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NUREG/CR-3950
PNL-5210
Vol. 8

Fuel Performance Annual Report for 1990

Manuscript Completed: September 1993
Date Published: November 1993

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Prepared for
Division of Systems Technology
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

Abstract

This annual report, the thirteenth in a series, provides a brief description of fuel performance during 1990 in commercial nuclear power plants. Brief summaries of fuel design changes, fuel surveillance programs, fuel operating experience and trends, fuel problems, high-

burnup fuel experience, and items of general significance are provided. References to additional, more detailed information, and related NRC evaluations are included where appropriate.

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Acknowledgment

The authors would like to express our appreciation for support from the Office of Nuclear Reactor Regulation, NRC. Special thanks go to B. Thomas, NRC/NRR, and M. E. Cunningham, PNL for providing their administrative and technical guidance. The authors would also like to thank J. A. Gibbons,

an Inquiry Into Science and Engineering Student (IIS&ES) assigned to the Reactor Systems and Materials Section, for her effort and perseverance in properly preparing this report for publication.

List of Acronyms

ABB	ASEA Brown Boveri
AECL	Atomic Energy of Canada Limited
AEC	Atomic Energy Commission
AFD	Axial Flux Difference
ANO	Arkansas Nuclear One
ANS	American Nuclear Society
ANF	Advanced Nuclear Fuels - now Siemens Nuclear Power Corporation
AP&L	Arkansas Power & Light
ARI	Alternate Rod Injection System
ARPI	Analog Rod Position Indicator
BWFC	Babcock & Wilcox Fuel Company
BWR	Boiling Water Reactor
ABB CENF	ABB Combustion Engineering Nuclear Fuel
CANDU	Canadian Deuterium-Uranium Reactor
CEA	Control Element Assembly
CFR	Code of Federal Regulations
CILC	Crud Induced Localized Corrosion
COFFEC	Cost of Fuel Failure Evaluation Code
CRD	Control Rod Drive
DFBN	Debris Filter Bottom Nozzle
DOE	Department of Energy
DRPI	Digital Rod Position Indication System
EPRI	Electric Power Research Institute
FRI	Fuel Reliability Indicator
GE	General Electric
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)
HTGR	High Temperature Gas Reactor
IFBA	Integral Fuel Burnable Absorber
IFM	Intermediate Flow Mixing
INPO	Institute of Nuclear Power Operations
IRM	Intermediate Range Monitor
IRPIS	Individual Rod Position Indicators
ISFSI	Independent Spent Fuel Storage Installation
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
NRC	Nuclear Regulatory Commission
OFA	Optimized Fuel Assembly
ORNL	Oak Ridge National Lab
PC	Personal Computer
PCI	Pellet-Cladding Interaction
PHWR	Pressurized Heavy Water Reactor
PNL	Pacific Northwest Laboratories
PWR	Pressurized Water Reactor
RCCA	Rod Control Cluster Assembly
RCIC	Reactor Core Isolation Cooling System
RPCS	Rod Pattern Control System

Acronyms

RPI	Rod Position Indicator
RPS	Reactor Protection System
SAR	Safety Analysis Report
SCC	Stress Corrosion Cracking
SFP	Spent Fuel Pool
SNP	Siemens Nuclear Power Corporation
UFSAR	Updated Final Safety Analysis Report
USAR	Updated Safety Analysis Report
<u>W</u>	Westinghouse
WNP	Washington Nuclear Power
Zr	Zirconium
Zry	Zircaloy

1.0 Executive Summary

Fuel performance in 1990 focused on extending burnup while maintaining fuel rod reliability. Optimizing fuel use by extending burnup is advantageous for several reasons. Extending burnup generates less waste, reduces fuel costs, and improves plant capacity by extending operating cycles thereby reducing shutdown time.

The sections that follow discuss the burnup levels achieved in 1990 and fuel rod reliability for each of the reporting vendors. Fuel and core related problems along with corrective and preventative procedures are also reviewed with specific focus on trends. It should be noted that Babcock & Wilcox Fuel Company (BWFC) did not provide information regarding their fuel operating experience for 1990 and, therefore, fuel operating experience for BWFC is not included.

Identifying trends involving fuel failure mechanisms (refer to Section 6) has become more difficult in recent years since the majority of vendors and utilities no longer report the number of failed fuel rods and their related causes.

Utilities have taken the position that they are not required to report fuel failures as unusual events, providing that measured coolant/offgas activities do not exceed the plant's technical specification limits. From the information that is available, it may be concluded that the overall number of failed fuel rods is decreasing. However, information does not exist to determine if an individual plant is experiencing continuing fuel performance problems, if specific fuel failure mechanisms are becoming more or less significant, or if new fuel failure mechanisms are developing as fuel burnup is extended.

1.1 Extended Burnup

A summary of the highest fuel burnup levels achieved by utilities using fuel fabricated by the four nuclear fuel vendors who reported to the Nuclear Regulatory Commission (NRC) in 1990 is provided in Table 1. The vendors that have provided information are ABB Combustion Engineering Nuclear Fuel (ABB CENF),⁽¹⁾ General Electric (GE),⁽²⁾ Siemens Nuclear Power Corp. (SNP),⁽³⁾ and Westinghouse Electric Corporation (W).⁽⁴⁾

Table 1. Highest Burnup Fuel Experience by Vendor

Vendor	Plant or Test	Type	Burnup (GWd/MTU)	Comment
ABB CENF ⁽¹⁾	ANO-2	PWR	44.8	Batch Avg.
	St. Lucie-2	PWR	44.0	Batch Avg.
	702 rods discharged		56-59.9	Batch Avg.
GE ⁽²⁾		BWR	>45.0	Bundle Average
		BWR	60.0	Peak Pellet Exp.
SNP ⁽³⁾	R. E. Ginna	PWR	52.1	Assembly Avg.
	D. C. Cook-2	PWR	46.4	Assembly Avg.
	Big Rock Point	BWR	45.1	Assembly Avg.
	Gundremmingen-C	BWR	40.0	Assembly Avg.
<u>W</u> ⁽⁴⁾	Zion-1 & 2	PWR	55.0	4 Assemblies Avg.
			60.0	Peak Rod Burnup
	North Anna-1	PWR	58.4	Lead Assembly Avg.
			>60.0	Lead Fuel Rod Avg.
BWFC	BWFC did not supply information as per Reference 5			

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BWFC declined to provide information for calendar year 1990 (see Reference 5) and, therefore, no BWFC extended burnup information was presented in Table 1.

1.2 Fuel Reliability

Fuel reliability has steadily increased over the past several years in Pressurized Water Reactors (PWRs) and has remained relatively good for Boiling Water Reactors (BWRs). Siemens Nuclear Power Corporation has reported the following data for industry medians during the period from 1988-1990.

Table 2. Industry Median Fuel Reliability Indicator (FRI)⁽³⁾

Year	PWRs ($\mu\text{Ci/ml}$) ^a	BWRs ($\mu\text{Ci/sec}$) ^b
1990	1.20×10^{-3}	99.0
1989	2.00×10^{-3}	83.0
1988	4.80×10^{-3}	--

^a INPO Standard FRI for PWRs is the coolant iodine-131 activity normalized to a standard cleanup rate and corrected for tramp uranium.

