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NUREG/CR-3950
PNL-5210
Vol. 9

Fuel Performance Annual Report for 1991

Manuscript Completed: July 1994
Date Published: August 1994

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Prepared for
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
NRC FIN L1864

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Previous Reports in Series

Reports	Calendar Year	Date Issued
NUREG-0633	1978	December 1979
NUREG/CR-1818 (PNL-3583)	1979	January 1981
NUREG/CR-2410 (PNL-3953)	1980	December 1981
NUREG/CR-3001 (PNL-4342)	1981	December 1982
NUREG/CR-3602 (PNL-4817)	1982	March 1984
NUREG/CR-3950 (PNL-5210) Vol. 1	1983	March 1985
NUREG/CR-3950 (PNL-5210) Vol. 2	1984	March 1986
NUREG/CR-3950 (PNL-5210) Vol. 3	1985	February 1987
NUREG/CR-3950 (PNL-5210) Vol. 4	1986	March 1988
NUREG/CR-3950 (PNL-5210) Vol. 5	1987	March 1989
NUREG/CR-3950 (PNL-5210) Vol. 6	1988	May 1990
NUREG/CR-3950 (PNL-5210) Vol. 7	1989	June 1992
NUREG/CR-3950 (PNL-5210) Vol. 8	1990	November 1993

Abstract

This annual report, the fourteenth in a series, provides a brief description of fuel performance during 1991 in commercial nuclear power plants and an indication of trends. Brief summaries of fuel operating experience, fuel design changes, fuel

surveillance programs, high-burnup experience, problem areas, and items of general significance are provided. References to more detailed information and related U.S. Nuclear Regulatory Commission evaluations are included.

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Executive Summary

This report is the fourteenth in a series that provides a compilation of information regarding commercial nuclear fuel performance. The series of annual reports were developed as a result of interest expressed by the public, advising bodies, and the U.S. Nuclear Regulatory Commission (NRC) for public availability of information pertaining to commercial nuclear fuel performance.

During 1991, the nuclear industry's focus regarding fuel continued to be on extending burnup while maintaining fuel rod reliability. Utilities realize that high-burnup fuel reduces the amount of generated spent fuel, reduces fuel costs, reduces operational and maintenance costs, and improves plant capacity factors by extending operating cycles.

Fuel Operating Experience

As of December 31, 1991, there were 111 operable nuclear power plants licensed in the United States. Domestic commercial nuclear plants generated some 612.6 TWh of electricity and achieved a record high average capacity factor of 70.2% for the year. These commercial power plants depend primarily on five vendors to supply their fuel requirements: ABB Combustion Engineering Nuclear

Fuels (ABB CENF), Babcock & Wilcox Fuel Company (BWFC),^(a) General Electric Company (GE), Siemens Power Corporation (SPC), and Westinghouse Electric Corporation (W). By the end of 1991, these fuel vendors had supplied to the commercial nuclear industry over 112,000 fuel assemblies, representing approximately 12 million individual fuel rods.

The assembly-average burnup associated with commercial pressurized-water reactor (PWR) fuel discharged during the period 1980 to 1987 was fairly constant at about 30 GWd/MTU. In 1991 this assembly-average burnup increased to 35 GWd/MTU—the highest annual average ever achieved. Boiling-water reactor (BWR) assembly-averaged burnup for fuel being discharged in 1991 also reached a new high of 27.8 GWd/MTU.

Fuel rod reliability for the industry as a whole, excluding failures from debris-induced fretting wear, continues to be better than 99.98%. Fuel reliability continually improved during the 1980s but has appeared to level off in the last few years. The industry's median fuel reliability^(b) for the past 3 years is summarized below.

Industry Median Fuel Reliability Indicator

Year	PWRs ($\mu\text{Ci/ml}$)	BWRs ($\mu\text{Ci/sec}$)
1991	1.4×10^{-3}	37.0
1990	1.2×10^{-3}	99.0
1989	2.0×10^{-3}	83.0

(a) BWFC declined to provide information for this report.

(b) This data is based on the Institute of Nuclear Power Operations (INPO) standard fuel reliability indicator (FRI) method. INPO standard FRI for PWRs is the coolant iodine-131 activity normalized to a standard cleanup rate and corrected for tramp uranium. INPO standard FRI for BWRs is the rate of fission gas release at the steam air ejector.

Fuel Design Changes and Summary of Fuel Surveillance Programs

Because the trend in fuel design is to extend burnup and increase fuel reliability, fuel vendors have focused on these major areas in fuel research and development. Fuel manufacturers for PWRs have focused on extending fuel burnups, decreasing cladding corrosion, incorporating burnable absorbers, and preventing fuel damage. Fuel manufacturers for BWRs have also focused on extending fuel burnup and preventing cladding corrosion while reducing failure due to pellet-cladding interaction.

"Fuel System Design," Section 4.2 of the NRC Standard Review Plan (NUREG-0800) requires that plans for testing, inspection, and surveillance

of fuel be submitted and reviewed for each domestic nuclear power plant. These plans are subsequently referenced and/or summarized in a plant's safety analysis report (SAR). Supplementary fuel surveillance programs for new fuel designs are also required by the NRC. A brief summary of various fuel surveillance programs maintained by fuel vendors is provided within this report.

Extended Burnup

The number of utilities planning to incorporate fuel specifically designed for extended burnup in current and future cores is rapidly increasing. A summary of the highest fuel burnup levels achieved by utilities in their cores as of December 31, 1991, is provided in the table below.

Highest Fuel Burnup Achieved by Vendor in 1991

Vendor	Plant or Test	Type	Burnup (GWd/MTU)	Comment
ABB	ANO-2	PWR	46.5	Batch Average
CENF	Calvert Cliffs-2	PWR	43.0	Batch Average
	St. Lucie-2	PWR	41.5	Batch Average
GE		BWR	>45.0	Bundle Average
		BWR	60.0	Peak Pellet Exposure
SPC	R. E. Ginna	PWR	52.1	Assembly Average
	Untweser	PWR	51.8	Assembly Average
	Big Rock Point	BWR	45.1	Assembly Average
	Gundremmingen-C	BWR	40.1	Assembly Average
<u>W</u>	Zion-1	PWR	55.0	4 Assemblies Average
	North Anna-1	PWR	58.0	4 Assemblies Average
			>58.0	Lead Assembly Average
BWFC	Information was not provided by BWFC			

Problems Observed During 1991

This report reviews and categorizes significant events which occurred during 1991 that either adversely effected, or had the potential to adversely effect, commercial nuclear fuel performance/integrity. The following is a brief summary of events reported in Licensee Event Reports (LERs) or discussed in various technical publications.

- There were fourteen fuel handling oriented events reported by nuclear power plants in the United States in 1991. Three events involved fuel assemblies that were either dropped, broken, or damaged in fuel handling mishaps, nine events resulted from procedural deficiencies, and two events were related to other problems.
- There were ten events resulting from fuel storage-related problems of which three involved the ability to remove heat from spent fuel pools, two involved ventilation system upsets, two involved storage racking issues, and three involved spent fuel pool problems.
- There were 46 control rod oriented events reported by nuclear power plants in the United States in 1991. Fourteen events were associated with control rod system problems, eighteen with control rod operation, ten with control rod position indicators, one with control system failure, and three involved control rod swelling or cracking.
- There were five core power/coolant oriented events reported by nuclear power plants in the United States in 1991. One of the events was a violation of technical specifications of 100% power limits, one involved the axial shape index, two dealt with lowering of the water level, and one was a boron related problem.

Although fuel failures did occur during 1991 there were no specific in-reactor failure incidents reported in LERs.

Trends

The commercial nuclear power industry's fuel rod reliability, excluding failures from debris-induced fretting wear, continues to be better than 99.98%. Fuel rod reliability continually improved during the 1980s and appears to have leveled off in the last few years. Over the past decade, considerable attention has been given to reducing the number of fuel failures in commercial reactors. Previously observed failure mechanisms are reasonably well understood (e.g., PCI, CILC, Debris Fretting, and Hydriding). As a result, utilities in many instances no longer consider common fuel failures to be an "off-normal" event, reportable to the NRC. Furthermore, only one fuel vendor reports the number and causes of fuel failures in their fuel operating experience report to the NRC. Because neither the utilities nor the vendors are reporting the specific details regarding fuel failures, an accurate analysis of fuel failure trends is difficult. For example, in past years, when a fuel rod failed it was often reported by the utility and/or the vendor. However, in recent years the problems have been analyzed, their causes determined, and steps taken to prevent them. Therefore, failures that are not unusual are not as likely to be reported even though failures continue to occur. Failures may even be anticipated because of the large amount of failure data at a given reactor or reactors. This allows the issue to fade away even though fuel failures are still occurring, thus making it difficult to determine what effect, if any, the various solutions have had on the problem or if new failure mechanisms have developed in recent years.

Debris-induced fretting is the fuel failure mechanism seen most often in the past several years. Debris-induced fretting damage occurs as a result of metallic debris that is introduced into the reactor coolant system during refueling outages and system repair operations. This debris is carried into the fuel rod channels where it can become trapped and eventually wear through the cladding due to flow-induced vibration. Utilities have become aware of the consequences of debris in the system and have

Executive Summary

taken action to prevent the introduction of debris into the reactor coolant system. Although recent progress has been made to reduce the number of debris-induced fretting failures by introducing debris trapping fuel assembly design modifications, debris still represents the most significant failure mechanism.

Grid-to-rod fretting induced cladding failures were also observed in 1991. At least one of these failures was believed to be different from previously observed grid-to-rod fretting failures. Previously, grid-to-rod fretting failures have been caused by loose grid cells due to grid damage incurred either during fabrication or handling. There is a possibility that as fuel assembly burnups increase, relaxation of the grid cells may be occurring faster than previously observed. Changes in fuel assembly spacer designs to increase assembly performance may have also influenced this observed problem. Further examinations of the grid-to-rod fretting observed in 1991 are planned for 1992 by at least one vendor.

Further improvements in fuel reliability will require vendors to continue to improve quality control and fuel manufacturing processes. Fuel reliability in the 90's may be affected by more complex water chemistry and operating conditions as utilities move towards hydrogen water chemistry, zinc injection, chemical cleaning of BWR fuel, and lithium injection in PWR plants. Better fuel inspection methods will be needed as well as more accurate methods of predicting the number of failed rods.

Crud induced localized corrosion (CILC), a BWR fuel failure mechanism, has diminished as utilities replace their brass condensers. However, current operational trends could be expected to lead to continued corrosion concerns including cladding embrittlement. The attainment of higher burnups, higher operating coolant temperatures, newer fuel

designs using thinner wall zirconium alloys, and various water chemistry changes will all pose additional risk to cladding corrosion. In fact, BWR vendors have not reported large defect failures in their zirconium barrier clad fuel in the 1991 vendor reports to the NRC although some utilities have observed large axial cracks in these fuel designs resulting in high primary coolant activities. The source of the large axial cracks is thought to be due to secondary hydriding promoted by a small initial defect in the rod.

Pellet-cladding interaction has not been reported in PWRs and has significantly been reduced in BWRs by the introduction of barrier fuel designs. However, due to the observed decrease in cladding ductilities at higher burnup levels the likelihood of PCI fuel failures may increase as a larger number of fuel rods achieve these burnup levels in the future. In addition, Zircaloy growth is identified as a possible future problem as burnup levels continue to increase.

Future fuel design modifications could reduce the margin to safety limits. For example, for better PCI resistant fuel in BWRs there has been a trend to reduce fuel rod linear heat generation rate (LHGR) by decreasing the rod's outer diameter. While this results in a rod with a lower stored energy content, therefore an increased LOCA margin, it reduces the margin that relates to thermal-hydraulic core stability.

The industry is looking towards continuing the evolution of fuel assembly lattice designs and the optimization of burnable absorbers to further extend fuel burnup levels. Moreover, neutronically efficient fuel assembly designs will save on uranium and enrichment costs and a continued effort to develop higher burnup fuel will reduce future spent fuel storage and disposal costs, result in longer fuel cycles, and increase plant capacity factors.

Acronyms

ABB CENF	ASEA Brown Boveri Combustion Engineering Nuclear Fuel
AEC	U.S. Atomic Energy Commission
AECL	Atomic Energy of Canada Limited
AFD	axial flux difference
ANO	Arkansas Nuclear One
ANF	Advanced Nuclear Fuels (now Siemens Power Corporation)
ANS	American Nuclear Society
API	Absolute Position Indicator
AP&L	Arkansas Power & Light Co.
ARI	Alternate Rod Injection System
ARPI	Analog Rod Position Indicator
ASI	Axial Shape Index
BWFC	Babcock & Wilcox Fuel Company
BWR	boiling-water reactor
CANDU	Canadian Deuterium-Uranium Reactor
CDT	Central Daylight-Savings Time
CEA	control element assembly
CEAC	control element assembly calculator
CEDM	control element drive mechanism
CFR	Code of Federal Regulations
CIAS	containment isolation actuation signal
CILC	crud induced localized corrosion
COFFEC	Cost of Fuel Failure Evaluation Code
CRD	control rod drive
CRDM	control rod drive mechanism
CST	Central Standard Time
DFBN	debris filter bottom nozzle
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOE	U.S. Department of Energy
DRPI	Digital Rod Position Indication System
EDT	Eastern Daylight-Savings Time
EPRI	Electric Power Research Institute
EST	Eastern Standard Time
ESF	engineered safety feature
FGR	fission gas release
FHB	fuel handling building
FRI	Fuel Reliability Indicator
GE	General Electric Company
GPU	General Public Utilities Corp.
GWd/MTU	number of Gigawatt days of thermal energy released by fuel containing one metric ton (10^3 kg) of heavy-metal atoms (e.g., uranium)

HBE ²	High Burnups Effects Program
HCU	hydraulic control unit
HTGR	high-temperature gas reactor
IFBA	integral fuel burnable absorber
IFM	intermediate flow mixer
INPO	Institute of Nuclear Power Operations
IRM	intermediate range monitor
IRPIS	individual rod position indicator
ISFSI	independent spent fuel storage installation
LCO	limiting condition for operation
LER	licensee event report
LFA	lead fuel assembly
LHGR	linear heat generation rate
LTA	lead test assembly
LUA	lead use assembly
LWR	light-water reactor
MOX	mixed-oxide fuel
MSIS	main steam isolation signal
MST	Mountain Standard Time
NPAR	Nuclear Plant Aging Research
NRC	U.S. Nuclear Regulatory Commission
OFA	optimized fuel assembly
ORNL	Oak Ridge National Laboratory
PCI	pellet cladding interaction
PHWR	pressurized heavy-water reactor
PIE	Post Irradiation Examination
PNL	Pacific Northwest Laboratory
PWR	pressurized-water reactor
RCCA	rod control cluster assembly
RCIC	Reactor Core Isolation Cooling System
RCS	Reactor Coolant System
RPCS	Rod Pattern Control System
RPI	rod position indicator
RPS	reactor protection system
SAR	safety analysis report
SCC	stress corrosion cracking
SFP	spent fuel pool
SPC	Siemens Power Corporation
SR	surveillance requirement
SRO	senior reactor operator

Acronyms

TS	technical specification
TSO	technical specification operability notification
UFSAR	Updated Final Safety Analysis Report
USAR	Updated Safety Analysis Report
<u>W</u>	Westinghouse Electric Corporation
WNP	Washington Nuclear Plant
Zry	Zircaloy

1.0 Introduction

This report is the fourteenth in a series that provides a compilation of the available information on nuclear reactor fuel performance. The series of annual reports were developed as a result of interest expressed by the public, advising bodies, and the U.S. Nuclear Regulatory Commission (NRC) for public availability of information pertaining to commercial nuclear fuel performance. A listing of the previous reports in this series can be found in the front matter of this report.

As noted in the first report (NUREG-0633) of this annual series, the U.S. Atomic Energy Commission (AEC), and later the NRC, requested operating nuclear reactor fuel performance details through the reporting requirements of Regulatory Guide 1.16. However, over the years the material presented in these reports has evolved. The 1971 version of Regulatory Guide 1.16 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, however, only abnormal degradation of fuel cladding and an indication of failed fuel were reportable items. Reporting requirements were further reduced in 1977 when it was decided that only abnormal degradation of fuel cladding was to be included, and the requirement for an annual operating report was eliminated.

In May 1982, the NRC proposed amending its regulations to improve the information received in licensee event reports (LERs) from nuclear power plant licensees (47 FR 19543; 48 FR 39039). The new regulation became effective January 1, 1984, and Paragraph 50.73(a)(2)(ii) of the new LER rule required events to be reported where the plant, including its principal safety barriers (i.e., fuel cladding, reactor coolant,

system boundary, or the containment), was seriously degraded or was determined to be in an unanalyzed condition. Examples of situations required to be addressed by this paragraph are "fuel cladding failures in the reactor or the storage pool that exceed expected values, that are unique or widespread, or that resulted from unexpected factors."

Other reports that deal with topics similar to this one are available. The NUREG series reports, *Nuclear Power Plant Operating Experience*, were printed through 1980 but did not address normal operation surveillance results, generic problems, and design trends. Results of plant operating experience were also screened by the Electric Power Research Institute (EPRI).

This report, which was prepared by Pacific Northwest Laboratory (PNL),^(a) specifically focuses on events that occurred in 1991 and the information that was available from vendor reports and various nuclear power oriented publications at that time.

There are seven sections in this report. Fuel operating experience of the five vendors that supply nuclear fuel to nuclear power plants licensed in the United States is presented in Section 2.0. Fuel design changes and fuel surveillance programs of these vendors are described in Section 3.0. The high-burnup fuel experience of these vendors and an industry overview is summarized in Section 4.0. Described in Section 5.0 are problems (especially generic ones) that occurred during the reporting period, and fuel failure trends and trends of reported events are summarized in Section 6.0. An extensive reference list (Section 7.0) is provided so that readers may refer to the actual publications to acquire further details regarding the various topics presented herein.

(a) Pacific Northwest Laboratory is operated for the U.S. Department of Energy by Battelle Memorial Institute.

2.0 Fuel Operating Experience

As of December 31, 1991, there were 111 operable nuclear power plants licensed in the United States. Domestic commercial nuclear plants generated some 612.6 TWh of electricity and achieved a record high average capacity factor of 70.2% for the year. In comparison, nuclear plants generated 576.9 TWh of electricity and had an average capacity factor of 66.1% in 1990 (DOE/EIA-0436 1992). These commercial nuclear power plants depend primarily on five vendors to supply their nuclear fuel needs: ASEA Brown Boveri Combustion Engineering Nuclear Fuels (ABB CENF), Babcock & Wilcox Fuel Company (BWFC),^(a) General Electric Company (GE), Siemens Power Corporation (SPC), and Westinghouse Electric Corporation (W). As of the end of 1991, these fuel vendors had supplied to the commercial nuclear industry over 112,000 fuel assemblies, representing approximately 12 million individual fuel rods. This section provides a brief synopsis of domestic nuclear fuel operating experience for 1991.

The assembly-average burnup associated with commercial pressurized-water reactor (PWR) fuel discharged during 1980-1987 was fairly constant at about 30 GWd/MTU. In 1991 this assembly-average burnup increased to 35 GWd/MTU—the highest annual average ever achieved. Boiling-water reactor (BWR) assembly-averaged burnup for fuel being discharged in 1991 also reached a new high of 27.8 GWd/MTU (SR/CNEAF/93-01 1993). The increasing trend in average burnup of discharged nuclear fuel from 1968 to 1991 is illustrated in Figure 2.1. (Refer to Section 4.0 of this report for a summary of in core high-burnup experience.)

Fuel rod reliability for the industry as a whole, excluding failures from debris-induced fretting wear, continues to be better than 99.98%. Different vendors report fuel reliability information differently. An increase in the overall fuel reliability typically means a decrease in the number of failed rods. Fuel integrity ratings have been developed to quantify fuel performance in terms of

reliability. These ratings are normally obtained from iodine-131 activity levels in the primary coolant initially, followed, where possible, by gas sipping or ultrasonic measurements. It should be noted that the definition of failed fuel has not been uniformly applied throughout the years. In many cases the number of fuel failures was inferred from indirect evidence, while in other cases only directly observed failures were counted (Bailey and Tokar 1985).

A common standard to assess the overall performance of fuel was developed by the Institute of Nuclear Power Operations (INPO). For PWRs, the fuel reliability indicator (FRI) is obtained using the reactor coolant iodine-131 activity level normalized to a standard cleanup system flow rate (referred to as "uncorrected activity") and correcting for tramp uranium.^(b) This corrected iodine-131 level is alternately referred to as the "corrected activity," or FRI value, and is typically reported in units of $\mu\text{Ci/ml}$ or $\mu\text{Ci/g}$. For BWRs, the FRI value is determined from the fission gas release measured at a steam jet air ejector and is reported in units of $\mu\text{Ci/sec}$. Smaller FRI values are indicative of fewer failed fuel rods in the core. The average coolant iodine-131 activity is about $1.2 \times 10^{-3} \mu\text{Ci}$ of iodine-131 per gram; levels above this value usually mean the presence of leaking fuel rods. The specific coolant activity technical specification limit for each reactor depends on such factors as reactor power and coolant purification flow rate.

A historical perspective on fuel reliability is provided in Appendix A. Overall, commercial reactor fuel reliability continues to be excellent. However, debris-induced fretting wear and grid-to-rod fretting continue to be a problem (refer to Section 6.0 for a more detailed discussion). Also, sporadic events involving damage to, or failure of, nuclear fuel continues to exist. Many of these events are summarized in Section 5.0 of this report.

(a) BWFC declined to submit information for this years annual report.

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

Figure 2.1 Summary of Average Burnup Associated with Spent Nuclear Fuel Discharged from 1968 through 1991
(SR/CNEAF/93-01 1993)

2.1 ABB Combustion Engineering Nuclear Fuel

As of December 31, 1991, ABB CENF had supplied a total of 8840 fuel assemblies to U.S. utilities operating commercial PWRs (30% in core, 70% discharged). This represents a total of 1,789,392 ABB CENF manufactured fuel rods. The ABB CENF fuel experience for 1991 is summarized in Table 2.1.

2.1.1 ABB CENF Fuel Burnup Experience

Batch-averaged burnups achieved by ABB CENF fuel in core, or discharged, during 1991 are presented in Table 2.1. The highest batch-averaged burnup achieved at discharge in 1991 was 46.5 GWd/MTU for 17 assemblies at Arkansas-2. The highest batch-average burnup achieved to date for ABB CENF fuel is 56.8 GWd/MTU for four assemblies during 1988 at Calvert Cliffs-1 (Buley and Wu 1990). The number of active and discharged ABB CENF assemblies as of December 31, 1991, is shown in Table 2.2.

Table 2.1 Summary of ABB Combustion Engineering Nuclear Fuel, Fuel Irradiated and/or Discharged in 1991
(Permission to reproduce this material is granted by ABB Combustion Engineering Nuclear Fuel)

Reactor/ (Fuel Cycle)	Fuel Batch	Number of Assemblies		Number of Fuel Rods		Batch-Averaged Burnup	
		In Reactor at End of Year	Discharged During Year	In Reactor at End of Year	Discharged During Year	GWd/MTU On Dec. 31, 1991	At Discharge
Arkansas-2/ (Cycles 8 & 9)	F	0	17	0	4,012	----	46.5
	H	0	28	0	6,352	----	42.8
	J	45	23	10,252	5,060	42.5	37.0
	K	64	0	14,416	0	28.9	----
	L	68	0	15,344	0	10.7	----
Calvert Cliffs-1/ (Cycle 10)	K	69	0	12,144	0	39.3	----
	L	52	0	9,152	0	30.5	----
	M	92	0	15,280	0	20.7	----
Calvert Cliffs-2/ (Cycles 8 & 9) ^(a)	H	1	69	176	12,144	33.7	43.0
	J	36	24	6,336	4,224	34.4	34.0
	K	88	0	14,800	0	27.1	----
	L	92	0	15,584	0	7.0	----
Fort Calhoun/ (Cycle 13)	M	41	0	7,048	0	39.0	----
	N	44	0	7,552	0	29.5	----
	P	40	0	6,784	0	15.4	----
Maine Yankee/ (Cycle 12)	P	72	0	12,400	0	40.5	----
	Q	72	0	12,464	0	32.5	----
	R	72	0	12,448	0	16.9	----
Palo Verde-1/ (Cycle 3)	B	1	0	220	0	33.0	----
	C	52	0	12,016	0	34.0	----
	D	80	0	18,528	0	30.0	----
	E	108	0	24,240	0	20.0	----

Fuel Operating Experience

Table 2.1 (contd)

Reactor/ (Fuel Cycle)	Fuel Batch	Number of Assemblies		Number of Fuel Rods		Batch-Averaged Burnup	
		In Reactor at End of Year	Discharged During Year	In Reactor at End of Year	Discharged During Year	GWd/MTU On Dec. 31, 1991	At Discharge
Palo Verde-3/ (Cycle 2 & 3)	B	33	40	7,260	8,800	21.5	28.0
	C	16	48	3,776	10,944	28.2	30.0
	D	104	0	23,584	0	24.9	-----
	E	88	0	19,904	0	9.0	-----
St. Lucie-2/ (Cycle 6)	E	12	0	2,800	0	40.4	-----
	F	49	0	11,380	0	41.5	-----
	G	80	0	18,448	0	31.5	-----
	H	76	0	17,456	0	15.9	-----
San Onofre-2/ (Cycles 5 & 6)	A	1	1	236	236	15.0	26.0
	F	0	108	0	24,112	-----	41.3
	G	108	0	24,112	0	23.4	-----
	H	108	0	24,176	0	1.5	-----
San Onofre-3/ (Cycle 5)	A	1	5	236	1,180	23.0	-----
	F	108	0	24,112	0	40.5	-----
	G	108	0	24,112	0	22.5	-----
Waterford-3/ (Cycles 4 & 5)	C	1	1	224	224	34.6	40.9
	D	0	48	0	11,232	-----	42.6
	E	48	36	10,976	7,920	39.0	38.0
	F	84	0	18,896	0	27.6	-----
	G	84	0	18,896	0	8.7	-----
Yankee Rowe/ (Cycle 21) ^(a)	C	36	0	8,222	0	26.0	-----
	D	40	0	9,090	0	10.0	-----

(a) Mid-cycle maintenance shutdown in October 1991 at Yankee Rowe has been made permanent.

Table 2.2 ABB Combustion Engineering Nuclear Fuel Burnup Experience With All-Zircaloy Assemblies Status as of December 31, 1991 (Permission to reproduce this material is granted by ABB Combustion Engineering Nuclear Fuel)

Fuel Assembly Batch Burnup, GWd/MTU	In-Core Fuel Assemblies with Pressurized Fuel Rods								Discharged Fuel							
	Number of Assemblies				Number of Fuel Rods				Number of Assemblies				Number of Fuel Rods			
	14x14	16x16	Other ^(a)	Total	14x14	16x16	Other ^(a)	Total	14x14	16x16	Other ^(a)	Total	14x14	16x16	Other ^(a)	Total
0 to 3.999	0	108	0	108	0	24,176	0	24,176	0	0	0	0	0	0	0	0
4.000 to 7.999	92	0	0	92	15,584	0	0	15,584	6	0	0	6	1,048	0	0	1,048
8.000 to 11.999	0	240	40	280	0	54,144	9,090	63,234	97	0	136	233	16,148	0	28,752	44,900
12.000 to 15.999	40	76	0	116	6,784	17,456	0	24,240	247	386	72	705	42,935	90,984	15,284	149,203
16.000 to 19.999	72	97	0	169	12,448	21,852	0	34,300	256	192	0	448	44,344	44,448	0	88,792
20.000 to 23.999	92	325	0	417	15,280	72,700	0	87,980	151	104	8	263	25,276	23,392	1,728	50,396
24.000 to 27.999	88	188	36	312	14,800	42,480	8,222	65,502	476	479	0	955	81,034	107,492	0	188,526
28.000 to 31.999	96	301	0	397	16,704	68,300	0	85,004	794	373	100	1,267	136,376	82,800	22,090	242,266
32.000 to 35.999	109	82	0	191	18,976	18,972	0	37,948	536	511	36	1,083	93,660	118,120	8,222	220,002
36.000 to 39.999	110	101	0	211	19,192	23,212	0	42,404	246	210	0	456	43,232	48,296	0	91,528
40.000 to 43.999	72	250	0	322	12,400	57,040	0	69,440	385	395	0	780	66,618	89,888	0	156,506
44.000 to 47.999	0	0	0	0	0	0	0	0	0	21	0	21	0	4,956	0	4,956
48.000 to 51.999	0	0	0	0	0	0	0	0	2	1	0	3	349	230	0	579
52.000 to 55.999	0	0	0	0	0	0	0	0	1	0	0	1	176	0	0	176
56.000 to 59.999	0	0	0	0	0	0	0	0	4	0	0	4	702	0	0	702
	771	1,768	76	2,615	132,168	400,332	17,312	549,812	3,201	2,672	352	6,225	551,898	611,606	76,076	1,239,580

(a) ABB CENF or W 15x15 lattice with cruciform control blades (Palisades and Yankee Rowe).

