods

Reactor Type	In-Core Fuel: Number of Rods	In-Core Fuel: Burnup, MWd/MTU	Discharged Fuel: Number of Rods	Discharged Fuel: Burnup, MWd/MTU	Total Number of Rods	Failure Rate, %,(a)	Total Failures: Number	Total Failures: Rate, %
BWR	90,425	39,100(b)	26,466	30,400	116,891	0.006	14	0.012
PWR	312,548	46,800	98,417	40,400	410,965	0.001	52	0.013
Total	402,973		124,883		527,856	0.002	66	0.013

(a) Failures not directly attributable to external causes and occurring below warranted burnup.
(b) Average of 60 Extended Burnup Demonstration Program fuel rods.

3.6 ELECTRIC POWER RESEARCH INSTITUTE (EPRI)

EPRI initiated a fuel surveillance program in operating LWRs in 1974 that involves projects with each of the domestic fuel vendors (Ref. 25 and 53). Those projects monitor only standard product-line fuel (however, EPRI is also sponsoring studies on new improved fuel design concepts-see Section 5.0). Under the projects, data are compiled on the performance of standard fuel rods and assemblies, control and burnable poisons, and BWR fuel channels. The EPRI program is directed toward solving specific industry problems such as fuel failures due to pellet-cladding interaction, fuel rod bow, and fission gas release or on improving overall fuel assembly performance (e.g., by employing extended burnup fuel cycles). Concerning the last item, DOE is conducting similar studies under their Extended Burnup Program. The EPRI fuel surveillance program is discussed further in Section 5.0. EPRI has made a critical review (Ref. 54) of the problems and progress with LWR materials, including core materials.

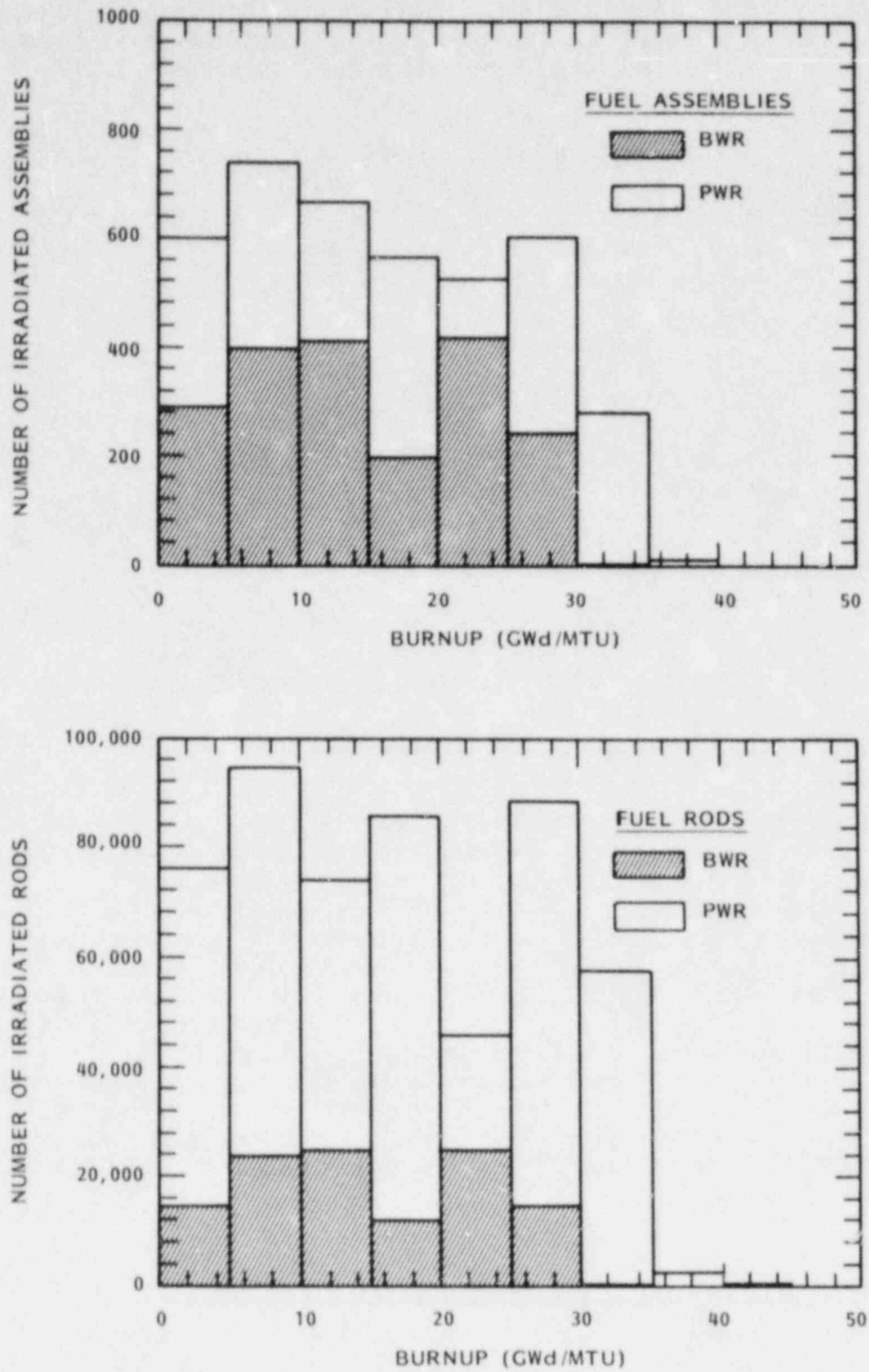

FIGURE 1. Exposure of Irradiated Exxon Nuclear Fuel as of December 31, 1981

TABLE 9. Exxon Nuclear Fuel Failures During 1981

Reactor	Number of Fuel Assemblies	Fuel Type	Date Found
BWRs:			
Barseback-1	3	UO_2	07/81
PWRs:			
Biblis-A	1	UO_2	05/81
Palisades	1	UO_2	10/81
Tihange-1	4	UO_2	03/81
Yankee Rowe-1	2	UO_2	06/81

TABLE 10. General Electric BWR/2-5 Fuel Experience Summary (January 1982)

	All 8 x 8
Total rods in core or discharged	1,489,246[a]
Estimated sound rods, %	>99.98
Lead assembly exposure, MWd/MTU	40,000
Peak linear heat generation rate, kW/m (kW/ft)	44.0 (13.4)

(a) Cumulative.

TABLE 11. Summary of General Electric Lead Test Assembly and Extended Burnup Bundles

Program	Reactor	No. of Bundles	Completed Cycles of Operation	Average Burnup At Last Outage, MWd/MTU	Objectives
8x8 Lead Test Assemblies	Monticello	1	5	40,000	Lead 8x8 Performance, Extended Burnup
	Quad Cities-1	1[a]	4	29,000[a]	Lead 8x8 Performance
8x8R Lead Test Assemblies	Peach Bottom-2	4[b]	4	35,000	Lead 8x8R Performance, Extended Burnup
	Vermont Yankee	2[a]	5	21,000[a]	Lead 8x8R Performance
P8x8R Lead Test Assemblies	Peach Bottom-3	1	3	25,000	Lead P8x8R Performance
Lead Test Assemblies	Quad Cities-1	5[c]	5	31,000	Extended Burnup, Fuel Performance
Lead Test Assemblies (Barrier Cladding)	Quad Cities-1	4	1	12,000	Barrier Clad
Lead Test Assemblies (Extended Burnup)	Monticello	4[d]	6	39,000	Extended Burnup
Barrier Reload Demonstration	Quad Cities-2	144			Power Ramp Testing

(a) Bundle(s) now discharged.
(b) One bundle discharged.
(c) Three bundles now discharged.
(d) One bundle discharged. The remaining three bundles include the 8x8 Lead Test Assembly.

TABLE 12. Westinghouse Zircaloy-Clad Fuel Burnup Status as of December 31, 1981: Assemblywise Burnup Distribution of Fuel Rods

Active Fuel Rods[a]

Assemblywise Burnup, MWd/MTU	No. of Fuel Rods
0 - 3,900	180,974
4,000 - 7,900	251,338
8,000 - 11,900	119,090
12,000 - 15,900	97,122
16,000 - 19,900	118,961
20,000 - 23,900	128,069
24,000 - 27,900	92,667
28,000 - 31,900	70,241
32,000 - 35,900	55,407
36,000 - 39,900	6,582
40,000 - 43,900	895
44,000 - 47,900	0
48,000 - 43,900	816

Previously Discharged Fuel Rods

Assemblywise Burnup, MWd/MTU	No. of Fuel Rods
0 - 3,900	0
4,000 - 7,900	3,222
8,000 - 11,900	26,130
12,000 - 15,900	70,889
16,000 - 19,900	221,103
20,000 - 23,900	135,476
24,000 - 27,900	192,188
28,000 - 31,900	220,437
32,000 - 35,900	104,330
36,000 - 39,900	38,229
40,000 - 43,900	1,790
44,000 - 47,900	0
48,000 -	0

Newly Discharged Fuel Rods[b]

Assemblywise Burnup, MWd/MTU	No. of Fuel Rods
0 - 3,900	0
4,000 - 7,900	707
8,000 - 11,900	28,942
12,000 - 15,900	4,870
16,000 - 19,900	12,900
20,000 - 23,900	18,100
24,000 - 27,900	72,587
28,000 - 31,900	90,730
32,000 - 35,900	46,579
36,000 - 39,900	10,234
40,000 - 43,900	204
44,000 - 47,900	0
48,000 -	0

Total Discharged Fuel Rods[c]

Assemblywise Burnup, MWd/MTU	No. of Fuel Rods
0 - 3,900	0
4,000 - 7,900	3,929
8,000 - 11,900	55,072
12,000 - 15,900	75,759
16,000 - 19,900	234,003
20,000 - 23,900	153,576
24,000 - 27,900	264,775
28,000 - 31,900	311,167
32,000 - 35,900	150,909
36,000 - 39,900	48,463
40,000 - 43,900	1,994
44,000 - 47,900	0
48,000 -	0

(a) In-core fuel rods that were not discharged from operating plants.
(b) Fuel rods discharged during calendar year 1981.
(c) Fuel rods discharged from 1969 through December 1981.

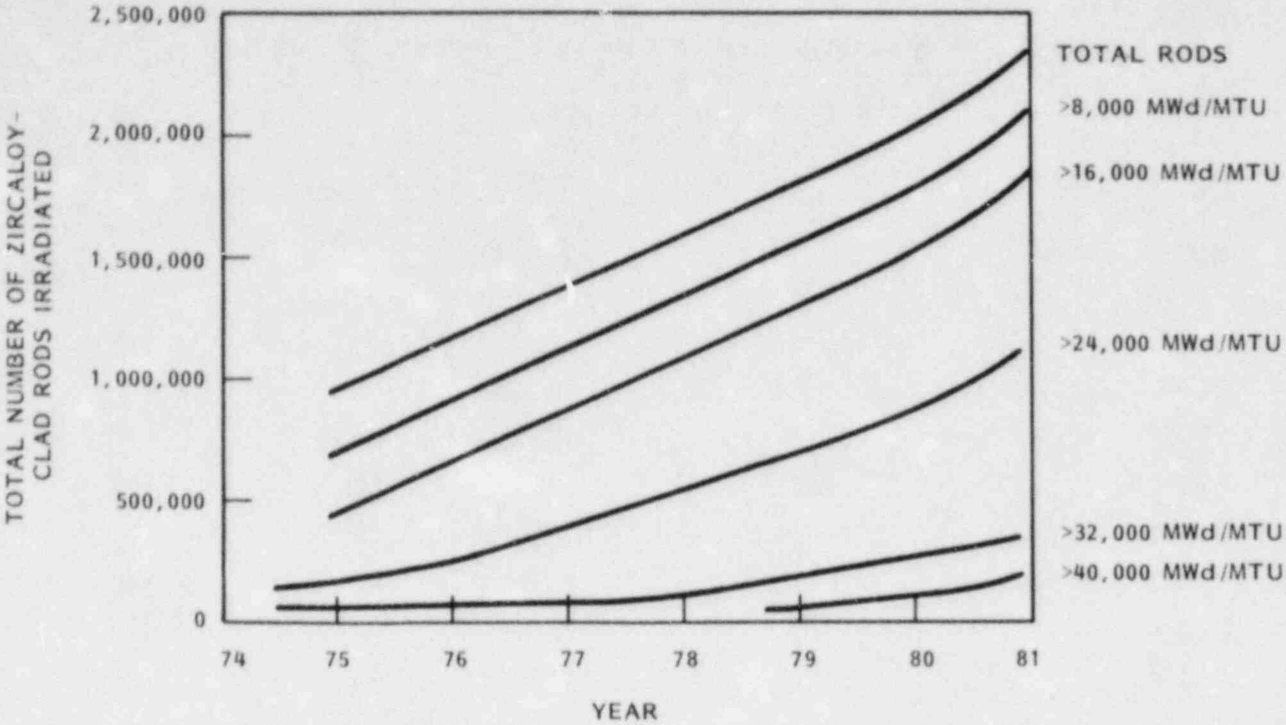

FIGURE 2. Burnup Performance of Westinghouse Zircaloy-Clad Fuel (Representative of All Zircaloy-Clad Fuel Operating and Discharged)

TABLE 13. Westinghouse Fuel Performance Status Report (Fourth Quarter 1981)

Reactor	Location	Date of Initial Criticality	Current Cycle No.	Peak Region Average Burnup as of Dec. 31, 1981, (MWd/MTU)	Fourth Quarter Percent of Design Basis Activity Release Rate[a]	Comment
Jose de Cabrera	Spain	06/68	11	30,500	2.1	
Beznau-1	Switzerland	06/69	11	33,800	0.75	
Mihama-1	Japan	07/70	3	26,000	No Data[b]	(c)
Point Beach-1	U.S.A.	11/70	10	34,700	2.4	(c)
Mihama-2	Japan	04/72	4	28,400	No Data	
Point Beach-2	U.S.A.	05/72	8	35,100	0.3	
Surry-1	U.S.A.	07/72	6	26,300	6.38	
Turkey Point-3	U.S.A.	10/72	8	31,300	No Data	(d)
Surry-2	U.S.A.	03/73	5	33,300	0.14	(c)
Zion-1	U.S.A.	06/73	6	37,000	1.1	
Indian Point-2	U.S.A.	05/73	5	33,400	2.4	
Turkey Point-4	U.S.A.	06/73	8	33,900	No Data	(c)
Prairie Island-1	U.S.A.	12/73	7	34,900	0.9	(c)
Zion-2	U.S.A.	12/73	6	36,700	< 0.08	(e)
Kewaunee	U.S.A.	03/74	7	31,900	≤ 0.01	
Prairie Island-2	U.S.A.	12/74	6	35,000	0.01	
Trojan	U.S.A.	12/75	4	34,100	4.2	
Indian Point-3	U.S.A.	04/76	3	33,140	0.2	(c)
Beaver Valley	U.S.A.	05/76	2	27,400	0.25	
Salem-1	U.S.A.	12/76	3	29,600	0.2	
Ko-Ri-1	Korea	06/77	3	27,000	No Data	
Farley-1	U.S.A.	08/77	4	32,700	0.33	(c)
Ohi-1	Japan	12/77	3	~24,000	No Data	(c)
D. C. Cook-2	U.S.A.	03/78	3	27,000	1.8	
North Anna-1	U.S.A.	04/78	3	17,000	2.8	
Ohi-2	Japan	09/78	2	14,700	No Data	(c)
North Anna-2	U.S.A.	06/80	1	14,000	0.6	
Sequoyah-1	U.S.A.	07/80	1	7,300	0.2	(d)
Ringhals-3	Sweden	07/80	1	6,700	(e)	(e)
Salem-2	U.S.A.	08/80	1	3,600	---	
Almarez-1	Spain	04/81	1	1,000	(e)	
Farley-2	U.S.A.	05/81	1	1,000	0.02	
McGuire	U.S.A.	06/81	1	1,000	(e)	(e)
Sequoyah-2	U.S.A.	10/81	1	1,000	(e)	(e)
Millstone-2	U.S.A.	12/75	5	---	~0.04	(c,f)

(a) Activity release rate calculated from coolant activity averaged over the quarter and presented as percent of that iodine-131 release rate which establishes the basis for design of plant shielding and coolant cleanup system equipment.
(b) No data reported (No Data).
(c) Plant refueling during period of report.
(d) Maintenance and inspection or repair during period of report.
(e) Reported information reflects last period of operation or inferrence from short periods of operation during period of report.
(f) This reactor is currently fueled by Westinghouse.

4.0 PROBLEM AREAS OBSERVED DURING 1981

4.1 PROBLEMS IN 1981 THAT ARE SIMILAR TO THOSE IN 1980

4.1.1 Iodine Spiking and Gross Gas Release

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

The NRC has developed Standard Technical Specifications (Table 14) for primary coolant iodine concentrations that make allowance for iodine spikes by permitting temporary excursions (not to exceed 48 hours) above the "equilibrium" concentration limit. For each excursion above the equilibrium limit, a Licensee Event Report is required. Four BWRs (Brunswick-1 and -2, Hatch-2, La Crosse) and approximately one-half of the operating PWRs have this type of technical specification.