^b INPO Standard FRI for BWRs is the rate of fission gas release at the steam jet air ejector.

The gradual decrease in iodine-131 activity levels for PWRs indicates increasing fuel reliability. The 1990 fuel reliability percentages (fuel related problems only) for the four reporting domestic vendors were all 99.998% or above for Zry-4 clad fuel in PWRs,^(1,3,4) and 99.994% or above for BWR zircaloy clad fuels (99.981% with Crud Induced Localized Corrosion (CILC) failures).^(2,3)

Primary fuel failure causes during 1990 were debris-induced fretting (see below) for PWR fuel and pellet-cladding interaction (PCI - see next page) and CILC for BWR fuel. Descriptions of the major failure mechanisms and summaries of the major non-fuel core related problems are provided below. However, the

cause of a large percentage (approximately 50%) of both PWR and BWR fuel failures is unknown.

No conclusions may be drawn regarding the reliability of BWFC fuel performance since no information was provided from BWFC.

1.2.1 Primary Causes of Fuel Failure in PWRs - Debris Fretting

The major cause of fuel failures in PWRs continues to be debris-induced fretting. Westinghouse estimates that this mechanism causes 70% of the identified fuel breaches.⁽⁴⁾ Similarly, ABB CENF estimates that approximately 75% of their fuel failures are due to debris-induced fretting.⁽¹⁾ This process occurs when bits of metallic debris in the primary coolant, resulting from maintenance operations, wear products, or broken parts of reactor components, are swept through the system and get trapped by the orifices at the bottom of the fuel assembly spacer grids or other restricted areas of the fuel assembly. Vibrations induced by coolant flow cause the debris to rub against the fuel cladding until a breach develops. Following detectable increases in iodine-131 coolant activity during reactor operation, the fuel failure is generally confirmed during reactor shutdown periods by sipping, ultrasonic testing, eddy current testing and/or visual observation.

In recent years utilities have taken aggressive actions to halt debris-induced fretting failures of fuel. First, a major effort has been made throughout the industry to prevent further introduction of debris into the system, with apparently good success. And second, a number of design changes are being tested by the various fuel vendors to minimize the effects of fretting once debris is introduced into the system. SNP⁽³²⁾ and BWFC have extended the length of end fittings, most of which is solid stock. BWFC has also lowered the spacer grid to take advantage of the solid portion of the end cap. ABB CENF⁽²⁴⁾ mainly employs long bottom end caps on the fuel rods or the GUARDIANTM bottom spacer grid design to combat debris-induced fretting failures (see section 3.1.1.2). Westinghouse employs the Debris Filter Bottom Nozzle (DFBN) with smaller holes for the same purpose.

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

General Electric has found PCI of non-barrier fuel and CILC to be the two major causes of cladding perforation in BWRs in recent years.⁽²⁾ Pellet-cladding interaction occurs when the fuel pellet and fuel rod cladding interact together causing the cladding to achieve localized stresses greater than the ultimate tensile strength of the cladding. This interaction occurs primarily during thermal expansion associated with power changes. The effects of PCI can thus be reduced by a slow ascent to full reactor power, but efficiency is lost by this method. GE has developed and fully implemented Zr-lined cladding, commonly referred to as barrier cladding. Since 1983 GE has not reported fuel failures due to PCI in fuel rods with barrier cladding, with over 920,000 rods having been irradiated up to December 1990.

General Electric estimates that CILC has caused approximately 80% of all GE BWR fuel rod failures. CILC only occurs in plants with copper alloy condenser tubes and filter demineralizer condensate cleanup systems, under very specific conditions. One accepted solution is to monitor the coolant chemistry carefully and eliminate sources of copper from the coolant loops. Research is also being pursued to develop Zirconium alloys that are more resistant to corrosion.

1.3 Non-Fuel Core Related Problems

In addition to the fuel integrity problems, several recurring non-fuel related incidents continue to occur and require mention. These problems have been associated with non-fuel core components: fuel handling, and control rod systems.

- Non-Fuel Core Components - There was one reported event involving damaged guide pins. There were no events reported in 1990 relating to the thinning of in-core instrumentation tubes.
- Fuel Handling - There were 21 domestic and 9 foreign events reported that were related to fuel handling problems in 1990.
- Control Rod Systems - There were 35 domestic and 7 foreign events reported that were related to control rod systems in 1990.

2.0 Introduction

This report is the thirteenth in a series which provides a compilation of the available information on nuclear reactor fuel performance. These reports were developed as a result of interest expressed by the

public, advising bodies, and the U.S. Nuclear Regulatory Commission (NRC) for the public availability of nuclear fuel performance information. The previous reports are listed below in Table 3.

Table 3. List of Previous Fuel Performance Annual Reports

Report Number	Reference Number	Year Reported On
NUREG-0633	6	1978
NUREG/CR-1818 (PNL-3583)	7	1979
NUREG/CR-2410 (PNL-3953)	8	1980
NUREG/CR-3001 (PNL-4342)	9	1981
NUREG/CR-3602 (PNL-4817)	10	1982
NUREG/CR-3950 (PNL-5210) Vol. 1	11	1983
NUREG/CR-3950 (PNL-5210) Vol. 2	12	1984
NUREG/CR-3950 (PNL-5210) Vol. 3	13	1985
NUREG/CR-3950 (PNL-5210) Vol. 4	14	1986
NUREG/CR-3950 (PNL-5210) Vol. 5	15	1987
NUREG/CR-3950 (PNL-5210) Vol. 6	16	1988
NUREG/CR-3950 (PNL-5210) Vol. 7	17	1989

As noted in the first report of this annual series (Ref. 6), 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.⁽¹⁸⁾ 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.⁽²⁾

In May 1982, the NRC proposed amending its regulations to improve the information received in Licensee Event Reports (LERs) from nuclear power plant licensees (Ref. 19a-d). 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

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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*^(20a-g) was 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)^(21a-c).

As a result, the primary intent of this report series is to summarize fuel design changes, fuel surveillance programs, fuel operating experience, fuel system problems (especially generic ones) that developed during the reporting period, high-burnup fuel experience, and items of general significance. The reports contain extensive reference lists so that readers may refer to the actual reports and publications to acquire further details regarding the topics included in these annual reports.

This report specifically focuses on calendar year 1990 events and information. The information contained in this report was gathered from vendor reports and

various nuclear power oriented publications. It is important to note that only four of the five fuel vendors submitted reports this year. BWFC did not respond and, therefore, 1990 data from BWFC were not included in this report. However, data from BWFC's 1989 report are included where appropriate.