2.1.2 ABB CENF Fuel Reliability

ABB Combustion Engineering Nuclear Fuel has estimated its overall reliability of fuel fabricated since 1983, excluding failures due to debris-induced fretting wear, to exceed 99.998% (Toelle 1992). A comparison of corrected iodine-131 activities in ABB CENF-fueled plants for 1987 and 1991 is presented in Table 2.3. The corrected activities were derived using the INPO standard FRI method. The downward trend in iodine-131 coolant activity for ABB CENF fueled PWRs during 1987-1991 is illustrated in Figure 2.2. The average plant activity by the end of 1991 was 0.0074 $\mu\text{Ci/g}$, and the median was 0.0024 $\mu\text{Ci/g}$. These values compare well with industry standards as reported by INPO.

It was estimated that 75% of the leaking fuel that was fabricated after 1983 (current fabrication process) and operated during 1987-1991, was caused by debris-induced fretting wear of the Zry-4 fuel rod cladding. Many of these leaking fuel rods were removed and replaced with non-fueled rods during refueling outages using ABB CENF fuel assembly reconstitution methods (Toelle 1992).

Table 2.3 Comparison of Corrected Coolant Iodine-131 Activities in 1987 and 1991 for Fuel Manufactured by ABB Combustion Engineering Nuclear Fuel (Permission to reproduce this material is granted by ABB Combustion Engineering Nuclear Fuel)

Corrected ^a Iodine-131 Activity Range, $\mu\text{Ci/g}$	Percentage of Plants in Range	
	End of 1987	End of 1991
> 0.05	23	0
0.005 - 0.05	38.5	25
0.0005 - 0.005	38.5	50
< 0.0005	0	25
Average Plant Corrected Iodine-131 Activity, $\mu\text{Ci/g}$	0.0304	0.0074
Median Plant Corrected Iodine-131 Activity, $\mu\text{Ci/g}$	0.0181	0.0024

(a) Corrected for tramp uranium and normalized to the same cleanup rate using the standard INPO method with a reference date of August 1989.

Figure 2.2 Corrected Coolant Activity* Versus Time (Permission to reproduce this figure is granted by ABB Combustion Engineering Nuclear Fuel)

2.2 General Electric Company

General Electric commercial nuclear fuel includes both zirconium liner (barrier) and conventional (non-barrier) fuel. Since the introduction of the barrier design feature by GE in 1983, all fuel designs offered by GE include barrier or non-barrier options. In recent years, the majority of the fuel fabricated by GE has been of the barrier type.

As of December 31, 1991, over 4.3 million GE 8x8 fuel type production Zircaloy-clad UO_2 rods were in, or had completed operation in, commercial BWRs. Approximately 1.55 million of these were pellet cladding interaction (PCI) resistant barrier fuel rods. The cumulative number of fuel rods in GE 8x8 bundles, loaded as a function of calendar year, is illustrated in Figure 2.3. As of the end of 1991, over 1.4 million GE fuel rods were in operation. The cumulative number of GE rods loaded into BWRs, by fuel version, is presented in Figure 2.4. A cross reference for GE fuel assembly types, including specific attributes, is provided for the reader in Table 2.4.

In 1991, fifteen domestic and nine overseas GE BWR plants containing GE fuel had refueling outages with over 3700 new GE 8x8 fuel bundles loaded. Nearly 73% of this new fuel was GE's latest production designs: GE8X8EB and GE8X8NB (Klapproth 1992).

2.2.1 GE Fuel Burnup Experience

As of December 31, 1991, GE's fuel experience included the successful operation of fuel bundles to greater than 45 GWd/MTU bundle-averaged exposure. This equates to approximately 60 GWd/MTU peak pellet exposure (Klapproth 1992).

2.2.2 GE Fuel Reliability

General Electric Company reports that as of 1991 GE fuel rod reliability remained better than 99.98% (Klapproth 1992). General Electric Company does not report its fuel reliability using the INPO standard for comparison against the other vendors.

Figure 2.3 General Electric Company 8x8 BWR Fuel Rod Experience
(Permission to use figure is granted by General Electric Company)

Figure 2.4 General Electric Company BWR Fuel Rods in Operation as of December 31, 1991
(Permission to use figure is granted by General Electric Company)

Table 2.4 Cross Reference for GE Fuel Assembly Types

Fuel Assembly	Assembly Version	Helium Backfill Pressure (ATM)	Barrier Cladding Option ^(a)	Hydrogen Getter ^(b)	Radially and Axially Zoned Enrichments	Radially and Axially Distributed Gadolinium
8X8	GE4	1	No	Yes	(Radial Only)	Yes
8X8R	GE5	1	No	Yes	Yes	Yes
P8X8R	GE6,7	3	No	Yes	Yes	Yes
BP8X8R	GE6B,7B	3	Yes	Yes	Yes	Yes
GE8X8EB	GE8B	(c)	Yes	Yes	Yes	Yes
GE8X8NB	GE9B	(c)	Yes	Yes	Yes	Yes
GE8X8NB-1	GE10	(c)	Yes	Yes	Yes	Yes
GE11	GE11	(c)	Yes	Yes	Yes	Yes

(a) Cladding consists of Zircaloy-2 with an inner lining of pure zirconium. The lining is mechanically bonded to the base material during manufacture. Since pure zirconium is softer than Zircaloy the barrier material inhibits PCI crack formation. Barrier fuel was introduced by GE as an answer to the PCI failure mechanism.

(b) Consists of small chips of a zirconium alloy, loosely packed in a stainless steel tube with one end capped and the other end covered with a wire screening. The getter is located in the rod plenum space and prevents chemical attack from moisture or hydrogenous impurities inadvertently admitted into the fuel rod during manufacture.

(c) Proprietary information.

2.3 Siemens Power Corporation

Siemens Power Corporation (formerly Advanced Nuclear Fuels Corporation) was originally formed as a unit of Exxon Corporation in 1969. Siemens Power Corporation designs and manufactures nuclear fuel for utilities throughout the United States, Belgium, France, Sweden, Taiwan, and the Federal Republic of Germany. As of December 31, 1991, fuel manufactured by SPC had been loaded into 51 commercial LWRs, including 25 BWRs and 26 PWRs. Siemens Power Corporation fuel has also been supplied to the LOFT test reactor.

A total of 19,915 fuel assemblies comprising 2,378,153 SPC fuel rods were irradiated by the end of 1991. Of these, 12,826 assemblies were irradiated in BWRs and 7089 assemblies were irradiated in PWRs. Siemens Power Corporation fuel experience is summarized in Table 2.5 (Copeland 1993).

2.3.1 SPC Fuel Burnup Experience

The highest assembly-averaged burnups reached by SPC fuel to date are 52.1 GWd/MTU in the R. E. Ginna

(PWR) in New York and 45.1 GWd/MTU at the Big Rock Point (BWR) in Michigan. The BWR 9x9 and PWR 17x17 fuel assemblies reached new high burnups during 1991. The BWR 9x9 fuel reached an exposure of 40.1 GWd/MTU at Gundremmingen-C in Germany, and PWR 17x17 fuel reached an exposure of 51.8 GWd/MTU at Unterweser, also in Germany. Siemens Power Corporation reports that its BWR 9x9 and PWR 17x17 fuels are more resistant to failure because of reduced linear heat generation rates (Copeland 1993). Siemens Power Corporation burnup distributions for BWR and PWR irradiated fuel are presented in Figures 2.5 and 2.6, respectively (Copeland 1993).

2.3.2 SPC Fuel Reliability

Throughout 1991 SPC fuel rod integrity remained better than 99.998%. Siemens Power Corporation reports fuel reliability using the INPO FRI standard. The FRI distribution for SPC PWR and BWR fuel is shown in Figure 2.7. Siemens Power Corporation reported no fuel failures attributed to design or manufacturing in 1991 (Copeland 1993). The 5-year trend in the FRI value indicates continued improvement in SPC fuel performance.

Table 2.5 Summary of Siemens Power Corporation Fuel Experience Through December 31, 1991
(Permission to reproduce this material is granted by Siemens Power Corporation)

Reactor Type	In Core Fuel Assemblies		Discharged Fuel Assemblies		Total Quantity
	Quantity	Max Burnup GWd/MTU	Quantity	Max Burnup GWd/MTU	
BWR	8,171	40.1	4,655	45.1 ^(a)	12,826
PWR	2,200	51.8	4,889	52.1	7,089
Total	10,371		9,544		19,915

Reactor Type	In Core Fuel Rods	Discharged Fuel Rods	Total
BWR	575,997	294,224	870,221
PWR	493,590	1,014,342	1,507,932
Total	1,069,587	1,308,566	2,378,153

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

During 1991, breaches in fuel rod cladding attributable to causes other than fuel design or manufacturing were determined to be the result of debris fretting (Copeland 1993); however, no specific data or information was provided regarding these fuel failures.

2.4 Westinghouse Electric Corporation

Westinghouse Electric Corporation first introduced Zircaloy clad fuel for use in commercial PWRs in 1968. Since then, 75 commercial PWRs have operated using W Zircaloy clad fuel. A chronological listing of nuclear power plants that have operated using W fuel assemblies is provided in Table 2.6. The W commercial nuclear fuel experience includes the design and production of 14x14, 15x15, 16x16, and 17x17 fuel assemblies for light water PWRs.

During 1991, fifty-seven plants operated using W-supplied fuel. As of December 31, 1991, a total of about 2.68 million Zircaloy clad fuel rods were in operation, representing 10,938 fuel assemblies. The addition of discharged fuel brings the total number of irradiated Zircaloy clad fuel rods to about 7.9 million. This

represents a total of approximately 33,600 fuel assemblies containing roughly 14,700 metric tons of UO_2 (Slagle 1993).

2.4.1 W Fuel Burnup Experience

Westinghouse Electric Corporation PWR fuel generated 390 million MWd (thermal) energy throughout 1991. Out of approximately 33,600 W irradiated fuel assemblies, 6151 assemblies have achieved average burnups in excess of 36 GWd/MTU. Of these extended burnup assemblies, 1949 had burnups greater than 40 GWd/MTU, and 444 had burnups greater than 44 GWd/MTU.

The average burnup of all W discharged fuel is about 29.3 GWd/MTU, and the average burnup of all W fuel (in-core plus discharged) is about 26.6 GWd/MTU. Thirty-three W-fueled plants have operated with fuel region average burnups in the range of 36 to 46.1 GWd/MTU. Presented in Figures 2.8, 2.9, and 2.10 are burnup summaries of active and discharged W fuel rods as of December 31, 1991.

Figure 2.5 Distribution of Irradiated Siemens Power Corporation Fuel By Assembly-Averaged Burnup as of December 31, 1991, for BWRs (Permission to reproduce this figure is granted by Siemens Power Corporation)

2.4.2 W Fuel Reliability

Westinghouse Electric Corporation reports that the average coolant Iodine-131 activity level ($\mu\text{Ci/g}$) for W-fueled plants continued to decrease in 1991. Shown in Figure 2.11 is the distribution of coolant activities in W-fueled plants, both domestic and international, from 1982 to 1991 for uncorrected iodine-131 and from 1988 to 1991 for corrected iodine-131. Although this information does not provide an accurate representation of plant specific fuel reliability, it does demonstrate an overall trend in the performance of W-supplied fuel.

During 1991, ultrasonic testing examinations were performed at 12 reactor sites. Sixty-one leaking rods were identified in 21 assemblies. Fuel assembly reconstitution was performed on 10 of the 21 assemblies. A TV camera examination of the rods removed from the 10 reconstituted assemblies found 9 rod failures due to debris-induced fretting, 40 rod failures due to grid-rod fretting, and 3 rod failures had no primary failure mechanism identified.

Grid-to-rod fretting was observed on the bottom inconel grid at one site in 1991. This is believed to be different from previously observed grid-to-rod fretting caused by

Figure 2.6 Distribution of Irradiated Siemens Power Corporation Fuel By Assembly-Averaged Burnup as of December 31 1991, for PWRs (Permission to reproduce this figure is granted by Siemens Power Corporation)

grid damage incurred during fabrication or handling at the site. Grid-to-rod fretting is usually observed at the bottom grid, because the vibration excitation is greater in that area due to the inlet flow turbulence. The damage that was observed in 1991 is thought to have occurred by a different mechanism, since there was wear at upper grid spans as well as the bottom grid on some rods. Additional examinations of the grid-to-rod fretting of 1991 is planned for 1992 (Slagle 1993).

2.4.3 W Non-Fuel Core Components

Westinghouse Electric Corporation has approximately 3000 rod control cluster assemblies (RCCAs) in operation. In 1983 it was observed that RCCAs are susceptible to fretting wear against upper internal guide cards

while fully withdrawn and stationary and to hairline cracks at the tips. Upon completion of an EPRI sponsored hot cell examination program of the affected RCCAs, operational guidelines were issued suggesting that the RCCAs be periodically repositioned and inspected for replacement prior to wear-through. Presently, RCCA inspections are performed by utilities as part of a managed maintenance program. The inspection data are evaluated, and strategies are developed to maximize component life. Furthermore, design changes have subsequently been included in RCCAs to reduce susceptibility to tip-cracking.

In 1988 a routine wear inspection of a hafnium absorber RCCA revealed localized increases in cladding diameter. It was later determined that hafnium absorber

Figure 2.7 Siemens Power Corporation Fuel Reliability Indicator, Using the INPO Standard Method (Permission to reproduce this figure is granted by Seimens Power Corporation)

RCCAs are prone to damage from hydriding (hydrogen diffusion through stainless steel cladding). However, W has reported that "detailed eddy current inspections of the hafnium RCCAs have shown that safe operation of the affected plants is not compromised, at least through the third 18-month or a fourth annual cycle."

2.5 Babcock & Wilcox Fuel Company

Babcock & Wilcox Fuel Company declined to provide information regarding performance of their nuclear fuel during this reporting period.

Table 2.6 Westinghouse Electric Corporation Fuel Performance Status Report Through 1991 (Permission to use this copyrighted material is granted by Westinghouse Electric Corporation)

Reactor	Location	Owner	Date of First Electrical Power	Nominal MWe Gross	Current Cycle Number	Peak Region Avg. Burnup GWd/MTU ^(a)	Generation MWh(e) Cumulative
Jose'de Cabrera	Spain	Union Electrica S.A	09/68	160	15	30.8	22,446,710
Beznau 1	Switzerland	Nordostschweizerische Kraftwerke AG	08/69	364	21	36.1	55,598,633
R.E. Ginna	U.S.A.	Rochester Gas & Electric	12/69	517	21	40.6	70,341,144
Point Beach 1	U.S.A.	Wisconsin Electric Power	12/70	524	19	43.7	71,103,730
Point Beach 2	U.S.A.	Wisconsin Electric Power	08/72	524	18	40.8	71,033,380
Surry 1	U.S.A.	Virginia Electric Power	08/72	858	11	41.2	80,515,188
Turkey Point 3	U.S.A.	Florida Power & Light	11/72	728	12	35.3	70,031,963
Surry 2	U.S.A.	Virginia Electric Power	02/73	858	11	37.1	78,433,849
Turkey Point 4	U.S.A.	Florida Power & Light	06/73	728	13	35.1	66,715,524
Indian Point 2	U.S.A.	Consolidated Edison	07/73	1022	11	37.5	85,451,889
Zion 1	U.S.A.	Commonwealth Edison	08/73	1085	12	37.1	99,148,442
Prairie Island 1	U.S.A.	Northern States Power	12/73	561	15	39.5	69,047,310
Zion 2	U.S.A.	Commonwealth Edison	12/73	1085	12	36.7	100,445,441
Kewaunee	U.S.A.	Wisconsin Public Service	03/74	563	--	34.1	68,022,331
Prairie Island 2	U.S.A.	Northern States Power	12/74	561	14	38.1	67,851,950
D. C. Cook 1	U.S.A.	Indiana & Michigan Electric	02/75	1089	12	38.6	103,180,040
Millstone 2 ^(b)	U.S.A.	Northeast Utilities	11/75	902	--	33.5	81,576,674
Trojan	U.S.A.	Portland General Electric	12/75	1178	14	38.7	83,377,749
Indian Point 3	U.S.A.	Power Authority of the State of New York	05/76	1013	8	37.2	74,498,545
Beaver Valley 1	U.S.A.	Duquesne Light	05/76	897	9	34.5	63,798,493
Salem 1	U.S.A.	Public Service Electric & Gas	12/76	1170	10	37.8	86,335,810
Kori 1	Korea	Korea Electric Power	07/77	595	10	33.7	47,674,678
Farley 1	U.S.A.	Alabama Electric Power	08/77	873	11	45.8	77,784,762
D.C. Cook 2	U.S.A.	Indiana & Michigan Electric	03/78	1133	8	35.3	82,026,030
North Anna 1	U.S.A.	Virginia Electric Power	04/78	990	9	39.0	74,215,987
North Anna 2	U.S.A.	Virginia Electric Power	08/80	990	8	42.0	70,094,952
Sequoyah 1	U.S.A.	Tennessee Valley Authority	10/80	1183	6	38.4	53,511,057
Salem 2	U.S.A.	Public Service Electric & Gas	05/81	1170	6	37.6	59,203,288
Farley 2	U.S.A.	Alabama Power Company	05/81	872	8	39.4	65,295,966
Mcguire 1	U.S.A.	Duke Power Company	09/81	1225	8	38.4	64,913,689

Table 2.6 (contd)

Reactor	Location	Owner	Date of First Electrical Power	Nominal MWe Gross	Current Cycle Number	Peak Region Avg. Burnup GWd/MTU ^(a)	Generation MWh(e) Cumulative
Krsko	Yugoslavia	Nuklearna Elektrarne, Krsko	10/81	664	9	37.5	42,122,234
Sequoyah 2	U.S.A.	Tennessee Valley Authority	12/81	1183	5	39.8	51,785,085
V.C. Summer 1	U.S.A.	South Carolina Electric & Gas	11/82	954	7	46.1	45,026,370
Kori 2	Korea	Korea Electric Power	04/83	650	6	33.1	39,209,480
Mcguire 2	U.S.A.	Duke Power Company	05/83	1225	7	38.8	63,219,104
Maanshan 1	Taiwan	Taiwan Power Company	05/84	951	6	34.7	37,146,204
Callaway 1	U.S.A.	Union Electric Company	10/84	1219	5	39.3	60,737,576
Diablo Canyon 1	U.S.A.	Pacific Gas & Electric	11/84	1137	5	39.8	51,024,300
Kori 3 (was KNU 5)	Korea	Korea Electric Power	01/85	950	5	35.1	42,245,031
Catawba 1	U.S.A.	Duke Power Company	01/85	1205	6	35.3	47,768,954
Byron 1	U.S.A.	Commonwealth Edison	02/85	1175	5	39.1	46,802,010
Maanshan 2	Taiwan	Taiwan Power Company	02/85	951	6	36.3	38,122,320
Wolf Creek	U.S.A.	Kansas Gas & Electric	06/85	1192	5	36.6	49,681,162
Diablo Canyon 2	U.S.A.	Pacific Gas & Electric	10/85	1164	5	37.8	45,762,700
Millstone 3	U.S.A.	Northeast Utilities	02/86	1209	4	35.8	40,915,918
Kori 4 (was KNU 6)	Korea	Korea Electric Power	04/86	950	4	33.8	38,243,668
Catawba 2	U.S.A.	Duke Power Company	05/86	1205	5	37.0	37,715,669
Yonggwang 1 (was KNU 7)	Korea	Korea Electric Power	08/86	996	4	35.7	37,274,098
Yonggwang 2 (was KNU 8)	Korea	Korea Electric Power	11/86	996	3	33.2	31,925,900
Shearon Harris	U.S.A.	Carolina Power & Light	01/87	960	4	35.8	29,267,014
Byron 2	U.S.A.	Commonwealth Edison	02/87	1175	3	34.4	33,536,834
Vogtle 1	U.S.A.	Georgia Power	03/87	1210	4	33.0	36,214,453
Braidwood 1	U.S.A.	Commonwealth Edison	07/87	1175	3	30.4	25,904,201
Beaver Valley 2	U.S.A.	Duquesne Light	08/87	888	3	35.2	24,743,381
South Texas 1	U.S.A.	Houston Lighting	04/88	1315	4	20.0	24,750,853
Braidwood 2	U.S.A.	Commonwealth Edison	05/88	1175	3	28.8	28,131,008
Vogtle 2	U.S.A.	Georgia Power	04/89	1210	2	31.8	22,407,151
South Texas 2	U.S.A.	Houston Lighting	05/89	1315	3	22.9	17,630,053
Seabrook	U.S.A.	PSC of New Hampshire	05/89	1194	2	17.1	11,368,098
Comanche Peak 1	U.S.A.	Texas Utilities Generating Company	04/90	1130	2	15.4	9,299,785

(a) Highest region average burnup after Cycle 1 startup.

(b) Non-Westinghouse plant with Westinghouse fuel.

Figure 2.8 Assembly-wise Burnup Distribution for W Zircaloy Active Rods as of December 31, 1991 (Permission to use this copyrighted material is granted by Westinghouse Electric Corporation)

Figure 2.9 Assembly-Wise Burnup Distribution for W Zircaloy Discharged Rods as of December 31, 1991 (Permission to use this copyrighted material is granted by Westinghouse Electric Corporation)

Figure 2.10 Assembly-Wise Burnup Distribution for all W Zircaloy Rods as of December 31, 1991 (Permission to use this copyrighted material is granted by Westinghouse Electric Corporation)

Figure 2.11 Distribution of Coolant Activities in W-Fueled Plants (Permission to use this copyrighted material is granted by Westinghouse Electric Corporation)

3.0 Fuel Design Changes and Summary of Fuel Surveillance Programs

"Fuel System Design," Section 4.2 of the NRC Standard Review Plan (NUREG-0800) requires that plans for testing, inspection, and surveillance of fuel be submitted and reviewed for each domestic nuclear power plant. The plans should include a pre-irradiation verification of a) the as-fabricated fuel, including cladding integrity, fuel system dimensions, fuel enrichment, burnable poison concentration, and absorber composition, and b) post-irradiation surveillance. The post-irradiation 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 during use. These plans are then referenced and/or summarized in the plant's safety analysis report (SAR). A supplementary fuel surveillance program appropriate for new fuel designs is noted in Stello (1974).

Provided below is a summary of current research and development programs and fuel surveillance programs for each of the four reporting fuel vendors. Since the trend in fuel design is to extend burnup and increase fuel reliability, fuel vendors have focused on these major areas in fuel research and development. Fuel manufacturers for PWRs have focused on extending fuel burnups; decreasing cladding corrosion, including burnable absorbers; and preventing fuel damage. Fuel manufacturers for BWRs have also focused on extending fuel burnup and preventing cladding corrosion while reducing failure due to pellet-cladding interaction. The various fuel vendors have developed different methods and approaches to these issues. The following sections illustrate the various programs and their results. Following the discussions of fuel design improvements there is a section on fuel surveillance programs where some of the newer and more significant programs are discussed for each vendor.

Information presented in the following subsections is derived from the annual vendor reports provided to the NRC. Additional information is taken from various reports and periodicals published during 1991. Typical

fuel assembly parameters associated with each vendor's fuel are provided in Appendix B.

3.1 ABB Combustion Engineering Nuclear Fuel

The following information is taken from a 1991 ABB CENF letter report (Toelle 1992) and from the proceedings of the International Topical Meeting on LWR Fuel Performance held in Avignon, France, in 1991 (Corsetti et al. 1991).

3.1.1 Research and Development

No new fuel design changes were specifically noted by ABB CENF (in Toelle 1992); however, ongoing research and development programs were discussed in Corsetti et al. (1991), from which the following information was taken.

3.1.1.1 Burnable Absorbers

ABB Combustion Engineering Nuclear Fuel experience with mixed erbia(Er_2O_3)/ UO_2 fuel used to supplement reactivity control has found that by using erbia in a significant fraction of the fuel rods in a core, it is possible to keep individual fuel rod concentrations of the burnable poison low. ABB Combustion Engineering Nuclear Fuel believes that erbia has some advantages over gadolinia and boron as a burnable poison. Erbium has a smaller cross section than does gadolinium, resulting in a smaller effect on the energy distribution of neutron flux and, thus, minimizing power peaking. The cross section for erbium is similar to that of boron, but it depletes more slowly and thus prevents the large power changes that would occur if boron were used to provide the same moderator temperature coefficient necessary for long fuel cycles. Two experimental programs have been developed for erbia testing by ABB CENF. One involves four lead fuel assemblies (LFAs)^(a) with

(a) Lead fuel assembly (ABB CENF), lead use assembly (GE), and lead test assembly (BWFC) all denote the same concept.

0.9 wt% Er_2O_3 and 3.4 wt% enriched UO_2 fuel, which were fabricated in 1989 and loaded into Calvert Cliffs Unit 2. The other program involves another four LFAs that contain fuel pins of 1.5 wt% Erbium and 3.65 wt% enriched UO_2 and were inserted in San Onofre Unit 2 in 1991 (Corsetti et al. 1991).

3.1.1.2 Fuel Design Improvements

Estimates from ABB CENF state that approximately 75% of all fuel damage is caused by debris-induced fretting (Tbelle 1992). Three new designs have been developed by ABB CENF to combat this problem. These designs focus on blocking and trapping debris in structures located below the active fuel region of the fuel rods. One design uses smaller flow holes in the bottom to block more debris. Another design uses long, solid end caps extended between the bottom nozzle and the bottom spacer grid. Any debris that is caught in this portion of the core is more likely to damage the end cap rather than the hollow, fuel-containing portions located above. The third design, referred to as GUARDIANTM, incorporates a special bottom grid to trap and retain debris during no-flow conditions. According to ABB CENF, the GUARDIANTM design blocks 93% of debris and retains 76% of the trapped debris during no-flow conditions. The effectiveness of the various design changes is summarized in Table 3.1.

3.1.2 Surveillance and Performance Programs

High burnup and extended cycle operation concerns are being addressed by a series of LFA programs (Tbelle

1992; Corsetti et al. 1991). Both standard and advanced fuel designs are being evaluated. The performance programs currently in progress will provide hot cell evaluation of fuel and cladding with peak local burnup approaching 70 GWd/MTU. These and other programs are summarized in Table 3.5.

3.2 General Electric Company

The following GE design and surveillance information was obtained from a 1991 GE letter report (Klapproth 1992) and from the proceedings of the International Topical Meeting on LWR Fuel Performance held in Avignon, France, in 1991 (Baily et al. 1991).

3.2.1 Research and Development

Over the years GE has made a variety of design modifications to improve fuel corrosion resistance and overall fuel performance. Modified features include water rod configuration, spacer and upper tie plates, cladding surface treatment (involving material and heat treatment), axial zoning of gadolinia, fuel rod helium prepressurization, pellet dimensions, and pellet density. Pellet-cladding interaction and crud-induced localized corrosion (CILC) are the primary cladding failure mechanisms that have affected GE fuel performance in recent years. Product improvements have been developed with the goal of eliminating these two failure mechanisms, as described below.

Table 3.1 Flow Test Results for Different ABB CENF Debris-Resistant Fuel Designs
(Copyright permission to use table is granted by ABB CENF)

Device	% Debris Blocked by Device	% Debris Caught Adjacent to Cladding	% Debris Retained During Zero Flow
Standard Bottom Nozzle	14	81	Not Measured
Small Flow Hole Nozzle	52	38	16
Long End Cap	81	15	65
GUARDIAN TM	93	7	76

3.2.1.1 Corrosion Resistance

Crud-induced localized corrosion, identified in 1979 as a failure mechanism, occurs under very specific conditions in plants with copper alloy condenser tubes and filter demineralizer condensate cleanup systems. General Electric Company (in Baily et al. 1991) estimates that approximately 80% of all GE8X8 fuel rod failures are a result of CILC. A reproducible out-of-reactor test was developed to confront this problem, and subsequent manufacturing processes have been developed and implemented. The extra corrosion protection has been added to the production of all GE fuel and provides more protection against corrosion in reactors that are not CILC susceptible. Various LUAs, discussed in Section 3.2.2, have been tested to determine the effectiveness of the manufacturing improvements.