During 1981, iodine spiking and/or gross gas release occurred at twelve plants. As shown below, two boiling water reactors (BWRs) and ten pressurized water reactors (PWRs) were involved:

Reactor	Reactor Type	
	BWR	PWR
Arkansas-1		X
Arkansas-2		X
Brunswick-2	X	
Cook-2		X
Crystal River-3		X
Davis Besse-1		X
La Crosse	X	
North Anna-1		X
Palisades		X
St. Lucie-1		X
Surry-1		X
Trojan		X

TABLE 14. Technical Specifications for Primary Coolant Activity[a]

Plant	Type[b]	Limits on Coolant Activity: Dose Equivalent Iodine-131, (μCi/g)	Other Isotopes, Value/$\bar{E}$[c]	Standard[d]
Arkansas-1[e]	PWR	3.5	72/$\bar{E}$	No
Arkansas-2[f]	PWR	1.0	100/$\bar{E}$	Yes
Beaver Valley-1	PWR	1.0	100/$\bar{E}$	Yes
Big Rock Point-1	BWR	(g)	(g)	--
Browns Ferry-1,-2, and-3	BWR	3.2	(g)	No
Brunswick-1 and-2	BWR	0.2	100/$\bar{E}$	Yes
Calvert Cliffs-1 and-2	PWR	1.0	100/$\bar{E}$	Yes
Cook-1 and-2	PWR	1.0	100/$\bar{E}$	Yes
Cooper Station	BWR	3.1	(g)	No
Crystal River-3	PWR	1.0	100/$\bar{E}$	Yes
Davis-Besse-1	PWR	1.0	100/$\bar{E}$	Yes
Dresden-1	BWR	20.0[h]	(g)	No
Dresden-2	BWR	20.0[h]	(g)	No
Dresden-3	BWR	20.0[h]	(g)	No
Duane Arnold	BWR	1.2[i]	(g)	No
Farley-1	PWR	1.0	100/$\bar{E}$	Yes
Fitzpatrick	BWR	3.1	(g)	No
Fort Calhoun-1	PWR	2.0	31/$\bar{E}$	No
Ginna	PWR	3.0	84/$\bar{E}$	No
Haddam Neck	PWR	(g)	68/$\bar{E}$	No
Hatch-1	BWR	10	(g)	No
Hatch-2	BWR	0.2	100/$\bar{E}$	Yes
Indian Point-2	PWR	(g)	60/$\bar{E}$	No
Indian Point-3	PWR	1.0	100/$\bar{E}$	Yes
Kewaunee	PWR	(g)	91/$\bar{E}$	No
La Crosse	BWR	0.2	100/$\bar{E}$	Yes
Maine Yankee	PWR	1.0	100/$\bar{E}$	Yes
Millstone-1	BWR	20.0[h]	(g)	No
Millstone-2	PWR	1.0	100/$\bar{E}$	Yes
Monticello	BWR	5.0	(g)	No
Nine Mile Point-1	BWR	25.0[h]	(g)	No
North Anna-1 and-2	PWR	1.0	100/$\bar{E}$	Yes
Oconee-1,-2, and-3	PWR	(g)	224/$\bar{E}$	No
Oyster Creek-1	BWR	8.0[h]	(g)	No
Palisades	PWR	1.0	100/$\bar{E}$	Yes
Peach Bottom-2 and-3	BWR	2.0	(g)	No
Pilgrim-1	BWR	20.0[h]	(g)	No
Point Beach-1	PWR	1.0	100/$\bar{E}$	Yes

TABLE 14. (contd)

Plant	Type[b]	Limits on Coolant Activity: Dose Equivalent Iodine-131, (μCi/g)	Other Isotopes Value/$\bar{E}$[c]	Standard[d]
Point Beach-2	PWR	(g)	162/$\bar{E}$	No
Prairie Island-1 and-2	PWR	(g)	27/$\bar{E}$	No
Quad Cities-1 and-2	BWR	5.0	(g)	No
Rancho Seco-1	PWR	(g)	43/$\bar{E}$	No
Robinson-2	PWR	1.0	100/$\bar{E}$	Yes
Salem-1	PWR	1.0	100/$\bar{E}$	Yes
San Onofre-1	PWR	1.0	100/$\bar{E}$	Yes
Saint Lucie-1	PWR	1.0	100/$\bar{E}$	Yes
Surry-1 and-2	PWR	1.0	100/$\bar{E}$	No[j]
Three Mile Island-1	PWR	(g)	130/$\bar{E}$	No
Three Mile Island-2	PWR	1.0	100/$\bar{E}$	Yes
Trojan	PWR	1.0	100/$\bar{E}$	Yes
Turkey Point-3 and-4	PWR	1.0[k]	135/$\bar{E}$	No
Vermont Yankee-1	BWR	1.1	(g)	No
Yankee Rowe-1	PWR	1.0	100/$\bar{E}$	Yes
Zion-1 and-2	PWR	(g)	58/$\bar{E}$	No

(a) As of December 1981.
(b) Pressurized water reactor (PWR), boiling water reactor (BWR).
(c) $\bar{E}$ = average disintegration energy, (MeV).
(d) Standard Technical Specifications:
BWR Activity:
0.2 μCi/g, Dose Equivalent Iodine-131, and 100/$\bar{E}$ μCi/g, equilibrium
4.0 μCi/g, Dose Equivalent Iodine-131 during iodine spikes
PWR Activity:
1 μCi/g, Dose Equivalent Iodine-131, and 100/$\bar{E}$ μCi/g, equilibrium
60 μCi/g, Dose Equivalent Iodine-131 during iodine spikes (some plants have higher spiking limits for decreasing power levels).
(e) Arkansas Nuclear One-Unit 1.
(f) Arkansas Nuclear One-Unit 2.
(g) Not available or nonexistent.
(h) Technical Specification stated as total iodine (μCi/g instead of Dose Equivalent Iodine-131.
(i) Increased sampling required when Dose Equivalent Iodine-131 is >0.012 μCi/g.
(j) Other aspects are not standard.
(k) Not specified as either total iodine or Dose Equivalent Iodine-131

Arkansas-1 (Refs. 55-69)

Following a routine shutdown and subsequent startup of the reactor on March 18, 1980, the average gross gas release rate for the first quarter of 1980 reached 4.3%, which exceeded the 4.0% allowed by Engineering Technical Specification (ETS) 2.4.2.1 (Ref. 55). At that time, there had been no previous similar occurrences.

During a plant shutdown on July 25, 1980, the average gross gas release rate for the third quarter of 1980 exceeded ETS 2.4.2.3.a. Previous releases, when compared to a revised and retroactive ETS, were found to exceed ETS 2.4.2.1 (Refs. 56-58). Failed fuel was to be replaced or relocated at the next refueling.

The reactor coolant system activity levels during Cycle 4 operation, after the fuel failures occurred, varied around 0.3 to 0.4 μCi/g with a December 30, 1980 level 0.196 μCi/g prior to shutdown for refueling (the ETS limit is 3.5 μCi/g) (Ref. 61). The reactor coolant activity (iodine-131 dose equivalent) at the end of Cycle 4 was estimated to represent roughly 70 failed fuel rods out of 36,816 total fuel rods in the core (Ref. 57).

Corrective Action for and Cause of Problem. During the January 1981 refueling outage, all 177 fuel assemblies were sipped. Of the 177 assemblies, 24 were identified as leakers by wet sipping. Of the 24 assemblies, 19 were removed before Cycle 5 operation and 5 were shuffled to low power density locations on the core periphery.

Since achieving full power operation following refueling (returned to power operation for Cycle 5 on March 19, 1981), the iodine levels in the reactor coolant system were monitored periodically during the months of April, May, June, July, and November 1981. The indicated activity level was estimated to represent 6 failed fuel rods out of 36,816 total fuel rods in the core. The next scheduled refueling outage is in January 1983.

B&W's studies (Ref. 61) indicated that the defect mechanism was limited to the assemblies that had been identified as having failed in Cycle 4 and was not progressive in nature. B&W reviewed the operating conditions around the time when the failures occurred and found no conditions that should have resulted in fuel failure. Also, B&W reviewed the quality assurance records associated with manufacture of the failed fuel assemblies and found no correlations that would indicate a generic type problem with respect to manufacturing defects.

Arkansas Power and Light Company (AP&L) worked with B&W to try to determine the cause of fuel failures in Cycle 4 (Ref. 62). The failed fuel evaluation plan (Ref. 60) was submitted to the NRC and included visual inspection. A report on the progress of the findings and some photographs were submitted to the NRC--the cause of the failures was not identified and AP&L had no plans to continue the investigation (Ref. 63). Subsequently, the licensee met with the NRC for discussions about the leaking fuel rods (Ref. 69).

NRC Action. The NRC reviewed the "preliminary status report" and the photographs (Ref. 63). Because a) a substantial number (far higher than "usual") of fuel failures occurred, b) they all appeared to have occurred at about the same time, c) the cause of the failures remains unknown, and d) existing regulations and regulatory guides provide both explicit and implicit requirements for a licensee to detect, report, replace, and identify the cause of failed fuel, the staff of the Core Performance Branch of the NRC strongly recommended in September 1981 that a further attempt should be made to identify the cause of the fuel failures (Ref. 63). At the request of the NRC staff, AP&L and B&W met with the NRC in March 1982 to discuss the final report concerning the investigation of leaker fuel at Arkansas-1 and the results of that meeting are provided in Ref. 69. In March 1982, the NRC completed its investigation (Ref. 70) of the Arkansas-1 fuel failures and reached the following conclusions:

1. No definite cause of the failures has been established. Since most of the failures appear to have occurred at about the same time, everyone involved (including the NRC staff) believes that there must have been an initiating event. A thorough investigation, however, has failed to reveal any power transient, water chemistry anomaly, or other trigger that could have initiated the failures.

2. There is no observable correlation of failures with manufacturing batches or lots of material (Zircaloy tubing or ingots). We believe that this conclusion is unambiguous.

3. It appears that there may have been at least two separate mechanisms of failure, inasmuch as several Batch-6 (first cycle of operation) rods appeared to have failed early in life (<2000 MWd/MTU burnup), whereas the other rods (from Batches 4 and 5) failed after appreciable burnup. Thus, it is possible that the Batch-6 failures occurred at random times following a major failure episode that was indicated by a fairly abrupt increase in coolant activity early in Cycle 4.

4. The early-in-life Batch-6 failures may have been due to primary hydriding. Although there is no direct evidence for this, it is plausible in light of the fact that the Batch-6 fuel had a slightly lower density and higher moisture specification than current generations of B&W fuel.

5. If the Batch-6 failures are attributed to a separate failure mechanism from the rest of the failures, the power and burnup correlation is consistant with pellet-cladding-interaction (PCI) failures, so this mechanism should not be ruled out as a possible cause.

6. AP&L and B&W agreed with NRC's conclusion that there was little evidence to suggest that the failures were due to a sudden occurrence of (waterside) corrosion.

7. The licensee appears to have made a conscientious effort to determine the cause of the failures and to remove the failures at the earliest practical opportunity. All but five known leakers were removed at the last refueling outage, and the remainder will be removed at the next refueling even though the failed fuel has not been completely burned. Considering the fact that B&W fuel assemblies are not reconstitutable, and thus do not lend themselves to easy examination, removal, and replacement of failed rods, the NRC believes that AP&L has acted responsibly.

The NRC staff concluded that the licensee had followed up on the investigation in an acceptable manner and that no further work is necessary (Refs. 69 and 70).

Arkansas-2 (Refs. 71-80)

During Cycle 1 in 1981, the primary coolant iodine-131 dose equivalent exceeded the limit of Technical Specification 3.4.8.a. on two occasions (1.59 μCi/g on February 17 and 1.66 μCi/g on March 10) (Refs. 71 and 73). The specific activity increase was caused by an unexpected spike in the primary coolant activity following a reactor trip. The occurrences are similar to those in Licensee Event Reports (LERs) 80-067 and 80-009.

Cycle 2 operation started on June 29, 1981. Based on a comparison of iodine-131 activities in Cycle 2 and Cycle 1, the licensee inferred that the number of perforated fuel rods in the Cycle 2 core is about one-half of the number present in the core at the end of Cycle 1 (Ref. 75). The next scheduled refueling outage is in September 1982.

Corrective Action for and Cause of Problem. During Cycle 1, the reactor coolant system iodine levels indicated the presence of some 20 to 30 failed fuel rods. At the refueling outage at the end of Cycle 1, the entire core was sipped. Seven leaking fuel assemblies were identified by wet sipping. Of the seven, five were to be reloaded for Cycle 2 operation. Eddy current testing was used to locate the failed rods in the five assemblies (Ref. 80). A total of 66 fuel rods and 1 burnable poison rod were replaced in the five assemblies, which were then resipped to verify integrity. From preliminary observations (Refs. 79 and 80), AP&L and C-E speculate that more than one damage mechanism may have been operating. The investigation is continuing. AP&L transmitted a summary of the fuel status at Arkansas-2 at the end of Cycle 1, including preliminary findings, to the NRC in May 1981 (Ref. 81).

NRC Action. The NRC reviewed the operating experience with the C-E 16 x 16 fuel design (Ref. 80). The only actual experience with that design comes from operation in Arkansas-2. Not all of the damage mechanisms responsible for the fuel problems (see above and also Section 4.1.3 on "Spacer Grid Damage") are fully understood at this time. The NRC believes that the fuel problems that have been encountered are typical of those experienced with other designs and are probably unrelated to the design change to a 16 x 16 geometry.

The NRC has performed a safety evaluation on the Arkansas-2 reload analysis for Cycle 2 operation (Ref. 82). The NRC has concluded that the reload fuel system design is acceptable as conditioned. The licensee is to provide a final report describing the results of fuel inspections (the licensee has reported by two letters in 1981 the preliminary results and partial conclusions from fuel assembly inspections). Since the reasons for some Cycle 1 fuel failures are not presently known and hence additional failures cannot be ruled out during Cycle 2 operation, the licensee has agreed to notify the NRC by letter in the event that plant instrumentation indicates additional fuel failures.

Brunswick-2 (Refs. 83-90)

The reactor coolant activity exceeded the technical specification limit of 0.2 μCi/g for iodine-131 dose equivalent on November 14, December 10, December 12, and December 18, 1980, and on July 3 and November 9, 1981.

Corrective Action for and Cause of Problem. The high coolant activity was a result of a reactor transient (startup, shutdown, or scram) with a subsequent increase in coolant fission product inventory originating from leaking fuel. The corrective actions taken were to reduce power (three cases), increase power (one case), or maintain constant coolant temperature (one case). The licensee also sought regulatory relief to allow higher off-gas activity (Ref. 88). The fuel bundles are to be sipped during the next refueling outage (currently scheduled to take place in September 1982)and the leaking fuel bundles removed from the core. In April 1982, the licensee indicated that an iodine spike on March 14, 1982, was associated with 7 x 7 fuel (Ref. 90).

NRC Action. The NRC evaluated the requests from the licensee concerning the reporting of off-gas activity. On December 14, 1981, the NRC issued Amendments 43 and 66 to the licenses for Brunswick-1 and -2, respectively, which involved changes (modified methods used) to the technical specifications. On February 23, 1982, the NRC issued Amendments 45 and 68 to the licenses for Brunswick-1 and -2, respectively, which allowed the licensee to shift the accounting period from 12 months to a calendar year.

Cook-2 (Refs. 91 and 92)

Following a shutdown, the reactor coolant system dose equivalent iodine concentration on October 2,1981, was found to have exceeded the limit of 1.0 μCi/g in Technical Specification 3.4.8. Previous similar occurrences include LERs 78-026 and 76-059 for Cook-1. The next scheduled refueling outage is in May 1982.

Corrective Action for and Cause of Problem. Upon discovery of the high dose equivalent iodine, the CVSC letdown purification flow was maintained at the maximum available.

Crystal River-3 (Refs. 93-102)

The reactor coolant activity exceeded the limit of 1.0 μCi/g for dose equivalent iodine-131 in Technical Specification 3.4.8 on December 8, 1980, and on February 17, June 18, June 27, July 1, July 31, and September 28, 1981. The occurrence on September 28 was the twenty-fifth event to be reported for this plant under this specification. The plant was shut down for a refueling outage on September 28 and returned to power operation on December 13, 1981.

Corrective Action for and Cause of Problem. The licensee attributes the events to leaking fuel or "tramp uranium" (i.e., uranium in the cladding within one recoil radius of the surface) and an anticipated dose equivalent iodine transient following a reactor coolant system transient. In the 1980 event, the dose equivalent iodine-131 level was reduced below the limit by recirculating the reactor coolant system through the makeup and purification system. In the six events in 1981, the reactor coolant system flow rate through the makeup and purification system was increased to correct the dose equivalent iodine-131 level. The licensee is also conducting an evaluation to determine if failed fuel is present (Refs. 96, 99, and 100). During the September-December 1981 outage, only new fuel was inspected; the utility indicated to the NRC that they believe they have no fuel failures and that the reason for reporting iodine spikes is that the limit is set too low.

NRC Action. Starting in 1980, the Core Performance Branch of NRC attempted to learn more about the nature of the fuel failures but has been unable to determine the exact number or types of fuel failures (Ref. 102).

Davis Besse-1 (Refs. 101-115)

The dose equivalent iodine-131 level in the reactor coolant system exceeded the limit of 1.0 μCi/g in Technical Specification 3.4.8.a. on December 3, 1980, and on March 8, May 12, July 30, September 2, October 16, and October 23, 1981. Before the event on December 3, 1980, there had been no previous reports of high iodine levels. During the refueling outage in early 1982, one broken hold-down spring was observed (see Section 4.1.2) but there was no other comment about fuel inspection (Ref. 116).