Modifications to the report's format were made this year in an effort to present information clearly and enhance readability. An effort has been made in Section 3 (Fuel Design Changes and Summary of Surveillance Programs) to better compare the information submitted by the vendors. Although the fuel manufacturers have differing approaches to improving their products, they deal with similar problems. The section is therefore a summary of all the design changes reported by each vendor with a focus on the common problem areas. Section 5 has been changed to include all events of interest (previous reports had split data into two or more sections). The events are listed in sections according to the problem that occurred. For example, if a fuel assembly is dropped and the accident is attributed to a defective procedure, the event is listed as a fuel handling incident, not a procedural incident. At the end of the section is a list providing cross-references to all event categories that were not specifically listed.

3.0 Fuel Design Changes and Summary of Fuel Surveillance Programs

Section 4.2, "Fuel System Design", of the NRC Standard Review Plan⁽²²⁾ requires that plans for testing, inspection, and surveillance of fuel be submitted and reviewed for each domestic nuclear power plant. The plans should include 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 Ref. 23.

Provided below is a summary of current research and development programs and fuel surveillance programs for each of the four reporting fuel vendors. This year's report presents the data differently than previous reports. The nuclear fuel fabrication vendors are continually developing new designs and upgrades for their fuel. Certain trends have emerged in this development that allows organizing the resulting data in a more appropriate manner. Since the trend in fuel design is to extend burnup, fuel reliability, and safety, fuel vendors have focused on a few major areas in fuel research and development. Fuel manufacturers for PWRs have focused on extending fuel burnups, cladding corrosion, burnable absorbers and fuel damage prevention while fuel manufacturers for BWRs have focused on extending fuel burnup, prevention of cladding corrosion and pellet-cladding interaction. The various fuel vendors have developed different methods and approaches to these issues. The following sections serve to illustrate the various programs and their results. Following the discussions of fuel design improvements there is a section on fuel surveillance programs. Some of the newer and more significant programs are discussed for each vendor and the programs are summarized in Table 8 at the end of this section.

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 1990.

Typical fuel assembly parameters associated with each vendor's fuel are provided in Appendix A. Information was not submitted to the NRC by BWFC for the 1990 calendar year as per Reference 5. Details about BWFC's research and development programs have therefore been omitted.

3.1 ABB Combustion Engineering Nuclear Fuel (ABB CENF) -(PWRs)

The following information is taken from ABB CENF's 1990 letter report.⁽¹⁾ Some more recent information is found in the transcript for the American Nuclear Society's (ANS) meeting in Avignon, France from April 21-24, 1991.⁽²⁴⁾

3.1.1 Research and Development

No specific new design changes were noted by ABB CENF in their 1990 letter report.⁽¹⁾ However, ongoing research and development programs were discussed in Reference 24, from which the following information was taken.

3.1.1.1 Burnable Absorbers

ABB Combustion Engineering Nuclear Fuel has developed experience with mixed erbia(Er_2O_3)/ UO_2 fuel used to supplement reactivity control. ABB CENF has found that by using erbia in a significant fraction of the fuel rods it is possible to keep individual fuel rod concentrations low. ABB CENF believes that erbia has some advantages over gadolinia and boron as a burnable poison. Erbia has a smaller cross sections 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 4 Lead Fuel Assemblies (LFAs)⁽²⁵⁾ with 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 4 LFAs that contain fuel pins of 1.5

Design/Surveillance

wt% Erbium and 3.65 wt% enriched UO_2 , and were inserted in San Onofre Unit 2 in 1991.⁽²⁴⁾

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.⁽¹⁾ Three new designs have been developed by ABB CENF to combat this problem and 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 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 discussed above is summarized in Table 4.

3.1.2 Surveillance and Performance Programs

High burnup, extended cycle operation concerns are being addressed by a series of LFA programs as discussed in ABB CENF's 1990 report,⁽¹⁾ and in the

1988 and 1991 International Topical Meetings on LWR Fuel Performance.^(24,26) Both standard and advanced fuel designs are being evaluated. The performance programs currently in progress will provide hot cell evaluation of fuel and cladding with peak local burnup approaching 70 GWd/MTU. These programs include:

- Zry-4 fuel rod and assembly guide tube growth
- Zry-4 fuel rod corrosion behavior
- Erbium-Uranium fuel behavior was discussed above in Section 3.1.1.1

These and other programs are summarized in Table 8 at the end of this section.

3.2 General Electric (GE) - (BWRs)

The following GE design and surveillance information was obtained from References 2 and 27.

3.2.1 Research and Development

GE has made a variety of design modifications over the years to improve fuel corrosion resistance and overall fuel performance. Modified features include water rod configuration, spacer and upper tie plates, cladding surface treatment (involving material and

Table 4. Flow Test Results for Different ABB CENF Debris-Resistant Fuel Designs⁽²⁴⁾

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

heat treatment), axial zoning of gadolinia, fuel rod helium prepressurization, pellet dimensions, and pellet density.

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. GE estimates that approximately 80% of all GE 8x8 BWR fuel rod failures are a result of CILC.⁽²⁷⁾ 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 lead use assembly's (LUAs), discussed below, have been tested to determine the effectiveness of the manufacturing improvements.

3.2.1.2 Fuel Design Improvements

Another form of fuel failure that occurs in BWRs is PCI. This problem was first addressed by GE in 1979 with the introduction of barrier cladding fuel design. This fuel design incorporates a zirconium lining on the inside of the Zircaloy-2 (Zry-2) cladding. The cladding design alleviates stresses caused by the fuel expanding faster than the cladding during power ramping. GE approximates that 14% of all GE 8x8 fuel failures occurred due to PCI in non-barrier fuel.⁽²⁷⁾ As of December 1990, over 920,000 GE barrier rods have operated for at least one cycle with no observed failures due to PCI⁽²⁾.

3.2.2 Surveillance and Performance Programs

The fuel surveillance program adopted by GE and accepted by the NRC is described in four reports.^(28,29,30,31) A summary of the GE LUA surveillance program is provided in Table 8, located at the end of this section. A more detailed description of GE LUA programs is presented below.

- 1983 LUAs - Four LUAs were loaded into Peach Bottom-3 in 1983 at the beginning of cycle 6 to test improved spacer and upper tie plate designs, axial zoning of gadolinia, variations in cladding thickness, pellet dimensions, and fuel rod helium prepressurization. Poolside examination after one cycle in August 1985 and after two cycles in November 1987 showed characteristics of normal operation. Peach Bottom-3 returned to service in December 1989. The third poolside examination is planned for October 1991.^(2,17)
- 1984 LUAs - Five LUAs were loaded into Duane Arnold in 1985 at the beginning of cycle 8 to test water rod configuration, improved spacer and upper tie plate designs, cladding surface treatment axial zoning of gadolinia, and variations in fuel rod helium prepressurization, pellet dimensions and pellet density. 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.^(2,17)
- 1987 LUAs - Four GE 8x8NB LUAs were loaded into Hatch-1 in 1987. 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. The third cycle of operation is scheduled for 1991.
- Four GE 8x8NB LUAs were loaded into Peach Bottom-2 in 1989. The first examination of these bundles is anticipated in 1991.^(2,17)
- Cladding Corrosion Performance LUAs - Two programs have been developed in this area. Six LUAs were loaded into Hatch-2 in early 1988 for cycle 8 and six were loaded into Hatch-1 in late 1988 for cycle 12. The LUAs were designed to test cladding material process variables such as cladding material, heat treatment, and surface