3.2.1.2 Pellet-Cladding Interaction Resistance

Pellet-cladding interaction was first addressed by GE in 1983 with the introduction of the barrier cladding fuel design. This fuel design incorporates a pure zirconium lining on the inside of the Zircaloy-2 cladding. This cladding design alleviates stresses caused by the fuel expanding faster than the cladding during power ramping. General Electric Company (in Baily 1991) estimates that 14% of all GE8X8 fuel failures occurred due to pellet-cladding interaction (PCI) in non-barrier fuel. As of December 1991, over 1.2 million GE barrier rods have operated for at least one cycle with no observed failures due to PCI (Klapproth 1992).

3.2.2 Surveillance and Performance Programs

A summary of the GE LUA surveillance program is provided in Table 3.5. A more detailed description of GE LUA programs are presented below.

- 1983 LUAs - Four LUAs were loaded into a BWR/4 in 1983 to test improved spacer and upper tie plate designs, axial zoning of gadolinia, variations in cladding thickness, pellet dimensions, and fuel rod helium prepressurization. Poolside examination after one cycle in August 1985, and after two cycles in November 1987, showed characteristics of normal operation. The LUAs were returned to service in December 1989. The third poolside examination was completed in October 1991 and showed characteristics typical of three cycles of normal operation.
- 1984 LUAs - Five LUAs were loaded into a BWR/4 in 1985 to test water rod configuration; improved spacer and upper tie plate designs; cladding surface treatment; axial zoning of gadolinia; and variations in fuel rod helium prepressurization, pellet dimensions, and pellet density. Four of the LUAs were loaded in central core positions, and one was placed near the edge of the core. Poolside examinations were made after one cycle in April 1987 and after two cycles in October 1988, showing characteristics of normal operation. The third cycle of operation ended in July 1990. The four central bundles were exposed to about 40 GWd/MTU. The edge LUA achieved about 25 GWd/MTU and was re-inserted into the core in September 1990 (Klapproth 1992). Additional inspections are planned in late 1992.
- 1987 LUAs - Four LUAs were loaded into a BWR/4 in 1987. These fuel assemblies represent lead- use GE8X8NB fuel. Poolside examinations were completed in October 1988 after one cycle of operation and in March 1990 after a second cycle of operation and showed normal characteristics of operation after both examinations. In October 1991, the third poolside examination was conducted, and after three cycles of operation at bundle average exposures of about 32 GWd/MTU, the rods showed characteristics of normal operation. The LUAs were re-inserted in November 1991 for continued operation (Klapproth 1992).
- 1989 LUAs - A group of four GE8X8NB LUAs were loaded into a BWR/4 in 1989. Examination of these bundles was completed in February 1991 after the first cycle of operation. The bundles revealed characteristics of normal operation and were re-inserted in April 1991 (Klapproth 1992).
- Cladding Corrosion Performance LUAs - Two programs have been developed in this area. Six LUAs were loaded into a BWR/4 early in 1988, and six were loaded into another BWR/4 in late 1988. The LUAs were designed to test cladding material process variables such as cladding material, heat treatment, and surface conditioning. The Hatch reactors

have historically exhibited highly variable cladding corrosion performance. Examination of these LUAs in March and October of 1991, after bundle average exposures up to 24 GWd/MTU, showed excellent corrosion resistance along the entire length of the fuel rods. In June and November of 1991 the LUAs were re-inserted for continued operation (Klapproth 1992).

- GE8X8NB-1 Channel LUAs - Four LUAs were loaded into BWR/4 in 1988. In October 1991, after three cycles of operation at bundle average exposures of about 25 GWd/MTU, the bundles showed characteristics of normal operation. The bundles were re-inserted for continued operation and the next inspection is scheduled in early 1993 (Klapproth 1992).
- GE11 LUAs - Three reactors (two BWR/4s and one BWR/5) were loaded with four LUAs each. The first cycle of operation was completed in 1991 with bundle average exposures of about 7-13 GWd/MTU. In April 1991, poolside examinations at one of these reactors revealed characteristics of normal operation, and the bundles were re-inserted for continued operation. Three more BWR/4s were loaded with four GE11 LUAs each, in 1991. Poolside examinations of these bundles are planned to begin in 1992 (Klapproth 1992).

3.3 Siemens Power Corporation

The following information was obtained from an SPC annual report (Copeland 1993) and from the proceedings of the International Topical Meeting on LWR Fuel Performance held in Avignon, France, in 1991 (Ritter 1991).

3.3.1 Research and Development

Siemens Power Corporation manufactures fuel for both PWR and BWR type reactors. Although SPC (in Copeland 1993) did not mention any new fuel design changes in their 1991 report, several developments for each type were in progress and were discussed during the 1991 fuel performance meeting in Avignon, France (Ritter et al. 1991).

3.3.1.1 Corrosion Resistance

Siemens Power Corporation has approached the corrosion issue by beta-quenching their fuel cladding. This involves rapidly quenching the hollow fuel rod tube from the beta phase region before the last two size reduction steps. This improves corrosion performance in plants susceptible to CILC and also eliminates water-side corrosion as a fuel burnup limiting factor. Siemens Power Corporation beta-quenched cladding has been used in seven PWRs and six BWRs. Exposures as high as 48.9 GWd/MTU have been reached with beta-quenched cladding, which has exhibited superior corrosion performance in BWRs (Copeland 1993).

Siemens Power Corporation (in Ritter et al. 1991) is also actively pursuing alloy chemistry studies to improve Zry-4 corrosion behavior and developing new alloys that may have performance superior to Zry-4 for PWRs.

3.3.1.2 Burnable Absorbers

The SPC research with burnable poisons has been conducted with gadolinia. Siemens Power Corporation obtained PWR burnups of 30 GWd/MTU using 10 wt% Gd_2O_3 and 50 GWd/MTU using 8 wt% Gd_2O_3 . No difference has been found in the operating characteristics of this fuel when compared with standard UO_2 fuel rods (Ritter et al. 1991).

3.3.1.3 Fuel Design Improvements

The SPC 9x9 fuel type for BWRs has been developed with several different configurations of water rods that are available. Current types have one, two, or five water rods, and a nine water rod type is undergoing testing. The various configurations produce lower linear heat generation rates to reduce fission gas release and the likelihood of damage due to PCI. Siemens Power Corporation has used axial zoning of gadolinia to improve uranium utilization and cold shutdown margins (in Ritter et al. 1991).

There have been several developments in SPC's 17x17 fuel type for PWRs to improve fuel performance and prevent some types of fuel failures. Baffle jetting is a

problem that occurs in PWRs when water flow causes vibration of fuel rods and subsequent cladding failure. Low cost fuel rod clips have been developed by SPC to prevent baffle jetting and, according to SPC, essentially eliminate that problem. Another improvement has been the development of high thermal performance spacers to improve heat transfer and lower cladding temperatures. This improves the fuel's departure from the nucleate boiling margin, which is necessary to prevent undesirable steam buildup caused by boiling on the surface of the cladding (Ritter et al. 1991).

A major area of development at SPC has centered around reducing damage to fuel from debris. Siemens Power Corporation (in Ritter et al. 1991) has tested two design changes for lower grid plates that will reduce fuel susceptibility to damage from debris.

Since most fuel failures were found to occur near the bottom of the fuel rod, SPC developed extended-length solid end caps in an attempt to solve debris fretting problems. The solid end cap is then exposed to the highest risk area for fretting, thus protecting the fuel containing portion of the fuel rod (Ritter et al. 1991). The first design incorporates small flow holes to trap particles. A large number of 6mm diameter holes (as opposed to the previous 11 mm diameter hole size) was found to be 15% more efficient at trapping debris (67% compared to 52%). The small holes helped to filter out large pieces of debris but were not as effective with smaller particles (Ritter et al. 1991). Siemens Power Corporation has also developed a new lower tie plate that was determined to be 97% efficient at blocking all major types of debris. The grid does not increase hydraulic resistance and is better at trapping small pieces of debris before they reach the fuel elements. The tie plate utilizes a curved grid to eliminate straight flow paths, is only 2 mm wide at the curved portion of the grid, and does not increase hydraulic resistance. This design has proven to be effective in blocking all major types of debris from entering the fuel assembly (Ritter et al. 1991). The effectiveness of the different designs to trap different types of debris is shown in Table 3.2.

3.3.2 Surveillance and Performance Programs

The SPC surveillance programs are summarized in Table 3.5.

3.4 Westinghouse Electric Corporation

The following information was obtained from a 1991 W annual report (Slagle 1993) and from the proceedings of the International Topical Meeting on LWR Fuel Performance held in Avignon, France, in 1991 (Balfour et al. 1991).

3.4.1 Research and Development

No new fuel design changes were specifically noted in the 1991 W annual report (Slagle 1993). Ongoing developments are presented below.

3.4.1.1 Debris Filter Bottom Nozzle

Westinghouse Electric Corporation has implemented several design improvements over the past years. The problem of fuel damage caused by debris was addressed with the advent of the debris filter bottom nozzle (DFBN). The DFBN has smaller flow holes through the bottom nozzle to block debris more efficiently but maintains the same pressure drop as earlier fuel designs. In 1991 the DFBN was used in at least one region of fuel in 44 of the 57 fueled reactors (Slagle 1993).

3.4.1.2 ZIRLO™

Westinghouse Electric Corporation has developed a new zirconium based alloy, known as ZIRLO™, to enhance fuel reliability and achieve extended burnup. This alloy provides significant improvements in fuel rods; guide thimble tubes; instrumentation tubes; mid-span grid and IFM corrosion resistance; and dimensional stability under irradiation (Slagle 1993).

3.4.2 Surveillance and Performance Programs

Westinghouse Electric Corporation fuel surveillance programs are summarized in Table 3.5. Additional information about some of these programs is presented below.

- EPRI 1973-1980 - Assembly burnups of 39-55 GWd/MTU were obtained in this program. Four assemblies were discharged from Zion Unit 1 (Cycle 6) with average burnups of 55 GWd/MTU (Slagle 1993).

Table 3.2. Comparison of SPC Standard and Debris Resistant PWR Fuel Assembly Designs
(Permission to reproduce this material is granted by Siemens Power Corporation)

Fuel Type	Debris Type and % Captured				Overall Efficiency
	Electrical Connectors ^(a)	Machining Chips ^(b)	Short Pins ^(c)	Long Pins ^(d)	
Standard Design (11 mm Flow Holes)	42	75	10	80	52
Small Hole Grid (6 mm Flow Holes)	83	74	63	47	67
Curved Blade Grid	100	91	96	100	97

(a) Ring and butt stake-on connectors 15-18 mm long.

(b) Milling and drilling chips 7 to 45 mm long.

(c) Spiral-wound gasket pieces and pins 12 to 50 mm long and ≥ 1.0 mm diameter.

(d) Spiral-wound gasket pieces, drill bits and pins 40 to 100 mm long and ≥ 1.0 mm diameter.

- EPRI 1983-1989 - Eight fuel assemblies completed their fourth 18-month cycle. Four of the assemblies were in relatively high power positions at North Anna Unit 1 (cycle 7) and attained assembly-average burnups of 58 GWd/MTU (Slagle 1993).
- ZIRLO™ Demo Assemblies - Two assemblies began operations in North Anna Unit 1 in June 1987. Other plants that have been installed with ZIRLO™ demonstration assemblies are listed in Table 3.3. In October 1991, V. C. Summer substituted a full region with ZIRLO™ clad fuel rods (Slagle 1993).

Table 3.3 Reactors using ZIRLO™ (Copyright permission to reproduce this material is granted by Westinghouse Electric Corporation)

Plant	Number of assemblies
North Anna 1	2
Vogtle 1	2
Three Mile Island 1	2
Almaraz 2	4
Ringhals 3	4
V.C. Summer	Full Region

3.5 Babcock & Wilcox Fuel Company

Babcock & Wilcox Fuel Company (in Taylor 1992) declined to provide information for fuel design changes and fuel surveillance programs and, therefore, no information is provided in these sections for 1991.

3.6 Electric Power Research Institute Programs

The current direction of the EPRI fuel surveillance program is given in a 1991 paper (Yang et al. 1991) that describes EPRI's current projects and the trends that EPRI predicts for fuel research and development. The main objectives of the EPRI programs are listed below:

- Solving current fuel failure mechanisms.
- Determining "unknown" failure mechanisms.
- More closely monitoring system parameters to determine effects on fuel operations and to identify fuel problems early in their development.
- Researching further for extended burnup operations (i.e., modeling and data compilation).
- Making recommendations to industry on how to help reduce the problems associated with failed fuel.

Other topics discussed by EPRI (in Yang et al. 1991) are the goal of zero-failures, which problems need to be further addressed in order to be able to realistically approach the "no-leaker" limit, and the development of the Cost of Fuel Failure Evaluation Code (COFFEC). This code is used to evaluate the cost impact on the U.S. nuclear industry that is incurred by fuel failures. The COFFEC study of the 1988-1989 period indicated that most of the incurred industry-wide cost resulted from fewer than 10% of the plants.

Additional information on EPRI research programs has been published over the past several years in numerous documents, which are listed in Table 3.4.

Table 3.4 EPRI Fuel Research and Development Publications

Title	Author
<i>Guidelines for Improving Fuel Reliability</i>	EPRI 1989a
<i>Methodology Estimating Number of Failed Fuel Rods and Defect Size</i>	EPRI 1989b
<i>EPRI Research and Development Program Plan, 1987-1989</i>	EPRI 1987a
<i>Review of Pacific Northwest Laboratories' Test Program Results</i>	EPRI 1987b
<i>Advances in Light Water Reactor Fuels</i>	Franklin 1987
<i>Hydrogen Water Chemistry for BWRs</i>	EPRI 1986a
<i>Lifetime of PWR Silver-Indium-Cadmium Control Rods</i>	EPRI 1986b
<i>Collection and Formatting of Data on Reactor Coolant Activity and Fuel Rod Failures</i>	EPRI 1986c
<i>Utility Experience with the BWR Power Shape Monitoring System</i>	Emrich 1986
<i>Demonstration of 9x9 Assemblies for BWRs</i>	EPRI 1984a
<i>Comparison of Advanced BWR Fuel Designs with Current Standard Designs</i>	EPRI 1984b
<i>LWR Core Materials Program: Progress in 1983-1984</i>	EPRI 1985
<i>Phenomena Associated with Extending Fuel Burnup</i>	Franklin 1982a
<i>The Schedule for Extending Fuel Burnup</i>	Franklin 1982b
<i>The EPRI LWR Fuel Surveillance Program</i>	Roberts 1982
<i>LWR Core Materials Program: Progress in 1981-1982</i>	EPRI 1983

Table 3.5 Major Fuel Surveillance Programs: Status Through 1991

Vendor	Fuel Type ^a	Power Plant	Planned (Completed) Operating Cycles	Scheduled Completion of Program	Inspections to Date
Siemens Power Corporation	14x14	R. E. Ginna	5(5)	1990	3
	17x17	Blayais 3	4(4)	1990	3
	8x8	WNP 2	4(3)	1991	3
	14x14	Calvert Cliffs	3(0)	1993	0
	15x15	Palisades	3(1)	1993	1
	9x9	Hatch 2	3(1)	1994	1
	9x9	Hatch 1	3(1)	1995	1
Babcock & Wilcox Fuel Company ^(b)	15x15 ^(c)	Rancho Seco	3	1990	2
	15x15 ^(d)	Oconee 1	4	1990	3
	17x17 ^(e)	McGuire 1	3	1991	2
ABB Combustion Engineering Nuclear Fuel	14x14 ^(f)	Calvert Cliffs 1	5(5), Pt 2 ^(k)	1993	5
	14x14 ^(g)	Calvert Cliffs 2	3(0)	1997	0
	16x16 ^(h)	Arkansas 2	5(5) ^(k)	1992	5
	16x16 ⁽ⁱ⁾	Palo Verde 1	3(2)	1994	2
	16x16 ⁽ⁱ⁾	Palo Verde 3	3(0)	1997	0
	16x16 ^(g)	San Onofre 2	2(0)	1995	0
	14x14 ^(j)	Maine Yankee	12(12)	1991	3
General Electric Company	Barrier LUAs	BWR/3	--	--	5
	1983 LUAs	BWR/4	--	--	3
	1984 LUAs	BWR/4	--	--	3
	1987 LUAs	BWR/4	--	--	3
	Corrosion Performance	BWR/4	--	--	2
	GE 8x8NB	BWR/4	--	--	3
	Corrosion performance	BWR/4	--	--	2
	1987 LUAs	BWR/4	--	--	1
	GE11 LUAs	BWR/4	--	--	1
	GE11 LUAs	BWR/4	--	--	1
	GE11 LUAs	BWR/5	--	--	1
	GE11 LUAs	BWR/4	--	--	0
	GE11 LUAs	BWR/4	--	--	0
	GE11 LUAs	BWR/4	--	--	0

Table 3.5 (contd)

Vendor	Fuel Type ^a	Power Plant	Planned (Completed) Operating Cycles	Scheduled Completion of Program	Inspections to Date
Westinghouse Electric Corporation	IFBA Demo Fuel Rods ^(l)	Turkey Point 3	(2)	--	--
	IFBA Demo Fuel Rods ^(m)	Turkey Point 4	(2)	--	--
	IFM Demo Assembly ⁽ⁿ⁾	McGuire 1	(2)	--	--
	DFBN Assembly ^(o)	3 Plants	--	--	--
	Zircaloy-Clad Fuel Rod Assembly ^(p)	North Anna 1	3(1) ^(r)	--	--
	MO ₂ ^(q)	R. E. Ginna	4(4) ^(s)	--	--

- (a) LTA = lead test assembly, LUA = lead use assembly, MO₂ = mixed oxide (UO₂-PuO₂) fuel, R = retrofit fuel design, D = demonstration, optimized fuel assembly (OFA)-Demo = Demonstration optimized fuel assembly, IFBA = integral fuel burnable absorber, IFM = intermediate flow mixing, FPIP = Fuel Performance Improvement Program, DFBN = debris filter bottom nozzle, ZIRLO = an advanced zirconium alloy cladding that contains niobium.
- (b) BWFC data through 1989.
- (c) Current-design assemblies containing axially-blanketed fuel columns.
- (d) Gadolinia LTAs of an advanced, extended-burnup design.
- (e) Four 17x17 LFAs (Mark-BW LA).
- (f) Standard and advanced fuel design LTAs.
- (g) Burnable poison irradiation program.
- (h) Standard and advanced fuel design, high-burnup program.
- (i) Advanced cladding designs.
- (j) Hot cell exam of high exposure control element assembly.
- (k) Hot cell examination of high burnup fuel yet to be performed.
- (l) The 4 IFBA rods were monitored during irradiation by in-core instrumentation.
- (m) There were 28 IFBA rods in each of 4 demonstration assemblies, which allowed removal of some of the rods for postirradiation examination.
- (n) One characterized IFM spacer grid demonstration assembly.
- (o) Three fuel assemblies with DFBNs.
- (p) Two demonstration fuel assemblies with ZIRLO-clad fuel rods began irradiation in June 1987. ZIRLO is an advanced zirconium alloy that contains niobium. ZIRLO is a trademark of Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.
- (q) Four assemblies with W mixed oxide fuel rods were involved. The mixed oxide (UO₂-PuO) fuel rods for R. E. Ginna were manufactured by W, but their irradiation was not part of a W development program.
- (r) The fuel rods attained a burnup of over 21.0 GWd/MTU in their first cycle, which was completed during February 1989. The rods are expected to surpass a burnup of 57.0 GWd/MTU at the completion of a third irradiation cycle.
- (s) The four assemblies were irradiated for the fourth cycle (i.e., they were in the Cycle 11-14 cores) and were discharged. Average burnup was 38.5 GWd/MTU.

4.0 Summary of High-Burnup Fuel Experience

The most recent trend in fuel design has been to extend burnup and increase fuel reliability. Each of the fuel vendors has approached this design effort differently. The following section illustrates the industry's high-burnup experience and related research and development programs.

4.1 Industry Overview

In 1978, DOE established an extended burnup program to demonstrate the technology necessary to extend discharge burnup levels to 45 GWd/MTU for BWRs and 50 GWd/MTU for PWRs (Lang 1986). Although reduction in fuel costs to utilities was the initial objective of the program, the industry has subsequently added other reasons for extending fuel burnup: the reduction of generated spent fuel by LWRs (i.e., waste minimization) and improved plant capacity/availability factors.

The U.S. Department of Energy has estimated that if target burnups of 50 GWd/MTU for PWRs and 45 GWd/MTU for BWRs were reached in all U.S. reactors, the volume of spent fuel generated annually would be reduced by 40% (Nuclear Fuel 1984). It is estimated that 26 commercial reactors will be required to expand their spent fuel storage capacity by the year 2000 (SR/CNEAF/92-01). An industry wide trend toward higher burnups may alleviate some spent fuel storage problems until a permanent and safe disposal facility can be constructed as required by the Nuclear Waste Policy Act of 1982.

The 1978 extended burnup program has resulted in newer fuel designs and improved manufacturing processes that have reduced many fuel failure mechanisms which limited fuel performance in the past. The program's burnup goals were partially achieved in 1982 by the discharge of two BWR assemblies at approximately 45 GWd/MTU and five PWR assemblies at approximately 50 GWd/MTU (SR/CNEAF/92-01). The extended burnup goals were further achieved in 1985 by the NRC's review and approval of vendor topical reports that addressed extended burnup experience, analytical methodology and fuel testing. While several utilities have achieved some extended burnup

experience on whole fuel batches, the industry on average has yet to achieve high burnups. However, longer fuel cycles (18-24 months) are beginning to be seen in the industry. The longer fuel cycles result in higher capacity/availability factors because the plant has less outage time per year of operation.

In 1987, EPRI set a goal of 60 GWd/MTU assembly-average burnup to be reached by 1997 (EPRI 1987). These goals are being spurred by the utilities' desires to achieve longer fuel cycles (18-24 months). However, the lack of high burnup data regarding cladding corrosion, ductility, fuel thermal conductivity, and the behavior of fuel and cladding during power transients and accidents is a significant safety concern to the NRC as the industry seeks to go beyond the 60 GWd/MTU rod average burnup level.

4.2 Summary of 1991 High Burnup Experience

The highest annual average burnups for all BWR and PWR discharged assemblies were reached in 1991 (27.8 and 35.0 GWd/MTU, respectively). The number of utilities planning to incorporate fuel specifically designed for extended burnup in current and future cores is rapidly increasing. The highest burnups achieved in 1991 by four of the nuclear fuel vendors are presented in the following subsections and summarized in Table 4.1.

A historic perspective of domestic BWR and PWR fuel burnup experience is given in Figures 4.1 and 4.2, respectively (a majority of the data for Figures 4.1 and 4.2 were obtained from Turner et al. (1979)). These figures are updated annually by PNL using information provided to the NRC in annual fuel vendor reports. Discharged fuel assembly burnup statistics for domestic BWRs and PWRs are summarized in Table 4.2 and illustrated in Figure 4.3. The statistics were obtained from U.S. DOE Energy Information Administration report SR/CNEAF/93-01. As can be seen from this data, there is indeed an increasing number of fuel assemblies being irradiated at higher burnup levels.

Summary of High-Burnup Fuel Experience

Table 4.1 Highest Burnup Fuel Experience by Vendor

Vendor	Plant or Test	Type	Burnup (GWd/MTU)	Comment	References
ABB CENF	ANO-2	PWR	46.5	Batch Average	Toelle, 1992
	Calvert Cliffs-2	PWR	43.0	Batch Average	
	St. Lucie-2	PWR	41.5	Batch Average	
GE		BWR	>45.0	Bundle Average	Klapproth, 1991
		BWR	60.0	Peak Pellet Exposure	
SNP	R. E. Ginna	PWR	52.1	Assembly Average	Copeland, 1992
	Untweser	PWR	51.8	Assembly Average	
	Big Rock Point	BWR	45.1	Assembly Average	
	Gundremmingen-C	BWR	40.1	Assembly Average	
<u>W</u>	Zion-1,-2	PWR	55.0	4 Assemblies Average	Slagle, 1992
	North Anna-1	PWR	58.0	4 Assemblies Average	
			>58.0	Lead Assembly Average	
BWFC	Information was not provided by BWFC

(a) EXTENDED BURNUP (ASSEMBLY AVERAGED) GENERIC APPROVALS BY NRC, 1985
 (b) 1984 PERFORMANCE SPECIFICATION

Figure 4.1 Domestic BWR Fuel Burnup Experience

Figure 4.2 Domestic PWR Fuel Burnup Experience

Table 4.2 Spent Fuel Burnup (Assemblies) (SR/CNEAF/93-01)

Burnup (GWd/MTU)		0-5	5-10	10-15	15-20	20-25	25-30	30-35	35-40	40-45	45-50	50-60	Total	Average
Boiling Water Reactors														
1991 Alone	#	96	132	1	60	205	1,295	1,500	68	0	0	0	3,357	
	%	2.8	3.9	0.030	1.8	6.1	38.6	44.7	2.0	0.0	0.0	0.0	100.0	27.8
Since 1968	#	2,502	2,481	3,995	7,294	12,042	15,634	4,608	108	7	0	0	48,671	
	%	5.1	5.1	8.2	15.0	24.7	32.1	9.5	0.22	0.014	0.0	0.0	100.0	21.8
Pressurized Water Reactors														
1991 Alone	#	0	44	115	3	161	164	378	1,398	573	149	8	2,993	
	%	0.0	1.4	3.8	0.10	5.4	5.5	12.6	46.7	19.1	5.0	0.27	100.0	35.0
Since 1968	#	49	184	1,286	3,217	2,305	6,828	10,404	8,113	2,202	219	25	34,929	
	%	0.14	0.53	3.7	9.2	6.6	19.5	29.8	23.2	6.3	0.063	0.072	100.0	30.0

Figure 4.3 Spent Fuel Burnup - Comparison of All Fuel Discharged Since 1968 with Fuel Discharged in 1991

4.2.1 ABB Combustion Engineering Nuclear Fuel

As of December 31, 1991, ABB CENF had 322 fuel assemblies (69,440 fuel rods) that had achieved burnups greater than 40 GWd/MTU still in-core. Burnups in excess of 40 GWd/MTU were achieved in 780 discharged fuel assemblies with 156,506 fuel rods. Burnups in excess of 44 GWd/MTU were achieved in 21 discharged fuel assemblies with 4956 fuel rods. Three discharged assemblies (579 fuel rods) had burnups in excess of 48 GWd/MTU and 5 discharged assemblies (878 fuel rods) had burnups in excess of 50 GWd/MTU (Toelle 1992).

4.2.2 General Electric Company

General Electric Company has had successful commercial reactor operation of fuel bundles to greater than 45 GWd/MTU bundle average exposure, with peak pellet average exposure reaching approximately 60 GWd/MTU (Klapproth 1992).

4.2.3 Siemens Power Corporation

The highest exposures reached by SPC fuel assemblies to date are 52.1 GWd/MTU in Rochester Gas and Electric Corporation's R. E. Ginna (PWR) and 45.1 GWd/MTU in Consumers Power Company's Big Rock Point (BWR). The highest exposure reached by a BWR 9x9 assembly is 40.1 GWd/MTU at Gundremmingen-C in Germany. Individual BWR peak exposures reached 48.9 GWd/MTU. The new 17x17 fuel assemblies, designed to reach higher burnups, reached 51.8 GWd/MTU at Unterweser in Germany (Copeland 1993).

4.2.4 Westinghouse Electric Corporation

Burnups in excess of 40 GWd/MTU were achieved in 1949 assemblies with approximately 450,000 fuel rods. Burnups in excess of 44 GWd/MTU were achieved in 444 assemblies with approximately 105,000 fuel rods. Four assemblies irradiated in Commonwealth Edison's Zion Units 1 and 2 reached an extended burnup of 55 GWd/MTU with a peak fuel rod burnup of 60 GWd/MTU. Four assemblies irradiated in Virginia Power's North Anna Unit 1 reached an average burnup of approximately 58 GWd/MTU, with the lead assembly

burnup approximately 58.4 GWd/MTU and the lead fuel rod average burnup greater than 60 GWd/MTU (Slagle 1993).

4.2.5 Babcock & Wilcox Fuel Company

Babcock & Wilcox Fuel Company (in Taylor 1992) declined to provide high-burnup fuel experience information and, therefore, no information is provided in these sections for calendar year 1991.