Corrective Action for and Cause of Problem. The iodine-131 level was monitored until it dropped below the limit. The licensee stated that a corrective action was not applicable, iodine spikes are typical following a reactor transient (in one case a power reduction was involved; the other cases involved reactor trips), and no damage to the fuel was indicated. The licensee submitted Facility Change Request 81-163, which involves a technical specification revision to change the limit (Refs. 107 and 108).

NRC Action. The request to change the limit is being evaluated by the NRC (Refs. 107 and 108) and will be discussed in the next annual report. As of early May 1982, no decision has been made.

La Crosse (Ref. 117)

The plant completed the refueling outage and Core 7 reload and returned to operation on January 31, 1981. Reactor coolant iodine concentrations and off-gas iodine releases in excess of the technical specification limits occurred five times between January and March 1981. (Also see Section 4.1.6-"Alpha Activity in Coolant") (Refs. 118-121). The next refueling outage was in April 1982. During that outage, the fuel was visually inspected and two-thirds of the core was dry sipped (Ref. 237). In August 1982, the licensee provided NRC with a report (Ref. 122), which indicated that abnormal degradation was discovered on April 20, 1982, in a fuel rod in a fuel assembly made by Allis Chalmers. The broken rod (it had a degraded segment that was about 5.5 in. long) was detected during visual inspection with an underwater TV camera. This was one of two remaining Allis Chalmers assemblies in use.

Corrective Action for and Cause of Problem. The utility believes the stainless steel-clad fuel rod failed by the same mechanism (i.e., pellet-cladding interaction with oxygen-assisted stress corrosion cracking) as earlier failures. The licensee indicates that reactor transients contributed to the problem. Power changes exceeding 15% per hour occurred on those occasions. See note in Ref. 117. The two Allis-Chalmers fuel assemblies were replaced with ENC fuel assemblies (Ref. 122).

North Anna-1 (Refs. 123-126)

On July 10, July 12, and August 4, 1981, the reactor coolant activity exceeded the dose equivalent iodine-131 limit in Technical Specification 3.4.8.As of December 31, 1981, the date of the next scheduled refueling outage has not yet been established.

Corrective Action for and Cause of Problem. The events were caused by a known fuel element defect in the core. Post reactor trip conditions in the core enhanced the release of fission fragments to the reactor coolant system, which caused the iodine spike. Sampling frequency was accelerated until the reactor coolant system specific activity returned to less than the limit in Technical Specification 3.4.8.a.

NRC Action. The NRC telephoned the licensee in June 1982. The licensee indicated that licensee event report was inadvertantly misworded: there are probably more failed fuel rods than the one mentioned (there may be several dozen failed rods).

Palisades (Refs. 127-131)

On January 15, 1981, a primary coolant system sample revealed a dose equivalent iodine-131 value that exceeded the limit of 1.0 μCi/g in Technical Specification 3.1.4. The event was not repetitive. The fuel in the core was supplied by Combustion Engineering and Exxon Nuclear.

Corrective Action for and Cause of Problem. Sampling frequency was increased as stipulated by the technical specification. Low-level cladding failure combined with transient conditions following a plant trip resulted in the high activity. No operating history (e.g., power escalation rates) that would contribute abnormally to fuel cladding failure has been identified.

During fuel inspection conducted subsequent to refueling, a single failed assembly was found (Ref. 29). Four defects were found in the cladding of one fuel rod on October 12, 1981. The most significant defect was a hole measuring approximately 0.5 inch in length and 0.25 inch in width. No fuel pellet was visible in the hole. The defective fuel rod was in an Exxon Nuclear fuel assembly that had attained a burnup of approximately 35,300 MWd/MTU. Extensive deterioration of the primary failure site prevented Exxon Nuclear from positively identifying the cause of the failure. Exxon Nuclear stated that the failure is thought to represent an isolated occurrence and not a generic problem (Ref. 29).

NRC Action. The NRC contacted the licensee on November 12, 1981, for information concerning the fuel rod failure (Ref. 131). The licensee indicated at that time that "they don't believe pellet-cladding interaction (PCI) was the cause of the failure." Additional information on the results of the fuel vendor's inspection is available in a proprietary report (see p. 5 in Ref. 29).

St. Lucie-1 (Refs. 132-135)

Following a plant trip from 100% power after an extended period of operation with a nominal level of fuel leakage, the dose equivalent iodine-131 level exceeded the technical specification limit of 1.0 μCi/g on September 8, 1981. This is the fifth occurrence of this type (fourth was on August 14, 1980). The plant remained shut down for the scheduled refueling outage. The plant returned to operation on December 3, 1981. The next scheduled refueling outage is in March 1983.

Corrective Action for and Cause of Problem. The dose equivalent iodine-131 level decreased to below the limit after 12 hours. After an extended period of power operation (140 days) with a nominal level of fuel leakage, a plant trip was a sufficient transient to cause iodine build-up (i.e., iodine spiking phenomenon).

Surry-1 (Refs. 136-144)

Following reactor trips on August 22 and 23, November 25, and December 16, 1981, and on April 25, 1982, and during a return to power after reactor trips on September 12 and November 26,1981, the dose equivalent iodine-131 levels of samples of the reactor coolant exceeded the limit in Technical Specification 3.1.D.2. After a reactor trip on November 29, 1981, the specific activity samples of the reactor coolant indicated a dose equivalent iodine-131 level greater than the Technical Specification limit. On

June 5, 1982, after a short reduction in power (down to 5%) while increasing power, the dose equivalent iodine-131 exceeded the Technical Specification 3.4.8.A limit.

Corrective Action for and Cause of Problem. With the two events in August, the sampling frequency was accelerated and maximum primary coolant purification was implemented. With the events in September, November, and December 1981 and in April 1982, the sampling frequency was increased. With the event in June 1982, the dose equivalent iodine-131 level decreased to below the limit after eight hours. The iodine spikes were caused by known, yet not specifically located, fuel element defects in the reactor core. The conditions following a trip, power increase, or power reduction enhanced the release of fission products (specifically iodine-131) to the reactor coolant system, causing an increase in coolant specific activity level.

NRC Action. The NRC telephoned the licensee in June 1982. The licensee indicated that the fuel vendor estimates there are about 20^{+30}_{-10} failed fuel rods in the core. It is planned to sip the fuel at the next outage (February 6, 1983) and to remove the leakers.

Trojan (Refs. 145-151)

In August 1981, it was reported that the steady-state, dose equivalent iodine-131 and gross gamma levels in the reactor coolant system were about 15 to 20% of the technical specification limits. An apparent step increase in the dose equivalent iodine-131 and gross gamma levels occurred sometime between the reactor trip on October 12 and the return to full power on October 27, 1981. During October and November the gross iodine level ranged up to about 50% of the limit and the gross gamma level ranged from 25 to about 40% of the limit. In December 1981, the reactor coolant system gross gamma activity was about 35 to 40% of the technical specification limit, with the coolant gross iodine level at approximately 50 to 60% of the limit (the limit is 1.0 μCi/g).

Corrective Action for and Cause of Problem. The licensee initiated an investigation into the nature and extent of the fuel failures. Prior to that time, the licensee had visually examined 44 fuel assemblies at the end of Cycle 3 and found no significant degradation (Ref. 151). (See Section 4.1.5 on "PWR Baffle Jetting").

In October 1981, the results of radiochemistry analyses by the licensee indicated failure of 10 to 15 fuel rods in fuel assemblies (Ref. 145). On November 4, the licensee indicated that they were continuing to closely monitor the primary coolant activity levels and would consider possible corrective measures if the activity levels approach the technical specification limits (Ref. 146). At that time, the licensee estimated that between 6 and 25 fuel rods are involved, the defects are "open" rather than "capillary" in nature, and, based on the cesium-137/cesium-134 ratio, the defects appear to be fuel that is in its second burnup cycle (Ref. 146). The last conclusion, however, did not rule out the possibility of defects involving both older and fresh fuel.

Fuel inspection in April 1982 revealed that 17 fuel assemblies have degraded fuel cladding (Ref. 150). The apparent cause of the fuel rod damage is baffle jetting. See "Trojan" in Section 4.1.5. The next scheduled refueling outage is in May 1983.

4.1.2 PWR Hold-Down Springs

In 1981, only one plant, Arkansas-1, reported hold-down spring damage. There were follow-up activities in 1981 on the 1980 spring failures at Davis Besse-1 and Oconee-1 and -2. This subject will be discussed further in the next annual report as broken hold-down springs were observed in 1982 on an assembly at Davis Besse-1 (Ref. 116) and on assemblies at Oconee-1 and -2 (Ref. 152).

Arkansas-1 (Refs. 153-155)

During the January 1981 refueling outage at the end of Cycle 4, a 100% inspection of hold-down springs on core fuel assemblies was performed. One fuel assembly out of the 177 examined was found to contain a broken hold-down spring. At the previous refueling outage all fuel assemblies had been inspected and no broken hold-down springs were observed.

Corrective Action for and Cause of Problem. The broken spring was replaced. The broken spring had a single, clean, torsional-type, through fracture in the top active coil about 30 degrees away from the active/dead coil transition area. The break was normal to the coil of the spring and there was no apparent chemical degradation. The position of the break resulted in the spring maintaining a preloaded condition. The associated fuel assembly (Batch 5) was examined for evidence of axial motion and none was observed.

The failed spring and all springs used in 56 Batch 5 fuel assemblies were from the same heat. Springs from that same heat have been irradiated in other reactors: 57 springs at Oconee-3, 56 springs at Oconee-1, 50 springs at Oconee-2, and 21 springs at Three Mile Island-1. No other failures have been reported.

This spring failure at Arkansas-1 is similar in frequency and visual appearance to the spring failure at Crystal River-3 (Refs. 156-159), which hot cell examination showed to be caused by fatigue, initiating at a surface anomaly. Because of the similarity, the licensee has no plans to have the failed spring at Arkansas-1 analyzed for metallurgical condition (Ref. 155).

The replacement spring (and all springs in Batch 7 fuel assemblies) is from a heat that has demonstrated an acceptable grain structure and has undergone stringent surface examination, both now design requirements (these should reduce or eliminate any future hold-down spring failures).

The licensee plans to conduct a 100% inspection of the core fuel assembly hold-down springs at the next refueling outage and to report on the results of the inspection to the NRC.

NRC Action. The NRC reviewed the licensee's response to an NRC request for information regarding springs at Arkansas-1 (Ref. 154). The NRC considered that the matter of failed springs was addressed satisfactorily in all respects except for future surveillance. The NRC requested that the licensee perform a 100% inspection of core fuel assembly hold-down springs at the next refueling outage (scheduled for January 1983) and to provide a report on the inspection results. The NRC also requested (and received) a report on the failed spring.

Davis Besse-1 (Refs. 116,160, and 161)

In August 1981, the licensee submitted a revised report concerning the 20 broken hold-down springs that were found in 1980 on fuel assemblies from Cycle 1. There had been no previous similar reportable occurrences. This event, the cause of the problem, the corrective action by the licensee, and the NRC action were described in the fuel performance annual report for 1980 (Ref. 162). The springs on all 133 fuel assemblies from Cycle 1 that were to be used in Cycle 2 were replaced. Forty-four fresh fuel assemblies were also to be used in Cycle 2. All 177 springs in use on fuel assemblies in Cycle 2 operation were manufactured to current specifications, which incorporate improvements in grain size control, annealing process, coiling and dimensional standards, and several other areas.

One broken hold-down spring was observed during the refueling outage in early 1982 (Ref. 116). The fuel assembly is being discharged and the broken spring is being sent to the vendor for evaluation.

Oconee-1,-2, and -3 (Refs. 152 and 163-165)

In December 1980, the licensee submitted a letter containing supplemental information concerning the broken hold-down springs on 5 of 686 fuel assemblies at the Oconee Nuclear Station. This event, the cause of the problem, and the corrective action by the licensee, and the NRC action were described in the fuel performance annual report for 1980 (Ref. 162). The licensee inspected the springs at Oconee-3 during the recent Cycle 6 reload.

In February 1982, video inspection found broken hold-down springs on three Oconee-2, Batch 7 fuel assemblies and on one Oconee-1, Batch 4 fuel assembly (Ref. 152).

Corrective Action for and Cause of Problem. The apparent cause of the spring failures is fatigue-induced cracking at an existing surface flaw, which then propogated by fatigue (Ref. 152). The hold-down spring inspection program will be continued until inspection results justify ending the program.

NRC Action. In February 1981, the NRC requested a commitment by the licensee to perform surveillance on hold-down springs presently in Oconee-1 and -2 cores during the next refueling outage and to report any spring failures noted. As noted above, the surveillance program is continuing because of the event in February 1982.

4.1.3 Spacer Grid Damage

Spacer grid damage was reported in 1981 by licensees for two PWRS: Arkansas-2 and Indian Point-2.

Arkansas-2 (Refs. 76-78, 80, and 166)

Sixty discharged Batch A fuel assemblies were visually examined with underwater television equipment. Damage to Zircaloy spacer grid perimeter straps was observed on five fuel assemblies. The spacer grid damage on two assemblies was extremely minor but the damage on three assemblies was significant (Ref. 78).

Corrective Action for and Cause of Problem. The licensee indicated that the damage occurred during fuel handling subsequent to Cycle 1 operation (i.e., after March 27, 1981). The damage resulted from grid-to-grid interaction with adjacent fuel assemblies during fuel handling within the core. It is estimated that 16 assemblies with damaged grids remain in the core and that fewer than 24 fuel rods are adjacent to the damaged grid sections, which means the potential damage due to fretting during Cycle 2 is limited. The licensee concluded that operation of Arkansas-2 with potentially degraded fuel assembly grid straps does not constitute a safety concern and is acceptable for Cycle 2 operation (Ref. 78).

The licensee's evaluation has not yet identified any problems that would have been expected to cause the observed grid damage. The investigation was continued and the licensee transmitted a final report (Ref. 166) to the NRC in June 1982 that describes the fuel inspection results, conclusions reached, and preventive measures to be employed in the future.

NRC Action. The NRC requested (and received) written documentation concerning the spacer grid damage. The NRC also requested that AP&L continue to investigate the cause of the spacer grid damage and to take corrective action prior to the next refueling. The NRC has also required AP&L to provide a loose parts monitoring program and to be alert to additional fuel failures that might occur due to fretting from loose grid pieces or inadequately supported fuel rods at the damaged grid sections. As noted above, the NRC received the required final report (Ref. 166). This subject will be covered more fully in the NRC's next annual report.

Indian Point-2 (Refs. 167-172)

Cycle 4/5 fuel shuffle operations started on December 30, 1980. During fuel handling, two fuel assemblies were found to each have a damaged grid strap at one corner. As fuel shuffling operations continued, some other assemblies experienced handling difficulties.

Corrective Action for and Cause of Problem. The core was completely discharged. All the fuel assemblies from Core 4 (except for 14 new Region 7 assemblies that were not handled) plus 58 new Region 7 assemblies and 7 assemblies discharged from Cycle 3 were inspected. The 258 fuel assemblies were visually inspected using an underwater television camera and the videotaping system. Grid strap damage in the form of torn or missing corners was observed on 87 assemblies. No damage to the fuel cladding has been identified that would be attributable to the grid strap problem.

The grid strap damage was caused by corner-to-corner interaction between adjacent assemblies during fuel handling operations. Following the licensee's evaluation, the core was reloaded for Cycle 5 operation using modified refueling procedures that minimize corner interaction between adjacent assemblies. The Cycle 5 core contains 49 of these assemblies with damaged grids plus 144 assemblies with no grid damage. Of the 49 assemblies, 3 had damage that was minor and inconsequential, 36 were considered acceptable only on the basis of special handling to preclude deterioration of existing damage or propagation of existing damage to other assemblies through interactions, and 10 were damaged enough to affect rod support. Evaluation of the 10 assemblies indicated that fretting wear would exceed the normal design limit but would not result in fuel cladding failure. The 10 assemblies are to be reexamined after each cycle of operation to determine acceptability for duty beyond one additional cycle.

The licensee indicates from their evaluation of the grid strap damage that there will be no adverse effects on Cycle 5 normal operation or postulated accident conditions.

4.1.4 Accelerated Corrosion

In 1981, fuel rod failures due to accelerated corrosion were reported by one domestic plant, Hatch-1. Similar type fuel rod failures occurred in 1979 in Vermont Yankee: see fuel performance annual reports for 1979 (Ref. 173) and 1980 (Ref. 162).

Hatch-1 (Refs. 174-180)

Fuel element leakage has been increasing since the startup from fueling of the plant for Cycle 5 operation in June 1981. On September 23, 1981, when at 100% power, the off-gas levels increased sharply but no technical specification limits were exceeded.

Corrective Action for and Cause of Problem. The power level was dropped to control the off-gas rates. The licensee shut down the plant on October 9 (the derated operation continued until then) for the sole purpose of removing and relacing leaking fuel assemblies found by in-core sipping (Ref. 177). All 560 fuel assemblies in the core were sipped and 11 Reload-2 fuel assemblies were identified as containing approximately three dozen failed fuel rods (Ref. 180). The licensee attributes the failures to accelerated corrosion of the cladding (the failures are similar to those seen earlier at Vermont Yankee). At least two other nonleaking fuel assemblies exhibit incipient corrosion (Ref. 176). Further investigation indicated that the mechanism causing the fuel failures was crud-induced localized corrosion. Of the factors that contribute to that mechanism, two important ones are variable water chemistry (copper contamination) and Zircaloy cladding corrosion resistance (Ref. 180). In the fuel bundles that were inspected, all of the failed rods were burnable poison type that contained gadolinia pellets (Ref. 180).