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conditioning. The Hatch reactors have historically exhibited highly variable cladding corrosion performance. Three bundles from Hatch-1 were examined in late 1989 and another three bundles were examined from Hatch-2 in early 1990, both after one cycle of operation. Exposures were up to 13 GWd/MTU and revealed little or no nodular corrosion along the full length of the fuel rods. The next examinations are scheduled for March 1991 and October 1991.^(2,17)

- GE 8x8NB-1 Channel LUAs - Four LUAs were loaded into Cooper in 1988 for cycle 12. Examinations were completed in April 1989 after one cycle of operation and in March 1990 after two cycles of operation and both investigations showed characteristics of normal operation.^(2,17)
- GE11 LUAs - Four LUAs were loaded in each of three reactors (two BWR 4s and one BWR 5). The first evaluations of these bundles are scheduled to begin 1991.⁽²⁾

Barrier Program LUAs

- The first PCI resistant fuel was loaded at Quad Cities-1 in 1979 for irradiation in cycle 5. This test consisted of four LUAs. The LUAs operated for up to 5 cycles and underwent 5 poolside examinations, all of which demonstrate normal behavior. Six rods were then removed from one of the assemblies (at 43 GWd/MTU) and placed in another less exposed assembly and exposed for a sixth cycle. These rods were examined in November 1990 and were showed to have exhibited normal behavior. The rods were then re-inserted into the core for Quad Cities-1 cycle 12.⁽²⁾
- Quad Cities-2 had a group of 144 barrier bundles placed in its core in 1981 for cycle 6. These bundles were to undergo significant power increases to test the integrity of the fuel during such an event. At the end of cycle 6 in 1983 and again at the end of cycle 7 in 1985 power increases were performed on sixteen bundles in each case. The assemblies were tested by offgassing and had no detectable leaks. All PCI restrictions were then removed from barrier fuel

bundles as a result of this positive test. Of the 144 bundles, 32 operated for 3 cycles, 80 operated for 4 cycles, 32 operated for 5 cycles, and 16 are operating in their 6th cycle.⁽²⁾

3.3 Siemens Nuclear Power Corporation (SNP)

[formerly Advanced Nuclear Fuels Corporation (ANF)] - (PWRs and BWRs)

The following information was obtained from SNP's annual report⁽³⁾ and a report presented by SNP at the International Topical Meeting on Fuel Performance in Avignon, France.⁽³²⁾

3.3.1 Research and Development

Siemens Nuclear Power Corp. manufactures fuel for both PWR and BWR type reactors. Although SNP did not mention any new design changes in their 1990 report⁽³⁾ several developments for each type were in progress previously and were discussed during the 1991 Fuel Performance meeting in Avignon.⁽³²⁾

3.3.1.1 Corrosion Resistance

Siemens Nuclear Power Corp. 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. Siemens Nuclear Power Corp. believes that this improves corrosion performance in plants susceptible to CILC and also eliminates waterside corrosion as a fuel burnup limiting factor.

Siemens Nuclear Power Corp. 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.⁽³²⁾

3.3.1.2 Burnable Absorbers

Siemens Nuclear Power Corp.'s research with burnable poisons has been conducted with gadolinia. SNP has obtained PWR burnups of 30 GWd/MTU using 10 wt% Gd₂O₃ and 50 GWd/MTU using 8 wt% Gd₂O₃. No difference has been found in the operating characteristics of this fuel when compared with standard UO₂ fuel rods.

3.3.1.3 Fuel Design Improvements

Siemens Nuclear Power Corp.'s 9x9 fuel type for BWR's has been developed with several different configurations of water rods 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. SNP has used axial zoning of gadolinia to improve uranium utilization and cold shutdown margins.⁽³²⁾

There have been several developments in SNP's 17x17 fuel type for PWR 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 SNP to prevent baffle jetting and, according to SNP, essentially eliminated that problem. Another improvement has been to develop high thermal performance spacers to improve heat transfer and lower cladding temperatures. This improves the fuel's departure from nucleate boiling margin.⁽³²⁾

A major area of development at SNP has centered around reducing damage to fuel from debris. SNP has

tested two design changes for the lower tie plates that will reduce fuel susceptibility to damage from debris.⁽³²⁾ In addition, SNP was the first to develop extended-length solid end caps to solve debris fretting problems because most of the failures occurred at the bottom of the fuel rod. The solid end cap is then exposed to the highest risk area for fretting, thus protecting the fuel containing portion of the fuel rod.⁽³²⁾ The first change in the lower tie plate design incorporates small flow holes to trap particles. A large number of 6 mm 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.⁽³²⁾ SNP has also developed a second 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.⁽³²⁾ The effectiveness of the different designs to trap different types of debris is shown in Table 5.

Table 5. Comparison of SNP Standard and Debris Resistant PWR Fuel Assembly Designs⁽³²⁾

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.

3.3.2 Surveillance and Performance Programs

Siemens Nuclear Power Corp.'s surveillance programs are summarized in Table 8 at the end of this section.

3.4 Westinghouse Electric Corporation (W) - (PWRs)

The W report WCAP-8183 Rev. 19, "Operational Experience with Westinghouse Cores"⁽⁴⁾ is the basis for the following section.

3.4.1 Research and Development

No new design changes were specifically noted in W's 1990 report⁽⁴⁾ but an overview of recent design changes and improvements is provided and additional design information is given in an article by M. G. Balfour et al.⁽³³⁾ Ongoing developments are presented in the following sections.

3.4.1.1 Burnable absorbers

Westinghouse uses Integral Fuel Burnable Absorber (IFBA) rods in their VANTAGE 5 and VANTAGE 5H fuel types. Approximately 80% of the fuel pellets in an IFBA rod are coated with ZrB_2 which serves as a burnable absorber.⁽⁴⁾

3.4.1.2 Fuel Design Improvements

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 1990 the DFBN was used in at least one region of fuel in 34 of the 59 W fueled reactors.⁽⁴⁾

Optimized Fuel Assemblies (OFAs) were developed for 14x14 and 17x17 fuel assemblies to improve fuel utilization by enhancing neutron moderation and reducing parasitic capture. Optimized Fuel Assemblies have a slightly smaller rod diameter than

normal, but retain the same rod pitch. The top and bottom grids in the OFAs are Inconel and the intermediate grids are made of Zry-4.⁽⁴⁾

Westinghouse has also developed VANTAGE 5 and VANTAGE 5H fuel. Each of the VANTAGE fuels were designed to enhance cost effectiveness, improve core operating margins, and improve operating flexibility. The features of the VANTAGE 5 include all of those in the OFA and also the following:

- Integral Fuel Burnable Absorbers (IFBAs) - described above in Section 3.4.1.1.
- Intermediate Flow Mixing Grids (IFMs) - these are grids located between the Zircaloy structural grids in the upper parts of the assembly. They enhance flow turbulence resulting in an increase in the departure from nucleate boiling margin.
- Axial Blankets
- Reconstitutable Top Nozzle
- Increased Discharge Burnup