4.3 High Burnup Effects Program

The High Burnup Effects Program (HBEP) was an international group sponsored program managed by Battelle, Pacific Northwest Laboratories. The sponsors included fuel manufacturers, utility representatives, and government entities from Europe, Japan, and the United States. The principal objective of the HBEP was to obtain well characterized data for typical LWR fuel irradiated to high burnup levels (Barner et al. 1990).

4.3.1 Review of Data

The HBEP was organized into three tasks (Barner et al. 1990):

- Task 1 - High Burnup Effects Evaluations. The objective of the work performed under this task was to compile and assess the publicly available information on fission gas release from high burnup UO_2 fuel.
- Task 2 - Fission Gas Sampling. The objective of the work performed under this task was to obtain fission gas release data from existing commercial fuel rods with peak-pellet burnup levels up to 55 GWd/MTU. This included acquiring existing irradiated fuel rods, continuing the irradiation of some of these rods to higher burnup levels, performing power-bumping on selected rods, and then performing extensive post-irradiation examinations (PIE). Data were acquired from 45 rods with rod-average burnup levels ranging from 22 to 47 GWd/MTU. A number of techniques were employed in the PIE of the irradiated fuel rods to obtain detailed data on both macro and micro scales. These techniques included cladding metrology, axial gamma scanning, rod

Summary of High-Burnup Fuel Experience

puncture and gas analysis, ^{148}Nd -based burnup analysis, whole pellet fuel density, electron probe micro-analysis, x-ray fluorescence, and scanning electron microscopy.

- Task 3 - Parameter Effects Study. The objective of the work performed under this task was to obtain well characterized data from fuel rods specifically built, characterized, and irradiated for the HBEP to peak pellet burnup levels in excess of 80 GWd/MTU. Extensive PIE was performed on 37 rods with rod-average burnup values ranging from 25 to 69 GWd/MTU.

4.3.2 Technical Findings

The following conclusions resulted from the HBEP (Barner et al. 1990):

- It was confirmed that the dependency of FGR upon design and irradiation history parameters, known to be applicable at lower burnup levels, was also applicable at higher burnup levels; specifically, for rods of similar design, less FGR was found in rods operated at lower linear heat generation rate (LHGR) levels than rods operated at high LHGR levels. Since fuel temperatures are dependent upon the LHGR, FGR can be related to fuel LHGR. In general, with increasing fuel temperature, FGR will increase. Also, rods with high internal helium gas pressure had less FGR than rods with low helium pressures. This result is principally an effect of released fission gas having a smaller diluting effect on the fill gas and thus lessening the effect on fuel temperatures.
- The net end of life transient FGR, due to power bumping, was found to be dependent on four factors: terminal LHGR level, burnup level, length of the hold period, and the pre-transient grain boundary bubble network.
- Depletion of matrix xenon and precipitation of bubbles at grain interfaces are thermally-driven FGR mechanisms. This result confirmed the burnup and time dependence of grain boundary bubble precipitation.
- At pellet-edge burnup levels greater than 60 GWd/MTU, it was found that there was a loss of definable grain structure and the formation of a high volume of porosity at the periphery of the pellet. There is a transfer of fission gas from the matrix to the pores in this high porosity rim region, which might lead to increased FGR from the fuel to the open volume of the rod. This rim effect seems to be the only observed high burnup effect.

During the course of the HBEP, 82 fuel rods from a variety of sources were characterized, irradiated, and examined in detail after irradiation. The study of fission gas release at high burnup levels concluded that no significant enhancement of FGR at the higher burnup levels was observed for the examined rods. The rim effect, an as yet unquantified contributor to athermal FGR, was concluded to be the one truly high burnup effect (Barner et al. 1990). A full understanding of the rim effect has not yet emerged, and this is an area for potential research. For further information on the High Burnup Effects Program see Barner et al. (1990), Barner et al. (1993), and Cunningham et al. (1992).

5.0 Problem Areas Observed During 1991

This section contains brief summaries of problems/events resulting in the degradation of fuel performance or having the potential to degrade fuel performance. All summarized problems/events pertain to calendar year 1991 and were obtained from open literature or NRC Licensee Event Report (LER) Compilation Reports. The events reported from LER compilation reports are reprinted in their entirety. In general, only problems/events associated with domestic plants are included; however, the authors have presented information regarding problems/events occurring at foreign plants where appropriate. A reference is provided for each summary directing the reader to original material for further information. Also included in this section are items of general significance that related to fuel performance not covered in the previous sections.

Section 5 is divided into the following categories: fuel; fuel handling; spent fuel; control rod; core or coolant; personnel or equipment; and miscellaneous. In general, events are listed alphabetically by reactor or organization within each category. Because some problems/events are the result of personnel errors or equipment problems, reference lists are included in Tables 5.1 and 5.2 as a cross reference for the specific events. Appendix C is provided to allow the reader to quickly identify all reported events associated with a specific commercial power plant. Also provided in Appendix C is a brief summary of the total number of events (per category) reported during the current and previous years.

5.1 Fuel Oriented

This section includes information on fuel damage events resulting from all of the potential fuel failure mechanisms such as swelling, oxidation, and corrosion of fuel cladding. Events pertaining to other fuel assembly related problems such as thinning of in-core instrumentation, iodine spiking incidents, and miscellaneous or unknown fuel problems are also included.

5.1.1 Fuel Failure or Damage

Fuel assemblies or fuel rods that fail by being damaged before, during, or after operation are generally discussed in this section. Although no specific events were identified, one item of interest was reported regarding barrier cladding performance.

Davies and Potts (1991) claim that in-reactor testing of defective GE barrier and non-barrier fuel rods reveal that, while the barrier fuel rod zirconium liner corrodes in the defected condition, there is no difference in the post-defect behavior of the barrier fuel as compared to the non-barrier fuel. It is also claimed that this has been confirmed by GE's commercial reactor fuel irradiation experience. However, ABB Atom observed (in Jonsson et al. 1991) a significant difference in post defect behavior between barrier fuel and non-barrier fuel. The barrier fuel was observed to have experienced extensive secondary hydriding with several longitudinal cracks along the length of the rod following the formation of a small defect due to debris fretting. It was concluded that the large amount of secondary hydriding was enhanced by the presence of the zirconium sponge barrier. The reason for the conflicting differences in performance between GE and ABB Atom barrier rods is not known at this time.

5.1.2 Flow-Induced Fretting and Wear

Two principal causes of flow-induced fretting and wear exist. Debris induced-fretting results from loose debris in the primary coolant becoming trapped against the fuel cladding causing damage. Vibrational flow-induced fretting results from coolant flow inducing fuel rods to vibrate against spacer grids, other fuel rods, or other components of the core. No items of interest in the United States were identified while one event in Sweden involving flow-induced fretting wear was identified in the literature.

Table 5.1 Cross-Reference for Events Related to Personnel Errors

	Fuel Oriented	Fuel Handling Oriented	Spent Fuel Oriented	Control Rod Oriented	Core/Coolant Oriented
Procedural Deficiency			5.3.4.2 5.3.2.2	5.4.2.2, 5.4.2.15 5.4.3.3, 5.4.3.5	5.5.3.2
Administrative Deficiency				5.4.3.2	
Nonconservative Assumption		5.2.3.2			
Human Error		5.2.1.3, 5.2.1.5 5.2.3.1	5.3.2.3, 5.3.3.1	5.4.1.10, 5.4.1.11 5.4.2.6, 5.5.3.7 5.4.3.10	5.5.3.1, 5.5.5.1
Procedural Violation		5.2.2.1, 5.2.2.2 5.2.2.3, 5.2.2.4 5.2.2.5, 5.2.2.6 5.2.2.7, 5.2.2.8 5.2.2.9	5.3.1.2, 5.3.1.3	5.4.1.5, 5.4.1.12 5.4.3.4	5.5.5.2
Fatigue					
Training					
Inadequate Communication					

Table 5.2 Cross-Reference for Events Related to Equipment Problems

	Fuel Oriented	Fuel Handling Oriented	Spent Fuel Oriented	Control Rod Oriented	Core/Coolant Oriented
Component Failure		5.2.1.4, 5.2.3.3 5.2.4.1, 5.2.4.2		5.4.1.7, 5.4.2.1 5.4.2.3, 5.4.2.5 5.4.2.6, 5.4.2.7 5.4.2.8, 5.4.2.9 5.4.2.11, 5.4.2.12 5.4.2.13, 5.4.2.14 5.4.2.16, 5.4.2.17 5.4.2.18, 5.4.3.6 5.4.3.9, 5.4.4.1 5.4.1.8	5.5.1.1
Computer Problems					
Installation Deficiency		5.2.1.1			
Maintenance Deficiency			5.3.2.1	5.4.1.1	
Manufacturing Deficiency					
Design Inadequacy				5.4.1.13	

5.1.2.1 Oskarshamn-2, Ringhals-1

"During annual outages at the 605-MWe Oskarshamn-2 and 750 MWe Ringhals-1 reactors, it was found that wear damage had occurred in the fuel channels. At Oskarshamn-2, the wear damage appears to have been caused by vibration of in-core detectors against the channel walls of a small number of fuel assemblies, and in two cases the walls have been penetrated. At Ringhals-1 the wear was associated with control rod guide pads; damage and penetration has been caused on several channels." No fuel was damaged at either reactor (*Nuclear News*, September 1991a).

5.1.3 Fuel Rod Internal Pressures

The release of gaseous fission products may result in the expansion of fuel rods, if the internal fuel rod pressure exceeds the reactor system pressure, excessive stresses along the length of the rods. One item of significance regarding fuel rod internal pressures was reported.

5.1.3.1 ROPE Program (Fuel Rod End-of-Life Internal Pressures)

Studsvik Nuclear (Sweden) has investigated end-of-life fuel rod internal pressure as a potential burnup limitation with its Rod Over Pressure Experiments (ROPE I for BWR and ROPE II for PWR fuel). ROPE I has been completed while ROPE II, which began in 1990, is expected to be completed in 1993. The ROPE I tests used irradiated BWR fuel (about 35 MWd/kgU), which was "refabricated and pressurized with a fill gas release at high burnup." Next, the rods were placed in an in-reactor loop and irradiated while examining cladding creep, gap conductance, cold gap, fuel swelling, and general fuel performance. The ROPE II test is to be conducted in a similar manner. The test rods were supplied by ABB Atom and Siemens. The PWR fuel was irradiated in Swedish and German PWRs to burnups over 40 GWd/MTU. These test rods were to be "refabricated" and pressurized at Studsvik's Hot Cell Laboratory. The test rods will have overpressures at power up to 10 MPa. Cladding creep, changes in gap conductance, fuel swelling, and other aspects of fuel performance will be evaluated (Grounes and Tomani 1992).

5.1.4 Guide Pin or Alignment Pin Problems

Guide pins are used to place fuel assemblies on the lower grid plate. Fuel alignment pins are located in the upper reactor internals and provide exact radial positioning of the fuel assemblies in the core by fixing their top and bottom heads. When these guide pins are not aligned properly they may be damaged. There was one item of interest in Germany during 1991 involving guide/alignment pin problems that was identified in the literature.

5.1.4.1 Replacing Fuel Alignment Pins in German PWRs

The material commonly used for alignment pins is Inconel X-750 which provides excellent wear resistance. The susceptibility of materials with high nickel content (e.g., Inconel X-750) to intergranular stress corrosion cracking became apparent in the late 1970s. Intergranular stress corrosion cracking is typically encountered when high nickel content alloys experience high stress concentrations combined with high temperature environments. By 1989, 80 broken fuel alignment pins had been detected in German PWRs. Failure of alignment pins during power operations is not considered a significant safety concern since the fuel assemblies are axially restrained in the reactor pressure vessel. However, high radial loads could be exerted on the alignment pins during a reactor accident. Although calculations demonstrated adequate safety margins to ensure safe reactor shutdown, the German authorities decided that non-destructive examinations of alignment pins should be made and equipment for replacing defective pins should be made available. As a result, ABB conducted several campaigns to replace pins made with Inconel X-750. In September of 1991 ABB completed its fourth complete alignment pin replacement program. To complete the work ABB qualified an ultrasonic inspection method, produces the necessary pin replacement equipment, and developed and tested a new modified alignment pin. The new modified pin is manufactured from 1.4571 K 700 (strain-hardened) material and coated with chromium carbide to improve wear resistance. For more specific information regarding the replacement efforts, and associated time and man-rem, refer to Poetz and Kalthoff (1991).

5.1.5 Miscellaneous - Fuel Related

All fuel related problems that do not fit into any of the above categories are listed below. There were three items of interest involving miscellaneous fuel oriented topics that were identified in the literature.

5.1.5.1 Cleaning Plugs Left in Fuel Rods

An incorrect valve setting caused felt cleaning plugs, used to clean fuel rods before pellet insertion, to remain in some BWFC rods after the procedure. The rods were destined for Northwest Utilities' Connecticut Yankee plant, but were found by BWFC workers before the rods left the manufacturing plant. If undiscovered, the potential production of hydrides in the rods could have led to fuel failure through corrosion of the Zircaloy cladding. Babcock & Wilcox Fuel Company is researching the effects of leaving the felt plugs in and is planning to modify the fuel and cleaning process (*Inside N.R.C.*, September 9, 1991).

5.1.5.2 Fission Products Release Simulation for High Burnup Fuel

Osborne and Lorenz (1991) provides a brief summary of fission product release studies at Oak Ridge National Laboratory (ORNL) under LWR severe accident conditions. Tests of high-burnup LWR fuel and simulated fuel with fission-product tracers were conducted in an induction furnace (2700 K). Release rate values for Kr-85 and Cs-137 were determined along with total releases. The most volatile fission-product elements (i.e., krypton, iodine, and cesium) were released almost totally at high temperatures, with little variation due to the test atmosphere. However, other fission products releases (e.g., strontium, molybdenum, ruthenium, tellurium, antimony, barium, and europium) were found to be dependent on the test atmosphere. Total releases and release rates for krypton and cesium were used to develop the ORNL Diffusion Release Model, a model that predicts the release of volatile fission products.

5.1.5.3 Rifled Cladding Design

In a report titled "Theoretical evaluation of rifled cladding for LWR fuel," (Montgomery et al. 1991) a theoretical evaluation of "Rifled Cladding" is presented. Rifled cladding deviates from regular cladding design by

the inner surface of the cladding tube being prismatic rather than cylindrical. Their are 40 faces on the inner surface of BWR 8x8 fuel and 30 on BWR 9x9 fuel. In addition to the prismatic cladding, fuel pellets are to be chamfered at both ends. It is reported that "Rifled cladding aims at reducing both the fission gas release and the maximum cladding stresses, thus helping to increase the attainable average rod burnup and decrease the risk of failure during operational maneuvers involving local power increases." The theoretical studies investigating the fuel-cladding mechanical interaction in rods of rifled cladding design were performed using the two performance codes INTERPIN and FREY-01. The studies conclude that improved failure resistance of rifled cladding rods can be explained by the reduction of the strong azimuthal mechanical interaction between fuel and cladding.

5.2 Fuel Handling Oriented

There is a potential for fuel assembly damage due to fuel handling operations. This section includes fuel handling events with dropped, broken, damaged, or potential for damage to fuel; fuel in incorrect position; crane operations; fuel handling procedural violations; fuel production plants; and miscellaneous or unknown fuel handling events.

5.2.1 Dropped, Broken, Damaged, or Potential for Damage

The reported events that involved dropped, broken, damaged, or potential for damage fuel rods as a result of fuel handling are listed in this section. There were three events in the United States and one event in a foreign country identified.

5.2.1.1 Brunswick-1

On January 2, 1991, a fuel bundle was dropped 127 inches into the core. As the bundle was being lowered, resistance from an adjacent control blade led to a momentary loss of power. The grapple opened, and the bundle was released. No releases from the fuel resulted (Brunswick-1, 91-001). An investigation revealed that air lines to the grapple solenoid were installed backwards, and the loss of power caused the grapple to fail in an open position (not closed as designed). The control

system was repaired, and the fuel bundle was removed. There are no plans to reuse the fuel bundle (*Atomic Energy Clearing House*, January 18, 1991).

5.2.1.2 Dresden-3

On October 18, 1991, failure to follow fuel movement procedures at the Dresden-3 plant led to the bending of lifting handles on two fuel assemblies and damage to the device used to lift the assemblies from the core (*Nuclear News*, March 1992).

5.2.1.3 Pilgrim Nuclear Power Station

An apparent malfunction in the equipment used to move fuel halted the refueling of the Pilgrim Nuclear Power Station, which had been initiated June 21, 1991. The process was stopped and was not resumed until the cause of the malfunction was thoroughly understood and corrected. The equipment problem was identified when the grapple used to move fuel prematurely disengaged while a fuel assembly was only partially lowered into a rack in the spent fuel pool (SFP). All fuel movements are conducted under twenty feet of water to preclude the risk of radiation exposure. The fuel assembly was placed into its storage rack in the SFP with no apparent damage to the assembly (*Atomic Energy Clearing House*, June 28, 1991).

5.2.1.4 Tricastin-3

"At Tricastin-3 on July 12, a fuel handling incident caused slight deformation of a fuel assembly. The fuel element was being lowered slowly towards the upending device of the fuel storage pool when by mistake the operation was switched to high speed. Although the operator stopped the maneuver immediately, the leg of the fuel element ended up pressing against the upending device and the element was slanted about 10 degrees off vertical" (*Nucleonics Week*, September 12, 1991b).

5.2.2 Fuel Handling Procedure Violations

Strict specifications for handling fuel at nuclear power plants are required by plant procedures according to technical specifications. There were nine events identified in the United States involving fuel handling procedure violations.

5.2.2.1 Diablo Canyon-1

On February 13, 1991, at 0225 Pacific Standard Time, with the Unit 1 core offloaded (not in any operational mode), all three diesel generators were inoperable while the spent fuel crane was operating with heavy loads over the fuel storage pool. A quality assurance audit team identified this condition, and an extensive review determined that the intent of Technical Specification (TS) 3.8.1.2(b) was not met. The root cause of this event was lack of written guidance for control of diesel generator operability. The TS was unclear with regard to a requirement for diesel generator operability during movement of irradiated fuel within the spent fuel storage pool or crane operation with heavy loads over the spent fuel storage pool and no fuel in the reactor vessel. Procedural guidance was issued to clarify the requirements of TS 3.8.1.2(b). This guidance requires that during fuel movement within the fuel storage pool, or crane operation with loads over the fuel storage pool, at least one fuel handling building (FHB) ventilation train must be capable of being powered from an operable emergency diesel generator (Diablo Canyon-1, 91-014).

5.2.2.2 Limerick-2

On April 10, 1991, a fuel handling supervisor observed the uptravel mechanical stop on the refueling platform monorail mounted auxiliary hoist to be mispositioned outside the limits of surveillance test procedure ST-6-107-633-2, "Core Alteration Testing." During the previous shift the uptravel mechanical stop was moved 24 in. to permit movement of a control rod blade from the SFP to the reactor. On April 12, 1991, during investigation into the event, it was determined that the top of a control rod blade was 1.5 in. less than the required 6 ft 6 in. below the normal fuel storage pool water level; a condition prohibited by TS SR 4.9.6.3B. Technical Specification SR 4.9.6.3B was amended on August 16, 1990. Prior to August 16, 1990, the interpretation of TS SR 4.9.6.3B did not limit control rod blade movement. The cause of this event was an inadequate maintenance procedure M-041-061 and an inadequate procedure ST-6-107-633-2. A contributing cause to the event was inadequate training of the fuel handling supervisors on in-vessel maintenance activities following the change to TS SR 4.9.6.3B. The maintenance procedure has been

changed to delete the provision to reposition the mechanical stop. Procedure ST-6-107-633-2 has also been revised (Limerick-2, 91-007).

5.2.2.3 Palo Verde-2

A core alteration during refueling at Palo Verde-2 on October 27, 1991, was the result of a control assembly being lifted by at least 1 ft without a senior reactor operator present, and without direct communications between the control room and the refueling station. On October 27, 1991, Palo Verde Unit 2 was in Mode 6 (refueling) with the reactor vessel head removed during a planned refueling outage. At approximately 1220 Mountain Standard Time (MST), the control room shift supervisor was notified that the control element assemblies had been withdrawn approximately 1 ft out of the core during defueling operations without direct communication established with the control room during this core alteration and without the refueling senior reactor operator present to observe and directly supervise the core alteration. This is contrary to the requirements of TS Limiting Condition for Operation (LCO) 3.9.5 and TS Administrative Control 6.2.2B. Core alterations were immediately suspended per the action requirement of LCO 3.9.5. Following resumption of core alterations after implementation of immediate corrective actions, core alterations were not suspended at 1716 MST contrary to the action requirements of TS 3.8.3.2 when one of the two required 120 volt vital AC busses transferred from its normal power source to its alternate power source. The cause of these events was due to a combination of cognitive personnel errors. The control room shift supervisor suspended core alterations, and plant management suspended all work affecting or involving core alterations. There have been no previous similar events reported pursuant to 10CFR50.73 (Palo Verde-2, 91-006).

5.2.2.4 Riverbend-1

At 1424 on June 4, 1991, with the reactor in Operational Condition 1 (power operation), irradiated fuel was moved in the fuel building without prior performance of surveillance requirements under TS 3/4.6.5.2 (secondary containment integrity - fuel building). Since the plant was in compliance with TS 3/4.6.5.1 (secondary containment integrity operating) and plant staff personnel took conservative actions to verify fuel building integrity, secondary containment integrity was ensured prior to fuel

movement. Thus, the intent of TS 3/4.6.5.2 was met. However, this report is submitted pursuant to 10CFR50.73(A)(2)(i)(b) to document non-compliance with the surveillance requirement. The procedures used during this evolution provided inadequate guidance to ensure requirements of TS 3.6.5.2 were met. The lack of checklists and/or prerequisite steps requiring sign off, coupled with the duration of the work, led the shift supervisor to assume that all pertinent requirements had been met during preparations made in previous shifts (Riverbend-1, 91-011). The fuel building pressure was less than the 0.25 in. of water vacuum required by the technical specifications, and the fuel building charcoal filtration train was in service. If a fuel drop accident occurred during fuel movement, the fuel building ventilation exhaust would have filtered as designed (Riverbend-1, 91-011 Rev. 1).

5.2.2.5 Riverbend-1

On December 23, 1991, with the reactor in Operational Condition 4, it was discovered that from June 4, 1991, to June 6, 1991, fuel was moved in the fuel building while resistance temperature detectors installed in the fuel building heater trains were not environmentally and seismically qualified. Thus, the fuel building filter trains are considered to have been inoperable. Therefore, this report is submitted pursuant to 10CFR50.73(A)(2)(i)(b) as operation prohibited by the technical specifications. The subject resistance temperature detectors have been replaced with qualified models. Training will be provided on the importance of communication when performing operability analyses. Gulf States Utilities Co. has performed an ambient humidity analysis and a decay time analysis for the fuel in the SFP. The results show that it was unlikely that fuel building charcoal filter efficiency was degraded and that if a fuel handling accident had occurred, offsite doses would have been bounded by the design basis fuel handling accident calculation (Riverbend-1, 91-022).

5.2.2.6 South Texas-1

On February 18, 1991, at 1334 hours, Unit 1 was in Mode 6. The FHB truck door was opened while operations personnel were involved in fuel handling as part of the core reload. At approximately 1340 hours, an operator on the FHB fuel bridge noted that the door was opened and secured all fuel movement. The doors were closed at 1359 hours. The FHB exhaust air system was

rendered inoperable when the FHB truck doors were opened. Fuel movement was suspended immediately upon discovery, as required by TS 3.9.12, until the FHB ventilation system was restored to an operable condition. The cause of this event was incomplete administrative controls on the FHB truck door. There were no controls in place to ensure that the appropriate TS requirements for the FHB exhaust air system were followed. Corrective actions included placement of additional locks on the FHB truck doors; issuance of a bulletin/night orders to security and health physics personnel to ensure that in addition to security and health physics that operations personnel are also present at the door prior to opening; issuance of a memorandum, which discussed this incident, to licensed operators; and an evaluation to determine the effect of the FHB doors on the operability of the FHB exhaust air system (South Texas-1, 91-005).

5.2.2.7 St. Lucie-1

Between 1430 on November 10, 1991, and 0210 on November 11, 1991, containment integrity was not in compliance with technical specifications. St. Lucie Unit 1 was in Mode 6 (refueling), and fuel assemblies were being moved in the reactor core. Containment integrity is required for core alterations. The cause of the incomplete containment integrity was personnel error by mechanical maintenance department personnel for failure to follow an approved procedure. Plant procedure requires that nuclear plant work orders be approved by the nuclear plant supervisor prior to starting work. Without obtaining this prior approval, maintenance workers removed a relief valve on one side of a containment penetration while a drain valve was open on the other side. This created a direct flow path from containment to the reactor auxiliary building. Permission to start work was requested after the relief valve had been removed by the previous shift. The corrective actions were that the containment integrity was re-established, the responsible maintenance supervisor was counseled on the need for strict adherence to procedures, and the event will be incorporated into maintenance journeyman training. This event was also discussed with all maintenance crews, and the importance of receiving proper authorization prior to beginning work on plant systems was stressed (St. Lucie-1, 91-008).

5.2.2.8 Three Mile Island-1

On October 8, 1991, Three Mile Island-1 was in refueling shutdown. Licensed operators were performing the 1303-11.4, "Refueling Systems Interlocks," test of the main fuel bridge hoist. This test is normally performed in conjunction with 1505-1, "Fuel and Control Component Shuffles." Section 6.3.3.1 of 1303-11.4 requires fuel movement, however, it also requires, by reference, that no movement of fuel take place until the prerequisites of 1505-1 are complete. In this event, the bridge crew did not adequately prepare for the evolution about to take place and moved fuel to test the hoist interlocks without having verified containment isolation. All of the 1505-1 prerequisites for containment isolation were completed, except for the open reactor building personnel and emergency airlock doors (NH/AL). Technical Specification 3.8.6 requires that at least one door in each airlock be closed when moving irradiated fuel in the reactor building. This event was caused by personnel error, due to a lack of understanding that the test involved fuel movement; and, procedural inadequacy, because 1303-11.4 did not clearly caution the operators that containment isolation was required prior to testing of the interlocks. The causes of the event were reviewed with all fuel handling personnel prior to the commencement of the fuel shuffle (Three Mile Island-1, 91-004 Rev. 1).

5.2.2.9 Waterford-3

On April 4, 1991, Waterford Steam Electric Station Unit 3 was in Mode 6, Refuel 4, when the TS Surveillance OP-903-099, "Spent Fuel Handling Machine Operability," was not performed satisfactorily. The acceptance criteria for the surveillance was not completed within 72 hours prior to the start of fuel movement. Technical Specification 3.9.7, Surveillance Requirement (SR) 4.9.7.1, states that the fuel handling machine shall be demonstrated operable within 72 hours prior to the start of fuel assembly movement. Fuel movement began before the supervisor reviewed and approved the surveillance results. Therefore, the plant operated in a condition prohibited by technical specifications. The root cause of this event is inadequate attention to details. Technical Specification SR 4.9.7.1, to perform the surveillance within 72 hours prior to the

start of fuel assembly movement, was not properly completed. To prevent recurrence, the individual who performed the surveillance was counseled on the requirements of procedure OP-O903-099 and TS 3.9.7. Because the fuel handling machine was subsequently tested satisfactory and demonstrated operable, this event did not threaten the health and safety of the general public or plant personnel (Waterford-3, 91-004).

5.2.3 Fuel Production Plants

This section lists events occurring at fuel production plants in 1991. There were three items of interest identified in the United States.

5.2.3.1 ABB CENF Fuel Fabrication Facility Accident

A small release of uranium hexafluoride at an ABB CENF nuclear fuel fabrication plant in Hematite, MO, slightly injured one worker, and 25 other employees were evacuated. The spill occurred after an employee failed to turn off a valve before disconnecting sampling equipment from a cylinder of uranium hexafluoride being tested for enrichment (*Nuclear News*, February 1991).