All of the affected bundles were determined to be unacceptable for further use and were replaced (Ref. 180). Also, all of the burnable poison rods (failed and nonfailed ones) were replaced. Of the 168 bundles removed from the core, the vendor expected to reconstitute 130-140 for reuse (Ref. 179). Several fuel assemblies from the remaining and replacement batches were inspected and found acceptable for continued use. The next scheduled refueling outage for Hatch-1 will begin in September 1982.

Hatch-2 contains similar fuel but there has been no evidence to indicate that the corrosion is occurring there (i.e., the off-gas levels do not indicate fuel failures in Hatch-2). The Hatch-2 fuel assemblies will be examined on an as-appropriate basis.

NRC Action. The NRC issued a preliminary notification, PNO-II-81-85 (Ref. 174), about the occurrence and subsequently described the event in an entry in NRC's bi-monthly newsletter (Ref. 180).

4.1.5 PWR Baffle Jetting

During 1981, baffle joint jet flow caused bowing of a fuel rod at Trojan. Fuel rods in two Japanese PWRs, Ohi-2 and Takahama-1, were damaged by baffle jetting in 1981. Failed rods were also observed in four fuel assemblies at Korea Nuclear-1 (Ko-Ri) in 1981 (Ref. 49). Some fuel rods failed in four assemblies at Tihange-1 (Ref. 29). (Also, see Section 4.2.8.)

Trojan (Refs. 150, 151, and 162)

During 1980, only one domestic plant, Trojan, reported the occurrence of the baffle jetting problem (two fuel rods with cladding damage were observed). The cause of the problem, the corrective action by the licensee, and the NRC action were described in the fuel performance annual report for 1980 (Ref. 162).

The licensee reported in June 1981 that all 12 fuel assemblies subject to cross-flow baffle jetting were visually inspected by television at the end of Cycle 3. Only one rod in one of the 12 fuel assemblies showed evidence of being affected by baffle joint jet flow. The rod was bowed in the span between the first and second grids from the bottom of the assembly but did not appear damaged or fretted.

The preplanned fuel inspection conducted during a refueling outage in April 1982 identified 17 fuel assemblies that have degraded fuel cladding (Ref. 150). Visual inspection revealed that some perimeter rods in eight Region F fuel assemblies have sustained severe damage. Portions of rodlets are missing and loose pellets were found. Sipping identified nine other nonobvious damaged fuel assemblies with failed fuel rods.

Corrective Action for and Cause of Problem. An augmented fuel inspection program was conducted (Ref. 150). All fuel assemblies to be used in the subsequent cycle were leak checked (sipped) and visually inspected to be damage free. Accessible loose pellets and debris will be retrieved from the reactor vessel internals and refueling cavity. Damaged assemblies adjacent to the baffle will be replaced with new modified fuel assemblies using stainless steel pins and/or other devices (e.g., clips) to ensure fuel integrity.

NRC Action. This subject will be discussed more fully in the next NRC annual report as the baffle jetting problem has become worse at Trojan in 1982 and it is also now occurring at Farley.

Ohi-2 and Takahama-1 (Ref. 181)

In July 1981, the activity level of the primary coolant increased in both of these Japanese PWRs (Westinghouse was the contractor for the reactor systems and cores at both plants). In August and September during regular inspection for fuel loading at both plants, some damaged fuel rods were found by sipping.The damage to the fuel rods was caused by fretting, probably due to baffle jet impingement. The damaged fuel rods were located near the corner of the baffle plates. This kind of fuel damage has been controlled by the Japanese with the aid of maximum momentum flux criteria; however, it has been found that damage occurs even if the baffle gap is maintained to satisfy the criteria.

NRC Action. In response to a request from the Japanese, the NRC discussed with them the U.S. experience and the regulatory view concerning this matter. The Japanese believe that rod bowing is related to baffle jetting and that rod bowing affects rod spacing and the phenomenon of momentum flux.

4.1.6 Alpha Activity in Coolant

In 1981, one licensee reported high alpha activity in the primary coolant of a BWR.

La Crosse (Refs. 118 and 119)

During startup from a refueling outage, the indicated gross alpha activity of the primary coolant exceeded the limit of 5×10^{-6} μCi/g in Technical Specification 4.2.2.22.a.3. Similar occurrences are described in LERs 80-006, 80-005, 79-010, and 78-005.

Corrective Action for and Cause of Problem. Operation of the primary coolant purification system reduced the alpha concentration to an allowable value. The high alpha activity indication was temporary and did not indicate recent fuel failures. The alpha activity came from residual irradiated fuel material that entered the system as a result of degraded stainless steel cladding on fuel rods primarily experienced during Cycle 4 and to a much lesser degree during Cycle 5 and subsequent handling of the irradiated fuel assemblies during refueling.

4.1.7 Dimensional Error

Dimensional errors were noted at one domestic PWR in 1981.

Millstone-2

Out-of-tolerance guide tube extensions were discovered in fuel delivered by Westinghouse for a December 15, 1981, refueling outage at Millstone-2 (Ref. 182). In 1980, 72 fuel assemblies at Millstone-2 were found to be the wrong size and were returned to Westinghouse for modification (Ref. 183).

4.2 NEW PROBLEMS IN 1981

4.2.1 Fuel Damage During Refueling Operations

A licensee reported on damage sustained by one fuel assembly during refueling operations in 1981.

Cook-1 (Refs. 184-188)

During Cycle 6 refueling operations on June 19, 1981, a 15 x 15 spent fuel assembly was damaged during movement from the reactor to the fuel transfer system containment side upender. There was no increase in normal containment background. As a result of the collision of the end of the fuel assembly with the shield wall retaining lip in the refueling cavity, one of the fuel rods was jarred loose from the assembly, fell out, and lodged behind a ladder in the cavity. Three rods were out of their normal positions, but were intact (Ref. 188). The rest of the fuel assembly was slightly distorted.

Corrective Action for and Cause of Problem. The damaged fuel assembly was removed from the upender, inspected, and transferred without incident to the spent fuel storage area outside the containment. No detectable increase in radiation levels occurred. The loose fuel rod was recovered. The licensee is assessing the problem. Westinghouse indicated to the licensee that in over 100 refuelings, a similar incident has not previously occurred to the best of their knowledge (Ref. 188).

NRC Action. The NRC Senior Resident Inspector observed all preparations, inspection, and transfer. The matter is expected to be dealt with in a later report by the resident inspector.

4.2.2 Loose Guide Tube Nut

A loose guide tube nut was reported at one plant in 1981.

Crystal River-3 (Refs. 189 and 190)

On March 26, 1981, a loose part was detected by the loose parts monitoring system atop the tube sheet in the "B" Once-Through Steam Generator.

Corrective Action for and Cause of Problem. The part was removed and forwarded to B&W for analysis. B&W identified the part to be a control rod upper guide tube nut. Approximately 46,000 nuts of this design have been or are in service in B&W plants with no reported problems. This is a unique isolated occurrence. The effects of continued operation of a fuel assembly missing the upper nut from 1 of 16 control rod guide tubes was evaluated and it was concluded that the unit was safe to operate (Refs. 189). The licensee will increase the level of attention given to loose parts monitoring.

NRC Action. The NRC has reviewed the information concerning the loose guide tube nut and concluded that there is no evidence of a generic defect in B&W's guide tube nut design. The NRC concluded that the licensee has taken the appropriate actions to resolve the concerns of the NRC staff and that resumption of power operation is acceptable (Ref. 190).

4.2.3. Hydride Defect

Indian Point-2 was the only plant to report a fuel rod with a hydride defect in 1981.

Indian Point-2 (Refs. 191-193)

During a review of the video inspection tape of a Westinghouse 15 X 15 (9-grid) fuel assembly, No. E-42, the licensee determined that there was an apparent perforation of the cladding on a fuel rod (second row in).

Corrective Action for and Cause of Problem. The licensee judged the perforation to be a hydride defect. It is an isolated instance of a single defect of a well known type. The licensee indicates that it is not abnormal and that the expected coolant activity from this single defect is about 0.005 μCi/g of iodine-131. Removal of this single defect would not noticeably reduce the 0.08 μCi/g of iodine-131 in Indian Point-2 in Cycle 4 or that expected in Cycle 5. The defect is not expected to degrade during further assembly use nor would it be expected to result in significant coolant activity. The fuel manufacturer has reviewed the defect, determined that no corrective is required, and recommended that the assembly is suitable for further use.

4.2.4 Fuel Assembly Dropped

Two fuel assemblies (one domestic, one foreign) were dropped in 1981.

Millstone-1

A BWR fuel assembly was dropped during core loading at Millstone-1 on March 26, 1981 (Ref. 194).

Lovissa-1

It was also reported (Ref. 195) in 1981 that a PWR fuel assembly at a Finnish reactor, Loviisa-1, was dropped approximately 0.5 m. The fuel rods were hermetic but the shroud tube had some dimensional changes because of deformation. A fuel assembly in this Soviet type VVER-440 reactor has a hexagonal cross section and is covered by a shroud tube (individual fuel rods cannot be examined because the shroud is not designed to be removable).

4.2.5 Top Nozzle Broken Off

One licensee reported in 1981 that a top nozzle broke off of a spent PWR fuel assembly during handling.

Prairie Island-1 (Refs. 196-200)

On December 16, 1981, the top nozzle of a spent Westinghouse fuel assembly separated from the remainder of the assembly as it was being lifted out of a storage rack in the spent fuel pool. The assembly was almost out of the rack when the top nozzle broke off. The assembly did not fall, but tipped over approximately 20 degrees from the vertical. The top of the fuel assembly is leaning against the wall of the spent fuel pool and is resting in a slot in the wall, making it difficult for the assembly to be accidentally moved such that it would fall. No radiation releases and no fuel rod damage occurred. The fuel assembly had attained a burnup of 29,000 MWd/MTU and was discharged in April 1979 (Ref. 49).

Corrective Action for and Cause of Problem. All movements of heavy objects over the area were stopped. The licensee contacted Westinghouse for recommendations for grasping the assembly so that it could be lifted and restored to a more stable condition. The fuel assembly was subsequently lifted and inserted into a storage position.

Short stainless steel sleeves, which are welded to the top nozzle, are mechanically joined (bulge joint) to the Zircaloy control rod guide thimbles in this fuel assembly (the sleeves and thimbles are the load-bearing members when the assembly is lifted). The separation occurred at the upper-most bulge joint in all 16 of the stainless steel sleeves. The Zircaloy thimbles remained intact. Based on the preliminary results from the metallurgical examinations, the vendor designates the apparent cause of the bulge joint failures to be intergranular stress corrosion cracking of the stainless steel in the vicinity of the bulge (Refs. 49 and 198). The cracking is believed to have developed during pool storage. This is the first incident of this type with a Westinghouse fuel assembly (Ref. 49). More information became available in October 1982 (see below).

NRC Action. On December 16, 1981, the NRC issued a preliminary notification, PNO-III-81-116, about the occurrence. The notification indicated that both reactors were operating and that all spent fuel area safety-related systems were operational.

In October 1982, the NRC contacted the Prairie Island Resident Inspector and Westinghouse personnel (Refs. 199 and 200). The subject will be covered briefly here but more fully in the NRC's next annual report. Westinghouse performed postirradiation examinations at Prairie Island and TV examinations at Trojan, Kewaunee, Point Beach (all three have the same design as Prairie Island), and Zion. The same material lot was used in fuel assemblies at Prairie Island, Point Beach, and Kewaunee. Results from the TV examinations at all plants except Prairie Island showed no evidence of stress corrosion cracking in the top nozzle.

At Prairie Island, a total of 27 assemblies were examined and 12 showed evidence of corrosion but there were no additional nozzle failures when the assemblies were moved (Refs. 199 and 200).

NRC's Office for Analysis and Evaluation of Operational Data (AEOD) does not have specific recommendations for immediate action related to correcting the problem because the root causes have not been identified. AEOD does suggest (Ref. 200) that the Office of Nuclear Reactor Regulation consider implementing the following items:

1. Notify licensees that may have plans for imminent transfer of Westinghouse fuel in the spent fuel pool.

2. Review the need for and feasibility of monitoring fuel in the spent fuel pool to detect stress corrosion cracking.

3. Review the adequacy of current guidance and/or the need for and feasibility of developing additional guidance for transfer (elevation, path, etc.) of fuel assemblies in the spent fuel pool.

4. Review whether spent fuel pool water chemistry specifications are adequate.

AEOD plans to report this event in the next issue of Power Reactor Events and to send a report to foreign countries through the NEA Incident Reporting System (IRS).

4.2.6 Refueling Integrity Breached

In 1981, one licensee reported that refueling integrity had been breached.

Surry-2 (Ref. 201)

With the unit at cold shutdown for refueling, an operator discovered that the refueling integrity had been breached while fuel was being moved, which is contrary to Technical Specification 3.10.A.1.

Corrective Action for and Cause of Problem. Contractor personnel had removed (for maintenance) an inside and an outside trip valve, which resulted in an open pathway from the containment atmosphere to the auxiliary building atmosphere. Fuel movement was terminated until one of the valves was replaced.

The auxiliary building atmosphere was being monitored and containment air samples showed contaminant levels remained below maximum permissible concentration values.

4.2.7 Control Rod Follower Bowed

During 1981, one licensee reported bowing of a control rod follower.

Yankee Rowe (Refs. 202 and 203)

During a controlled plant shutdown for refueling while inserting shutdown rods, one rod became inoperable (contrary to Technical Specification 3.1.3.1) and would not scram upon loss of stationary gripper coil voltage (contrary to Technical Specification 3.1.3.3). Similar occurrences were reported as LER 79-02, AO 72-11, AO 67-5, AO 63-11, and AO 63-1.

Corrective Action for and Cause of Problem. The actions required by the technical specification were met and the shutdown margin was equal to or greater than 4.72%. The rod was inserted by using the pulldown coil. The root cause of this event was a bowed Zircaloy control rod follower on a cruciform-shaped hafnium control rod. The bowing appears to have been caused by a Zircaloy growth phenomenon, which may occur in rods that were manufactured in the 1960s and is due to the method of stress relieving.

During 1972 all control rods were relaced with the exception of two hafnium control rods. This change was due to control rod follower distortion (see Changes 97 and 104). The rods replaced in 1972 have not shown signs of bowing. The two hafnium rods showed no distortion at that time and were left in the core.

Since one of the two hafnium rods now exhibits this phenomenon, both rods will be replaced with spare Ag-In-Cd rods of the same design during the present Core 14-15 refueling.

4.2.8 Vibration-Induced Fretting

In 1981, fuel rod failures in two fuel assemblies due to vibration-induced fretting were reported.

Yankee Rowe (Refs. 29 and 204-210)

Some fuel rod failures were suspected because of high coolant activity levels measured during Cycle 14 operation.

Corrective Action for and Cause of Problem. An examination was initiated by the licensee during the reload outage. All fuel assemblies except one were sipped. The exception was assembly B574, which had a visible indication of fuel failure. The sipping results indicated fuel failure in another assembly, which was discharged.

Fuel assembly B574, which had been manufactured by Exxon Nuclear, had nine damaged fuel rods: four rods had circumferential breaks (also two end caps are missing) and eddy-current testing of all rods indicated that five had fretting damage. The sound fuel rods were placed in a new fuel assembly cage and the nine failed or damaged rods were replaced with Zircaloy-clad, Zircaloy-filled dummy rods. The reconstituted assembly was to be reinserted for Cycle 15 operation.

The licensee attributes the damage to vibration-induced fuel rod fretting. The damage was confined to the upper portion of the fuel assembly and appears to be related to previous fuel failures (Ref. 209). A similar occurrence at Yankee Rowe was reported earlier as LER 78-31 (at that time, degradation in four fuel assemblies was observed and the cause of the problem was being investigated). Both the current and previously reported failures appear to be limited to fuel assemblies positioned next to the core baffle. The fuel vendor indicated (Ref. 29) that "although damage of the type observed is generally associated with crossflow, there is as yet no consensus on the cause of the failure of the assemblies."

As a result of the fretting problem, the licensee has replaced fuel rods in several suspect locations with dummy rods. These dummy, or sacrificial, rods are Zircaloy-clad stainless steel pins in the outer row of fresh fuel assemblies, which also have a slightly modified grid design.

NRC Action. The NRC reviewed and evaluated the licensee's application to reload the plant and operate it for Cycle 15. Although the NRC agrees with the licensee on the method used to reduce the number of new failures, there is no assurance that all susceptable rod locations are now filled with dummy rods. Furthermore, the actual damage process has not been identified yet and may not be affected by the modified grid design. The NRC concludes that potential fuel failures during Cycle 15 operation, although small in number, may not be entirely eliminated and that continued surveillance of this problem is required. The licensee has committed (Ref. 210) to continue monitoring coolant activity levels during Cycle 15 operation and to notify the NRC staff of significant changes that may indicate additional failures. As in the past, the licensee will continue an investigation into the cause and prevention of such failures, and make further repairs to damaged fuel as necessary. In view of the small number of failures observed to date, and the licensee's continued efforts to resolve this problem, the NRC finds the issue of fuel failures during Cycle 15 operation adequately addressed.