The features of the VANTAGE 5H are the same as the VANTAGE 5 except that a standard fuel rod is used instead of the OFA configuration and the VANTAGE 5H also employs a new low-pressure-drop Zircaloy grid design.⁽⁴⁾

Westinghouse is currently testing ZIRLOTM cladding containing niobium to provide better corrosion resistance, allowing for longer burnups and/or higher operating temperatures.^(4,33) The first two demonstration assemblies were irradiated up to 21 Gwd/MTU in their first cycle and one has been re-inserted. The second cycle is expected to achieve 37 Gwd/MTU in early 1991.⁽³⁴⁾

3.4.2 Surveillance and Performance Programs

Westinghouse's fuel surveillance programs are summarized in Table 8. 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.⁽⁴⁾
- 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.⁽⁴⁾
- Optimized Fuel Assemblies - From 1979 to 1986, ten demonstration OFAs (six 17x17 and four 14x14) were irradiated in four reactors (Point Beach-2, Beaver Valley-1, Salem-1, and Farley-1) to assembly-average burnups ranging from 33.8 to 53.0 GWd/MTU. All assemblies were discharged in good condition except one that suffered fretting wear due to a nonstandard step in the manufacturing process. In 1990 thirty plants operated with at least one region of OFA fuel. Observations of OFA fuel at more than 20 plants showed good performance.⁽⁴⁾ The 1990 statistics presented in Table 6 are for at least one OFA region in-core:

Table 6. Number of Plants with Westinghouse OFAs⁽⁴⁾

Cycle	Number of Plants	Peak Batch Avg. Burnup (GWd/MTU)
1	30	27
2	26	38
3 or 4	24	45

- VANTAGE 5 and VANTAGE 5H Fuel Assemblies - Four VANTAGE 5 17x17 assemblies began testing at V. C. Summer Unit 1 during cycle 2 in December, 1984. The assemblies completed three cycles and were discharged in September 1988 with average burnups of 46.0 GWd/MTU. No mechanical wear or damage was evident on any of the

assemblies and the IFM grids did not have adverse effects on any adjacent fuel assemblies.⁽⁴⁾

- Integral Fuel Burnable Absorber demonstration fuel rods have been tested at Turkey Point Units 3 and 4.⁽⁴⁾
- Intermediate Flow Mixing grids have been used at McGuire Unit 1.⁽⁴⁾
- Full regions of reload fuel with at least one VANTAGE 5 fuel feature were in operation in 38 plants during 1990. This includes:
 - 15 plants using axial blankets
 - 36 plants using reconstitutable top nozzles
 - 14 plants with IFBAs
 - 6 plants with IFM grids
 - 25 plants with assembly modifications for higher burnups
 - 5 plants operated with a full region of VANTAGE 5 fuel
 - 6 plants operated with a full region of VANTAGE 5H fuel
- ZIRLO™ Demo Assemblies - Two assemblies began operations in North Anna Unit 1 in June, 1987. The ZIRLO™ rods were examined in February 1989 and achieved a burnup of 21 GWd/MTU in their first cycle. One assembly was re-inserted and should achieve an average burnup of 37 GWd/MTU in early 1991.⁽⁴⁾

3.5 Babcock & Wilcox Fuel Company (BWFC) - (PWRs)

BWFC has declined to provide information for this report as per Reference 5 and, therefore, no information is provided in these sections for calendar year 1990.

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3.5.1 Research and Development

BWFC has declined to provide information for this report as per Reference 5 and, therefore, no information is provided in these sections for calendar year 1990.

3.5.2 Surveillance and Performance Programs

BWFC has declined to provide information for this report as per Reference 5 and, therefore, no information is provided in these sections for calendar year 1990.

3.6 Electric Power Research Institute (EPRI) Programs

The current direction of the EPRI fuel surveillance program is given in a 1991 paper published in the Proceedings from the International Topical Meeting on LWR Fuel Performance in Avignon, France.⁽³⁵⁾

The paper describes EPRI's current projects and the trends that EPRI predicts for fuel research and development. The program's main objectives are listed below:

Table 7. References for EPRI Fuel Programs

Title	Reference #	Year
Guidelines for Improving Fuel Reliability	288	1989
EPRI Research and Development Program Plan, 1987-1989	289	1987-1989
Review of Pacific Northwest Laboratories' Test Program Results	290	1987
Advances in Light Water Reactor Fuels	291	1987
Hydrogen Water Chemistry for BWRs	292	1986
Lifetime of PWR Silver-Indium-Cadmium Control Rods	293	1986
Collection and Formatting of Data on Reactor Coolant Activity and Fuel Rod Failures	294	1986
Utility Experience with the BWR Power Shape Monitoring System	295	1986
Demonstration of 9x9 Assemblies for BWRs	296	1984
Comparison of Advanced BWR Fuel Designs with Current Standard Designs	297	1984
LWR Core Materials Program: Progress in 1983-1984	298	1983-1984
Phenomena Associated with Extending Fuel Burnup	299	1982
The Schedule for Extending Fuel Burnup	300	1982
The EPRI LWR Fuel Surveillance Program	301	1982
LWR Core Materials Program: Progress in 1981-1982	302	1981-1982

1. Solving current fuel failure mechanisms.
2. Determining "unknown" failure mechanisms.
3. Monitoring more closely system parameters to determine effects on fuel operations and to identify fuel problems early in their development.
4. Further research for extended burnup operations (i.e. modeling and data compilation).
5. Recommendations for industry to help reduce the problems associated with failed fuel.

Other topics discussed in the paper are:

- The goal of zero-failures and which problems need to be further addressed in order to be

able to realistically approach the "no-leaker" limit.⁽³⁵⁾

- Development of the Cost of Fuel Failure Evaluation Code (COFFEC) - This code is used to evaluate cost impact on the U.S. nuclear industry incurred by fuel failures. The COFFEC study of the 1988-1989 period indicated that most of the incurred industry-wide cost results from fewer than 10% of the plants.⁽³⁵⁾

Additional information on EPRI research programs has been published over the past several years in numerous documents whose topics are listed in Table 7.