5.2.3.2 GE Fuel Fabrication Facility Accident

A chemical incident at the GE commercial nuclear fuel manufacturing plant in Wilmington, N.C., led GE officials to declare an alert at the plant on May 29, 1991. The incident occurred when about 300 lb of uranium was misrouted from normal processing and flowed into a 20,000 gal waste storage tank. The greater than expected amounts of uranium in the waste system tank posed a potential for a localized criticality. The U.S. Nuclear Regulatory Commission specialists removed the uranium solids, and the alert status was terminated on June 1, 1991 (*Nuclear News*, July 1991). While GE discounted the possibility of a criticality accident, an NRC incident investigation team (IIT) claimed that such an accident could "yield a burst of neutron and gamma radiation that would likely be fatal to anyone within 10 feet" (*Inside N.R.C.*, August 12, 1991). The IIT determined the causes to be 1) a lack of belief in the possibility of a nuclear criticality at GE; 2) GE management did not provide effective guidance and oversight of its NRC-licensed activities to assure that operations

were conducted in a safe manner; and 3) a deep-seated, production-minded orientation within GE was not sufficiently tempered by a safety-first attitude, particularly regarding nuclear criticality safety (*Atomic Energy Clearing House*, August 16, 1991a).

5.2.3.3 SPC Fuel Production Facility Accident

"Three men were contaminated with radioactivity on June 17, 1991, at the plutonium processing area at the Siemens fuel element factory at Hanau, Germany." The contamination stemmed from a 4-mm crack in a plastic cover of a 3.3 kg fissile material container (*Nuclear News*, August 1991).

5.2.4 Miscellaneous - Fuel Handling Related

Reported events that are fuel handling related but do not fit into any of the above categories are listed in this section. There were two events in the United States which fall into this category that were identified in the literature.

5.2.4.1 Braidwood-2

At 2239 on March 8, 1991, the containment fuel handling incident area Radiation Monitor 2RT-AR012 momentarily went into high alarm and interlock actuation, due to sudden spiking. This initiated a Train B containment ventilation isolation signal. No components repositioned as all valves were secured in the closed position. At 2216 on March 9, a second spike occurred on Radiation Monitor 2RT-AR012, which also resulted in a ventilation isolation signal. On March 6, prior to these actuations, the monitor was identified as slowly drifting. It was concluded that detector replacement was required. On March 11, 1991, the detector for Radiation Monitor 2RT-AR012 was replaced. The cause of this event was component failure. The faulty detector was creating spurious high radiation spikes that resulted in the monitor initiating the containment ventilation isolation signals. The detector for this monitor is normally replaced on an 18 month frequency in accordance with the environmental qualification program. The failed detector in this event had been in service since August of 1990. This detector has been returned to the vendor for failure analysis (Braidwood-2, 91-002).

5.2.4.2 San Onofre-3

At 2255 on November 7, 1991, with Unit 3 at 100% reactor power, FHS Train "A" spuriously actuated as Radiation Monitor 3RI-7822 was being returned to service following a filter change-out for the particulate/iodine channel. There was no indication of increased radiation levels in the FHB. At 2314, after the airborne activity levels in the FHB were confirmed to be normal, FHS Train "A" was reset and the FHB ventilation system was returned to normal. The cause of the FHS Train "A" actuation was an intermittent failure internal to the normal/bypass switch associated with the FHS circuitry. The planned corrective actions include 1) replacement of the FHS normal/bypass handswitch and 2) performance of a failure analysis on the replaced handswitch to determine the root cause of the failure. A supplement to this LER will be submitted to update the cause and corrective action if the root cause is determined to be other than an isolated failure. There is no safety significance to this event since FHB airborne activity levels remained normal, and all FHS Train "A" components operated in accordance with design (San Onofre-3, 91-007).

5.3 Spent Fuel Oriented

Items of interest that are related to spent fuel are presented in this section. Specific areas included in this section include SFP problems; heat removal and ventilation; dry storage; spent fuel consolidation; miscellaneous spent fuel storage issues; and miscellaneous problems.

5.3.1 Spent Fuel Pool Problems

This section lists the events that involved the storage of spent fuel in SFPs that do not involve heat removal or ventilation problems. There were three events or items of interest that involved SFP problems identified in the literature.

5.3.1.1 Millstone-1

On May 1, 1991, at 1030 hours, with the plant in cold shutdown for the scheduled refueling outage (0% power, 90°F), source range monitor and intermediate range monitor dry tube replacement was in progress when an irradiated dry tube assembly momentarily

broke the surface of the fuel pool during transfer from the reactor vessel. The event resulted in the automatic initiation of standby gas treatment system due to refuel floor high radiation. All systems functioned as designed, and there were no significant safety consequences from this event. The dry tube was subsequently disposed of using a special procedure (Millstone-1, 91-015).

5.3.1.2 Millstone-2

On February 25, 1991, with the plant in Mode 1 at 100% power, it was determined that SR 4.9.16.1 was not performed within the required 72 hour time interval and that SR 4.0.2.B, "the combined time interval for any 3 consecutive surveillance intervals not to exceed 3.25 times the specified surveillance interval," was not satisfied. The missed surveillance was performed approximately 96 hours after the previously completed surveillance. Surveillance Requirement 4.9.16.1 states that "the decay time of all fuel within a distance "I" from the center of the SFP cask set-down area shall be determined to be ≥ 120 days within 24 hours prior to moving a shielded cask to the refueling floor and at least once per 72 hours thereafter." Distance "I" equals the major dimension of the shielded cask. There are no safety consequences as a result of this event. Since at the time of the missed surveillance there were no fuel assemblies in the SFP with a decay time of less than 120 days (Millstone-2, 91-007).

5.3.1.3 San Onofre-2

On January 25, 1991, with Unit 2 at 100% power, during the performance of the Train B FHB post accident clean-up unit (PACU) (E-3371) monthly surveillance, the PACU inlet duct heater (E-465) was found to be "off". Contrary to the surveillance requirements, investigation revealed that the local pilot switch for E-465 was in the "off" position, de-energizing the heater. The pilot switch was turned "on," and the surveillance was re-initiated and completed satisfactorily. It was determined that a previous E-371 monthly surveillance performed on January 15, 1991, was incorrectly accepted as satisfactory since the completed surveillance procedure indicates that the heater was "off," contrary to procedural and TS requirements during the conduct of the test. Consequently, the operability requirements for PACU E-371 were not satisfied when movement of irradiated fuel in the Unit 2 SFP occurred during the period

of January 19-22, 1991, which resulted in violating the requirements of TS 3.9.12. The erroneous acceptance of E-371 monthly surveillance performed on January 15, 1991, is attributed to procedural deficiencies, inadequate knowledge level of PACU heater operation, and inadequate review of test results (San Onofre-2, 91-001).

5.3.2 Heat Removal

Reported events that involve heat removal in SFPs are listed in this section. Adequate heat removal of residual spent fuel energy is necessary in order to prevent damage to the spent fuel. There were four events, or items of interest, identified in the United States.

5.3.2.1 Connecticut Yankee

On July 17, 1991, at 1000 hours, with the plant in Mode 1 at 100% power, operations department personnel isolated service water to the SFP heat exchangers to support preplanned maintenance on the service water system. The root cause of this event was the need to remove the spent fuel cooling system from service to replace several pipe nipples in the service water system. Corrective action consisted of returning the SFP cooling system to service at 1755 hours following completion of the maintenance activities. This event is reportable under 19FR50.73(A)(2)(V)(B) as a condition that alone could have prevented the fulfillment of the safety function of system needed to remove residual heat (Connecticut Yankee, 91-014).

5.3.2.2 Turkey Point-4

On March 13, 1991, at 1528 Eastern Standard Time (EST), with Unit 3 and Unit 4 defueled, the Unit 4 startup transformer locked out, which caused the 4A-4160 volt bus to de-energize. The 4B-4160 volt bus was already out of service. Two of four available nonsafety-related diesel generators were started and made available to provide power to the 4A bus. Power was restored to the 4A bus at 1635. Loss of power to the 4A bus resulted in the loss of normal SFP cooling. The pool temperature increased by approximately 3°F to 87°F. During the time the cooling system was not in operation. An unusual event was declared at 1550 and terminated at 1722 when the SFP cooling was restored. State and NRC notifications were made as required. An event response team (ERT) could not determine a root

cause for this event. The ERT concluded that the most probable root cause of this event was inadequate process control. Construction waste materials (metal shavings) fell into a control panel that was still energized. These metal shavings may have caused a short that initiated the event. The waste materials were removed. Applicable procedures were revised to ensure clean work practices. Another procedure was revised to identify and provide additional controls for work on, and around, sensitive equipment (Turkey Point-4, 91-001 Rev. 1).

5.3.2.3 Turkey Point-4

This voluntary LER is being submitted following the guidance provided by NUREG-1022, Supplement 1, item 19.1. On June 26, 1991, at 0308 Eastern Daylight-Saving Time (EDT), with both units defueled, the unit 4 4D-4KV bus locked out. The bus lockout resulted in the de-energizing of the 4C intake cooling water pump and the 4C component cooling water (CCW) pump. CCW pump 4A was in service and remained in service during this event. With 4A and 4B ICW pumps out of service for maintenance, the de-energizing of the 4C ICW pump made the SFP cooling system inoperable since the normal heat sink was not available. On 6/26/91, at 0403, the 4C, CCW pump was started and at 0412, 4C ICW pump was started restoring the normal heat sink for the SFP cooling system. The root cause of this event was inadequate administrative controls. A sensitive equipment list used in preparing a work package was superseded which resulted in a bumped relay 174, device RB inside cubicle 4AD06. Relay 174, device RB tripped causing the lockout of the 4D-4KV bus. All work in the area of the relay that actuated (relay 174, device RB) was stopped so the cause of the actuation could be evaluated. Work will not resume until the event has been reviewed with applicable contractor personnel emphasizing the sensitivity of the equipment in the areas being worked and also emphasizing attention to detail (Turkey Point-4, 91-003).

5.3.2.4 Beyond design basis accidents in spent-fuel pools — Generic Issue 82

An article published in *Nuclear Engineering and Design* by Throm (1991) presents Generic Issue 82 (GI-82) that relates to the NRC's concern for a postulated accident sequence resulting in the loss of water from a LWR

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spent-fuel storage pool causing a Zircaloy cladding fire that could propagate to older stored fuel. The issue was originally identified in the late 1970s during hearings related to spent-fuel pool reracking amendments. The studies conclude that "the risk due to beyond design basis accidents in spent-fuel pools, while not negligible, is sufficiently low that the added cost involved with further risk reduction is not warranted."

5.3.3 Ventilation

Ventilation in the SFP building or fuel building is not necessary to ensure that airborne contamination is controlled, reducing the possibility of contamination to workers. The events that were caused by faulty or defective ventilation in these areas are listed in this section. There were two events identified in the United States.

5.3.3.1 McGuire-1

On July 24, 1991, at 1000, while Unit 1 was in Mode 1 (power operation) at 100% power, operations control room personnel were informed by performance (PRF) personnel that fuel handling exhaust (VF) system had been technically inoperable since 5/27/91. Additionally, fuel had been moved, and crane loads had been moved over the fuel pool on 7/91, 7/18/91, and 7/22/91 through 7/24/91 while the VF system was technically inoperable. The VF system was subsequently declared inoperable and activity in fuel pool was terminated. Appropriate notification was made to the NRC at 1324, on 7/91. This event is assigned causes of inappropriate actions. Operations control room personnel were subsequently directed to declare the VF system inoperable if either the supply or exhaust equipment is out of service (McGuire-1, 91-011).

5.3.3.2 Prairie Island-1

On March 15, 1991, both units were operating at full power. Surveillance Procedure SP-1115, "Spent Fuel Pool Special Ventilation System Test," was being performed. This test verifies that the SFP special ventilation system is activated and the normal SFP ventilation system isolated by actuation of redundant radiation monitors. Spent fuel pool special ventilation Train A had been tested satisfactorily. When Train B was to be tested, the Train A radiation monitor was put in reset per the procedure. After high radiation was simulated

on the Train B radiation monitor, control room operators observed that SFP normal ventilation had isolated as required but that both trains of the SFP special ventilation system were running. Actuation of Train A was a non-engineered safety feature (ESF) actuation of an ESF system. Troubleshooting of the Train A radiation monitor revealed no problems. Surveillance Procedure SP-1115 was then performed again in its entirety; the monitors and the actuation circuitry operated as designed. Further investigation was done. No cause for the event could be assigned, and the spurious actuation of Train A of the SFP special ventilation system could not be duplicated: possible causes were investigated, but no cause could be assigned. Observation of the equipment is continuing (Prairie Island-1, 91-001).

5.3.4 Reracking/Storage Rack Issues

Reported events that involved improper storage of spent fuel are listed in this section. There were two events in the United States that involved storage rack issues.

5.3.4.1 Hatch-2

On May 5, 1991, at 0930 Central Daylight-Saving Time (CDT), Unit 2 was in the refueling mode with the core completely loaded when a discrepancy between Unit 2 TS 3.9.10 and the design of the high density fuel storage system (HDFSS) (EIIS code DB) racks in the Unit 2 SFP was identified. Technical Specification 3.9.10 requires at least 23 ft of water be maintained over the top of irradiated fuel assemblies seated in the storage racks. However, rack design is such that 23 ft of water can be maintained only above the top of the active fuel region of assemblies seated in the racks. The top of each fuel assembly, i.e., top of the upper tie plate, is approximately 12.125 in. above the top of the active fuel (TAF). The General Electric Company was contacted to assess the safety significance of the discrepancy and provide a basis for the 23-ft requirement. The General Electric Company indicated the discrepancy was not a safety concern, but was unable to document the 23-ft depth was intended to be measured with respect to TAF. Plant and corporate support personnel analyzed the impact of this discrepancy and concluded that a safety concern did not exist. The cause of this event is a less-than-adequate design change safety evaluation (SE) in that the SE for the HDFSS design change developed

in the early 1980s did not address compliance with TS 3.9.10. On May 20, 1991, LCO 2-91-E424 was initiated to ensure compliance with Action Statement 3.9.10 (Hatch-2, 91-014).

5.3.4.2 McGuire-1

While reviewing TS Section 3.9.12, McGuire reactor unit personnel identified 11 fuel assemblies that had been stored in the Unit 1 SFP in a manner contrary to the requirements of TS 3.9.12. This LCO requires, in part, that fuel stored in Region 2 of the SFP shall undergo 16 days of decay, and if a checkerboard pattern is employed for unqualified fuel, one row between normal storage locations and checkerboard storage locations will be vacant. The vacant row provision of the specification was not satisfied from March 23, 1990, through October 23, 1991. At the time of discovery at 0900 on October 24, 1991, Unit 1 was defueled, and Unit 2 was in Mode 1 (power operation) at 100% power. This event has been assigned a cause of defective procedure. The fuel assemblies in question were immediately moved to positions to establish the required vacant row (McGuire-1, 91-016).

5.3.5 Miscellaneous Spent Fuel Storage Issues

Events of general nature involving spent fuel storage that do not fit into any of the above categories are listed below. There were two items of interest identified in the United States involving spent fuel storage issues.

5.3.5.1 EPRI Research for On-Site Spent Fuel Storage Increases

The Electric Power Research Institute has started several demonstration projects to validate options for on-site spent fuel storage capacity increases. The licensing and technological challenges for spent fuel consolidation were investigated in a program with Northeast Utilities, ABB CENF, and DOE. The program showed that a 2:1 consolidation ratio was possible. Another major program was the demonstration of metal cask storage. In cooperation with DOE and Virginia Power, cask designs were evaluated and led to the NRC licensing of cask storage at Surry. The other project EPRI participated in was the development, demonstration, and licensing of NUHOMS storage of spent fuel.

NUHOMS is a dry storage option that horizontally stores spent fuel inside a concrete silo (*Nuclear Engineering International*, September 1991).

5.3.5.2 Fort St. Vrain Spent Fuel Storage Licensed

"The NRC, in late November 1991, issued a license authorizing Public Service Company of Colorado (PSCC) to place spent nuclear fuel and related components from its Fort St. Vrain Nuclear Generating Station in an independent spent fuel storage facility." (*Nuclear Safety*, April-June 1992)

5.4 Control Rod Oriented

This section includes control rod system problems; control rod operation; control rod position indications; control element assembly problems; control rod swelling, wear, corrosion and cracking; guide tube problems; and miscellaneous or unknown control rod problems.

5.4.1 Control Rod System Problems

This section lists all reported events that involved control rod system problems that are not categorized into other more specific sections. There were 14 events or items of interest identified in the United States and one event in Sweden involving control rod system problems.

5.4.1.1 Big Rock Point

Big Rock Point TS 7.5.7 requires that "...the control rod drive that was removed shall without delay be replaced by a spare control rod drive or the original control rod drive shall be reinstalled." Contrary to the above, on December 10, 1991, at 0330, a control rod drive was removed from the reactor, and a spare drive nor the original drive could be reinstalled. Problems with alignment of the drives caused the delay to occur. Since the alignment problem required additional time to investigate, a blank flange was installed as a temporary safeguard to prevent a path for possible drainage of the reactor. Later that day, following review of the installation difficulties, personnel re-installed the spare drive. The facility is presently in a major refueling outage, and the reactor was defueled at the time of occurrence. The cause of the alignment trouble was with the hoist and

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pulley assembly used to install the control rod drive to the reactor vessel. The location of the work and first time experience for some of the personnel performing the installation also contributed to the delay. To prevent recurrence, the event was reviewed with plant personnel stressing proper installation technique, prior to resuming further drive change-outs. Procedure improvements are also being investigated (Big Rock Point, 91-008).

5.4.1.2 Brunswick-1

On June 12, 1991, at approximately 0451 hours, Unit 1 was operating with control rod drive hydraulic control unit inoperable due to maintenance, when the control received a low nitrogen pressure alarm on another hydraulic control unit making it inoperable. With two or more control rod scram accumulators inoperable, TS 3.0.3 was invoked, placing the plant in a 6 hour to hot shutdown condition. An auxiliary operator was dispatched to recharge the accumulator with the low pressure. There was no intent to commence a unit shutdown unless problems were encountered during the recharging evolution. On June 12, 1991, at approximately 0531, the accumulator was successfully recharged and TS 3.0.3 was exited. The cause of this event is attributed to the simultaneous inoperability of two Hydraulic 1 units. Corrective actions included recharging the hydraulic control unit (HCU) accumulator with the low pressure, completing repairs on the HCU with the nitrogen leak. Safety significance of this event is minimal. With the reactor vessel at full operating pressure and the scram discharge volume at atmospheric pressure, the accumulator is not needed to meet scram time requirements. Standby liquid control system was available as a backup to the control rod drive hydraulic control unit accumulators. A review of records revealed no similar events (Brunswick-1, 91-015).

5.4.1.3 Brunswick-1

On November 5, 1991, Unit 1 was at 100% reactor power. At 0040, as control rod drive (CRD) HCU 34-19 was being recharged with nitrogen, a second HCU accumulator 46-27 low nitrogen pressure caused a CRD accumulator low pressure/high level alarm in the control room. With two or more control rod drive scram accumulators inoperable, TS 3.0.3 was invoked. This required placing the unit in a 6 hour to hot shutdown condition. After verifying the second alarm was due to

low nitrogen pressure, the auxiliary operator continued recharging the first HCU 34-19 with nitrogen. There was no intent to commence a unit shutdown unless problems were encountered during the recharging evolution. At 0055, on November 5, 1991, the accumulator RHCL 34-19 was successfully recharged, and TS 3.0.3 was exited. The cause of this event is the simultaneous inoperability of two control rod drive hydraulic control units due to the drop in the reactor building ambient temperature. A cold front resulted in areas of the reactor building reaching new seasonal lows (as recorded building temperatures dropped approx. 7°F over the previous 3 days). Previous LER was 1-91-015 (Brunswick-1, 91-027).

5.4.1.4 Dresden-2

At 0327 hours on September 21, 1991, with Unit 2 at 85% power, CRD exercising was being conducted on CRD R-10 per Dresden operating surveillance (DOSS 300-1, daily/weekly control rod drive exercise. While attempting to exercise CRD R-10 from Position "48" (full out) to Position "46," the CRD failed to latch at Positions "46" and "44." With qualified nuclear engineer permission, an attempt to latch at Position "42" was also made, and failure to latch again occurred. The CRD was inserted and held at Position "00" by continuously applying an emergency-in signal. At 0535 hours, CRD R-10 began drift out; the CRD was scrambled to Position "00" (full in) and held in position by charging water pressure. Tests were conducted per special procedure (SP) 91-9-114 to relatch CRD R-10 and flush the CRD collet piston. During testing, CRD R-10 was successfully relatched. This testing indicated that there was no leakage in the directional control valves. The CRD collet housing was determined to be intact, and latching failure was determined to be a result of collet piston binding. Upon completion of flushing, latching failure again occurred, and CRD R-10 was scrambled to Position "00" where it successfully latched. This event had minimal safety significance (Dresden-2, 91-031).

5.4.1.5 Dresden-2

On October 6, 1991, with Unit-2 at 35% power, "failure to follow proper procedures" by reclosing valves in the control rod drive system during routine testing led to the drainage of about 10,600 liters of radioactive water into the reactor building (*Nuclear News*, March 1992).

5.4.1.6 Farley-2

At 1704 on August 6, 1991, while operating at approximately 100% power, a reactor trip occurred. Apparently, a lightning induced transient in the rod control system caused some of the control rods to drop which resulted in a negative flux rate reactor trip. Testing and inspection of the rod control system revealed no damage. It is postulated that a lightning strike induced voltage on the control rod drive mechanism cables which momentarily de-energized the power supply to one or more rod control cabinets and caused the rods associated with that cabinet or cabinets to drop. An engineering study has been initiated to determine how to prevent further lightning induced reactor trips. The unit returned to power operation, at 2049 on August 7, 1991 (Farley-2, 91-005).

5.4.1.7 Oconee-3

Oconee 3 was shut down the month of January for maintenance actions. The maintenance actions required repair of a control rod drive motor that failed due to moisture in the stators, and a low pressure injection line piping leak repair (*Nucleonics Week*, February 6, 1992).

5.4.1.8 Palo Verde-3

At approximately 1449 MST on November 14, 1991, Palo Verde Unit 3 was in Mode 1 (power operation) at approximately 100% power when a lightning-induced electrical fault on the A phase main transformer caused a generator trip, turbine trip, and reactor power cut-back. Approximately 35 seconds following these events, the reactor tripped on low departure from nucleate boiling ratio (DNBR) signals. The plant was stabilized in Mode 3 (hot standby) at normal operating temperature and pressure. The control room shift supervisor classified the event as an uncomplicated reactor trip in accordance with the emergency plan implementing procedure. No other safety system responses occurred and none were required. The post trip review found that the low DNBR reactor trip was due to a control element assembly (CEA) subgroup deviation. During the post trip investigation, a PS engineering discovered that a problem with the control element assembly calculator (CEAC) software design may have delayed the reactor trip for up to 16 seconds when a second CPC time delay was initiated. At the time of the second time delay, one

CEA subgroup was thought to be misaligned greater than the allowed limit in the CPCs. The cause of the second time delay was that the CEAC software design did not anticipate that there would be CEA slips lasting less than 0.5 seconds (Palo Verde-3, 91-008).

5.4.1.9 Perry-1

On January 29, 1991, at 2305, maintenance activities on the rod control and information system (RC&IS) resulted in a condition prohibited by TS 3.1.3.3. Control room operators were made aware that the replacement of power supplies would result in the loss of scram accumulator fault indication and also the loss of RC&IS interface capabilities at the main control panel. Due to a misapplication of TS 3.1.3.3.A.2, control room operators believed the plant was in a 12-hour shutdown LCO action statement; however, the on-coming shift, after reviewing the situation, determined that the TS 3.1.3.3.A.2 action could not be taken and that the plant was in TS LCO 3.0.3. Actions were initiated in accordance with TS 3.0.3 to shut down the plant, and then RC&IS was returned to service. Operators declared the scram accumulators and RC&IS operable, and TS 3.0.3 was exited on January 30, 1991, at 0200. The cause of this event was personnel error, inattention to detail. The unit supervisor interpreted the technical specifications and incorrectly applied the 12-hour shutdown action requirement. To prevent recurrence, this event has been discussed in detail with the control room operators involved (Perry-1, 91-005).

5.4.1.10 St. Lucie-1

On September 18, 1991, with Unit 1 in Mode 1 at 100% power, operators prepared to return the 1A control element drive mechanism (CEDM) motor generator (MG) to service following maintenance on its flywheel bearing. At 1043, while a utility non-licensed operator was manually synchronizing the 1A MG to the CEDM bus, the generator output breaker was closed, out of phase with the running MG. This resulted in an instantaneous overcurrent condition on the CEDM bus. The overcurrent condition opened seven trip circuit breakers (TCBs) and removed power to the CEDMs. All CEAs were then fully inserted. The main turbine automatically tripped upon sensing an undervoltage condition on the CEDM bus and the two remaining TCBs opened. The operators then implemented standard post trip

actions, and the normal reactor trip recovery procedure. The root cause for the out of phase synchronization of the MG was that a non-licensed operator made a personnel error while attempting to manually synchronize the 1A MG to the CEDM bus. The non-licensed operator closed the generator output breaker out of phase while he was in an incorrect section of the procedure. Contributing factors to this event included that on-the-scene supervision of this load-threatening activity was absent, a lack of specific technical knowledge by the operator, and no pre-evolution brief (St. Lucie-1, 91-006).

5.4.1.11 Susquehanna-2

At 1200 hours on March 15, 1991, with Unit 2 in Condition S (refueling) operations personnel completed Surveillance 50-281-002, "Refueling Platform Main and Auxiliary Hoists Operability." The surveillance requires a control rod to be withdrawn one notch to test operability of the refueling bridge interlock. Per TS LCO 3.1.5, SLC is required to be operable in Conditions 1, 2, and 5' (- with any control rod withdrawn). A control rod may be withdrawn with SLC inoperable if the required actions of the LCO 3.9.10.2 are met. When the surveillance was performed the SLC system was out of service for maintenance, and the requirements for LCO 3.9.10.1 were not satisfied. This condition was determined reportable as an operation prohibited by technical specifications. A historical review determined that a similar event occurred on March 14, 1991. The root cause of these events is attributed to personnel error, in conjunction with several contributing factors. The reactor manual control system withdraw pushbutton had a yellow caution tag attached stating not to withdraw any control rods with SLC inoperable. The control room operators incorrectly evaluated that the restriction on the caution tag did not apply and the surveillance could be performed. There were no safety consequences or compromise to the public health or safety as a result of this event (Susquehanna-2, 91-004).

5.4.1.12 Three Mile Island-1

"GPU Nuclear's Three Mile Island-1 was forced down 51.9 hours when the reactor tripped automatically on low pressure after dropping Group 7 rods" (*Nucleonic Week*, September 12 1991a). On July 24, 1991, at 0520,

the reactor tripped on low RCS pressure due to Control Rod Group 7 dropping into the reactor core as a result of personnel error in combination with an equipment failure. During the performance of a weekly reactor protection system surveillance procedure, an instrument and controls technician failed to follow the procedure, which resulted in a failure to re-energize the primary power supply to the rod drive programmer assemblies. Failure of the "A" phase gate drive of the Group 7 control rods on the secondary power supply, resulted in a degradation of that power supply. During restoration of the integrated control system to auto upon completion of the surveillance, the Group 7 control rods received a command to move. The primary power supply remained in a de-energized state, and the programmer cycled through the failed gate drive on the secondary group power supply. Due to the inadequate number of phases on the secondary group power supply to support rod movement, the Group 7 control rods dropped into the core. The reactor protection system inserted the remaining rod groups when RCS pressure decreased to 1900 psig. The event was reported per 10CFR50.73(A)(2)(iv) (Three Mile Island-1, 91-002).

5.4.1.13 Waterford-3

On August 25, 1991, at 0214 hours, while performing OP-903-107, plant protection system functional test, the plant received a safety injection actuation signal (SIAS), containment isolation actuation signal (CIAS), and main steam isolation signal (MSIS). Subsequently, a reactor trip occurred on loss of power to the control element drive mechanism motor generator sets due to the SIAS. Operations control room personnel entered the emergency operating procedures and carried out the immediate operator actions. Recovery was performed in accordance with OP-902-006, loss of rain feedwater procedure. The root cause of the event was an unforeseen design problem. When the matrix relay hold pushbutton (Cutler-Hammer Model 10250t506) was pushed, the speed of insertion was rapid enough such that the energization of the test coils occurred before the energization of the hold coil, resulting in an actuation of the SIAS, CIAS, and MSIS functions under test. A modification is being considered by system engineering to replace the pushbutton with a three-position spring return handswitch (Waterford-3, 91-019).