4.2.9 BWR Channel Box Deflection

The NRC notified the licensees associated with all near-term BWR operating license applications of the acceptable resolution of the BWR channel box deflection issue proposed by Zimmer (Ref. 211). The licensees have been requested to commit to a similar plan. However, since the issue does not have immediate consequences, at least some of the applicants are pursuing an alternate resolution. The current position of NRC's Division of Licensing is that all pending near-term BWR operating licensees will be conditioned such that operation for a second fuel cycle will not be authorized until the issue is satisfactorily resolved.

Plants of the BWR-4, -5, and -6 types are included in the near-term operating license applications. Included among the BWR-4s and BWR-5s are La Salle, Fermi-2, Shoreham, Susquehanna, and WNP-2. The BWR-6s are Grand Gulf, Clinton,Perry, and River Bend.

Regarding operating reactors, the NRC staff has not determined that this issue is important enough to require backfitting. However, if subsequent evidence indicates that significant deflections of the channel boxes are occurring, the NRC would reconsider the possible need for implementing backfitting requirements.

4.3 OLD PROBLEMS (1980) THAT DID NOT RECUR OR THAT WERE SOLVED

4.3.1 Boron Loss from BWR Control Blades

The problem of BWR control blade cracking and the associated loss of boron was described in the fuel performance annual report for 1979 (Ref. 1). The NRC issued IE Bulletin 79-26 (Ref. 212) on that subject in November 1979 and issued Revision 1 to that bulletin in August 1980. There were no LERs involving this matter submitted by licensees in 1980 or 1981.

In 1981, licensees for Dresden-1, -2, and -3 (Ref. 213); Duane Arnold (Ref. 214); Quad Cities-1 and -2 (Ref. 213); and Vermont Yankee (Ref. 215) submitted responses (with proprietary attachments) to the revised bulletin. The boron depletion model is shown to be in good agreement with postirradiation examination data. The information confirms that the loss of boron based on 50% local boron depletion is preditable anu is not affected by plant operating parameters.

4.3.2 PWR Control Rod Guide Tube Support Pin Cracking

Inspections in 1980 at a foreign PWR revealed stress-corrosion cracking in Westinghouse-supplied control rod guide tube support pins (Ref. 162). Westinghouse notified the domestic plants that might be affected (Catawba-1, North Anna-1, Sequoyah-1, and Surry-1) and recommended that all pins be replaced with ones that had received a solution heat treatment of at least 1366 K (2000°F).

No domestic occurrences were reported in 1980 or 1981. In September 1981, France's Gravelines B-1 reactor was taken out of service to repair a broken control rod guide tube support pin (Refs. 216 and 217). Subsequently, a broken support pin was found at Fessenheim-1 in France (Ref. 216). At least eight support pin failures occurred earlier (first failures were detected in early 1978 at Japan's Mihama-3). The only domestic support pin failures occurred in May 1982 at North Anna-1 (Ref. 216). Two nuts from control rod gude tube support pins were found in steam generators at North Anna-1 (Refs. 218-220 and 238). The pins were replaced.

NRC Action. In 1982, the NRC issued three preliminary notifications (Refs. 218-220) concerning the support pin failures at North Anna-1. The NRC met with the vendor on June 2, 1982 (Ref. 221). The NRC issued IE Information

Notice No. 82-29 on July 23, 1982 (Ref. 216). The NRC arranged for the transfer of responsibility (Task Interface Agreement for Task No. 82-45 on August 4, 1982) for steam generator tube and tube sheet damage at North Anna-1 (as described in Refs. 218-220) from Region II to the Division of Licensing. On August 16, 1982, the NRC issued Board Notification No. 82-81 (Refs. 222 and 223) to the appropriate boards (Atomic Safety & Licensing Board, Atomic Safety Licensing Appeal Board) for Callaway-1, Comanche-1 and -2, Diablo Canyon-1 and -2, FNP 1-8, and Summer-1. This subject will be more fully covered in the next NRC annual report.

4.3.3 Underpressurized Gadolinia Fuel Rods

Some underpressurized gadolinia fuel rods (the NRC estimated less than 100) were inadvertantly installed in some BWR fuel assemblies that were shipped in 1980 (Ref. 162). Of the 124 suspect assemblies, 122 went to Quad Cities-1 and 2 to Vermont Yankee. The problem was caused by a malfunction at a weld station. The weld station was repaired and no new occurrences have been reported since then.

4.3.4 Misoriented Fuel Assemblies

In 1980, a total of four BWR fuel assemblies were found to be misoriented at three plants, Browns Ferry-1 and -2 and Oyster Creek-1 (Ref. 162). The licensees improved their procedural controls. There were no occurrences reported in 1981.

5.0 FUEL DESIGN CHANGES AND SUMMARY OF FUEL SURVEILLANCE PROGRAMS

The current design and warranty burnups for BWR and PWR fuel are approximately 28,000 MWd/MTU and 33,000 to 36,000 MWd/MTU, respectively and are based on limited fuel performance data and not on optimized fuel cycle costs (Ref. 25). The present fuel designs are limited to those burnups because of the uncertainties associated in extrapolating data obtained from PWR and BWR fuel after three and four cycles of exposure, respectively. In considering higher burnup goals for fuel, there are four principal areas of concern: corrosion of the cladding exterior surface (especially in the new plants with higher temperature coolant), release of fission gas from the fuel, failure of fuel rods due to pellet-cladding interaction, and distortion of fuel rods and fuel assemblies. A small number of fuel assemblies have reached burnups of 40,000 MWd/MTU (Refs. 25, 224, and 225) or higher and none of those areas of concern have appeared to be a potentially serious problem with those assemblies. However, there continues to be a need for additional data, especially from fuel that has attained higher burnups.

A wide variety of fuel design changes are being studied by domestic fuel vendors, EPRI, and DOE. Table 15 summarizes the major fuel surveillance programs currently in progress and includes information on a number of the associated fuel design changes being studied.

The status (through 1981) of B&W's major fuel performance programs are shown in Table 15. The B&W annual summary (Ref. 15) provides detailed information on the DOE/AP&L/B&W Extended-Burnup Program in Arkansas-1, the DOE/Duke/B&W Extended-Burnup Program in Oconee-1, the DOE/SMUD/B&W Axial Blanket Fuel Design and Development Program in Rancho Seco, the B&W/Duke Fuel Rod Bow Program (involves lifted rods) in Oconee-2, the B&W/Duke Low-Absorption Grid Program in Oconee-2, and the B&W/Duke 17 x 17 Lead Test Assembly (LTA) Program in Oconee-2. Additional information on B&W extended-burnup fuel in the DOE-sponsored programs is in Refs. 17-19 and 226.

The status of C-E's fuel surveillance programs is listed in Table 15. Details on the EPRI-sponsored surveillance program in Calvert Cliffs-1 are in Refs. 24-26. Data on the DOE-sponsored program in Fort Calhoun and Arkansas-2 are provided in Refs. 24, 27, and 28.

The fuel surveillance programs being conducted by ENC are included in Table 15. Additional information is available on the EPRI/ENC/Carolina Power & Light program in Robinson-2 (Refs. 29 and 34), the EPRI/ENC/General Public Utilities project in Oyster Creek (Refs. 25 and 29), the DOE/Consumers Power/ENC/PNL program in Big Rock Point (Refs. 29 and 35), the ENC/Northern States Power project in Prairie Island-2 (Ref. 29), and the EPRI/Empire State Electric Energy Research Corporation (ESEERCO)/ENC project (Ref. 25). The EPRI/ESEERCO/ENC project is a new one that involves the irradiation of lead test assemblies, the design of which incorporates graphite coated cladding and annular fuel pellets.

TABLE 15. Major Fuel Performance Programs: Status Through 1981

Vendor	Fuel Type[a]	Power Plant	Planned No. of Operating Cycles	Scheduled Completion	Interim Inspections To Date
Babcock & Wilcox	15 x 15	Oconee-1	5	1985	3
	15 x 15[b]	Arkansas-1[c]	3	1987	None
	15 x 15[d]	Rancho Seco	3	1986	None
	15 x 15[e]	Oconee-2	4	1982	4
	15 x 15[f]	Oconee-2	3	1982	3
	15 x 15[g]	Oconee-2	1	1982	None
	15 x 15[g]	Oconee-1	3	1987	None
	17 x 17 (LTAs)[h]	Oconee-2	3	1982	3
Combustion Engineering	14 x 14[i]	Calvert Cliffs-1	5	1982	
	14 x 14	Fort Calhoun	6	1982	
	16 x 16	Arkansas-2[j]	--	1983	
	16 x 16	Arkansas-2	--	1984	
Exxon Nuclear	15 x 15	H. B. Robinson-2	5	1982	
	8 x 8	Oyster Creek	--	1983	
	--[k]	Prairie Island-2	4	--	
	rods	Big Rock Point		1982	
General Electric	7 x 7 (MO_2)	Quad Cities-1	4	1982	
	8 x 8 (LTAs)	Monticello	5	--	
	8 x 8 (LTAs)	Quad-Cities-1	5	--	
	8 x 8R (LTAs)	Peach Bottom-2	4[l]	1983	
	8 x 8R (LTAs)	Vermont Yankee	5	--	
	P8 x 8R (LTAs)	Peach Bottom-3	3	--	
	LTAs (barrier cladding)	Quad Cities-1	1	--	
	LTAs (extended burnup)	Monticello	6	1982	
	Barrier Reload Demonstration	Quad Cities-2	--	--	
Westinghouse	15 x 15	Zion-1/Zion-2	5	1982	
	17 x 17	Trojan	5	1984	
	17 x 17 (LTAs)	Surry-2	4	1981	
	17 x 17 (OFA-Demo)	Farley-1	--	--	
	17 x 17 (OFA-Demo)	Salem-1	--	--	
	17 x 17 (OFA-Demo)	Beaver Valley-1			
	14 x 14 (OFA-Demo)	Point Beach-2	--	--	
	MO_2	R. E. Ginna	--	--	

(a) LTA = lead test assembly, MO_2 = mixed oxide (UO_2-PuO_2) fuel, R = retrofit fuel design, OFA-Demo = Demonstration Optimized Fuel Assemblies.
(b) Lead test assemblies of an advanced 15 x 15 extended burnup design.
(c) Arkansas Nuclear One-Unit 1 (Arkansas-1)
(d) Current-design 15 x 15 assemblies containing axially-blanketed fuel columns.
(e) Current-design 15 x 15 assemblies with special Zircaloy cladding materials and EPRI creep collapse specimen clusters.
(f) Current-design 15 x 15 assemblies with lifted rods and cladding having a known spiral eccentricity.
(g) Current-design 15 x 15 assemblies utilizing low absorption spacer grid material (Zircaloy-4).
(h) Two of these four LTAs are reconstitutable.
(i) Involves fuel pellets of three kinds (i.e., two nondensifying types and one densifying type).
(j) Arkansas Nuclear One-Unit 2 (Arkansas-2)
(k) Involves five characterized assemblies incorporating axial blankets.
(l) A fifth cycle is being considered.

The status of GE's fuel surveillance programs (Ref. 39) is shown in Table 11 and Table 15. Additional information is also available on the EPRI-sponsored project involving annular mixed oxide (UO_2-PuO_2) fuel pellets in Quad Cities-1 (Refs. 25 and 43), on the EPRI-sponsored project in Peach Bottom-2 (Refs. 25 and 43), on the DOE-sponsored program with extended burnup fuel in Monticello (Refs. 25, 43, and 44), and on the DOE/Commonwealth Research Corporation/GE program on barrier fuel in Quad Cities-1 and -2 (Ref. 45). Information on the DOE/Tennessee Valley Authority/GE program is provided in Ref. 48.

The status of the Westinghouse fuel surveillance programs is shown in Table 15. More detailed information is available on the EPRI-sponsored project in Zion-1/Zion-2 (Refs. 25 and 49-51). The four assemblies in the Zion reactors had attained a burnup of 53,000 MWd/MTU as of December 31, 1981. Additional information is also available on the EPRI-sponsored project in Trojan (Refs. 49 and 50); on the DOE-sponsored program in Surry-2 (Refs. 49-52); on the demonstration Optimized Fuel Assemblies (OFA) in Farley-1, Salem-1, and Beaver Valley-1 (Refs. 49 and 50) and in Point Beach-2 (Ref. 49); and on the mixed oxide (UO_2-PuO_2) fuel in Ginna (Ref. 49). Information on the DOE-sponsored program involving the irradiation of Westinghouse fuel rods in the CEN BR-3 reactor is provided in Ref. 51. Information on other Westinghouse studies is provided in Ref. 227.

The current status of the EPRI fuel surveillance program was presented at an international conference in November 1981 (Ref. 25). EPRI also recently published a paper (Ref. 228) that describes the phenomena associated with extending fuel burnup and a paper (Ref. 229) that discusses the schedule for extending burnup.

As shown in Table 16, a wide variety of fuel types are being irradiated in Big Rock Point (Refs. 230-232).

TABLE 16. Various Fuel Rod Types[a] That Have Been or Are Being Irradiated in Big Rock Point Reactor Under DOE's Fuel Performance Improvement Program (Ref. 230)

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

(a) The irradiation tests include full-length and segmented fuel rods. All rods are clad with cold-worked and stress-relieved Zircaloy-2 cladding.
(b) Product of Acheson Colloids Corp., Port Huron, Michigan.

6.0 SUMMARY OF HIGH-BURNUP FUEL EXPERIENCE

Domestic BWR fuel burnup experience is summarized in Figure 3. Domestic PWR fuel experience is summarized in Figure 4. Most of the data on fuel discharged prior to 1979 was obtained from Ref. 233.

The present optimum extended burnups (discharge batch averages) are typically about 50,000 MWd/MTU and 45,000 MWd/MTU for PWRs and BWRs on 12-month cycles, respectively, and about 10% higher for 18-month cycles (Ref. 234). EPRI anticipates that fuel assemblies designed for improved high burnup performance will not be available for full reloads until the late 1980s and that there will be a gradual increase in burnup of 1000-2000 MWd/MTU per year (Ref. 229). The projected benefits to the LWR fuel cycle from extended burnup are discussed in a recent paper by DOE (Ref. 235). The regulatory perspective on extended burnup fuel is discussed in Ref. 236. The NRC's position is that operation beyond 45,000 to 50,000 MWd/MTU probably can not be justified until tests of fuel under normal and transient conditions at that burnup level are completed.