Table 8. Major Fuel Surveillance Programs: Status Through 1990

Vendor	Fuel Type ^(a)	Power Plant	Planned # (Completed #) Operating Cycles	Scheduled Completion of Program	Inspections to Date
Siemens Nuclear Power Corporation (Formerly Advanced Nuclear Fuels)	15 X 15	Robinson-2	5(5)	Complete	3
	14 x 14	Prairie Island-2	3(3)	Complete	1
	8 x 8	Oyster Creek	5(5)	Complete	5
	11 x 11	Big Rock Point	4(4)	Complete	3
	14 x 14	Ginna	5(5)	1990	3
	17 x 17	Blayais-3	4(4)	1990	3
	8 x 8	WNP-2	4(3)	1991	3
	14 x 14	Calvert Cliffs	3(0)	1993	0
	15 x 15	Palisades	3(1)	1993	1
	9 x 9	Hatch-2	3(1)	1991	1
	9 x 9	Hatch-1	3(1)	1995	1
BWFC (partnership between Babcock & Wilcox and the American subsidiary of a French consortium) ^(yy)	15 x 15	Oconee-1	5 ^(b)	Complete	3
	15 x 15 ^(c)	Arkansas-1 ^(d)	4	Complete	3
	15 x 15 ^(e)	Rancho Seco	2	1990	2
	15 x 15 ^(f)	Oconee-2	4	Complete	4
	15 x 15 ^(g)	Oconee-2	3	Complete	3
	15 x 15 ^(h)	Oconee-2	1	Complete	1
	15 x 15 ^(h)	Oconee-1	3	Complete	3
	17 x 17 ⁽ⁱ⁾	Oconee-2	3	Complete	3
	15 x 15 ^(j)	Oconee-1	4	1990	3
	15 x 15 ^(k)	Oconee-2	3	Complete	2
	15 x 15 ^(l)	Arkansas-1	4	Complete	3
	15 x 15 ^(m)	Haddam Neck ⁽ⁿ⁾	3	Complete	2
	17 x 17 ^(o)	McGuire-1	3	1991	2

Table 8. (cont.)

Vendor	Fuel Type ^(a)	Power Plant	Planned # (Completed #) Operating Cycles	Scheduled Completion of Program	Inspections to Date
ABB Combustion Engineering Nuclear Fuel	14 x 14 ^(p)	Calvert Cliffs-1	5(5)	Complete	5
	14 x 14 ^(p)	Fort Calhoun	6(6)	Complete	4
	14 x 14 ^(q)	Calvert Cliffs-1	5(5), Pt 1	Complete	5
	14 x 14 ^(q)	Calvert Cliffs-1	5(5), Pt 2	1993 ^(r)	5
	14 x 14 ^(s)	Calvert Cliffs-2	3(0)	1997	0
	16 x 16 ^(s)	Arkansas-2 ^(t)	3(3)	Complete	3
	16 x 16 ^(u)	Arkansas-2	3(3)	Complete	3
	16 x 16 ^(v)	Arkansas-2	5(5)	1992 ^(r)	5
	16 x 16 ^(s)	St. Lucie-2	3(2)	Complete	1
	16 x 16 ^(u)	Palo Verde-1	3(2)	Complete	3
	16 x 16 ^(w)	Palo Verde-1	3(1)	1994	2
	16 x 16 ^(w)	Palo Verde-3	3(0)	1997	0
	16 x 16 ^(s)	San Onofre-2	2(0)	1995	0
	14 x 14 ^(ww)	Maine Yankee	12(12)	1991	3
General Electric	Barrier LUAs ^(x)	Quad Cities-1	7(6)	--	6
	Barrier LUAs ^(x)	Quad Cities-2	5(6)	--	--
	1981 LUAs ^(y)	Browns Ferry-3	(1)	--	--
	1983 LUAs ^(z)	Peach Bottom-3	3(2)	1991	2
	1984 LUAs ^(aa)	Duane Arnold	4(3)	--	3
	1987 LUAs ^(bb)	Hatch-1	3(2)	1991	2
	Corrosion performance ^(cc)	Hatch-1 & 2	1(2)	1991	1
	1988 LUAs ^(dd)	Cooper	(2)	--	--
	1989 LUAs ^(xx)	Peach Bottom-2	(1)	1991	2
	GE11 LUAs	3 Reactors	(1)	1991	0

Table 8. (cont.)

Vendor	Fuel Type ^(a)	Power Plant	Planned # (Completed #) Operating Cycles	Scheduled Completion of Program	Inspections to Date
Westinghouse	(ee)	North Anna-1	4(4)	complete	--
	17 x 17 (OFA- Demo) ^(hh)	Farley-1	4(4) ^(ff)	complete	4(hh)
			5(5) ⁽ⁱⁱ⁾	complete	5
	17 x 17 (OFA- Demo) ^(hh)	Salem-1	4(4) ^(ff)	complete	3
	17 x 17 (OFA- Demo) ^(hh)	Beaver Valley-1	3(3) ^(jj)	complete	3
	14 x 14 (OFA- Demo) ^(kk)	Point Beach-2	4(4) ^(ll)	complete	4(nn)
	17 x 17 (VANTAGE-5 Demo)	Summer-1	3(3) ⁽ⁿⁿ⁾	complete	1
	IFBA Demo Fuel Rods ^(oo)	Turkey Point-3	(2)	--	--
	IFBA Demo Fuel Rods ^(pp)	Turkey Point-4	(2)	--	--
	IFM Demo Assembly ^(qq)	McGuire-1	(2)	--	--
	DFBN Assembly ^(rr)	3 Plants	--	--	--
	ZIRLO-Clad Fuel Rod Assembly ^(tt)	North Anna-1	3(1) ^(ss)	--	--
	MO ₂ ^(uu)	R. E. Ginna	4(4) ^(vv)	--	--

- (a) LTA = lead test assembly, LUA = lead use assembly, MO₂ = mixed oxide (UO₂-PuO₂) fuel, R = retrofit fuel design. D = demonstration, OFA-Demo = Demonstration Optimized Fuel Assembly, IFBA = integral fuel burnable absorber, IFM = intermediate flow mixer, FPIP = Fuel Performance Improvement Program, DFBN = debris filter bottom nozzle, ZIRLO = an advanced zirconium alloy cladding that contains niobium.
- (b) For this entry, and the following entries for BWFC, scheduled completion means completion of irradiation.
- (c) LTAs of an advanced, extended-burnup design.
- (d) Arkansas Nuclear One-Unit 1 (also known as ANO-1).
- (e) Current-design assemblies containing axially-blanketed fuel columns.
- (f) Current-design assemblies with special Zircaloy cladding materials and EPRI creep collapse specimen clusters. (See Ref. 36a-h)
- (g) Current-design assemblies with lifted rods and cladding having a known spiral eccentricity in wall thickness. (See Refs. 37 and 38)
- (h) Current-design assemblies utilizing low-absorption spacer grid material (Zry-4).
- (i) Two of these four LTAs are reconstitutable. (See Ref. 39)
- (j) Gadolinia LTAs of an advanced, extended-burnup design.
- (k) Pathfinder LTA with 12 fuel rods with advanced Zircaloy cladding materials: 6 rods have cladding with pure zirconium liners on the inside surface of the Zircaloy cladding and 6 rods have beta-quenched, Zry-4 tubing (Ref. 40).
- (l) Same as (c); additional cycle of irradiation.
- (m) Four LTAs with Zry-4 clad fuel rods to replace fuel assemblies with stainless steel-clad fuel rods.
- (n) Haddam Neck is also known as Connecticut Yankee.
- (o) Four 17 x 17 LFAs (Mark-BW LA).
- (p) Standard-design, high-burnup program.
- (q) Standard and advanced fuel design LTAs.
- (r) Hot cell examination of high burnup fuel yet to be performed.
- (s) Burnable poison irradiation program.