5.4.1.14 Waterford-3

At 0324 hours on November 17, 1991, while performing OP-903-107, plant protection system functional test, the plant received a safety injection actuation signal (SIAS), containment isolation actuation signal (CIAS), and main steam isolation signal (MSIS). Subsequently, a reactor trip occurred on loss of power to the control element drive mechanism motor generator sets due to the SIAS. Operations control room personnel entered the emergency operating procedures and carried out the immediate operator actions. Recovery was performed in accordance with OP-902-006, loss of main feedwater procedure. Inspection of the test cabinet revealed a loose wire connection in the CD matrix test circuitry. The loose wire connection was tightened and a system retest was performed satisfactorily. The cause of this event was a test circuit malfunction that was caused by a loose connection on TB701 Terminal 2. The cause of the loose connection could not be positively determined and may have existed for some time. Waterford 3 is implementing a design change to install test circuit enhancements (Waterford-3, 91-022).

5.4.2 Control Rod Operation

Reported events that involved the movement or operation of control rods are listed in this section. There were 18 events, or items of interest, identified in the United States involving control rod operational problems.

5.4.2.1 Arkansas Nuclear-2

Unit 2 of the Arkansas Nuclear One was operating at 100% power. On August 23, 1991, at 2200 hours, while performing a monthly surveillance which exercises the control element assemblies to demonstrate that they are movable, CEA Group 6 (consisting of 5 CEAs) failed to respond to control room demands. The licensee entered a TS action statement requiring that the unit be placed in Hot Standby within 6 hours with more than one CEA inoperable. The licensee requested a temporary waiver of compliance to allow up to 72 hours to correct the problem. The waiver was granted by the Region based on the affected CEAs being capable of fulfilling their primary safety function by remaining trippable and aligned. Instrumentation and control technicians determined the cause of the failure to be a defective relay card in the Control Element Drive Mechanism Control

System. The card was replaced and the TS action statement was exited at 0118 hours on August 24. Without the temporary waiver of compliance, the unit would have been required to reduce power and shut down within 6 hours had the problem not been corrected. On July 24, the licensee had met with the staff to discuss draft guidance on a proposed TS line item improvement, based on a TS amendment request from the licensee, to allow continued plant operation for 72 hours for diagnosis and repair, with more than one CEA inoperable provided that the affected CEAs remain trippable and aligned. The licensee is to submit revised TS pages, based on the agreement reached at the meeting, for staff approval of the proposed TS amendment (*Atomic Energy Clearing House*, September 13, 1991).

5.4.2.2 Arkansas Nuclear-2

On January 18, 1991, at 1415, while performing CEA current traces, Group 6 CEA 46 dropped to its fully inserted position. Operations personnel reduced power to 97.5% in accordance with the "CEA misalignment" procedure (AOP 2203.03). At 1429, after determining that the cause of the dropped rod had been the inadvertent opening of its circuit breaker, the operators commenced withdrawing CEA 46 to realign it with the rest of the Group 6 CEAs. Reactor power was held constant (97.5%) during the recovery effort, as directed by AOP 2203.03. At 1443, CEA 46 reached realignment with group 6. A subsequent evaluation determined that the time-dependent reduced power requirements of technical specifications regarding dropped CEAs were not adhered to during the recovery. Technical specifications required that during the period of recovery, power should have been reduced by 5.8%. The root cause of this event was determined to be inadequate procedural guidance. Procedure AOP 2203.03 was ambiguous with respect to reduced power requirements. This event was discussed with the operations crews. A procedural change was implemented to remove the ambiguities regarding reduced power requirements (Arkansas Nuclear-2, 91-003).

5.4.2.3 Beaver Valley-1

On July 18, 1991, the plant was critical in Mode 2 while the testing and plant performance group was performing their core design check. Test procedure criticality had been achieved by the withdrawal of Control Bank D rods

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to 215 steps. Control bank rods were then moved in and out several times, per the procedure, in order to determine an isothermal temperature coefficient (ITC). At 0445 hours an ITC was determined and found to be within the acceptance criteria. In order to prepare for the next portion of the test procedure, Control Bank D was to be withdrawn to 228 steps, and Control Bank B was to be inserted while maintaining the reactor critical. At 0450, Control Bank D was withdrawn, adding a small amount of positive reactivity, but when Control Bank B was selected and demanded to insert, the rod control system urgent alarm came in. Only Group 2 of Control Bank B began to move. The reactor operator then bled approximately 50 gal to stop the power increase. This took the reactor subcritical and brought the plant into Mode 3. The test was suspended. An investigation revealed that a fuse and its fuse indicator were blown in the 1BD power cabinet. At 0723, all control rods were inserted by manually opening the reactor trip breakers. There were no safety implications as a result of this event. The shutdown margin was verified, and the control rods were able to be tripped (Beaver Valley-1, 91-021 Rev. 1).

5.4.2.4 Braidwood-1

On July 17, 1991, Surveillance 1BWOS 1.3.1.2-1, "Unit One Movable Control Assemblies Monthly Surveillance," was started to verify operability of each control rod assembly. At this time, all shutdown (S/D) banks (A,B,C,D and E) were fully withdrawn. Shutdown Banks A and B have eight control rods and are separated into two groups with four rods in each group. During testing of S/D Bank A, problems were encountered with the control rods in Group 1. No rod motion was indicated for two of the Group 1 rods. The other two rods in Group 1 were not fully withdrawn. At 0419, TS 3.0.3 was entered. This started a 6 hour time clock to enter Mode 3 (hot standby). At 0645, a unit shutdown commenced. Since this shutdown was required by technical specifications, an unusual event was declared in accordance with the Braidwood Station emergency plan. Troubleshooting revealed that a malfunctioning circuit card was not providing proper current to the control rod drive coils. The power reduction was stopped at 0926. At 0940, the circuit cards were replaced. At 0943, S/D Bank A Group 1 rods were realigned. The rods were satisfactorily tested and were declared operable at 1002. The unusual event classification was terminated at 1003.

The cause of the event was component failure. A defective transistor in the firing circuit control card was not generating the proper frequency (Braidwood-1, 91-007).

5.4.2.5 Braidwood-2

"Commonwealth Edison's Braidwood-2 was forced down for 86.1 hours due to a reactor trip caused by rod control drive system failure." (*Nucleonics Week*, October 10, 1991c) Unit 2 was operating at 88% reactor power in a coast down mode at the end of core life. On August 1, 1991, at 2354, the unit operator attempted to move the control rods in Bank D to control reactor coolant system temperature. A rod drive (RD) system failure released numerous control rods and allowed them to fall into the core. A reactor trip signal was generated because the nuclear instrumentation (NR) system detected a rapid power decrease. Control room personnel immediately recognized and responded to the reactor trip. The unit was stabilized with normal post-trip response conditions. During RD troubleshooting, the logic cabinet was analyzed to determine the amount of electric current being supplied to the stationary and movable grippers. The 2BD slave cyclor was intermittently giving erratic current demand orders. The rods dropped because the stationary gripper was unable to hold the rods before the moving gripper unlatched. The cause of the 2BD slave cyclor failure was a defective binary counter. The counter is required to properly sequence the energization/de-energization of the movable and stationary grippers during rod motion. The 2BD slave cyclor binary counter circuit card was replaced. The logic cabinet slave cyclor circuits were tested and analyzed for proper operation (Braidwood-2, 91-003).

5.4.2.6 Brunswick-2

On September 27, 1991, at 0443 Unit 2 was in a refueling outage with the reactor core partially off loaded to the SFP. All control rods were fully inserted and the CRD system was secured except for a temporary mechanical jumper to supply cooling water while CRD system maintenance was being performed. Several hours earlier, as the CRD system was being prepared to return to service, the reactor protection system (RPS) trip had been reset. The unit's senior reactor operator (SRO) while touring the reactor control panels found the scram discharge volume (SDV) hi-hi level trip

bypass switch was still in the bypass position with both SDV not drained annunciators cleared. As the reactor operators (RO) were busy, the SRO returned the switch to the normal position himself, without informing the ROs. The SDV hi-hi level RPS trip, whose annunciator he had not verified, resulted in the full RPS trip. There is no safety significance to this event, as the RPS responded as expected and no control rod motion occurred since all control rods were previously fully inserted. The longer reset times of the SDV electronic level switches does not affect their operability. Previous related LERs are 1-88-031 and 1-87-010 (Brunswick-2, 91-014).

5.4.2.7 Cook-1

On August 19, 1991, during routine surveillance testing of the full length rods, control Bank A was ordered into the core by the rod control system but appeared not to move into the core by observation of the analog rod position indication. Further positioning of control Bank A resulted in an urgent alarm being received. Initial troubleshooting of the rod control system found no reason for the urgent alarm and it was reset. The rods were moved per the test requirements and the surveillance completed. At approximately 1200, a rod misalignment in shutdown Bank A, Group 2 was suspected due to additional small amount of dilution required to keep thermal power at the pre-surveillance level and a decrease in the analog rod position indication for shutdown Bank A, Group 2 in relation to pre-surveillance readings. At 1305 hours the flux mapping system was used to determine that the four rods of shutdown Bank A, Group 2 were inserted approximately six steps into the core. Further troubleshooting of the rod control system discovered a failure of the 2AC power cabinet multiplexing relay, MXR-1. It was replaced and proper operation of the rod control system was verified (Cook-1, 91-007 Rev. 1).

5.4.2.8 Dresden-2

At 2147 hours on August 5, 1991, with Unit 2 at 98% rated power, during main turbine electro-hydraulic control (EHC) low pressure scram switch calibration per Dresden Instrument Surveillance (DIS) 500-8, ENC low pressure scram switch calibration, CRD F-3 suddenly inserted from Position "48" to Position "34." Previous to the unplanned rod insertion, a channel a 1/2 scram

signal had been received and reset per DIS 500-8. Upon initiation of a channel B 1/2 scram signal, CRD F-3 suddenly inserted. Shortly after rod insertion, operations personnel reported chattering in one of the two scram pilot solenoids (305-117 or 305-118). DIS 500-8 was suspended and the control rod was manually inserted to Position "00" and electrically disarmed. Troubleshooting indicated that the rod insertion was caused by air leakage past the scram pilot solenoid valves. The scram pilot solenoid valves were therefore replaced and CRD F-3 was placed back in service. Further inspection will be conducted on the pilot solenoid valves that were removed to confirm the failure mode. Safety significance was minimal because no core thermal and hydraulic limits were challenged and the scram function of CRD F-3 was not affected. Review of the system history records indicates that this type of event has not been an adverse trend (Dresden-2, 91-022).

5.4.2.9 Farley-2

"Alabama Power's Farley-2 was forced down for 224.5 hours due to a dropped rod caused by a failed circuit card, electro-hydraulic fuel line, and closure of an instrument air valve" (*Nucleonics Week*, June 13, 1991). At 1055 on April 1, 1991, while operating at approximately 100% power, a reactor trip occurred when rod H-10 dropped into the core. The reactor trip occurred due to a high negative flux rate as detected by the power range nuclear detectors. The operator was performing FNP-2-STP-5.0 (Full Length Control Rod Operability Test). When control rod group C was tested, rod H-10 dropped into the core. This event was caused by defective circuit card(s) in the rod control system. The suspect cards were replaced and the unit returned to power operation at 1208 on April 9, 1991 (Farley-2, 91-001).

5.4.2.10 McGuire-2

On July 12, 1991, operations personnel and instrumentation and electrical (IAE) personnel were performing the quarterly calibration on power range neutron flux channels for Unit 2 to correspond with current incore/excore correlation values. During this calibration, a quadrant power tilt ratio (QPTR) alarm was received after the first neutron flux channel was returned to service. To comply with technical specifications, a load reduction to 91% power was required. During the subsequent power reduction, Group 1 of control rod

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Bank D failed to move properly causing a control rod misalignment greater than TS limits. To comply with TSs, boron was added and the control rods were realigned. An axial flux difference (AFD) oscillation resulted from the movement of the control rods and power reduction. Since the control rods would not move properly they were unavailable to dampen the oscillation. IAE personnel attempted to repair the control rod problem during the next day, but were unable to do so. At 0658, on July 14, 1991, operations personnel initiated a manual reactor trip prior to exceeding its limits for AFD. Operations personnel implemented the reactor trip recovery procedure to recover from the transient. Unit 2 was in Mode 1 (power operation) at 90.5% power at the time of the reactor trip (McGuire-2, 91-007).

5.4.2.11 McGuire-2

On November 8, 1991, at approx. 1142, Operations control room personnel were performing the monthly procedure PT/2/A/4600/01, rod cluster control assembly movement test. Shutdown Banks A and B moved properly. However, shutdown Bank C dropped into the core as the reactor operator at the controls attempted to step this bank in. The solid state protection system subsequently generated a negative flux rate trip signal which automatically tripped the reactor. Operations control room personnel also initiated a manual reactor trip as required by the reactor trip procedure. Diesel generator 2B was running at the time and the emergency breaker for diesel generator 2B opened due to an overload when a low-low level in steam generator A caused the auxiliary feedwater and nuclear service water pumps to start. Unit 2 was in Mode 1 (power operation) at 100% power prior to the trip. The unit performed as expected during the transient and there were no nuclear safety significant problems. An investigation was performed and it was determined that an electronic equipment failure within the shutdown Bank C input logic circuit had caused the stationary rod gripper coils to stay de-energized too long. An independent safety review was performed and it was determined it was safe and prudent to restart the reactor. Unit 2 was returned to Mode 1 on November 9, 1991, at 0510 (McGuire-2, 91-012).

5.4.2.12 Millstone-1

On August 16, 1991, while at 23% power and 530°F, eight control rods exhibited significant delays in scram insertion times and approached the technical specifica-

tion requirement for a maximum scram time of 7 seconds for 90% insertion. Subsequent review of the two control rods previously reported as failing the 90% insertion time indicated that, although the full insertion times exceed 7 seconds, the 90% insertion times were acceptable. However, a 2 x 2 array was identified which would have failed the 2 x 2 array analysis. The affected array contained two of the slow control rods and would have failed the technical specification requirements for the 5, 20, and 50% times for the 3 fastest rods of a 2 x 2 array. The slow rods were identified during the performance of beginning of cycle control rod scram time testing. As each was identified, the control rod was left fully inserted and electrically disarmed. The delayed scram was determined to be caused by the slow venting of air from the scram pilot solenoid valves (117 and 118) located on the associated hydraulic control units. The solenoid valves were rebuilt and the control rods scram timed. All affected control rods successfully passed with nominal scram times and were returned to service. There were no safety consequences from this event (Millstone-1, 91-025 Rev. 1).

5.4.2.13 Oconee-3

"Oconee-3 was forced down for 15.38 hours when control rods dropped into the core." (Nucleonics Week, August 8, 1991) On June 9, 1991, with Unit 3 at 100% full power, a CRD system verification test required by technical specifications was being performed. At 1506 hours, after transferring a single control rod in group 5 to its auxiliary power supply, the entire rod group fell into the core. Reactor power dropped to approximately 30% full power prior to a reactor trip three seconds later on low pressure/temperature ratio. The plant post-trip response was normal. The unit was stabilized at hot shutdown conditions. The root cause of the event was found to be equipment malfunction: a failed transfer switch allowed the normal and alternate CRD power supplies to simultaneously energize the group 5 CRD mechanisms. The failed transfer switch was replaced. Further investigations are centered around the history of the transfer switch, methods to prevent its failure, and methods to give the operator indication when power supply transfer failures occur (Oconee-3, 91-006 Rev. 1).

5.4.2.14 Palo Verde-1

On September 14, 1991, at approximately 1722 MST, Palo Verde Unit 1 was in Mode 1 (power operation) at approximately 55% power when a reactor trip and a

main steam isolation signal (MSIS) engineered safety features actuation occurred due to high steam generator level. Prior to the reactor trip and MSIS, reactor power had been reduced as a result of a part-length control element assembly slipping part way into the core during monthly testing. This power reduction was required by technical specifications. There were no other safety system responses required. The reactor trip was diagnosed as an uncomplicated reactor trip. The plant was stabilized in mode 3 (hot standby). The reactor trip and MSIS on high steam generator level was due to the Number 1 steam generator economizer valve failing open which resulted in overfeeding the steam generator. The economizer valve failure was due to a failure of the economizer valve positioner. Troubleshooting was performed, the ability to withdraw the part-length CEA was restored, the economizer valve positioner replaced and tested, and both were returned to service. There have been no previous similar events reported pursuant to 10CFR50.73 (Palo Verde-1, 91-009).

5.4.2.15 Perry-1

On October 6, 1991, at 1037, control rod 46-23 exceeded its maximum scram time to Position 43. Operators then took action to demonstrate that the eight adjacent control rods surrounding control rod 46-23 could satisfy the maximum scram insertion time limits. Control rod 42-19 also exceeded its maximum scram insertion time to Position 43. Because control rods 46-23 and 42-19 are adjacent rods, the plant was required by technical specifications to be in at least Hot Shutdown within 12 hours. On October 6, 1991, at 1436, operators commenced a shutdown of the plant. The cause of the "slow" control rods is attributed to component failure. The Automatic Switch Company (ASCO) Model Number HVA176-816-1 Scram Solenoid Pilot Valve (SSPV) for each of the affected control rods was determined to be the cause of the "slow" control rods. A combination of contaminants found on the valve disk and seats is believed to have formed an adhesive which could have bound the valve seat. All of the suspect SSPVs were from the same lot remanufactured by ASCO in November 1990. After shutdown of the plant, all 49 of the SSPVs from lot number 184010001 were removed from their associated HCUs and replaced. A maintenance instruction will be revised to further restrict the contaminants that may have been introduced at Perry. As

part of the established requalification training program, all plant licensed operators will be instructed on the lessons learned from this event (Perry-1, 91-018 Rev. 1).

5.4.2.16 Prairie Island-1

At 0127 hours on August 16, 1991, Unit 1 tripped unexpectedly from 100% power. The "first out" annunciator was the negative flux rate trip. Review of rod position indication just prior to and after the trip indicated that the four control Bank A, Group 2 rods that are powered from power cabinet 2AC began to fall into the core prior to receipt of the reactor trip signal. Extensive testing of the rod control system was done. Testing of the failure detection circuitry (urgent failure alarm) showed some conditions exist under which some rods would unexpectedly drop. The system does not provide protection against dropping rods for all component malfunctions or failures. The most likely cause of the event was intermittent failure in a firing card for the stationary grippers supplied by power cabinet 2AC, causing interruption of power to the control Bank A, Group 2 rods. During post-trip testing which simulated high temperatures found on firing components during normal operation, the card developed an unacceptable wave form in an oscillator circuit. This card had previously tested satisfactorily at room temperature. The intermittent control circuit failure occurred such that the failure detection circuitry did not protect the affected rods against dropping (Prairie Island-1, 91-011).

5.4.2.17 Robinson-2

At 0741 hours on March 19, 1991, with no apparent warning and no control rod motion in progress, a rod control system urgent failure alarm was received. The condition was the result of failed printed circuit boards (PCB) in the phase control and firing control circuits for the control Bank C, Group 2 stationary gripper control in power cabinet 2AC. The urgent failure alarm within power cabinet 2AC prevented all overlap rod motion within that power cabinet, which resulted in the inoperability of 12 of the 45 control rod assemblies. Since this condition exceeded the requirements of the technical specification LCO, TS 3.0 was entered. The failed PCBs were replaced, and the affected control rods were successfully tested and declared operable at 1200 hours. This event is reportable pursuant to

10CFR50.73(A)(2)(i)(b), in accordance with the guidance provided within NUREG-1022, Supplement 1, as a condition prohibited by the plant's technical specifications (Robinson-2, 91-004).

5.4.2.18 San Onofre-1

At 2343 on May 28, 1991, with Unit 1 in Mode 1 at 91% power, two of the 12 control bank 1 control rods (H-6 and H-10) dropped into the core. Appropriate control room alarms were received and operators confirmed negative reactivity insertion per procedures. Operators properly responded (within 20 seconds) by manually tripping the reactor. At 2344, the auxiliary feedwater (AFW) system actuated on low steam generator (SG) water level which occurred due to expected SG water level shrinkage. At 0044 on May 29, 1991, the reactor trip response procedure was exited with the plant stabilized in mode 3. The control rods dropped as a result of an open connection in the "half-voltage" resistor network associated with this rod subgroup. The open connection de-energized the moveable gripper coils associated with the affected rods, thus permitting them to drop into the core. The open connection was caused by a conductor/lug (jumper) connection failure at one termination point. Destructive physical analysis has revealed that the open connection resulted from long-term localized heating resulting in oxidation of the copper wire in the vicinity of the failure site. Continued heating led to arcing melting, and eventual failure. The failed jumper was replaced with in-kind parts. In addition, other jumper wires/lugs within the control rod "half power" resistor bank were replaced in-kind (San Onofre-1, 91-010).

5.4.3 Control Rod Position Indicator

To facilitate the operation of a reactor an indicator is used to note the position of the control rod. The reported events that involved malfunctioning or defective indicators are listed in this section. There were ten events in the United States that were identified in the literature involving control rod position indicators.

5.4.3.1 Beaver Valley-2

On 4/20/91, after inserting control rods 14 steps to support reactor engineering, the operators noted the computer point for average control rod position disagreed

with the point that summed total rod steps by more than the allowed 12 steps. This should have activated the rod position deviation alarm, but no alarm had occurred. The computer engineer was consulted and determined that the computer's rod position deviation alarm had been bypassed, preventing the alarm from annunciating. Further investigation determined that the alarm had inadvertently been bypassed on 4/12/91 during a computer cold start/reboot sequence. On that date, the computer engineer performing the reboot, reinitiated the reboot procedure before the initial startup sequence was complete. By terminating the first startup prior to completion, the alarm bypass was activated and not restored by the second reboot. The station has modified the computer software to provide an alarm to the reactor operator whenever this or similar points are bypassed. All personnel in the computer group are being trained on the potential effects of terminating a computer startup prior to its completion. Review of the manual operator rod position logs show that no rod position deviation had occurred while the rod deviation alarm was inoperable (Beaver Valley-2, 91-001).

5.4.3.2 Catawba-1

On July 30, 1991, at approximately 1115 hours, with Units 1 and 2 in Mode 1, power operation, and operating at 100% power, operations personnel were notified by performance (PRF) personnel that both unit's rod deviation monitors (RDMS) were functionally inoperable. On July 25, 1991, production support department (PSD) personnel informed the station PRF personnel that a logic error discovered in the operator aid computer (OAC) rod position alarm monitor (RAM) program at McGuire Nuclear Station affected the RDMS at Catawba Nuclear Station also. Unit specific technical specification operability notification (TSO) sheets were issued on July 30 declaring the monitors inoperable and Operations began complying with the technical specification action and surveillance requirements. This incident is attributed to a management deficiency because no policy requiring functional verification of OAC programs existed during the initial startup of the units. Corrective actions included PSD personnel reprogramming the RAM programs and verifying that the RDMS operated properly. Operations declared the Unit 2 and 1 RDMS operable per TSO sheets on July 30, at 1500 hours, and July 31, at 1200 hours, respectively (Catawba-1, 91-016).

5.4.3.3 Crystal River-3

On September 5, 1991, Crystal River Unit 3 was operating at 100% rated thermal power and preventive maintenance on control rod drive power supplies was being conducted. During the course of this maintenance, three power supplies for control rod position indication were adjusted. After the maintenance was completed, several plant computer alarms were received. Based on conflicting computer and main control board indications and computer problems experienced on the previous day, the computer alarms were believed to be invalid. Approximately one hour later, control rod exercise surveillance testing was begun. The performance of this procedure led operators to determine that multiple absolute position indication (API) channels were outside the required 2% accuracy and, therefore, were declared to be inoperable. Technical Specification 3.0.3 was entered. The shift supervisor on duty (SSOD) ordered the utility electricians to restore the power supplies to the previous as found readings. API operability was restored and TS 3.0.3 exited. Later investigation revealed that guidance provided in the maintenance procedure for measuring power supply voltage was not sufficient to preclude the three power supplies from being improperly adjusted, thereby altering API. The procedure will be revised to ensure proper measurements in the future (Crystal River-3, 91-008).

5.4.3.4 Diablo Canyon-1

From approximately 1615 PST on November 14, 1991, to 2305 PST on November 16, 1991, conditional surveillances of the control rod positions required by TS 4.1.3.1.1 and TS 4.1.3.2, including the allowed extension of TS 4.0.2, were missed. Control room operators were unaware that the rod position deviation monitor (RPDM) of the plant process computer (PPC) was inoperable. On November 17, 1991, licensed operators completed surveillance test procedure (STP) I-42, "Rod position deviation monitor functional test," and discovered that the RPDM failed to initiate an alarm on the main annunciator panel, the PPC typewriter, and the PPC alarm video screen at the appropriate procedural step. Operations declared the RPD inoperable and initiated four-hour conditional surveillances in accordance with TS 4.1.3.1.1 and 4.1.3.2. The four-hour surveillances were missed from approximately 1615 PST on November 14, 1991, to 2305 PST on November 16, 1991. During that time, rod positions were checked every

12 hours as required by TS when the RPDM is operable. The immediate cause of this event was that after the PPC was shut down and restarted on November 14, the RPDM logic inhibited the alarm function because the PPC incorrectly statused the reactor trip breakers as open (Diablo Canyon-1, 91-017).

5.4.3.5 McGuire-1

On July 19, 1991, at 1657, operation control room personnel were informed that the operability of Units 1 and 2 rod deviation monitors were questionable. Prior to this event, licensee event report 370/91-05 documented a manual reactor trip on Unit 2. The investigation of the Unit 2 reactor trip revealed that Groups 1 and 2 of control rod Bank D had become misaligned by greater than 12 steps and no rod deviation alarms were received. Operations personnel declared Units 1 and 2 rod deviation monitors inoperable and issued a work request to investigate and repair the rod deviation monitor problem. During the investigation of the rod deviation monitor problem, production support personnel discovered Units 1 and 2 rod deviation monitors had been inoperable since December 7, 1983. Production support personnel corrected the computer program procedure and verified the rod deviation monitors operated properly. This event resulted from defective procedures because the procedure used to program the rod deviation monitor was in error. This event also resulted from a management deficiency because a procedure was not in place that required operator aid computer programs to be functionally verified when installed or modified. A procedure to functionally verify operator aid computer program installation and modifications had been developed and implemented prior to the discovery of this event (McGuire-1, 91-010).

5.4.3.6 Nine Mile Point

A loss of electrical power at Niagara Mohawk's Nine Mile Point on August 13, 1991, was started by a fault in a main transformer. As a result, "uninterruptable power supplies" tripped with a loss of some control room visual displays and sound annunciators (*Atomic Energy Clearing House*, August 30, 1991). In addition to other plant instrumentation and systems being disabled, the control rod position indicator was out. Operators manually shut down the reactor and the event was declared a "site-area emergency", the second highest level of alert (*Atomic Energy Clearing House*, August 16, 1991b).

Problem Areas Observed During 1991

5.4.3.7 Sequoyah-2

On March 29, 1991, at 1142 EST, with Unit 2 in Mode 1, TVA identified the plant process computer P-250 computer point C0019A (Control Bank C, Group 2, Position K-10) was out-of-scan. The computer point monitors a control rod cluster position, which provides input into the "Rod deviation and sequence/NIS power range tilts" alarm. Surveillance Requirement 4.1.3.2 requires the comparison of the demand position and rod position indication systems at least once every four hours if this alarm is inoperable. The computer point was taken out of scan on March 21, 1991, while troubleshooting the "rod deviation and sequence/NIS power range tilts" alarm, which came in during the performance of periodic instruction (PI) O-PI-NXX-085-001.0, E-0, "Bimonthly resetting of control rod fully withdrawn position." The cause of the event is failure to return the point to scan: configuration control during troubleshooting of the alarm was not implemented. Immediate corrective action was to place the computer point back into scan. Additional corrective actions include discussions with personnel concerning the importance of configuration control and implementation of a method to maintain configuration control (Sequoyah-2, 91-004 Rev. 1).

5.4.3.8 Vogtle-1

On November 18, 1991, Unit 1 was in Mode 3 (hot standby). The reactor trip breakers were closed, and withdrawal of Shutdown Bank (SDB) "A" was in progress per procedure 88006-C, "Rod Drop Time Measurement with Rod Drop Test Cart." After withdrawing SDB "A" to 53 steps, it was noticed that the digital rod position indication (DRPI) for SDB "A" Rod D-2 had stopped moving. Shutdown Bank "A" was further withdrawn, and at group step counter readings of 59 and 58, DRPI urgent alarms were received and the rod at bottom led illuminated for Rod D-2. The operators thought they had also observed indication of Rod D-2 dropping into the core. The abnormal operating procedure (AOP) for rod control system malfunctions was entered, and at 1010 EST, a reactor trip was conservatively initiated as discussed in evolution prebriefings and as directed by the AOP. It was verified that the reactor trip breakers opened properly, and the SDB "A" rods fully inserted into the core. Subsequent troubleshooting indicated Rod D-2 did not actually drop into the core

prior to initiation of the reactor trip. Rather, a DRPI data problem was indicated to exist. While the exact cause of this problem was not determined, the operability of the DRPI system per technical specification requirements was demonstrated and a reactor startup was successfully completed on November 19, 1991 (Vogtle-1, 91-012).