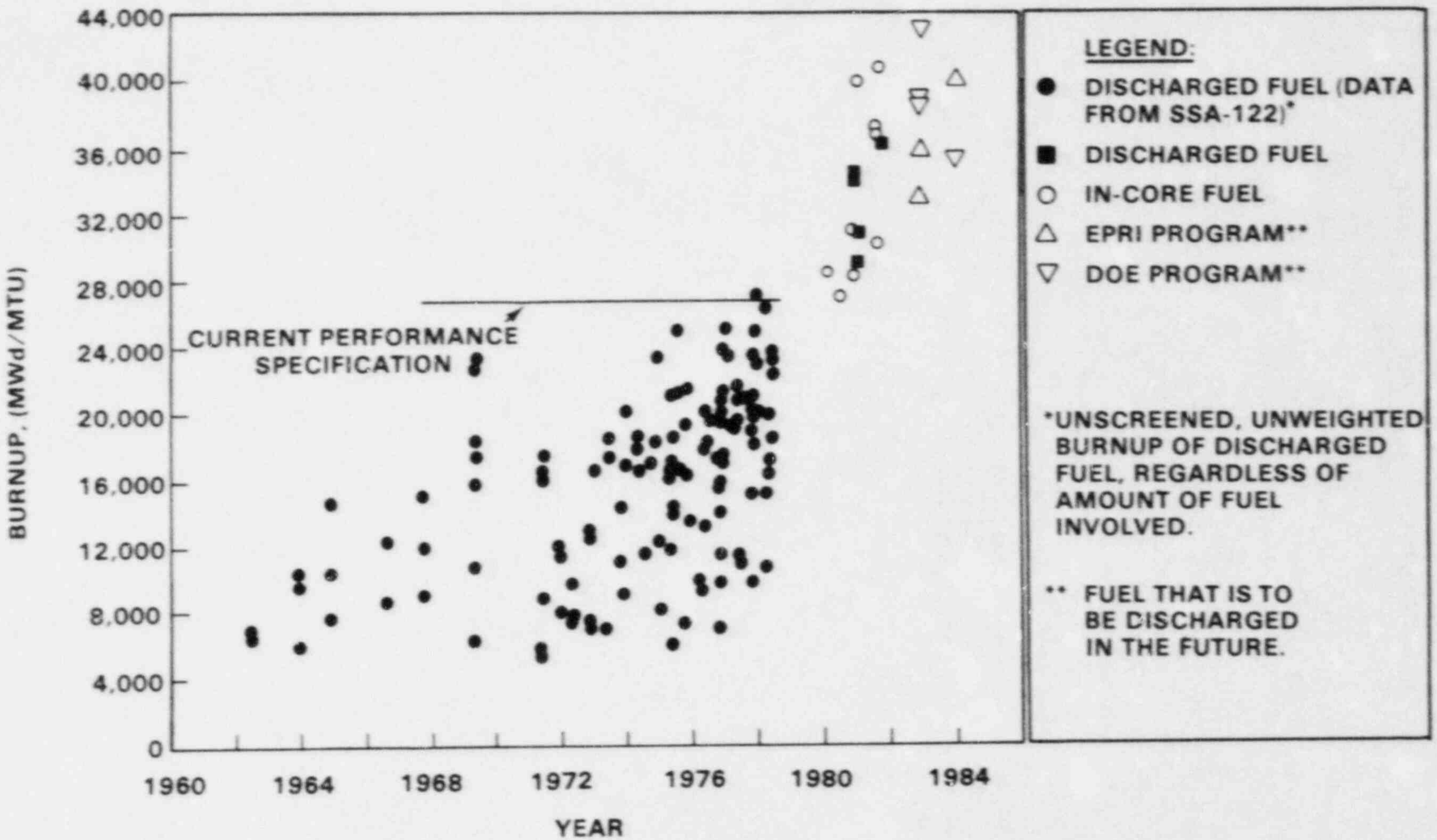

FIGURE 3. Domestic BWR Fuel Burnup Experience

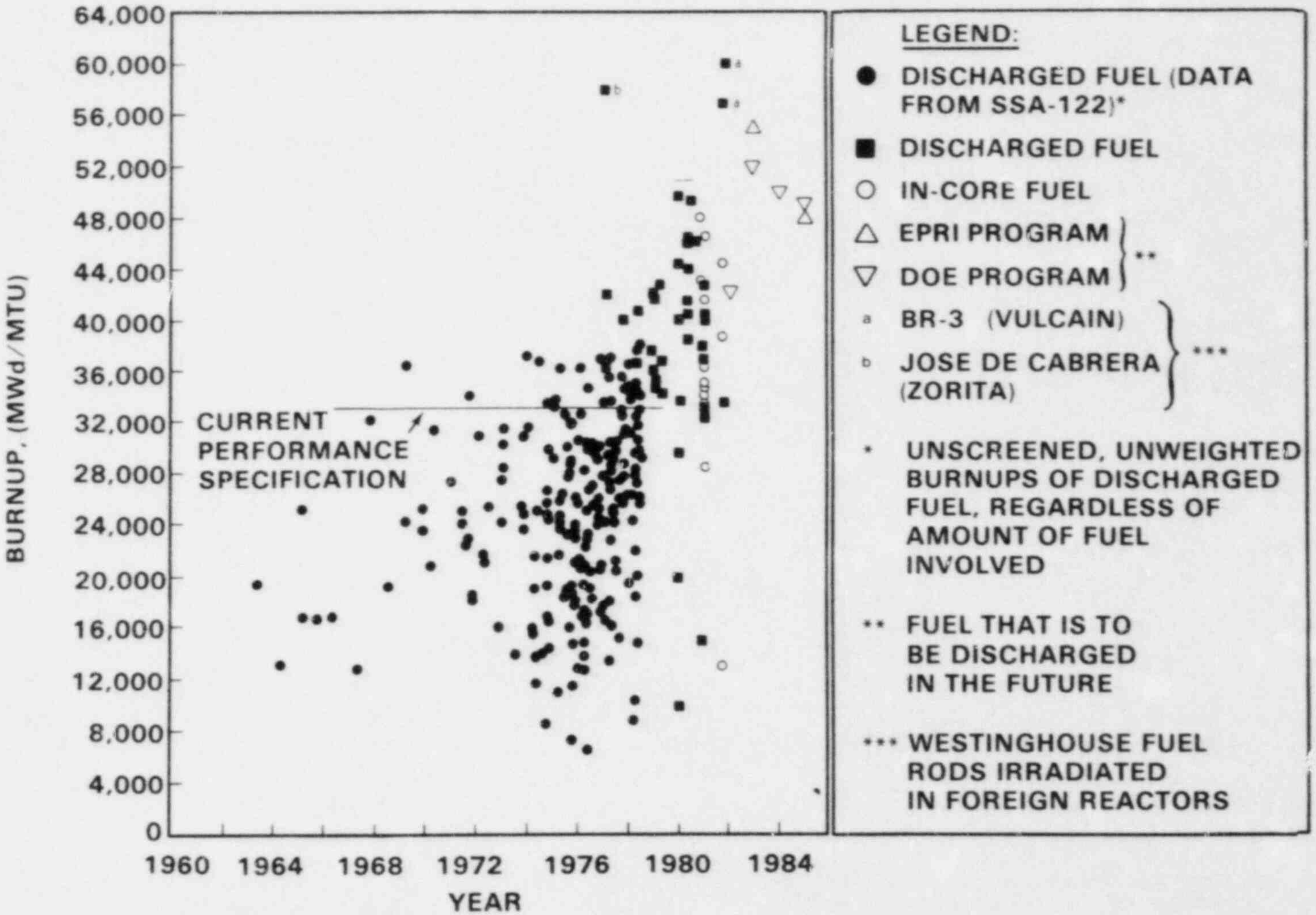

FIGURE 4. Domestic PWR Fuel Burnup Experience

7.0 REFERENCES

1. M. D. Houston, Fuel Performance Annual Report (Period through December 1978). NUREG-0633, U.S. Nuclear Regulatory Commission, December 1979.

2. M. Tokar, W. J. Bailey and M. E. Cunningham, Fuel Performance Annual Report for 1979. NUREG/CR-1818 (PNL-3583), U.S. Nuclear Regulatory Commission, January 1981.

3. W. J. Bailey, K. H. Rising and M. Tokar, Fuel Performance Annual Report for 1980. NUREG/CR-2410 (PNL-3953), U.S. Nuclear Regulatory Commission, December 1981.

4. Regulatory Guide 1.16 through Revision 5, "Reporting of Operating Information-Appendix A Technical Specifications," 1971 to 1979.

5. Office of Operations Evaluation, Nuclear Power Plant Operating Experience During 1973, OOE-ES-004, U.S. Atomic Energy Commission, December 1974.

6. Office of Management Information and Program Control, Nuclear Power Plant Operating Experience 1974-1975. NUREG-0227, U.S. Nuclear Regulatory Commission, April 1977.

7. Office of Management Information and Program Control, Nuclear Power Plant Operating Experience 1976. NUREG-0336, U.S. Nuclear Regulatory Commission, December 1977.

8. M. R. Beebe, Nuclear Power Plant Operating Experience-1977: Annual Report, NUREG-0483, U.S. Nuclear Regulatory Commission, February 1979.

9. M. R. Beebe, Nuclear Power Plant Operating Experience-1978: Annual Report, NUREG-0618, U.S. Nuclear Regulatory Commission, December 1979.

10. R. L. Scott, D. S. Queener and C. Kukielka, Nuclear Power Plant Operating Experience-1979: Annual Report, NUREG/CR-1496 (ORNL/NUREG/NSIC-180), Oak Ridge National Laboratory, Oak Ridge, Tennessee, May 1981.

11. Nuclear Safety Analysis Center, Screening and Evaluation of First Half 1981 Licensee Events Reports, NSAC-49, May 1982.

12. W. J. Bailey, C. J. Morris, F. R. Reich, and K. L. Swinth, Assessment of Current Onsite Inspection Techniques for Light-Water Reactor Fuel Systems: Executive Summary. NUREG/CR-1380 (PNL-3325), Vol. 1, U.S. Nuclear Regulatory Commission, July 1980.

13. U.S. Nuclear Regulatory Commission Standard Review Plan, Rev. 1. NUREG-75/087 (presently NUREG-0800).

14. V. Stello (NRC), Memorandum to R. C. DeYoung, "Supplemental Surveillance of 17 x 17 Fuel Assemblies," November 14, 1974.

15. J. H. Taylor (Babcock & Wilcox Company), Letter to C. H. Berlinger (NRC), "Performance of B&W Fuel for Calendar Year 1981," August 11, 1982. (The enclosure was: T. D. Pyecha, L. A. Walton and T. A. Coleman, Babcock & Wilcox 1981 Fuel Performance Report, July 1982.)

16. J. S. Tulenko, "PWR Fuel Performance and Manufacturing-An Integrated Approach," Trans. Am. Nucl. Soc. 40:248-249, April 1982.

17. L. A. Walton, et al., "Development of an Extended Burnup Fuel Assembly," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 2:5-21 through 5-32, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

18. T. A. Coleman, et al., "Extended Burnup Fuel Performance," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 1:1-39 through 1-57, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

19. L. W. Newman, et al., "Burnable Poisons for Extended Burnup Fuel Management-Present and Future," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 2:7-25 through 7-37, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

20. T. A. Coleman, et al., "Improved Uranium Utilization in Pressurized Water Reactors," Trans. Am. Nucl. Soc. 39:133-134, November-December 1981.

21. A. E. Scherer (Combustion Engineering, Inc.), Letter to C. H. Berlinger (NRC), "1981 Performance of C-E Fuel," July 13, 1982.

22. C. K. Anderson and R. N. Duncan, "Recent Design and Performance Features of Combustion Engineering All-Zircaloy Fuel Assemblies," Trans. Am. Nucl. Soc. 40:249-252, April 1982.

23. N. Fuhrman, P. A. Van Saun, and G. P. Smith, "The High Fluence Performance of Al_2O_3-B_4C Burnable Poison," Trans. Am. Nucl. Soc. 39:405, November-December 1981.

24. M. G. Andrews, S. C. Hatfield, and E. J. Ruzauskas, "Operating Experience with Combustion Engineering Fuel at High Burnups," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 1:1-5 through 1-20, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

25. J. T. A. Roberts, D. Franklin, H. Ocken, and S. T. Oldberg, "The EPRI LWR Fuel Surveillance Program," Specialists' Meeting on Examination of Fuel Assembly for Water Cooled Power Reactors, IWGFPT/12, pp. 62-78, International Atomic Energy Agency, November 1981.

26. S. R. Pati and N. Fuhrman, "Densification, Swelling and Microstructures of LWR Fuels Through Extended Burnups," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 1:4-39 through 4-56, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

27. R. Harris, P. O. Donnell, and C. Bagnal, "Fuel Management Schemes for PWR Extended Burnup Fuel Cycles with Gadolinium," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 2:7-39 through 7-48, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

28. R. Freeman, et al., "High Burnup Fuel Cycle Characteristics of Conventional and Low Leakage Fuel Management Schemes for Arkansas Nuclear One Unit 2," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 2:7-49 through 7-59, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

29. G. F. Owsley (Exxon Nuclear Company, Inc.), Letter to M. Tokar (NRC), "XN-NF-82-38 (NP)," July 12, 1982. (The enclosure was: C. A. Exarhos, Summary of Exxon Nuclear Company Fuel Performance Through December 31, 1981, XN-NF-82-38 (NP), June 1982).

30. C. A. Exarhos, Summary of Exxon Nuclear Company Fuel Performance Through December 31, 1981, XN-NF-82-38(P), June 1982. This document is not publicly available because it contains proprietary information.

31. G. A. Sofer, "Technology of Burnable Absorber Reactivity Control and Other Techniques to Increase Fuel Burnup in LWR Fuel Cycles," Trans. Am. Nucl. Soc. 40:183-184, April 1982.

32. F. B. Skogen, L. A. Nielsen, and R. G. Grummer, "Operating Experience with Exxon Nuclear-Supplied Gadolinia in Pressurized Water Reactor," Trans. Am. Nucl. Soc. 40:194-198, April 1982.

33. F. B. Skogen, R. B. Stout, and G. A. Sofer, "Experience with Gadolinia Burnable Absorbers in Light Water Reactors," Trans. Am. Nucl. Soc. 39:401-402, November-December 1981.

34. C. A. Exarhos, et al., "High Burnup Performance of Exxon Nuclear Company Fuel in the H. B. Robinson Pressurized Water Reactor," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, I:1-59 through 1-70, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

35. F. Buckman, et al., "Development of Annular-Coated-Pressurized and Sphere-Pac LWR Fuels," LWR Extended Burnups-Fuel Performance and Utilization, DOE/NE/34087, 2:5-83 through 5-101, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

36. R. J. Guenther, et al., "The Elongation Behavior of PCI-Resistant LWR Fuel Designs During Steady-State Irradiations," Trans. Am. Nucl. Soc. 38:272-273, June 1981.

37. J. O. Barner, et al., "Elongation Behavior of PCI-Resistant LWR Fuel Designs During Power Ramping," Trans. Am. Nucl. Soc. 38:273-275, June 1981.

38. L. R. Bunnell and M. D. Freshley, "The Function of Graphite Coatings to Mitigate LWR Fuel Failures Caused by Pellet-Cladding Interaction," Trans. Am. Nucl. Soc. 38:275-276, June 1981.

39. C. L. Hilf (General Electric Company), Telecopy to M. Tokar (NRC), August 23, 1982.

40. "GE Is Working to Correct a Design Flaw...," Nucleonics Week, 23(19):8 (May 13, 1982), McGraw-Hill Inc., New Nork.

41. "Power Reactors 1982," Nuclear Engineering International, 27(330):20-21 (August 1982 Supplement).

42. American Nuclear Society, "World List of Nuclear Power Plants," Nuclear News, 25(2):95 (February 1982).

43. J. A. Baumgartner, et al., "Extended Burnup Fuel Investigations in Commercial Reactors," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 1:1-71 through 1-82, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

44. D. E. Nevinski, "DOE/GE/NSP Extended Burnup Program," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 1:2-21 through 2-29, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

45. J. H. Davies, et al., "Irradiation Tests On Barrier Fuel in Support of a Large-Scale Demonstration," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 2:5-51 through 5-67, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

46. H. E. Bliss, et al., "A Large-Scale Demonstration Irradiation of Zirconium-Liner Barrier Fuel in Quad Cities-2," Trans. Am. Nucl. Soc. 39:392-393, November-December 1981.

47. R. S. Rosenbaum, et al., "Resistance of Zirconium Barrier Fuel to Pellet-Cladding Interaction," Trans. Am. Nucl. Soc. 38:270-272, June 1981.

48. P. D. Brown, et al., "Impact of Extended Burnup on the BWR Fuel Cycle," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 2:7-71 through 7-90, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

49. R. G. Jones and J. A. Iorii, Operational Experience with Westinghouse Cores (Up to December 31, 1981), WCAP-8183, Rev. 11, May 1982.

50. H. W. Wilson, et al., "Fuel Performance Characteristics at Extended Burnup," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 1:1-21 through 1-37, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

51. E. Roberts, "Extended Burnup Considerations and Experience," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 1:1-1 through 1-3, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

52. M. L. Smith, et al., "Extended Burnup Implementation and Experience," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 2:7-13 through 7-18, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

53. D. Franklin, H. Ocken, and S. T. Oldberg, LWR Core Materials Performance Program: Progress in 1979-1980, EPRI NP-1770-SR, Electric Power Research Institute, Palo Alto, California, March 1981.

54. J.T.A. Roberts, "Light Water Reactor Materials: Critical Review of Problems and Progress," Trans. Am. Nucl. Soc. 38:13-14, June 1981.

55. "LER No. 80-006/04X-1," LER Monthly Report-December 1981, February 19, 1982 (p.1).

56. "LER No. 80-027/04X-2," LER Monthly Report-August 1981, September 25, 1981 (p. 1).

57. David C. Trimble (Arkansas Power & Light Company), Letter to G. L. Madsen (NRC), "Arkansas Nuclear One-Unit 1, Docket No. 50-313, License No. DPR-1, Licensee Event Report No. 80-027/04X-2 (File: 0520)," June 25, 1981.

58. "LER No. 80-027/04X-3," LER Monthly Report-December 1981, February 19, 1982 (p.1).

59. Guy S. Vissing (NRC), Memorandum for H. Denton et al. (NRC), "Daily Highlight," February 9, 1981.

60. David C. Trimble (Arkansas Power & Light Company), Letter to J. F. Stolz (NRC), "Failed Fuel Evaluation Scope (File: 1510.5)," March 20, 1981.

61. David C. Trimble (Arkansas Power & Light Company), Letter to Robert W. Reid (NRC), "Cycle 5 Startup with Five Failed Fuel Assemblies (File: 1510.5)," February 19, 1981.

62. David C. Trimble (Arkansas Power & Light Company), Letter to Robert W. Reid (NRC), "Additional Cycle 5 Information on Failed Fuel Evaluations (File: 1510.5)," February 26, 1981.

63. Michael Tokar (NRC), Memorandum to Carl H. Berlinger (NRC), "ANO-1 Fuel Failures," September 25, 1981.

64. T. Cogburn (Arkansas Power & Light Company), Memorandum to B. Johnson (NRC), "ANO-1 Failed Fuel Analysis," May 8, 1981 (for April).

65. T. Cogburn (Arkansas Power & Light Company), Memorandum to B. Johnson (NRC), "Arkansas Nuclear One Failed Fuel Analysis," June 8, 1981 (for May).

66. T. Cogburn (Arkansas Power & Light Company), Memorandum to B. Johnson (NRC), "Arkansas Nuclear One Failed Fuel Analysis," July 7, 1981 (for June).

67. P. C. Rogers (Arkansas Power & Light Company), Memorandum to B. Johnson (NRC), "Arkansas Nuclear One-Unit 1 Failed Fuel Analysis," August 10, 1981 (for July).

68. D. B. Lomax (Arkansas Power & Light Company), Memorandum to B. Johnson (NRC), "Arkansas Nuclear One-Unit 1 Failed Fuel Analysis," December 10, 1981 (for November).