- (t) Arkansas Nuclear One-Unit 2 (also known as ANO-2).
- (u) Standard surveillance program.
- (v) Standard and advanced fuel design, high-burnup program.
- (w) Advanced cladding designs.
- (x) Four bundles with barrier cladding at Quad Cities-1 were involved. Cycles 6 & 7 involved 6 rods removed from the initial bundles and placed in another assembly for further irradiation. At Quad Cities-2, 144 barrier bundles were used, 16 of which continue to be irradiated in their sixth cycle.
- (y) Eight bundles with improved design features involved.
- (z) Four bundles with improved design features involved.
- (aa) Five bundles with improved design features involved.
- (bb) Four bundles. Program objective: lead use GE 8x8NB.
- (cc) Six fuel bundles each at Hatch 1 & 2. Program objective: cladding material process.
- (dd) Four fuel bundles. Program objective: lead use GE8x8NB-1 features.
- (ee) Eight fuel assemblies were irradiated as part of an EPRI program for their fourth consecutive 18-month operating cycle; four of the eight were in relatively high power positions and attained an assembly average burnup of about 58.1 GWd/MTU at discharge (May 1989); the IFBA average burnup was 58.4 GWd/MTU (Ref. 290).
- (ff) The two OFA-Demo assemblies in Farley-1 and the two assemblies in Salem-1 were discharged in 1984 after 4 cycles for examination. Burnup achieved: 39.1 GWd/MTU in Farley-1, and 34.4 GWd/MTU in Salem-1 (Ref. 18).
- (gg) Note deleted
- (hh) Two OFA-Demo assemblies.
- (ii) One of the two OFA-Demo assemblies was re-inserted for irradiation (fifth cycle) and achieved a burnup of 52.8 GWd/MTU (Ref. 18). One standard fuel assembly (the symmetric partner to the OFA-Demo assembly in Cycle 7) was also irradiated for a fifth cycle and attained an average burnup of 52.1 GWd/MTU (Ref. 18).
- (jj) The two assemblies achieved a burnup of 35.5 GWd/MTU (Ref. 18), were discharged in 1984 after 3 cycles, and were examined.
- (kk) Two assemblies.
- (ll) The four assemblies completed their second cycle of irradiation in 1983. Subsequent examination showed one assembly had nine failed fuel rods (cause: fretting wear at bottom Inconel spacer grid). The other three assemblies were in good condition, were returned to the core for a third and fourth cycle of irradiation, were discharged in 1985 and were examined (see Ref. 18). Average burnup achieved was 40.3 GWd/MTU (Ref. 18).
- (mm) Note deleted
- (nn) The four assemblies completed their third cycle of irradiation and were discharged in 1988 after attaining an accumulated average burnup of 46.0 GWd/MTU.
- (oo) The 4 IFBA rods were monitored during irradiation by in-core instrumentation.
- (pp) There were 28 IFBA rods in each of four demonstration assemblies, which allowed removal of some of the rods for postirradiation examination.
- (qq) One characterized IFM spacer grid demonstration assembly.
- (rr) Three fuel assemblies with DFBNs.
- (ss) 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.
- (tt) 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.
- (uu) Four assemblies with W mixed oxide fuel rods were involved. The mixed oxide (UO₂-PuO) fuel rods for Ginna were manufactured by W but their irradiation was not part of a W development program.
- (vv) 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.
- (ww) Hot cell exam of high exposure control element assembly.
- (xx) Four GE8x8NB bundles.
- (yy) BWFC data through 1989.

4.0 Fuel Operating Experience

The total number of fuel assemblies that were in, or that had completed, operation in U.S. commercial nuclear power plants increased from approximately 110,500 in 1989 to approximately 114,450 at the end of 1990. Of these assemblies 69,350 were used in BWRs and 45,100 were used in PWRs.⁽⁴¹⁾ The total number of fuel rods supplied to the world by the five U.S. nuclear fuel vendors through 1990 was over 16.3 million (11.5 million for PWRs and 4.8 million for BWRs). The data for the number of fuel rods does not include the number of rods that BWFC produced in 1990 since their operating report was not submitted. The number of fuel rods throughout the world in 1989 was over 15.0 million (10.8 million for PWRs and 4.2 million for BWRs).⁽¹⁷⁾

As of the end of 1990 there were 111 operable plants licensed in the United States. U.S. commercial nuclear plants generated 576.9 TWh and achieved a record high average capacity factor of 66.1% for the year.⁽⁴²⁾ In comparison, nuclear plants generated 529.4 TWh and had an average capacity factor of 62.3% in 1989.⁽⁴³⁾

An outline of fuel performance for four of the domestic vendors is provided in the sections that follow. Babcock & Wilcox Fuel Company did not provide an experience summary for 1990 and has therefore been omitted. This section also includes fuel integrity ratings reported by each vendor. These ratings are normally obtained from iodine-131 activity levels initially, followed where possible by gas sipping or ultrasonic measurements; these methods are described in References 44a-j. The Institute of Nuclear Power Operations (INPO) Fuel Reliability Indicator (FRI) has been adopted by several fuel vendors to assess the overall performance of fuel rods. The FRI for PWRs is determined by normalizing the iodine-131 coolant activity level to a standard cleanup system flow rate (also referred to as "uncorrected activity") and correcting for tramp uranium⁽⁴⁵⁾ (referred to as "corrected activity" or FRI value.) For BWRs, the FRI value is determined from the fission gas release measured at the steam jet air ejector. Smaller FRI values are indicative of fewer failed fuel rods in the core. The average coolant iodine-131 activity is typically 1.2×10^{-3} μCi per gram of iodine-131. Levels above this value usually indicate the presence of one or more 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.

Historical information on fuel failure rates is provided in Appendix B. It should be noted that the definition of failed fuel is not uniformly applied;^(46,47,48) in many cases the number of fuel failures is inferred from indirect evidence, while in other cases only directly observed failures are counted.⁽⁴⁹⁾

Overall commercial reactor fuel operating experience continues to be excellent; however, sporadic events involving damage to, or failure of, fuel continue to exist. These events are detailed in Section 5.0.

4.1 ABB Combustion Engineering Nuclear Fuel (ABB CENF)

A summary of ABB CENF fuel performance during 1990 is provided in Reference 1.

4.1.1 Fuel Performance - Fuel Utilization and Burnup

ABB Combustion Engineering Nuclear Fuel fuel that was either in a core, or discharged, in 1990 and the batch-averaged burnups achieved are presented in Table 9. The highest batch-averaged burnup achieved in 1990 was 44.8 GWd/MTU at Arkansas-2. The highest batch-averaged burnup at discharge in 1990 was 44 GWd/MTU at St. Lucie-2. Previous high burnups for ABB CENF fuel were 56.8 GWd/MTU for 4 assemblies during 1988 at Calvert Cliffs-1.⁽⁵⁰⁾

The number of active and discharged ABB CENF assemblies in 1990 is shown in Table 10. The total number of ABB CENF assemblies in 1990 was 8400 (31% in core, 69% discharged). The total number of ABB CENF fuel rods was 1,695,488 (32% in core, 68% discharged).