5.4.3.9 Vogtle-2

During the period between 1518 Central Daylight-Saving Time (CST) on January 2, 1991, and 0743 CST on January 7, 1991, the rod position deviation monitor alarm was inoperable and the special condition surveillance requirement of TS 4.1.3.2, which is applicable when the monitor is inoperable, was not implemented. The TS requires that the demand position indication system and the digital rod position indication system be compared at least once per 4 hours when the rod position deviation alarm is inoperable. The cause of this event was a cognitive personnel error on the part of the balance of plant (BOP) operator. The BOP operator failed to follow procedure on 1-2-91 when he improperly entered a computer point value. Contributing to this event was the failure of a relay output circuit card in the interface between the DRPI and the Proteus computer. The failed relay output circuit card has been replaced and the BOP operator has been counseled regarding the importance of procedural compliance (Vogtle-2, 91-001).

5.4.3.10 Wolf Creek-1

During the period of July 30, 1991, at approximately 0900 CDT through July 31, 1991, at approximately 0748 CDT, the 4-hour comparison of control rod demand position to digital position indication was not performed as the result of misinformation concerning the operability of the rod position deviation monitor. This condition is contrary to TS SR 4.11.3.2. Although the monitor was not operable, the annunciator fed by the rod position deviation monitor would have alarmed correctly. The misinformation occurred because personnel had overlooked four failed pulse counter cards. When the cards failed on July 18, 1991, a corrective work request was not initiated. The individuals involved have been counseled and procedural guidance will be enhanced (Wolf Creek-1, 91-012).

5.4.4 Control System Failure

This section lists the reported events that were caused by failure of the control rod drive system. There was one event in the United States identified in the literature involving control system failure.

5.4.4.1 San Onofre-1

At about 0430 on April 16, 1991, with Unit 1 at about 90% power and after successful completion of routine technical specification surveillance testing to determine control rod operability, a rod control (RC) system interlock relay (which prevents simultaneous control signals for insertion and withdrawal of control rod Banks 1 and 2) failed. The failure mode of the relay effectively caused the RC system logic to be in the withdraw mode, thereby precluding RC system control rod bank insertion. At the time of the failure, the position of all rods and banks were within the limits of applicable TSs. Furthermore, reactor power and power distribution were within the limits of applicable TSs. The unit remained fully capable of being automatically or manually tripped by the reactor protection system. Continued operation in the existing configuration was considered to be safer during the brief period required to effect repairs than to place the unit in a shutdown transient which would normally be performed with control rod insertion. Accordingly, a temporary waiver of compliance was requested and approved for a period of 36 hours, commencing at 0430 on April 16, 1991, and ending at 1630 on April 17, 1991. This provided sufficient time to replace the faulty relay and to demonstrate that the RC system functions as designed (San Onofre-1, 91-008).

5.4.5 Control Rod Swelling, Wear, Corrosion Cracking

Control rod swelling is caused by irradiation induced damage resulting in excessive stress to the control rod and leads to failure. Control rod wear is caused by control rod movement and also vibration caused by flow of the coolant. Corrosion cracking is induced by cracking of the cladding material and leads to failure. The reported events that were caused by these problems are listed in this section. There were three events in the United States involving control rod swelling, wear, corrosion, or cracking identified in the literature.

5.4.5.1 Calvert Cliffs-1

Baltimore Gas & Electric has requested an emergency TS change to allow continued operation for Cycle 10 of the Calvert Cliffs Unit 1 reactor with the center control rod excluded from operability requirements. Its failure to insert fully during a recent shutdown (and swelling observed in previously removed related rods) has raised doubts about its future movements. This is a "weak" rod with only one of its five fingers containing control material, and is located in a twice burned fuel assembly. The sticking is believed to be caused by swelling of the Zr slug at the ends of the unpoisoned fingers. This phenomena has been observed in other unpoisoned control-rod fingers in other reactors. The swelling is limited to the Zr slugs, which are only present in this particular rod. Since the control rod may cease moving at any axial position, BG&E has evaluated the reactivity and power distribution affects of the center rod position on normal operation, transients, and accidents. These evaluations have concluded that no significant deviations from previous analyses will occur. The staff review has concurred with this conclusion. The rod will be removed during the next reload (*Atomic Energy Clearing House*, March 1, 1991).

5.4.5.2 Monticello

During the 1991 refueling outage, 37 of 72 control rod drive housing flange cap screws that were examined, displayed linear indications just below the cap screw head. Upon discovery of the indications, a Safety Review Item addressing safety concerns was prepared. Four new, original equipment cap screws were installed in each control rod drive housing flange. A reactor vessel hydro was performed. Cap screws were sent to three labs for testing to determine the mode of failure. All labs found similar shallow, rounded, oxide-filled indications. These indications showed no signs of active crack propagation. In addition, all tensile tests done on bolts exhibiting the most extreme indications showed no decrease in tensile strength. Capscrews will be inspected during normal control rod drive maintenance activities, and replaced with original design cap screws as needed (Monticello, 91-008 Rev. 1).

5.4.5.3 Millstone-2

Northeast Utilities Company's Millstone-2 unit considered operating with replacement CEAs that have been known to experience center finger failures and circumferential cracking." The ABB CENF CEAs first encountered problems at Maine Yankee in 1990. Investigation of a stuck CEA "revealed three CEAs had missing center finger end cups and one had a circumferential crack in the center finger end cap weld area." Because of the Millstone-2 reactor design, CEAs in use there have a "significantly lower neutron exposure than in other plants" and therefore are not as susceptible to boron carbide swelling (*Nuclear Fuel*, July 8, 1991).

5.4.6 Miscellaneous - Control Rod Oriented

Unknown events or events that do not fall under one of the above specific categories are listed in this section. There was one item of interest relating to a control rod drive systems.

The Nuclear Plant Aging Research (NPAR) Program completed Phase 1 of the aging phenomena associated with boiling-water-reactor (BWR) control rod drive mechanisms (CRDMs) study (Greene 1992). The study, sponsored by the Office of Nuclear Regulatory Research (NRC), acquired information and data from a questionnaire distributed to each U.S. BWR utility, a CRDM aging and maintenance workshop, and an analysis of Nuclear Plant Reliability Data System (NPRDS) failure data associated with BWR CRDMs. The report also assessed the merits of various methods of managing control rod aging degradation.

Phase 1 of the study concluded that CRDM degradation, other than usual in-service wear, resulted from Graphitar seal embrittlement, fatigue fracture, thermal degradation, collet housing cracking, nitrided surface corrosion, and maintenance errors during changeout and rebuilding activities. The complete report published in NUREG/CR-5699, Vol. 1, entitled *Aging and Service Wear of Control Rod Drive Mechanisms for BWR Nuclear Plants*, also provides valuable recommendations for specific actions that utilities can implement to mitigate CRDM aging.

5.5 Core or Coolant Oriented

This section includes power limit exceeded; unexpected power fluctuation; coolant flow exceeded; lowering of water level; water chemistry issues; boron related issues; debris in coolant; miscellaneous or unknown core/coolant problems.

5.5.1 100% Power Exceeded

Reported events that involved exceeding technical specifications of power levels are listed below. There was one item of interest in the United States that involved 100% power exceeded.

5.5.1.1 Brunswick-1

On July 27, 1991, Unit 1 had completed a power increase to full power. On July 28, 1991, at approximately 0600 hours EDT, during routine surveillance, the shift supervisor observed that the gross megawatt recorder was indicating higher than expected (slightly above 790 MWE) for 100% power operation. The shift technical advisor performed a hand heat balance calculation to check the actual megawatt thermal (MWT) output which revealed that actual power was 100.59% and megawatts thermal (MWT) were 2450. (The plant is licensed for 2436 MWT.) Reactor power was reduced to approximately 2400 MWT. Investigations revealed that a feedwater (FW) inlet temperature point was inputting a high value into the process computer heat balance calculation indicating an erroneous power calculation (99.92% with actual power at 100.59%). An average value of the three other FW inlet temperature points was substituted for high point which resulted in an accurate power calculation by the process computer. The cause of the event was due to a failed temperature element which will be replaced during a unit outage when parts are available. The safety significance of this event was minimal due to the fact that the unit did not exceed 102% power (Brunswick-1, 91-021).

5.5.2 Axial Shape Index Related

This section lists events that were caused by an axial power shape that was exceeded. The axial power shape is the amount of power that is produced along the vertical axis of the core. There was one event in the United States that involved axial shape index problems. The event is described below.

5.5.2.1 Palo Verde-2

At approximately 0648 MST on August 9, 1991, Palo Verde Unit 2 was in Mode 1 (power operation) at approximately 40% power when the reactor was manually tripped due to axial shape index (ASI) approaching its trip setpoint on the core protection calculator. At the time of this event, the reactor was being shut down due to the loss of control element drive mechanism cooling fans. Following the trip, two of the four reactor coolant pumps (RCPS) did not transfer to their alternate source of power and stopped. The remaining two RCPS supplied adequate forced circulation. The reactor trip was diagnosed as an uncomplicated reactor trip. No safety system responses occurred and none were required. At approximately 0658 MST the plant was stabilized in Mode 3 (hot standby) at normal temperature and pressure. The cause of the manual reactor trip was ASI approaching its trip setpoint during shutdown. The corrective action was to trip the reactor and ensure ASI remained within limits. Previous similar events were reported in LER 528/87-018-01 and LER 529/90-001-00 (Palo Verde-2, 91-003).

5.5.3 Lowering of Water Level

Reported events that were caused by low water in the core are listed in this section. There were two events in the United States that involved lowering of the water level.

5.5.3.1 Hope Creek-1

On May 7, 1991, at 2102, a reactor scram occurred due to a low reactor water level. Conditions leading to the scram were the 'A' feedwater level control indication alarming with a high level, causing the feedpumps to back down. This was followed by the 'C' channel alarming at 36" and a subsequent reactor scram. All control rods were verified to be inserted and plant systems responded as expected, with minor exceptions as noted

in the text of this report. Follow-up investigation determined that the most probable cause of the event was personnel error when a controls technician inadvertently connected a current source to the wrong transmitter while performing a surveillance test. Although the most probable root cause was personnel error, a less than adequate cabinet design also contributed to the event. Immediate corrective actions included counseling the controls technician with respect to self-verification and attention to detail and implementing a design change to install test switches with input jacks and to label the input jacks with a channel designator. The surveillance procedure was revised in accordance with the design change and is being reviewed for additional enhancements (Hope Creek-1, 91-008).

5.5.3.2 Vogtle-1

On October 26, 1991, false instrument readings caused operators to lower the reactor water level four feet below the reactor flange, instead of the normal two feet. The water temperature rose 17°F because an intake for a residual heat removal pump was above the water level. An NRC augmented inspection team (AIT) found that a "sufficient safety margin existed." The AIT team also found that the improper installation of the instrumentation and a deficiency in the procedures were also responsible (*Inside N.R.C.*, December 16, 1991).

5.5.4 Water Chemistry

This section lists events that were caused by improper water chemistry in the reactor coolant. There was one item of interest in the United States and one item of interest in France that involved water chemistry.

5.5.4.1 Testing Higher Water pH in French PWRs

Electricite de France is currently testing a higher water pH in six French PWRs to investigate the potential for reductions in dose rates. The higher pH of 7.2 (versus 7.0 for the EdF standard) should result in up to 40% reduction in the dose rate measured in the primary coolant pipes and up to 28% reduction in the hot channel head dose rate. Another advantage to the higher pH appears to be smaller crud deposits on fuel elements (Beslu et al. 1991).

5.5.5 Boron Related Problems

Improper boron concentrations in the core were the cause of the events listed in this section. There were two items of interest relating to boron concentrations in primary coolant chemistry.

5.5.5.1 Belleville-2

"Belleville-2, which had a Level 2 event in May, had another, unrelated one July 10, when 10 cubic meters of non-borated water were injected by mistake into the reactor coolant-system (RCS) because of an error in estimating the level of water in a feed tank that had been used for a test. Although the average boron concentration of the RCS remained above specified limits at all times, the incident was classified at Level 2 because of its potential safety consequences." (*Nucleonics Week*, September 12, 1991a)

5.5.5.2 Shearon Harris-1

Technical Specification 3.1.2.1 requires one boration flowpath be available while in mode 5 and 6. It was identified that the required boration flowpath was functional but not operable for six days because refueling water storage tank boron concentration was 2217 ppm. This exceeded the required boron concentration range of 2000 - 2200 ppm. During this time a flowpath from the boric acid tank was available, if required. Although this flowpath was available, the surveillance on this flowpath was not current, therefore, this flowpath could not be considered operable. This event is being reported in accordance with 10CFR50.73(A)(2)(i)(b) as a technical specification violation (Shearon Harris-1, 91-011).

5.6 Personnel Errors and Equipment Problems

Events throughout this report that resulted from personnel errors or equipment problems have been listed in Tables 5.1 and 5.2 as a quick reference.

5.7 Miscellaneous

This section includes generic interests; inspection technology; hot particles; and airborne contamination.

5.7.1 Generic Issues/General Interest

This section lists reported events that are of general interest to the nuclear industry. There were two items of interest regarding generic issues or general interest topics.

5.7.1.1 Predicting Failure of Coated Fuel Particles

A model to predict failure of coated fuel particles during core heatup events has been developed by Japanese researchers. The model will be tested for validity at the High Temperature Engineering Test Reactor (HTTR), a 30 MWth gas-cooled reactor that will be built at the Oarai Establishment of the Japan Atomic Energy Research Institute (Ogawa et al. 1991).

5.7.1.2 Zirconium Hydride Blistering

Zirconium hydride blistering, which caused a pressure tube rupture in Ontario Hydro's Pickering-2 in 1983 and triggered an accelerated retubing program for Candu reactors, is not occurring in tubes of zirconium-niobium alloy, nor is hydride buildup due to primary side corrosion occurring as quickly as was previously predicted. Pickering-1 and -2 - since retubed - originally had fuel-carrying tubes made of a different metal, Zircaloy-2 (Silver March 28, 1991).

Canada's Pickering-4 "began a planned 19-month outage for fuel channel replacement" (*Nucleonics Week*, October 10, 1991b). The program was spurred by the discovery of zirconium hydride blistering in Pickering-2 in 1983 and is replacing the original tubes with tubes made of zirconium-niobium alloy.

5.7.2 Inspection Technology

Reported events that were a result of inspection of reactor components are listed below. There were two items of interest in the United States that involved inspection technology.

5.7.2.1 FFRDS System for Fuel Rod Examination

ABB Reaktor's Failed Fuel Rod Detection System (FFRDS) has examined over 2 million fuel rods, from its introduction in 1979 until the end of 1990. A success

rate of >99.99% is claimed, and about 740 defective assemblies (with approximately 1380 defective fuel rods) out of roughly 9000 assemblies were detected. The system works for PWR as well as BWR assemblies, and takes about 5 minutes for each assembly face. The system relies on water inside the fuel rod to detect a failed rod, and detection of a failed rod can be hampered if water is not present at the inspection level (usually at the bottom of the fuel assembly) (Forch and Baro 1991).

5.7.2.2 SCORE System for Rod Control Cluster Assembly Inspection

B&W Fuel Company has developed an ultrasonic/eddy current inspection device for examining wear in control rods and rod cluster control assemblies (RCCAs). The System for Control Rod Examination (SCORE), "precisely characterizes wear and wear rates, internal and external RCCA defects." An examination of an RCCA takes about 30 minutes, and the data from the inspection is stored and processed by an on-site computer (Losh 1991).

5.7.3 Hot Particles

Hot particles are a small amount of material with a high activity level. There was one event in the United States that involved hot particles.

5.7.3.1 Waterford-3

At approximately 0600 Hours on May 10, 1991, during refueling outage number 4, Waterford Steam Electric Station Unit 3 primary access point (PAP) portal radiation monitor alarmed when entered by a contaminated contractor technician. A subsequent investigation determined that the technician had become contaminated on the previous day and had passed through the personnel contamination monitor (PCM-1) and pap portal monitor (PM-7) without alarming either monitor on 5/9/91. The root cause of this event was the failure of personnel monitors to detect a hot particle with an activity level of 1600 counts per minute (cpm), due to the low level of activity and the attenuation associated with the location of the particle inside the technician's shoe. Corrective action involves an increase in the frequency of sensitivity checks on the PM-7 personnel monitors to improve the probability of detecting similar particles by minimizing instrument drift. There was no potential for exposure to the general public in excess of 10 CFR 20 limits and; therefore, this event did not threaten the health and safety of the general public (Waterford-3, 91-006).

6.0 Trends

The commercial nuclear power industry's fuel rod reliability, excluding failures from debris-induced fretting wear, continues to be better than 99.98%. Fuel rod reliability continually improved during the 1980s and appears to have leveled off in the last few years. Over the past decade, considerable attention has been given to reducing the number of fuel failures in commercial reactors. Previously observed failure mechanisms are reasonably well understood (e.g., PCI, CILC, Debris Fretting, and Hydriding). As a result, utilities in many instances no longer consider common fuel failures to be an "off-normal" event, reportable to the NRC. Furthermore, only one fuel vendor reports the number and causes of fuel failures in their fuel operating experience report to the NRC. Because neither the utilities nor the vendors are reporting the specific details regarding fuel failures, an accurate analysis of fuel failure trends is difficult. For example, in past years, when a fuel rod failed it was often reported by the utility and/or the vendor. However, in recent years the problems have been analyzed, their causes determined, and steps taken to prevent them. Therefore, failures that are not unusual are not as likely to be reported even though failures continue to occur. Failures may even be anticipated because of the large amount of failure data at a given reactor or reactors. This allows the issue to fade away even though fuel failures are still occurring, thus making it difficult to determine what effect, if any, various solutions have had on the problem or if new failure mechanisms have developed in recent years.

Debris-induced fretting is the fuel failure mechanism seen most often in the past several years. Debris-induced fretting damage occurs as a result of metallic debris that is introduced into the reactor coolant system during refueling outages and system repair operations. This debris is carried into the fuel rod channels where it can become trapped and eventually wear through the cladding due to flow-induced vibration. Utilities have become aware of the consequences of debris in the system and have taken action to prevent the introduction

of debris into the reactor coolant system. Although recent progress has been made to reduce the number of debris-induced fretting failures by introducing debris trapping fuel assembly design modifications (see Section, 3), debris still represents the most significant failure mechanism.

Grid-to-rod fretting induced cladding failures were also observed in 1991. At least one of these failures was believed to be different from previously observed grid-to-rod fretting failures. Previously, grid-to-rod fretting failures have been caused by loose grid cells due to grid damage incurred either during fabrication or handling. There is a possibility that as fuel assembly burnups increase, relaxation of the grid cells may be occurring faster than previously observed. Changes in fuel assembly spacer designs to increase assembly performance may have also influenced this observed problem. Further examinations of the grid-to-rod fretting observed in 1991 are planned for 1992 by at least one vendor.

A paper was presented at the International Topical Meeting on LWR Fuel Performance in Avignon France, 1991 (Strasser et al. 1991). The paper, which provides a utility perspective on fuel design requirements for the 90's, discussed fuel reliability, operational flexibility, fuel reactor safety issues, and economics. In regards to fuel reliability the paper concluded that if the utilities are to approach their desired goal of "zero defects," then fuel reliability must continue to improve.

Further improvements in fuel reliability will require vendors to continue to improve quality control and fuel manufacturing processes. Fuel reliability in the 90's may be affected by more complex water chemistry and operating conditions as utilities move towards hydrogen water chemistry, zinc injection, chemical cleaning of BWR fuel, and lithium injection in PWR plants. Better fuel inspection methods will be needed as well as more accurate methods of predicting the number of failed rods.

Trends

Cladding corrosion was discussed by Strasser. He noted that CILC, a BWR fuel failure mechanism, has diminished as utilities replace their brass condensers. However, Strasser warned that current operational trends can be expected to lead to continued corrosion concerns including cladding embrittlement. The attainment of higher burnups, higher operating coolant temperatures, newer fuel designs using thinner wall zirconium alloys, and various water chemistry changes will all pose additional risk to cladding corrosion. In fact BWR vendors have not reported large defect failures in their zirconium barrier clad fuel in the 1991 vendor reports to the NRC, although some utilities have observed large axial cracks in these fuel designs resulting in high primary coolant activities. The source of the large axial cracks is thought to be due to secondary hydriding promoted by a small initial defect in the rod.

Pellet-cladding interaction has not been reported in PWRs and has significantly been reduced in BWRs by the introduction of barrier fuel designs. However, due to the observed decrease in cladding ductilities at higher burnup levels the likelihood of PCI fuel failures may increase as a larger number of fuel rods achieve these burnup levels in the future. In addition, Zircaloy growth is identified as a possible future problem as burnup levels continue to increase.

Strasser cautioned that there may be future design modifications that could reduce the margin to safety limits. For example, for better PCI resistant fuel in BWRs there has been a trend to reduce fuel rod linear heat generation rate (LHGR) by decreasing the rods outer diameter. While this results in a rod with a lower stored energy content, therefore an increased LOCA margin, it reduces the margin that relates to thermal-hydraulic core stability. Fuel designers need to be vigilant in their analyses of adverse affects caused by fuel design modification.

Average fuel cycle costs associated with operating a commercial nuclear plant in the United States was 13% of the total operations and maintenance costs. Several opportunities exist for reducing the costs of fuel. Strasser reports that the utilities would like to see the continued evolution of fuel assembly lattice designs and the associated optimization of burnable absorbers to further extend fuel burnup levels. Also, neutronically efficient fuel assembly designs would save uranium and enrichment costs. Furthermore, a continued effort to develop higher burnup fuel would result in lower spent fuel storage and final disposal costs and result in longer fuel cycles thus increased plant capacity factors.

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Appendix A

Summary of Commercial LWR Nuclear Fuel Reliability

Appendix A

Summary of Commercial LWR Nuclear Fuel Reliability

The information presented in this appendix has been compiled from data reported in previous fuel performance reports in this series. A table has been developed

for each commercial nuclear fuel vendor to present information pertaining to the reliability of their commercial LWR fuel.

Appendix A

Table A.1 Reliability of Babcock and Wilcox Nuclear Fuel

Year	Zircaloy-Clad Fuel Rod Reliability^(a) (%)	Stainless Steel Clad Fuel Reliability^(a) (%)	Average Plant Coolant I-131 Activity (μCi/g)
1991 ^(b)			
1990 ^(b)			
1989	99.997	98.6	0.023
1988	99.99	99.98	0.035
1987	99.98	99.98	0.028
1986	99.999	99.998	0.014
1985	99.995	99.997	0.031
1984	99.990	100	0.051
1983	99.991	100	0.041
1982	99.994		0.031
1981	99.992		0.046
1980	99.997		0.086
1979	99.97		
1978	99.9 to 99.99		

(a) B&W has defined fuel reliability as follows:

$$\frac{[(\text{total \# of rods irradiated during calendar year}) - (\text{estimated \# of rods with defects})] \times 100\%}{(\text{total \# of rods irradiated during calendar year})}$$

(b) B&W has not provided information for these years.

Table A.2 Reliability of ABB Combustion Engineering Nuclear Fuel

Year	Zircaloy-Clad Fuel Rod Reliability (%)	Average Plant Corrected Coolant I-131 Activity ^(e) ($\mu\text{Ci/g}$)
1991	99.998 ^(a,b)	0.0074
1990	99.998 ^(a,b)	0.0055
1989	99.997 ^(a,b)	0.0096
1988	(c)	0.014
1987	(c)	0.0304
1986	(c)	--
1985	(c)	--
1984	99.98	--
1983	99.98	--
1982	99.98	--
1981	(d)	--
1980	(d)	--
1979	>99.99	--
1978	99.99	--
1977	99.98	--
1976	99.98	--
1975	99.97	--
1974	>99.75	--
1973	99.96	--
1972	99.99	--
1971	99.99	--

- (a) ABB CENF has defined fuel reliability for these years in terms of the overall reliability of ABB CE fuel fabricated since 1983.
- (b) Excluding failures caused by debris-induced fretting wear.
- (c) ABB CENF did not report fuel rod reliability for these years, instead they provided data on coolant activity. In their 1988 annual report, ABB CENF indicated that their overall fuel rod reliability for fuel fabricated after 1984 was estimated to be 99.997%--excluding failures caused by debris-induced fretting wear and by baffle jetting in the Yankee Rowe plant.
- (d) Reliability of 8 x 8 fuel if fuel failures involving CILC are excluded.
- (e) INPO Standard Method.

Table A.3 Reliability of General Electric Nuclear Fuel

Year	Zircaloy-Clad Fuel Rod Reliability (%)	Comments Regarding GE Fuel Reliability
1991	>99.98	GE 8x8 Fuel Types, 1974-1991
1990	>99.98	All Fuel Types, 1974-1990
1989	99.98	All Fuel Types, 1974-1989
1988	>99.97	All Fuel Types, 1974-1988
1987	>99.99 >99.999	All Fuel Types, 1974-1987 Barrier Types, 1987
1986	>99.99 99.994 >99.999	All Fuel Types, 1974-1986 All Fuel Types, 1986 Barrier Types, 1986
1985	>99.99	All Fuel Types, 1974-1985
1984	>99.99 100.00	All Fuel Types, 1974-1984 Barrier Types, 1984
1983	99.993 99.998	All Fuel Types, 1974-1983 All Fuel Types, 1974-1983 ^(a)
1982	>99.98	All Fuel Types, 1974-1982
1981	>99.98	All Fuel Types, 1974-1981
1980	>99.98	All Fuel Types, 1974-1980
1979	99.984 99.998	All Fuel Types, 1974-1979 For 8x8R plus 8x8R(PP) Types ^(b)

(a) Reliability of 8 x 8 fuel if fuel failures involving CILC are excluded.

(b) R = retrofit design, PP = prepressurized.

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Table A.4 Reliability of Siemens Power Corporation Nuclear Fuel

Year	Fuel Rod Reliability ^(a) (%)	Median Plant (PWR) Corrected Coolant I-131 Activity ^(e) (μ Ci/ml)	Median Plant (BWR) FGR Measured at Steam Jet Air Ejector ^(e) (μ Ci/sec)
1991	99.998	0.0023	46
1990	99.997	0.00166	159
1989	99.997	0.00126	370
1988	>99.994	0.000102	(g)
1987	$\geq$ 99.995 ^(b,c)	(f)	--
1986	99.995 ^(b,c)	--	--
1985	99.994 ^(b,c)	--	--
1984	99.995 ^(b,c)	--	--
1983	99.998 ^(b,c) 99.87 ^(b,d)	--	--
1982	99.998 ^(b,c)	--	--
1981	99.998 ^(b,c) 99.987 ^(b,d)	--	--
1980	100	--	--

(a) Includes the reliability for both PWR and BWR fuel rods. Please refer to previous annual fuel performance reports for specific PWR and BWR reliability values.

(b) On a cumulative basis.

(c) The fuel reliability value is based on fuel failures that were judged to be from fuel-related or unknown causes and were not directly attributable to external causes (e.g., plant-related causes such as baffle jetting, fretting from the presence of foreign objects or other off-normal core conditions).

(d) The fuel reliability value is based on fuel failures from all causes.

(e) INPO Standard Methods.

(f) SPC began reporting INPO fuel reliability indicator (FRI) values for PWR plants in 1988.

(g) SPC began reporting INPO FRI values for BWR plants in 1989.

Table A.5 Reliability of Westinghouse Electric Corporation Nuclear Fuel

Year	Zircaloy-Clad Fuel Rod Reliability (%)	Average Plant Corrected Coolant I-131 Activity ^(a) ($\mu\text{Ci/g}$)
1991		0.00019
1990	99.998 ^(b)	0.0021
1989	99.994 ^(b)	0.0047
1988	99.994 ^(b)	0.0049
1987	99.994 ^(b)	0.0045
1986	(c)	0.0044
1985	(c)	0.0086, ^(d) 0.0092 ^(e)
1984	(c)	0.008 ^(d)
1983	(c)	0.030 ^(d)
1982	(c)	0.0296, ^(d) 0.041 ^(e)
1981	(c)	--
1980	(c)	--
1979	~99.983	--
1978	(c)	--
1977	99.938-99.9999	--
1976	(c)	--
1975	99.75-100.00 ^(f)	--
1974	99.790-99.999 ^(f)	--
1973	99.91-99.999 ^(f)	--
1972	99.74-100 ^(f)	--
1971	99.23-100 ^(f)	--
1970	99.24-99.999 ^(f)	--
1969	99.64-100 ^(f)	--

(a) Westinghouse began reporting fuel performance in terms of coolant activity level starting June 30, 1976. In 1982 Westinghouse provided data on average coolant activity levels in terms of $\mu\text{Ci/g}$ for Westinghouse fueled plants. Beginning 1991, Westinghouse began to also report the distribution of coolant activities for Westinghouse-fueled plants.