69. G. S. Vissing (NRC), "Summary of Meeting with AP&L Concerning the ANO-1 Leaking Fuel Rod Investigation," Docket No. 50313, April 5, 1982 (Arkansas-1).

70. C. H. Berlinger (NRC), Memorandum for J. Stolz (NRC), "Completion of ANO-1 Failed Fuel Investigation (TACS 43771)," March 30, 1982.

71. David C. Trimble (Arkansas Power & Light Company), Letter to G. L. Madsen (NRC), "Arkansas Nuclear One-Unit 2, Docket No. 50-368, License No. NPF-6, Licensee Event Report NO. 81-008/03X-1 (File: 2-0520)," April 7, 1981.

72. "LER No. 81-008/03X-1," <u>LER Monthly Report-May 1981</u>, June 5, 1981 (p. 2).

73. David C. Trimble (Arkansas Power & Light Company), Letter to K. V. Seyfrit (NRC), "Arkansas Nuclear One-Unit 2, Docket No. 50-368, License No. NPF-6, Licensee Event Report No. 81-008/03L-0 (File: 2-0520)," March 12, 1981.

74. "LER No. 81-008/03L-0," <u>LER Monthly Report-April 1981</u>, April 29, 1981 (p. 2).

75. David C. Trimble (Arkansas Power & Light Company), Letter to Robert A. Clark (NRC), "Cycle 2 Iodine Activity (File: 2-1510)," November 15, 1981.

76. David C. Trimble (Arkansas Power & Light Company), Letter to G. L. Madsen (NRC), "Licensee Event Report No. 81-021/99X-0 (File: 2-0520)," June 22, 1981.

77. "LER No. 81-021/99X-0," LER Monthly Report-July 1981, August 12, 1981 (p. 4).

78. David C. Trimble (Arkansas Power & Light Company), Letter to Robert A. Clark (NRC), "NRC Request for Information on Fuel Assembly Spacer Grid Damage (File: 2-1510)," June 4, 1981.

79. David C. Trimble (Arkansas Power & Light Company), Letter to Robert A. Clark (NRC), "Preliminary Results of ANO-2 Fuel Inspection (File: 2-1510)," May 22, 1981. The attachment to this letter is not publicly available because it contains proprietary information.

80. D. Powers (NRC), Note to C. Grimes (NRC), "Operating Experience with the C-E 16x16 Fuel Design," November 18, 1981.

81. David C. Trimble (Arkansas Power & Light Company), Letter to Robert A. Clark (NRC), "Arkansas Nuclear One-Unit 2, Docket No. NPF-6, Preliminary Results of ANO-2 Fuel Inspection," May 22, 1981. This document is not publicly available because it contains proprietary information.

82. Lester S. Rubenstein (NRC), Memorandum for Thomas M. Novak (NRC), "CPB Fuels SE Input on ANO-2 Reload Safety Analysis," June 16, 1981.

83. "LER No. 80-082/03L-0," LER Monthly Report-January 1981, February 4, 1981 (p. 13).

84. A. C. Tollison, Jr. (Carolina Power & Light Company), Letter to James P. O'Reilly (NRC), "Brunswick Steam Electric Plant, Unit No. 2, Docket No. 50-324, License No. DPR-62, Licensee Event Report 2-80-82," December 8, 1980. (LER No. 80-082/03L-0).

85. "LER No. 80-111/03L-0," LER Monthly Report-February 1981, March 2, 1981 (p. 20). (Also NSIC Accession 163398)

86. A. C. Tollison, Jr. (Carolina Power & Light Company), Letter to James P. O'Reilly (NRC), "Brunswick Steam Electric Plant, Unit No. 2, Docket No. 50-324, License No. DPR-62, License Event Report 2-80-111," January 7, 1981. (LER No. 80-111/03L-0).

87. C. R. Dietz (Brunswick Steam Electric Plant), Letter to James P. O'Reilly (NRC), "Licensee Event Report 2-81-63," July 28, 1981. (LER No. 81-063/03L-0).

88. "Brunswick-2," Licensed Operating Reactors, Status Summary Report, Data as of 10-31-81, NUREG-0020, 5(11):2-045 (November 1981).

89. "LER No. 81-000S", Licensee Event Report (LER) Compilation for Month of February 1982, NUREG/CR-2000 (ORNL/NSIC-200), 1(2):7, April 1982 (Brunswick-2).

90. "LER No. 82-036," Licensee Event Report (LER) Compilation for Month of June 1982, NUREG/CR-2000 (ORNL/NSIC-200), 1(6):15, July 1982 (Brunswick-2).

91. "LER No. 81-049/03L-0," LER Monthly Report-December 1981, February 19, 1982 (p. 39).

92. D. V. Shaller (Indiana and Michigan Electric Company), Letter to J. G. Keppler (NRC), "RO 81-049/03L-0," October 28, 1981.

93. J. A. Hancock (Florida Power Corporation), Letter to J. P. O'Reilly (NRC), "Licensee Event Report 80-052/03L-0," January 5, 1981. Also NSIC Accession 163329.

94. "LER No. 81-010/03L-0," LER Monthly Report-April 1981, April 29, 1981 (p. 18).

95. J. A. Hancock (Florida Power Corp.), Letter to J. P. O'Reilly (NRC), "Licensee Event Report 81-010/03L-0," March 13, 1981. (Docket No. 50-302, Crystal River-3).

96. Patsy Y. Baynard (Florida Power Corporation), Letter to J. P. O'Reilly (NRC), "Licensee Event Report 81-036/03L-0," July 15, 1981.

97. "LER No. 81-036/03L-0," LER Monthly Report-September 1981, October 13, 1981 (p. 24).

98. "LER No. 81-050/03L-0," LER Monthly Report-October 1981, November 20, 1981 (p. 34).

99. Patsy Y. Baynard (Florida Power Corp.), Letter to J. P. O'Reilly (NRC), "Licensee Event Report 81-050/03L-0," August 27, 1981. (Docket No. 50-302, Crystal River-3).

100. Florida Power Corporation, Licensee Event Report No. 81-060/01T-0, October 14, 1981.

101. LER No. 81-060/01T-0," LER Monthly Report-December 1981, February 19, 1982 (p. 33).

102. M. Tokar (NRC), Memorandum for R. O. Meyer (NRC), "Repeated Violations of I-113 Coolant Activity Limit at Crystal River 3," May 22, 1980.

103. "LER No. 80-089/03L-0," LER Monthly Report-February 1981, March 2, 1981 (p. 29).

104. Terry D. Murray (Toledo Edison), Letter to James G. Keppler (NRC), "Reportable Occurrence 80-089, Davis-Besse Nuclear Power Station Unit 1, Date of Occurrence: December 3, 1980," January 2, 1981. (LER No. 80-089/03L-0).

105. "LER No. 81-016/03L-0," LER Monthly Report-May 1981, June 5, 1981 (p. 33).

106. Terry D. Murray (Toledo Edison), Letter to James G. Keppler (NRC), "Reportable Occurrence 80-016, Davis-Besse Nuclear Power Station Unit 1, Date of Occurrence: March 8, 1981," April 3, 1981. (LER No. 81-016/03L-0).

107. "LER No. 81-031/03L-0," LER Monthly Report-July 1981, August 12, 1981 (p. 20).

108. Terry D. Murray (Toledo Edison), Letter to James G. Keppler (NRC), "Reportable Occurrence 81-031, Davis-Besse Nuclear Power Station Unit 1, Date of Occurrence: May 12, 1981," June 11, 1981. (LER No. 81-031/03L-0).

109. "LER No. 81-031/03X-1," LER Monthly Report-September 1981, October 13, 1981 (p. 33).

110. Terry D. Murray (Toledo Edison), Letter to James G. Keppler (NRC), "Revision 1 to Licensee Event Report 81-031," July 13, 1981. (LER No. 81-031/03X-1).

111. "LER No. 81-031/03X-2," LER Monthly Report-October 1981, November 20, 1981 (p. 44).

112. Terry D. Murray (Toledo Edison), Letter to James G. Keppler (NRC), "Revision 2 to Licensee Event Report 81-031," August 25, 1981. (LER No. 81-031/03X-2).

113. "LER No. 81-031/03X-3," LER Monthly Report-November 1981, December 23, 1981 (p. 33).

114. Terry D. Murray (Toledo Edison), Letter to James G. Keppler (NRC), "Revision 3 to Licensee Event Report 81-031," October 2, 1981. (LER No. 81-031/03X-2).

115. "LER No. 81-031/03X-4," LER Monthly Report-December 1981, February 19, 1982 (p. 43).

116. Toledo Edison, Letter to NRC, "Monthly Operating Report for April 1982 for Davis Besse-1," May 7, 1982. (Docket No. 50346, Acession No. 8206010385R.)

117. Dairyland Power Cooperative, Letter to NRC (Region 3), "LER No. 80-003S," April 23, 1981. (NSIC Accession 166089. Event date: January 30, 1981.)

118. "LER No. 81-003/03L-0," LER Monthly Report-April 1981, April 29, 1981 (p. 37). Also NSIC Accession 164682.

119. Frank Linder (Dairyland Power Cooperative), Letter to James G. Keppler (NRC), "Dairlyland Power Cooperative, La Crosse Boiling Water Reactor (LACBWR), Provisional Operating License No. DPR-45, Reportable Occurrence No. 81-03," February 25, 1981. (LER No. 81-003/03L-0).

120. Frank Linder (Dairyland Power Cooperative), Letter to James G. Keppler (NRC), "Dairyland Power Cooperative, La Crosse Boiling Water Reactor (LACBWR), Provisional Operating License No. DPR-45, Reportable Occurrence Nos. 80-06 and 80-07," September 12, 1980. (LER Nos. 80-006/03L-0 and 80-007/03L-0).

121. Frank Linder (Dairyland Power Cooperative), Letter to James G. Keppler (NRC), "Dairyland Power Cooperative, La Crosse Boiling Water Reactor (LACBWR), Provisional Operating License No. DPR-45, Reportable Occurrence Nos. 79-05 and 79-06," April 19, 1979. (LER Nos. 79-005/01T-0, and 80-005/01T-1, and 79-006/01T-0).

122. Frank Linder (Dairyland Power Cooperative), Letter to James G. Keppler (NRC), "Dairyland Power Cooperative, LaCrosse Boiling Water Reactor (LACBWR), Provisional Operating License No. DPR-45, Reportable Occurence No. 82-06," August 10, 1982. (Includes LER No. 82-006/01X-1 and LAC-TR-112).

123. "LER No. 81-053/03L-0," LER Monthly Report-September 1981, October 13, 1981 (p. 81).

124. W. R. Cartwright (Virginia Electric and Power Company), Letter to James P. O'Reilly (NRC), "LER 81-053/03L-0," July 27, 1981. (Docket No. 50-338, North Anna-1).

125. "LER No. 81-065/03L-0," LER Monthly Report-October 1981, November 20, 1981 (p. 90).

126. W. R. Cartwright (Virginia Electric and Power Company), Letter to James P. O'Reilly (NRC), "LER 81-065/03L-0," August 17, 1981. (Docket No. 50-338, North Anna-1).

127. Steven R. Frost (Consumers Power Company), Letter to James G. Keppler (NRC), "Docket 50-255, DPR-20, Palisades Plant, Licensee Event Report 81-009, Dose Equivalent Iodine Exceeded Technical Specification Limit," February 26, 1981. (LER No. 81-009/03L-0).

128. "LER No. 81-009/03L-0," LER Monthly Report-April 1981, April 29, 1981 (p. 51). Also NSIC Accession 164269.

129. Brian D. Johnson (Consumers Power Company), Letter to James G. Keppler (NRC), "Docket 50-255, DPR-20, Palisades Plant, Licensee Event Report 81-41, Abnormal Degradation of Fuel Cladding," October 26, 1981.

130. "LER No. 81-041/01T-0," LER Monthly Report-November 1981, December 23, 1981 (p. 76).

131. M. Tokar (NRC), Telephone Conversation with R. W. Voll (Consumers Power Company), November 12, 1981.

132. "LER No. 81-041/03L-0," LER Monthly Report-November 1981, December 23, 1981 (p. 95).

133. J. W. Williams, Jr. (Florida Power & Light Company), Letter to James P. O'Reilly (NRC), "Reportable Occurrence 335-81-41, St. Lucie Unit 1, Date of Occurrence: September 8, 1981," October 6, 1981. (LER No. 81-041/03L-0).

134. "LER No. 80-039/03L-0," LER Monthly Report-January 1981, February 4, 1981 (p. 49).

135. A. D. Schmidt (Florida Power & Light Company), Letter to James P. O'Reilly (NRC), "Reportable Occurrence 335-80-39, St. Lucie Unit 1, Date of Occurrence: August 14, 1980," September 15, 1980. (LER No. 80-039/03L-0).

136. "LER No. 81-047/03L-0," LER Monthly Report-November 1981, December 23, 1981 (p. 99).

137. Virginia Electric and Power Company, "Licensee Event Report No. 81-047/03L-0," Docket No. 50-280, September 30, 1981. (Surry-1).

138. "LER No. 81-033/03L-0," LER Monthly Report-November 1981, December 23, 1981 (p. 97).

139. Virginia Electric and Power Company, "Licensee Event Report No. 81-033/03L-0," Docket No. 50-280, September 15, 1981. (Surry-1).

140. "LER No. 81-072," Licensee Event Report (LER) Compilation for Month of March 1982, NUREG/CR-2000 (ORNL/NSIC-200), 1(3):43, May 1982 (Surry-1).

141. "LER No. 81-079," Licensee Event Report (LER) Compilation for Month of February 1982, NUREG/CR-2000 (ORNL/NSIC-200), 1(2):43, April 1982 (Surry-1).

142. "LER No. 81-071," Licensee Event Report (LER) Compilation for Month of March 1982, NUREG/CR-2000 (ORNL/NSIC-200), 1(3):44, May 1982 (Suury-1).

143. "LER No. 82-051," Licensee Event Report (LER) Compilation for Month of July 1982, NUREG/CR-2000 (ORNL/NSIC-200), 1(7):124, August 1982 (Surry-1).

144. "LER No. 82-020," Licensee Event Report (LER) Compilation for Month of August 1982, NUREG/CR-2000 (ORNL/NSIC-200), 1(8):144, September 1982 (Surry-1).

145. "Trojan," Licensed Operating Reactors, Status Summary Report, Data as of 10-31-81, NUREG-0020, 5(11):2-319 (November 1981).

146. U.S. Nuclear Regulatory Commission, Daily Highlight-Region V, November 5, 1981.

147. U.S. Nuclear Regulatory Commission, Daily Highlight-Region V, August 27, 1981.

148. "Trojan," Licensed Operating Reactors, Status Summary Report, Data as of 11-30-81, NUREG-0020, 5(12):2-321 (December 1981).

149. "Trojan," Licensed Operating Reactors, Status Summary Report, Data as of 12-31-81, NUREG-0020, 6(1):2-309 (January 1982).

150. C. P. Yundt and R. L. Steele (Portland General Electric), Letter to R. H. Engelken (NRC), "Licensee Event Report No. 82-06," May 6, 1982. (LER No. 82-006/01T-0 for Trojan.)

151. Bart D. Withers (Portland General Electric Company), Letter to Robert A. Clark (NRC), "Preliminary Report on Trojan EOC-3 Fuel Examination," June 25, 1981.

152. "LER No. 82-003," Licensee Event Report (LER) Compilation for Month of May 1982, NUREG/CR-2000 (ORNL/NSIC-200), 1(5):59, June 1982 (Oconee-2).

153. Office of Nuclear Reactor Regulation, Daily Highlights, Week of January 26-30, 1981 ("Arkansas Nuclear One, Unit No. 1").

154. R. W. Reid (NRC), Letter to William Cavanaugh, III, (Arkansas Power & Light Company), February 9, 1981.

155. David C. Trimble (Arkansas Power & Light Company), Letter to Robert W. Reid (NRC), "Fuel Assembly Holddown Spring Failure Information (File: 4135, 0510)," February 13, 1981.

156. J. H. Taylor (B&W), Letter to M. Tokar (NRC), "B&W 1980 Fuel Performance," June 10, 1981.

157. L. S. Rubenstein (NRC), Memorandum for T. M. Novak (NRC), "B&W Fuel Assembly Holddown Spring Failures," December 30, 1980.

158. "LER No. 80-019/01T-0," LER Monthly Report-July 1981, August 12, 1981 (p. 17).

159. J. A. Hancock (Florida Power Corporation), Letter to J. P. O'Reilly (NRC), "Licensee Event Report 80-019/01T-0," May 29, 1980. (Docket No. 50-302, Crystal River-3).

160. Terry D. Murray (Toledo Edison), Letter to James G. Keppler (NRC), "Licensee Event Report 80-040/01X-1," August 24, 1981.

161. "LER No. 80-040/01X-1," LER Monthly Report-October 1981, November 20, 1981 (p. 43).

162. W. J. Bailey, K. H. Rising, and M. Tokar, Fuel Performance Annual Report for 1980. NUREG/CR-2410 (PNL-3953), U.S. Nuclear Regulatory Commission, December 1981.

163. "LER No. 80-015/01X-1," LER Monthly Report-January 1981, February 4, 1981 (p. 40).

164. William O. Parker, Jr. (Duke Power Company), Letter to Harold R. Denton (NRC), "Oconee Nuclear Station Docket Nos. 50-269,-270, -287," December 15, 1980. (LER No. 80-015; also see NSIC Accession 161760).