(b) The fuel reliability value is based on fuel failures from all causes.

(c) Westinghouse did not state a fuel rod reliability (integrity) value. Westinghouse continues to evaluate fuel performance in terms of coolant activity level.

(d) Uncorrected activity, excludes fuel failures due to baffle jetting.

(e) Uncorrected activity, includes fuel failures due to baffle jetting.

(f) The range of values for individual plants in all four quarters of the given year are provided.

Appendix B

Typical Fuel Assembly Parameters

Table B.1 Typical Fuel Assembly Parameters

Vendor	ABB CENF	ABB CENF	B&W	B&W	B&W	B&W	B&W
	PWR	PWR	PWR	PWR	PWR	PWR	PWR
Reactor Type	PWR	PWR	PWR	PWR	PWR	PWR	PWR
Reactor System	CE	CE	B&W	B&W	W	W	W
Fuel Rod Array	14x14	16x16	15x15	17x17	15x15	15x15	17x17
Version	CE	CE	Mark B ^(e)	Mark C	Haddam	Mark BW	Mark BW
Assemblies Per Core	217	217	177	205	157	157	193
Fuel Rod Positions per Assembly	176	236	225	289	225	225	289
Typical Number of Fueled Rods per Assembly	164	224	208	264	204	204	264
Rod Pitch, mm (in.)	14.7 (0.580)	12.9 (0.506)	14.4 (0.568)	12.8 (0.502)	14.3 (0.563)	14.3 (0.563)	12.6 (0.496)
System Pressure, MPa (psia)	15.5 (2250)	15.5 (2250)	15.2 (2200)	15.5 (2250)	13.9 (2015)	13.9 (2015)	15.5 (2250)
Core Average Power Density, kW/liter	78.5	96.4	91.4	107.3	82.25	82.25	82.25
Average LHGR, ^(a) kW/m (kW/ft)	20.0 (6.09)	18.2 (5.54)	20.3 (6.20)	18.8 (5.73)	18.1 (5.53)	18.4 (5.60)	17.8 (5.43)
Axial Peak LHGR in an Average Rod, kW/m (kW/ft)	24.0 (7.31)	21.0 (6.41)	24.4 (7.44)	22.6 (6.88)	25.1 (7.66)	25.5 (7.76)	27.6 (8.42)
Max. Peak LHGR, kW/m (kW/ft)	53.5 (16.3)	42.7 (13.0)	53.0 (16.16)	49.9 (15.20)	47.6 (14.5)	47.6 (14.5)	42.7 (13.0)
Max. Fuel Temp., °C (°F)	2140 (3890)	1880 (3420)	2340 (4244)	2290 (4155)	2149 (3900)	2149 (3900)	1927 (3500)
Core Average Enrichment wt% ²³⁵ U	3.89	2.36	3.30	3.15	4.00	3.41	3.40
Max. Local Exposure, GWd/MT	50	55	55	55	55	55	55
Cladding Material ^(b)	Zry-4	Zry-4	Zry-4	Zry-4	304SS	Zry-4	Zry-4

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Appendix B

Table B.1 (contd)

Vendor	ABB CENF	ABB CENF	B&W	B&W	B&W	B&W	B&W
Fuel Rod Length, m (in.)	3.71 (145.9)	4.09 (161.0)	3.904 (153.7)	3.878 (152.7)	3.218 (126.7)	3.197 (125.9)	3.848 (151.5)
Active Fuel Length, m (in.)	3.47 (136.7)	3.81 (150)	3.602 (141.8)	3.632 (143.0)	3.061 (120.5)	3.012 (118.6)	3.658 (144.0)
Plenum Length, m (in.)	0.22 (8.6)	0.25 (10.0)	0.298 (11.7)	0.242 (9.5)	0.122 (4.8)	0.159 (6.3)	0.164 (6.4)
Fuel Rod OD, mm (in.)	11.18 (0.44)	9.70 (0.382)	10.92 (0.430)	9.63 (0.379)	10.72 (0.422)	10.72 (0.422)	9.50 (0.374)
Cladding ID, mm (in.)	9.75 (0.384)	8.43 (0.332)	9.58 (0.377)	8.41 (0.331)	9.88 (0.389)	9.35 (0.368)	8.28 (0.326)
Cladding Thickness, mm (in.)	0.711 (0.028)	0.635 (0.025)	0.673 (0.0265)	0.610 (0.024)	0.419 (0.0165)	0.686 (0.027)	0.610 (0.024)
Diametral Gap, ^(c) micron (mil)	190.5 (7.5)	178 (7.0)	213.4 (8.4)	198.1 (7.8)	165 (6.5)	178 (7.0)	165 (6.5)
Fuel Pellet Diameter, mm (in.)	9.56 (0.3765)	8.26 (0.325)	9.362 (0.3686)	8.209 (0.3232)	9.715 (0.3825)	9.17 (0.361)	8.115 (0.3195)
Fuel Pellet Length, mm (in.)	11.43 (0.45)	9.91 (0.39)	11.05 (0.435)	9.53 (0.375)	11.63 (0.458)	10.80 (0.425)	10.16 (0.400)
Fuel Pellet Density, %TD ^(d)	95	95	95	95	95	95	96

Table B.1 (contd)

Vendor	ABB CENF	ABB CENF	B&W	B&W	B&W	B&W	B&W
Fill Gas and Pressure (psig)	He 300-450	He 300-450	He 415	He 435	He 40		

(a) LHGR = linear heat generation rate.

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

(c) Diametral gap = cladding ID - pellet diameter

(d) Theoretical density (TD) of stoichiometric UO_2 is 10.96 g/cm^3

(e) Distinguishing Features:

Mark B4: Prepressurized fuel

Mark BEB: Extended Burnup LTA

Mark B4Z: Mark B4 fuel with Zircaloy spacers

Mark BGd: LTA with integral Gadolinia absorbers in the fuel

Mark B5: Revised end fitting

Mark B7: Mark B6 features plus shorter lower end fitting, longer fuel rod, increased plenum volume

Mark B5Z: Mark B5 fuel with Zircaloy spacer

Mark B8: Debris fretting resistant fuel rod design, reduced prepressurization

Mark B6: Zircaloy spacers, skirtless upper end spacer, removable top nozzle

Table B.1 (contd)

Vendor	GE	GE	GE	GE	SNP	SNP	SNP	SNP	SNP	SNP
	BWR/2,3	BWR/2,3	BWR/4-6	BWR/4-6	BWR/2,3	BWR/2,3	BWR/2,3	BWR/2,3	BWR/4-6	BWR/4-6
Reactor Type	GE	GE	GE	GE	GE	GE	GE	GE	GE	GE
Reactor System	8x8	9x9	8x8	9x9	9x9	9x9	9x9	9x9	9x9	9x9
Fuel Rod Array	Barrier	GE-11	GE-4a	GE-11	JP-3	9-5	IX	9X	JP-4,5	9-5
Version	560		560		724					
Assemblies Per Core	64	81	64	81	81	81	81	81	81	81
Fuel Rod Positions per Assembly	62	77	63	77	79	76	72	72	79	76
Typical Number of Fueled Rods per Assembly	16.3 (0.640)		16.3 (0.640)		14.52 (0.572)		14.52 (0.572)	14.52 (0.572)	14.52 (0.572)	
Rod Pitch, mm (in.)	7.14 (1035)		7.14 (1035)		7.07 (1026)					
System Pressure, MPa (psia)	49.15		50.51		46					
Core Average Power Density, kW/liter	17.7 (5.38)		17.9 (5.45)		12.1 (3.68)					
Average LHGR, ^(c) kW/m (kW/ft)	21.24 (6.99)		21.48 (7.09)		17.5 (5.34)					
Axial Peak LHGR in an Average Rod, kW/m (kW/ft)	44.0 (13.4)		44.0 (13.4)		37.7 (11.5)					
Max. Peak LHGR, kW/m (kW/ft)	1890 (3435)		1830 (3325)		2040 (3705)					
Max. Fuel Temp, °C (°F)	1.99		2.54		2.8					
Core Average Enrichment, wt% ²³⁵ U	45		40		55					
Max. Local Exposure, GWd/MTU	Zry-2	Zry-2	Zry-2	Zry-2	Zry-2	Zry-2	Zry-2	Zry-2	Zry-2	Zry-2
Cladding Material ^(d)										

Table B.1 (contd)

Vendor	GE	GE	GE	GE	SNP	SNP	SNP	SNP	SNP	SNP
Fuel Rod Length, m (in.)	4.20 (165.4)		4.09 (161.1)		4.04 (159.07)				4.16 (163.84)	
Active Fuel Length, m (in.)	3.81 (145.24)		3.71 (146)		3.68 (145.24)				3.81 (150.0)	
Plenum Length, m (in.)	0.25 (9.48)		0.36 (14.0)		0.243 (9.580)				0.24 (9.578)	
Fuel Rod OD, mm (in.)	12.27 (0.483)		12.52 (0.493)		10.76 (0.424)				10.76 (0.424)	
Cladding ID, mm (in.)	10.64 (0.419)		10.80 (0.425)		9.25 (0.364)					
Cladding Thickness, mm (in.)	0.813 (0.032)		0.863 (0.034)		0.76 (0.030)				0.76 (0.030)	
Diametral Gap, ^(e) micron (mil)	229 (9.0)		229 (9.0)		190 (7.5)					
Fuel Pellet Diameter, mm (in.)	10.41 (0.410)		10.57 (0.416)		9.05 (0.356)					
Fuel Pellet Length, mm (in.)	10.41 (0.410)		10.67 (0.420)		10.41 (0.410)					
Fuel Pellet Density, %TD ^(f)	95		95		94.5				94.5	
Fill Gas and Pressure (psig)	He 30		He 0		He 60				He 60	

(a) Gadolinia in fuel rods.

(b) Bottom 42 inches of fuel rod is stainless steel.

(c) LHGR = linear heat generation rate.

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

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

(f) Theoretical density (TD) of stoichiometric UO₂ is 10.96 g/cm³.

Table B.1 (contd)

Vendor	SNP	SNP	SNP	SNP	SNP	SNP	W	W	W
Reactor Type	BWR/4-6	BWR/4-6	PWR	PWR	PWR	PWR	PWR	PWR	PWR
Reactor System	GE	GE	ABB CE	W	W	W	CE	B&W	W
Fuel Rod Array	9x9	9x9	14x14	14x14	15x15	17x17	14x14	15x15	17x17
Version	IX	9X	ABB CE	Top Rod	W	W	WE	WE	Vantage 5
Assemblies Per Core					193	193		193	193
Fuel Rod Positions per Assembly	81	81	176	196	225	289	176	225	289
Typical Number of Fueled Rods per Assembly	72	72	176	179	204	264	176	208	264
Rod Pitch, mm (in.)	14.45 (0.569)	14.45 (0.569)	14.73 (0.580)	14.12 (0.556)	14.30 (0.563)	12.60 (0.496)	14.73 (0.580)	14.3 (0.563)	12.60 (0.496)
System Pressure, MPa (psia)					15.5 (2250)	15.5 (2250)		15.5 (2250)	
Core Average Power Density, kW/liter					98.1	104.7		98.1	
Average LHGR, ^(c) kW/m (kW/ft)			20.76 (6.34)		22.0 (6.60)	17.8 (5.44)		22.0 (6.70)	
Axial Peak LHGR in an Average Rod, kW/m (kW/ft)					26.40 (8.04)	21.4 (6.53)		26.40 (8.04)	
Max. Peak LHGR, kW/m (kW/ft)					51.9 (15.8)	54.5 (16.6)		61.7 (18.8)	
Max. Fuel Temp, °C (°F)					2200 (3997)	1747 (3177)		2340 (4250)	
Core Average Enrichment, wt% ²³⁵ U			3.34		2.55	3.65	3.00	2.80	3.42
Max. Local Exposure, GWd/MTU					47.5	52		50	

Table B.1 (contd)

Vendor	SNP	SNP	SNP	SNP	SNP	SNP	W	W	W
	Zry-2	Zry-2	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4
Cladding Material ^(d)									
Fuel Rod Length, m (in.)			3.72 (146.4)	3.86 (152.0)	3.86 (152.0)	3.86 (152.0)	3.72 (146.4)	3.80 (149.7)	3.87 (152.3)
Active Fuel Length, m (in.)	3.81 (150.0)	3.81 (150.0)	3.40 (134.0)	3.65 (144.0)	3.65 (144.0)	3.65 (144.0)	3.47 (136.7)	3.66 (144)	3.65 (144.0)
Plenum Length, m (in.)				0.18 (7.28)	0.17 (6.80)	0.18 (7.26)		0.21 (8.2)	0.18 (7.41)
Fuel Rod OD, mm (in.)	(0.431)	(0.431)	11.17 (0.440)	10.59 (0.417)	10.76 (0.424)	9.14 (0.360)	11.17 (0.440)	10.72 (0.422)	9.14 (0.360)
Cladding ID, mm (in.)					9.25 (0.364)	7.87 (0.310)		9.48 (0.373)	
Cladding Thickness, mm (in.)	(0.025)	(0.029)	0.78 (0.031)	0.74 (0.029)	0.76 (0.030)	0.63 (0.025)	0.66 (0.026)	0.62 (0.024)	0.058 (0.023)
Diametral Gap, ^(e) micron (mil)					190 (7.5)	177.8 (7.0)		190 (7.5)	
Fuel Pellet Diameter, mm (in.)	9.49 (0.374)	9.47 (0.373)	9.39 (0.370)		9.05 (0.356)	7.69 (0.303)	9.68 (0.381)	9.29 (0.366)	7.85 (0.309)
Fuel Pellet Length, mm (in.)			10.80 (0.425)		6.93 (0.273)	8.84 (0.348)	15.24 (0.600)	15.24 (0.600)	12.95 (0.510)
Fuel Pellet Density, %TD ^(f)	96.26	94.5	94	94	94	94	95	95	95
Fill Gas and Pressure (psig)			He 375	He 305	He 290	He 290	He 275-400	He	He

(a) Gadolinia in fuel rods.

(b) Bottom 42 inches of fuel rod is stainless steel.

(c) LHGR = linear heat generation rate.

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

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

(f) Theoretical density (TD) of stoichiometric UO₂ is 10.96 g/cm³.

Appendix C

List of Domestic Events by Reactor

Table C.1 List of Domestic Events by Reactor (BWRs)

Power Plant, (Reactor Supplier) Utility	Fuel Oriented	Fuel Handling Oriented	Spent Fuel Oriented	Control Rod Oriented	Core/Coolant Oriented	Total
Arnold (GE) <i>Iowa Electric Light & Power Co.</i>				5.4.1.1		1
Big Rock Point (GE) <i>Consumers Power Co.</i>						0
Browns Ferry-1 (GE) <i>Tennessee Valley Authority</i>						0
Browns Ferry-2 (GE) <i>Tennessee Valley Authority</i>						0
Browns Ferry-3 (GE) <i>Tennessee Valley Authority</i>				5.4.1.2, 5.4.1.3	5.5.1.1	4
Brunswick-1 (GE) <i>Carolina Power & Light Co.</i>		5.2.1.1		5.4.2.6		1
Brunswick-2 (GE) <i>Carolina Power & Light Co.</i>						0
Clinton-1 (GE) <i>Illinois Power Co.</i>						0
Cooper Station (GE) <i>Nebraska Public Power District</i>				5.4.1.4, 5.4.1.5		3
Dresden-2 (GE) <i>Commonwealth Edison Co.</i>				5.4.2.8		1
Dresden-3 (GE) <i>Commonwealth Edison Co.</i>		5.2.1.3				0
Enrico Fermi-2 (GE) <i>Detroit Edison Co.</i>						0

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Appendix C

Table C.1 (contd)

Power Plant, (Reactor Supplier) <i>Utility</i>	Fuel Oriented	Fuel Handling Oriented	Spent Fuel Oriented	Control Rod Oriented	Core/Coolant Oriented	Total
FitzPatrick (GE) <i>New York Power Authority</i>						0
Grand Gulf-1 (GE) <i>Entergy Operations, Inc.</i>						0
Hatch-1 (GE) <i>Southern Nuclear Operating Co.</i>						0
Hatch-2 (GE) <i>Southern Nuclear Operating Co.</i>			5.3.4.1			1
Hope Creek (GE) <i>Public Service Electric & Gas</i>					5.5.3.1	1
LaSalle County-1 (GE) <i>Commonwealth Edison Co.</i>						0
LaSalle County-2 (GE) <i>Commonwealth Edison Co.</i>						0
Limerick-1 (GE) <i>Philadelphia Electric Co.</i>						0
Limerick-2 (GE) <i>Philadelphia Electric Co.</i>		5.2.2.2				1
Millstone-1 (GE) <i>Northeast Utilities</i>			5.3.1.1	5.4.2.12		2
Monticello (GE) <i>Northern States Power Co.</i>				5.4.5.2		1

Table C.1 (contd)

Power Plant, (Reactor Supplier) Utility	Fuel Oriented	Fuel Handling Oriented	Spent Fuel Oriented	Control Rod Oriented	Core/Coolant Oriented	Total
Nine Mile Point-1 (GE) <i>Niagara Mohawk Power Corp.</i>				5.4.3.6		1
Nine Mile Point-2 (GE) <i>Niagara Mohawk Power Corp.</i>						0
Oyster Creek (GE) <i>GPU Nuclear Corp.</i>						0
Peach Bottom-2 (GE) <i>Philadelphia Electric Co.</i>						0
Peach Bottom-3 (GE) <i>Philadelphia Electric Co.</i>						2
Perry-1 (GE) <i>The Cleveland Electric Illum. Co.</i>				5.4.1.9, 5.4.2.15		1
Pilgrim-1 (GE) <i>Boston Edison Co.</i>		5.2.1.4				0
Quad Cities-1 (GE) <i>Commonwealth Edison Co.</i>						0
Quad Cities-2 (GE) <i>Commonwealth Edison Co.</i>						2
River Bend (GE) <i>Gulf States Utility Co.</i>		5.2.2.4, 5.2.2.5				0
Susquehanna-1 (GE) <i>Pennsylvania Power & Light Co.</i>						

Table C.1 (contd)

Power Plant, (Reactor Supplier) Utility	Fuel Oriented	Fuel Handling Oriented	Spent Fuel Oriented	Control Rod Oriented	Core/Coolant Oriented	Total
Susquehanna-2 (GE) Pennsylvania Power & Light Co.				5.4.1.11		1
Vermont Yankee (GE) Vermont Yankee Nuclear Power Co.			5.3.5.3			1
Washington Nuclear-2 (GE) Washington Public Power Supply						0
Total	0	6	3	13	2	24

Table C.2 List of Domestic Events by Reactor (PWRs)

Reactor (Reactor Supplier) Utility	Fuel Oriented	Fuel Handling Oriented	Spent Fuel Oriented	Control Rod Oriented	Core/Coolant Oriented	Total
Arkansas Nuclear-1 (B&W) Entergy Operations, Inc.				5.4.2.1		1
Arkansas Nuclear-2 (CE) Entergy Operations, Inc.				5.4.2.2		1
Beaver Valley-1 (W) Duquesne Light Co.				5.4.2.3		1
Beaver Valley-2 (W) Duquesne Light Co.				5.4.3.1		1
Braidwood-1 (W) Commonwealth Edison Co.				5.4.2.4		1
Braidwood-2 (W) Commonwealth Edison Co.		5.2.4.1		5.4.2.5		2
Byron-1 (W) Commonwealth Edison Co.						0
Byron-2 (W) Commonwealth Edison Co.						0
Callaway-1 (W) Union Electric Co.						0
Calvert Cliffs-1 (CE) Baltimore Gas & Electric Co.				5.4.5.1		1
Calvert Cliffs-2 (CE) Baltimore Gas & Electric Co.						0
Catawba-1 (W) Duke Power Co.				5.4.3.2		1

Table C.2 (contd)

Reactor (Reactor Supplier) Utility	Fuel Oriented	Fuel Handling Oriented	Spent Fuel Oriented	Control Rod Oriented	Core/Coolant Oriented	Total
Catawba-2 (W) Duke Power Co.						0
Comanche Peak-1 (W) Texas Utilities Electric Co.						0
Cook-1 (W) Indiana/Michigan Power Co.				5.4.2.7		1
Cook-2 (W) Indiana/Michigan Power Co.						0
Crystal River-3 (B&W) Florida Power Company				5.4.3.3		1
Davis-Besse-1 (B&W) Toledo Edison Co.						0
Diablo Canyon-1 (W) Pacific Gas & Electric Co.		5.2.2.1		5.4.3.4		2
Diablo Canyon-2 (W) Pacific Gas & Electric Co.						0
Farley-1 (W) Southern Nuclear Operating Co.						0
Farley-2 (W) Southern Nuclear Operating Co.				5.4.1.6 5.4.2.9		2
Fort Calhoun-1 (CE) Omaha Public Power District						0

Table C.2 (contd)

Reactor (Reactor Supplier) Utility	Fuel Oriented	Fuel Handling Oriented	Spent Fuel Oriented	Control Rod Oriented	Core/Coolant Oriented	Total
						0
Ginna (W) Rochester Gas & Electric Corp.			5.3.2.1			1
Haddam Neck (W) Connecticut Yankee Atomic Power						0
Indian Point-2 (W) Consolidated Edison Co.						0
Indian Point-3 (W) New York Power Authority						0
Kewaunee (W) Wisconsin Public Service Corp.						0
Maine Yankee (CE) Maine Yankee Atomic Power Co.			5.3.4.2 5.3.3.1	5.4.3.5		3
McGuire-1 (W) Duke Power Co.				5.4.2.10 5.4.2.11		2
McGuire-2 (W) Duke Power Co.			5.3.1.2	5.4.5.3		2
Millstone-2 (CE) Northeast Utilities						0
Millstone-3 (W) Northeast Utilities						0
North Anna-1 (W) Virginia Power Co.						0
North Anna-2 (W) Virginia Power Co.						0

Table C.2 (contd)

Reactor (Reactor Supplier) Utility	Fuel Oriented	Fuel Handling Oriented	Spent Fuel Oriented	Control Rod Oriented	Core/Coolant Oriented	Total
Oconee-1 (B&W) Duke Power Co.						0
Oconee-2 (B&W) Duke Power Co.						0
Oconee-3 (B&W) Duke Power Co.				5.4.1.7 5.4.2.13		2
Palisades (CE) Consumers Power Company						0
Palo Verde-1 (CE) Arizona Public Service Co.				5.4.2.14		1
Palo Verde-2 (CE) Arizona Public Service Co.		5.2.2.3			5.5.2.1	2
Palo Verde-3 (CE) Arizona Public Service Co.				5.4.1.8		1
Point Beach-1 (W) Wisconsin Electric Power Co.						0
Point Beach-2 (W) Wisconsin Electric Power Co.						0
Prairie Island-1 (W) Northern States Power Co.			5.3.3.2	5.4.2.16		2
Prairie Island-2 (W) Northern States Power Co.				5.4.1.12		1

Table C.2 (contd)

Reactor (Reactor Supplier) Utility	Fuel Oriented	Fuel Handling Oriented	Spent Fuel Oriented	Control Rod Oriented	Core/Coolant Oriented	Total
Robinson-2 (W) Carolina Power & Light Co.				5.4.2.17		1
Salem-1 (W) Public Service Electric & Gas						0
Salem-2 (W) Public Service Electric & Gas						0
San Onofre-1 (W) Southern Cal Edison Co.				5.4.2.18 5.4.4.1		2
San Onofre-2 (CE) Southern Cal Edison Co.			5.3.1.3			1
San Onofre-3 (CE) Southern Cal Edison Co.		5.2.4.2				1
Sequoyah-1 (W) Tennessee Valley Authority						0
Sequoyah-2 (W) Tennessee Valley Authority				5.4.3.7		1
Seabrook (W) New Hampshire Yankee						0
Shearon Harris (W) Carolina Power & Light Co.					5.5.5.2	1
South Texas-1 (W) Houston Lighting & Power Co.		5.2.2.6				1
South Texas-2 (W) Houston Lighting & Power Co.						0

Table C.2 (contd)

Reactor (Reactor Supplier) Utility	Fuel Oriented	Fuel Handling Oriented	Spent Fuel Oriented	Control Rod Oriented	Core/Coolant Oriented	Total
St. Lucie-1 (CE) <i>Florida Power and Light Co.</i>		5.2.2.7		5.4.1.10		2
St. Lucie-2 (CE) <i>Florida Power and Light Co.</i>						0
Summer (W) <i>South Carolina Electric & Gas Co.</i>						0
Surry-1 (W) <i>Virginia Power Co.</i>						0
Surry-2 (W) <i>Virginia Power Co.</i>						0
Three Mile Island-1 (B&W) <i>GPU Nuclear Corp.</i>		5.2.2.8		5.4.1.12		2
Trojan (W) <i>Portland General Electric Co.</i>						0
Turkey Point-3 (W) <i>Florida Power and Light Co.</i>						0
Turkey Point-4 (W) <i>Florida Power and Light Co.</i>			5.3.2.2 5.3.2.3			2
Vogtle-1 (W) <i>Southern Nuclear Operating Co.</i>				5.4.3.8	5.5.3.2	2
Vogtle-2 (W) <i>Southern Nuclear Operating Co.</i>				5.4.3.9		1

Table C.2 (contd)

Reactor (Reactor Supplier) Utility	Fuel Oriented	Fuel Handling Oriented	Spent Fuel Oriented	Control Rod Oriented	Core/Coolant Oriented	Total
Waterford (CE) Entergy Operations, Inc.		5.2.2.9		5.4.1.13 5.4.1.14		3
Wolf Creek (W) Wolf Creek Nuclear Operating Corp.				5.4.3.10		1
Zion-1 (W) Commonwealth Edison Co.						0
Zion-2 (W) Commonwealth Edison Co.						0
Total	0	8	8	33	3	53

Distribution

No. of Copies		No. of Copies	
Offsite		Onsite	
1	Nuclear Regulatory Commission Division of Technical Information and Document Control 7920 Norfolk Avenue Bethesda, MD 20014	27	Pacific Northwest Laboratory J. M. Alvis, P8-34 C. E. Beyer, P8-34 (5) M. E. Cunningham, P8-34 M. D. Fieshley, K6-77 R. J. Guenther, P8-34 D. D. Lanning, P8-34 C. L. Painter, P8-34 (10) F. E. Panisko, P8-35 Publishing Coordination Technical Report Files (5)
24	Reactor Systems Branch Division of Engineering and Systems Technology Office of Nuclear Reactor Regulation U.S. Nuclear Regulatory Commission Mail Stop 8E23 Washington, DC 20555 ATTN: R. C. Jones (4) L. E. Phillips (10) E. D. Kendrick (10)		
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BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev.,
and Addendum Numbers, if any.)

NUREG / CR-3950

PNL - 5210

Vol. 9

2. TITLE AND SUBTITLE

Fuel Performance Report for 1991

3. DATE REPORT PUBLISHED

MONTH

YEAR

August

1994

4. FIN OR GRANT NUMBER

L1864

5. AUTHOR(S)

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6. TYPE OF REPORT

Technical

7. PERIOD COVERED (Inclusive Dates)

CY 1991

* U.S. Nuclear Regulatory Commission

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Pacific Northwest Laboratory
Richland, WA 99352

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This annual report, the fourteenth in a series, provides a brief description of fuel performance during 1991 in commercial nuclear power plants and an indication of trends. Brief summaries of fuel operating experience, fuel design changes, fuel surveillance programs, high-burnup experience, problem areas, and items of general significance are provided. References to more detailed information and related U.S. Nuclear Regulatory Commission evaluations are included.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Fuel Performance
Fuel Design Changes
Fuel Surveillance Programs
Fuel Operating Experience
Fuel Trends
Fuel Problems
High-Burnup Fuel Experience

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)

Unclassified

(This Report)

Unclassified

15. NUMBER OF PAGES

16. PRICE

DATE

FILMED

10 / 4 / 94

END