165. Robert W. Reid (NRC), Letter to William O. Parker, Jr. (Duke Power Company), "Surveillance on Holddown Springs in Oconee Units Nos. 1 and 2," February 20, 1981.

166. John R. Marshall (Arkansas Power and Light Company), Letter to Robert A. Clark (NRC), "Arkansas Nuclear One-Unit 2, Docket No. 50368, Licensee No. NPF-6, Fuel Inspection Results and Preventive Measures (File:2-1510)," June 2, 1982. This document is not publicly available because it contains proprietory information.

167. John D. O'Toole (Consolidated Edison Company of New York), Letter to Boyce H. Grier (NRC), "LER 81-002/99X-1," March 20, 1981.

168. "LER No. 81-005/01T-0," LER Monthly Report-March 1981, March 27, 1981 (p. 48).

169. John D. O'Toole (Consolidated Edison Company of New York), Letter to Boyce H. Grier (NRC), "LER 81-002/99X-1," March 20, 1981.

170. "LER No. 81-002/99X-1," LER Monthly Report-April 1981, April 29, 1981 (p. 35).

171. John D. O'Toole (Consolidated Edison Company of New York), Letter to Boyce H. Grier (NRC), "LER 81-002/99X-2," May 19, 1981.

172. "LER No. 81-002/99X-2," LER Monthly Report-June 1981, July 6, 1981 (p. 31).

173. M. Tokar, W. J. Bailey and M. E. Cunningham, Fuel Performance Annual Report for 1979. NUREG/CR-1818 (PNL-3583), U.S. Nuclear Regulatory Commission, January 1981.

174. U.S. Nuclear Regulatory Commission, Preliminary Notification of Event or Unusual Occurrence, PNO-II-81-85, October 7, 1981.

175. J. T. Beckman, Jr. (Georgia Power), Letter to Director of Nuclear Reactor Regulation (NRC), "Replacement of Leaking Fuel Assemblies," October 7, 1981.

176. M. Manry (Georgia Power), Letter to James P. O'Reilly (NRC), "Hatch Unit-1, Docket No. 50-321, Notification of Reportable Occurrence No. 50-321/1981-114," October 30, 1981.

177. M. Manry (Georgia Power), Letter to James P. O'Reilly (NRC), "Reportable Occurrence Report No. 50-321/1981-114," November 10, 1981 (LER No. 81-114/01T-0).

178. "LER No. 81-114/01T-0," LER Monthly Report-December 1981, February 19, 1982 (p. 57).

179. "Reactor Notes," Nucleonics Week 23(4):12 (January 28, 1982), McGraw-Hill Inc., New York.

180. S. A. Massaro, ed., Power Reactor Events, November-December 1981, 4(1):5-6, U.S. Nuclear Regulatory Commission, May 1982.

181. Bob Senseney (NRC), Note to Thomas Novak et al. (NRC), "Topics for Meeting on November 2, 1981," October 30, 1981. (The attachment, dated November 2, 1981, was "Discussion Paper" from the Washington Office of the Overseas Electrical Industry Survey Institute, Inc.; Item 1 was "Fuel-pins damage due to baffle jet impingement in PWR".)

182. "Millstone-2," Licensed Operating Reactors, Status Summary Report, Data as of 10-30-81, NUREG-0020, 5(11):2-179 (November 1981).

183. "Nuclear Plant Equipment Returned for Adjustment," Wall Street Journal, Western Edition, July 8, 1980 (p. 4). Reprinted by permission of the Wall Street Journal.© Dow Jones & Company, Inc. 1980. All rights reserved.

184. U.S. Nuclear Regulatory Commission, Preliminary Notification of Event or Unusual Occurance, PNO-III-8-57A, June 22, 1981.

185. "Cook-1," Licensed Operating Reactors, Status Summary Report, Data as of 07-31-81, NUREG-0020, 5(8):2-066 (August 1981).

186. LER No. 81-000/99X-0," LER Monthly Report-September 1981, October 13, 1981 (p. 27).

187. Indiana and Michigan Electric Company, Licensee Event Report No. 81-000/99X-0, Docket No. 50315, July 2, 1981 (Cook-1).

188. D. V. Shaller (Indiana and Michigan Electric Company), Letter to J. G. Keppler (NRC), "Event at Donald C. Cook Nuclear Plant on June 19, 1981," July 2, 1981. (Docket No. 50315).

189. P. Y. Barnard (Florida Power Corporation), Letter t John F. Stolz (NRC), "Missing Control Rod Upper Guide Tube Nut," April 2, 1981. (Crystal River-3)

190. G. A. Schwenk (NRC), Memorandum for E. L. Jordan (NRC), "Loose Guide Tube Nut Discovered in OTSG at Crystal River Unit No. 3," April 10, 1981.

191. N. R. Metcalf (Westinghouse Electric Corporation), Letter to M. L. Lee (Consolidated Edison Company of New York), "Indian Point Unit 2: Condition of Fuel Assembly E-42," January 22, 1981.

192. John D. O'Toole (Consolidated Edison Company of New York), Letter to Boyce H. Grier (NRC), "LER 81-005/01T-0," February 3, 1981.

193. "LER No. 81-005/01T-0," LER Monthly Report-March 1981, March 27, 1981 (p. 48).

194. "Millstone-1," Licensed Operating Reactors, Status Summary Report, Data as of 05/31/81, NUREG-0020, 5(6):M-4 (June 1981).

195. E. Patrakka and E. Tirri, "Fuel Examination Experience at Finnish Nuclear Power Plants," Specialist's Meeting on Examination of Fuel Assembly for Water Cooled Power Reactors, IWGFPT/12, pp. 52-59, International Atomic Energy Agency, November 1981.

196. U.S. Nuclear Regulatory Commission, Preliminary Notification of Event or Unusual Occurrence, PNO-III-81-116, December 16, 1981.

197. L. O. Mayer (Northern States Power), Letter to NRC (Region III), "Spent Fuel Assembly D-34 Top Nozzle Event," Dec. 30, 1981. The attachment was Licensee Event Report 81-031/01T-0 (Docket No. 50282).

198. L. O. Mayer (Northern States Power), Letter to NRC (Region III), "Update Report-Spent Fuel Assembly D-34 Top Nozzle Event", May 12, 1982. The attachment was Licensee Event Report 81-031/01X-1 (Docket No. 50282).

199. Earl J. Brown (NRC), Memorandum for Karl V. Seyfrit (NRC), "Fuel Assembly Degradation While in the Spent Fuel Storage Pool," October 21, 1982, (AEOD/E242).

200. Karl V. Seyfrit (NRC), Memorandum for Darrell G. Eisenhut (NRC), "Fuel Assembly Degradation While in the Spent Fuel Storage Pool," October 22, 1982.

201. J. L. Wilson (Virginia Electric and Power Company), Letter to James P. O'Reilly (NRC), "Licensee Event Reports for Surry Units 1 and 2," December 19, 1981 (LER 81-078/03L-0).

202. "LER NO. 81-005/03L-0," LER Monthly Report-August 1981, September 25, 1981 (p. 71).

203. Herbert A. Autio (Yankee Atomic Electric Company), Letter to B. H. Grier (NRC), "Licensee Report 50-29/81-05/03L, Failure of #17 Control Rod to Scram," June 1, 1981. (LER No. 81-005/03L-0).

204. "Yankee Rowe," Licensed Operating Reactors, Status Summary Report, Data as of 04-30-81, NUREG-0020, 5(5):Y-2 (May 1981).

205. "Yankee Rowe," Licensed Operating Reactors, Status Summary Report, Data as of 02-28-81, NUREG-0020, 5(3):Y-2 (March 1981).

206. L. S. Rubenstein (NRC), Memorandum to T. M. Novak (NRC), "Yankee Rowe Cycle 15 Reload," July 20, 1981.

207. Herbert A. Autio (Yankee Atomic Electric Company), Letter to B. H. Grier (NRC), "Licensee Event Report 50-29/81-08/01T, Mechanical Damage of Fuel Cladding," June 10, 1981.

208. "LER NO. 81-008/01T-0," LER Monthly Report-July 1981, August 12, 1981 (p. 72).

209. H. A. Autio (Yankee Atomic Electic Company), Letter to B. H. Grier (NRC), "Licensee Event Report 50-29/78-31/01T," November 25, 1978.

210. J. A. Kay (Yankee Atomic Electric Company), Letter to D. M. Crutchfield (NRC), "Reporting Requirements," July 15, 1981.

211. Darrell G. Eisenhut (NRC), Note to Harold R. Denton (NRC), "Implementation of Resolution for BWR Channel Box Deflection Issue," November 3, 1981.

212. Norman C. Moseley (NRC), Memorandum for B. H. Grier et al., IE Bulletin No. 79-26, "Boron Loss from BWR Control Blades," November 19, 1979.

213. Thomas J. Rausch (Commonwealth Edison), Letter to James G. Keppler (NRC), "Dresden Station Units 1, 2, and 3 and Quad Cities Station Units 1 and 2, Response to IE Bulletin 79-26 Item 4, 'Boron Loss from BWR Control Blades,' NRC Docket Nos. 50-10/237/249 and 50-254/265," April 15, 1981.

214. Larry D. Root (Iowa Electric Light and Power Company), Letter to James G. Keppler (NRC), "Information for Item 4 of IE Bulletin 79-26, Rev. 1," April 16, 1981.

215. L. H. Heider (Vermont Yankee Nuclear Power Corporation), Letter to Boyce H. Grier (NRC), "Response to IE Bulletin 79-26, Revision 1," April 14, 1981.

216. Edward L. Jordan (NRC), IE Information Notice No. 82-89: Control Rod Drive (CRD) Guide Tube Support Pin Failures at Westinghouse PWRs, July 23, 1982.

217. "Reactor Notes," Nucleonics Week, 23(25):10 (June 24, 1982), McGraw-Hill Inc., New York.

218. U.S. Nuclear Regulatory Commission, Preliminary Notification of Event or Unusual Occurrence, PNO-II-82-64, June 9, 1982.

219. U.S. Nuclear Regulatory Commission, Preliminary Notification of Event or Unusual Occurrence, PNO-II-82-64A, June 22, 1982.

220. U.S. Nuclear Regulatory Commission, Preliminary Notification of Event or Unusual Occurrence, PNO-II-82-64B, July 1, 1982.

221. R. Andrew Holland (NRC), Memorandum for William E. Mills (NRC), "Minutes of Westinghouse, NRC Meeting on Wednesday, June 2," June 2, 1982.

222, Thomas M. Novak (NRC), Memorandum for the Atomic Safety & Licensing Boards for Callaway Plant (Unit 1) and Comanche Peak Steam Electric Station (Units 1 and 2) and the Atomic Safety & Licensing Appeal Board for Virgil C Summer Nuclear Station (Unit 1), "Board Notification-Control Rod Drive Guide Tube Support Pin Failures at Westinghouse Plants (Board Notification No. 82-81)," August 16, 1982.

223. T. M. Novak (NRC), Memorandum for all Licensing Project Managers (NRC), "Board Notification Regarding Control Rod Drive Guide Tube Support Pin Failures at Westinghouse Plants," August 16, 1982.

224. E. J. Ruzauskas, et al., C-E/EPRI Fuel Performance Evaluation Program, Examination of Calvert Cliffs 1 Test Fuel Assembly after Cycle 4, CE NPSD-146, June 1981 (EPRI Project RP586-1).

225. J. B. Melehan, et al., Interim Report Zion Unit 2 Cycle 4, WCAP-9845, March 1981 (EPRI Project RP611-1).

226. J. T. Mayer, et al., "B&W Fuel Performance to 40,000 MWd/MTU Burnup," Trans. Am. Nucl. Soc. 38:286-287, June 1981.

227. K. C. Radfrod, R. C. Goodspeed, and H. W. Keller, "Fabrication, Development and Application of Annular Al_2O_3-B_4C Burnable Absorber," Trans. Am. Nucl. Soc. 39:405-407, November-December 1981.

228. D. G. Franklin and P. Knudsen, "Phenomena Associated with Extending Fuel Burnup," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 1:4-1 through 7-4, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

229. D. G. Franklin, "The Schedule for Extending Fuel Burnup," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 2:8-7 through 8-10, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

230. K. N. Woods and J. O. Barner, Fuel Performance Improvement Program, Semiannual Progress Report, October 1981-March 1982, DOE/ET/34215-30, August 1982.

231. F. W. Buckman, C. E. Crouthamel, and M. D. Freshley, "FCI-Remedy Development for the Fuel Performance Improvement Program," Proceedings of Am. Nucl. Soc. Topical Meeting on Light Water Reactor Fuel Performance, Portland, Oregon, April-May 1979. (pp. 158-168.)

232. F. W. Buckman, et al., "Development of Annular-Coated-Pressurized and Sphere-Pac LWR Fuels," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 2:5-83 through 5-101, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

233. S. E. Turner, W. J. Elgin, and R. P. Hancock, Historical Survey of Nuclear Fuel Utilization in U.S. LWR Plants. SSA-122 (DOE/ER/10020-T1), August 1979.

234. P. M. Lang, "Contribution to Panel Discussion on Extended Burnup--How Far? How Fast?," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 2:8-3 through 8-5, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

235. W. E. Murphie and P. M. Lang, "Projected National and World Benefits to the LWR Fuel Cycle from Extended Burnup," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 2:7-61 through 7-69, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

236. L. S. Rubenstein and M. Tokar, "Regulatory Perspective on Extended Burnup Fuel," LWR Extended Burnup-Fuel Performance and Utilization, DOE/NE/34087, 2:6-31 through 6-37, April 1982. (Proceedings of Am. Nucl. Soc. Topical Meeting in Williamsburg, VA.)

237. S. J. Raffety (Dairyland Power Cooperative), LACBWR Cycle 6 Fuel Performance and Finalized Refueling Plan for Cycle 7, LAC-TR-096, June 1, 1981. (Also see LAC-7572.)

238. "NRC Staff, Utilities Meeting on New Steam Generator Proposals," Nucleonics Week, 23(30):2 (July 29, 1982), McGraw-Hill Inc., New York.

DISTRIBUTION

No. of Copies

OFFSITE

295 Nuclear Regulatory Commission
Division of Technical Information and Document Control
7920 Norfolk Avenue
Bethesda, MD 20014

2 DOE Technical Information Center

4 C. H. Berlinger
Chief, Core Performance Branch
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

R. O. Meyer
Core Performance Branch
Division of Systems Integration
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

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

No. of Copies

ONSITE

39 Pacific Northwest Laboratory

W. J. Bailey (12)
J. O. Barner
C. E. Beyer
M. E. Cunningham
M. D. Freshley
E. R. Gilbert
R. L. Goodman
R. J. Guenther
C. R. Hann
D. D. Lanning
R. R. Lewis
R. K. Marshall
C. L. Mohr
F. E. Panisko
L. J. Parchen
W. N. Rausch
G. E. Russcher
R. A. Scoggin
J. W. Upton
R. E. Williford
C. L. Wilson
Technical Information (5)
Publishing Coordination (2)

NRC FORM 335 (7-77) U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET**	1. REPORT NUMBER *(Assigned by DDC)* NUREG/CR-3001 PNL-4342
4. TITLE AND SUBTITLE *(Add Volume No., if appropriate)* Fuel Performance Annual Report for 1981	2. *(Leave blank)*
	3. RECIPIENT'S ACCESSION NO.
7. AUTHOR(S) W.J. Bailey (PNL) and M. Tokar (USNRC)	5. DATE REPORT COMPLETED MONTH November YEAR 1982
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)* Pacific Northwest Laboratory P.O. Box 999 Richland, Washington 99352	DATE REPORT ISSUED MONTH December YEAR 1982
	6. *(Leave blank)*
	8. *(Leave blank)*
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)* Division of Systems Integration Office of Nuclear Reactor Regulation U.S. Nuclear Regulatory Commission Washington, DC 20555	10. PROJECT/TASK/WORK UNIT NO.
	11. CONTRACT NO. Fin B2501
13. TYPE OF REPORT Technical	PERIOD COVERED *(Inclusive dates)* 1981
15. SUPPLEMENTARY NOTES	14. *(Leave blank)*

16. ABSTRACT *(200 words or less)*

This annual report, the fourth in a series, provides a brief description of fuel performance during 1981 in commercial nuclear power plants. Brief summaries of fuel operating experience, fuel problems, fuel design changes and fuel surveillance programs, and high-burrup fuel experience are provided. References to additional, more detailed information and related NRC evaluations are included.

17. KEY WORDS AND DOCUMENT ANALYSIS	17a. DESCRIPTORS
Fuel Performance	Fuel Operating Experience Fuel Problems Fuel Design Changes Fuel Surveillance Programs High-Burnup Fuel Experience

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT	19. SECURITY CLASS *(This report)* unclassified	21. NO. OF PAGES
	20. SECURITY CLASS *(This page)* unclassified	22. PRICE $

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

FOURTH CLASS MAIL
POSTAGE & FEES PAID
USNRC
WASH D C
PERMIT No G 67

120555078877 1 ANR3
US NRC
ADM DIV OF TIDC
POLICY & PUBLICATIONS MGT BR
PDR NUREG COPY
LA 212
WASHINGTON DC 20555

NUREG/CR-3001

FUEL PERFORMANCE ANNUAL REPORT FOR 1981

DECEMBER 1982