Calendar year 1990 ABB CENF fuel surveillance programs completed or under development are summarized in Table 8 at the end of Section 3.

Operating Experience

Table 9. Summary of ABB Combustion Engineering Nuclear Fuel, Fuel Irradiated and/or Discharged in 1990⁽¹⁾

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, 1990	At Discharge
Arkansas-2/ (Cycle 8)	F	17	0	4,012	0	44.8	-----
	H	28	0	6,352	0	41.9	-----
	J	68	0	15,312	0	34.4	-----
	K	64	0	14,416	0	15.8	-----
Calvert Cliffs-1/ (Cycle 10)	K	69	0	12,144	0	33.5	-----
	L	52	0	9,152	0	21.3	-----
	M	92	0	15,280	0	10.6	-----
Calvert Cliffs-2/ (Cycle 8) ^a	H	69	0	12,144	0	43.0	-----
	J	60	0	10,560	0	34.0	-----
	K	88	0	14,800	0	22.0	-----
Fort Calhoun/ Cycles 12 & 13	M	41	3	7,048	504	31.8	33.0
	N	44	0	7,552	0	19.1	-----
	P	40	0	6,784	0	5.5	-----
Maine Yankee/ Cycles 11 & 12	N	0	64	0	10,880	-----	40.5
	P	72	0	12,400	0	33.4	-----
	Q	72	0	12,464	0	21.5	-----
	R	72	0	12,448	0	5.2	-----
Palo Verde-1/ Cycles 2 & 3	B	1	96	220	21,120	25.0	30.0
	C	52	12	12,016	2,704	27.0	34.0
	D	80	0	18,528	0	19.0	-----
	E	108	0	24,240	0	7.0	-----
Palo Verde-2/ Cycles 2 & 3	B	1	68	220	14,960	24.0	30.2
	C	36	28	8,496	6,224	26.0	33.5
	D	108	0	24,400	0	18.0	-----
	E	96	0	21,616	0	6.0	-----
Palo Verde-3/ Cycles 1 & 2	A	0	69	0	16,284	-----	15.3
	B	73	35	16,060	7,700	27.0	17.6
	C	64	0	14,720	0	25.0	-----
	D	104	0	23,584	0	15.0	-----
St. Lucie-2/ Cycles 5 & 6	D	0	4	0	944	-----	44.0
	E	12	45	2,800	10,412	36.0	42.0
	F	49	27	11,380	6,156	32.0	34.0
	G	80	0	18,448	0	18.0	-----
	H	76	0	17,456	0	1.0	-----

Table 9. (cont).

Operating Experience

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, 1990	At Discharge
San Onofre-2/ Cycle 5	A	1	0	236	0	21.0	-----
	F	108	0	24,112	0	33.0	-----
	G	108	0	24,112	0	12.5	-----
San Onofre-3/ Cycles 4 & 5	A	1	5	236	1,180	15.0	31.0
	D	0	16	0	3,776	-----	30.5
	E	0	88	0	20,320	-----	35.0
	F	108	0	24,112	0	27.5	-----
	G	108	0	24,112	0	5.5	-----
Waterford-3/ Cycle 4	C	1	0	224	0	34.6	-----
	D	48	0	11,232	0	39.0	-----
	E	84	0	18,896	0	27.6	-----
	F	84	0	18,896	0	8.7	-----
Yankee Rowe/ Cycles 20 & 21	B	0	36	0	8,222	-----	32.0
	C	36	4	8,222	868	17.0	20.0
	D	40	0	9,090	0	1.3	-----

^a Calvert Cliffs-2 did not operate during 1990

4.1.2 Fuel Rod Integrity

ABB Combustion Engineering Nuclear Fuel has made a comparison of the corrected iodine-131 activities in ABB CENF fueled plants during 1987 and 1990. This comparison is illustrated in Table 11.⁽¹⁾ The corrected activities were obtained using the INPO standard FRI method described at the beginning of Section 4.

In Figure 1 the corrected coolant iodine-131 activity is plotted vs. time over the period from 1987-1990. The average plant activity at the close of 1990 was 0.0055 $\mu\text{Ci/g}$ and the median was 0.0027 $\mu\text{Ci/g}$. These values compare well with industry standards as reported by INPO.⁽¹⁾

It is estimated that 75% of the leaking fuel that was fabricated after 1983 (current fabrication process) and operated during the 1987-1990 period 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.⁽¹⁾

The overall reliability of ABB CENF fuel used from 1983 to 1990, excluding failures due to fabrication processes, is estimated to exceed 99.998%.⁽¹⁾

Table 10. ABB Combustion Engineering Nuclear Fuel Burnup Experience With All-Zircaloy Assemblies
Status as of December 31, 1990⁽¹⁾

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	76	40	116	0	17,456	9,090	26,546	0	0	0	0	0	0	0	0
4.000 to 7.999	112	312	0	424	19,232	69,968	0	89,200	6	0	0	6	1,048	0	0	1,048
8.000 to 11.999	92	0	0	92	15,280	0	0	15,280	97	0	136	233	16,148	0	28,751	44,900
12.000 to 15.999	0	361	36	397	0	81,244	8,222	89,466	247	387	72	706	42,935	91,220	15,284	149,439
16.000 to 19.999	44	160	0	204	7,552	39,976	0	44,528	256	192	0	448	44,344	44,448	0	88,792
20.000 to 23.999	212	109	0	321	36,416	24,636	0	61,052	151	104	8	263	25,276	23,392	1,723	50,396
24.000 to 27.999	0	66	0	66	0	15,160	0	15,160	476	478	0	954	81,034	107,256	0	188,290
28.000 to 31.999	41	282	0	323	7,048	63,568	0	70,616	795	326	100	1,221	136,552	73,080	22,090	231,722
32.000 to 35.999	201	296	0	497	35,104	66,816	0	101,920	536	471	36	1,043	93,660	109,320	8,222	211,202
36.000 to 39.999	0	13	0	13	0	3,024	0	3,024	222	210	0	432	39,008	48,296	0	87,304
40.000 to 43.999	69	76	0	145	12,144	17,584	0	29,728	316	151	0	467	54,474	34,988	0	89,462
44.000 to 47.999	0	17	0	17	0	4,012	0	4,012	0	4	0	4	0	944	0	944
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	132,776	400,444	17,312	550,532	3,109	2,324	352	5,785	535,706	533,174	76,076	1,065,265

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

Table 11. Comparison of Corrected^a Coolant Iodine-131 Activities from 1987 and 1990 for Fuel Manufactured by ABB Combustion Engineering Nuclear Fuel⁽¹⁾

Corrected Iodine-131 Activity Range, $\mu\text{Ci/g}$	Percentage of Plants in Range	
	End of 1987	End of 1990
> 0.05	23	0
0.005 - 0.05	38.5	33
0.0005 - 0.005	38.5	67
< 0.0005	0	0
Average Plant Corrected Iodine-131 Activity, $\mu\text{Ci/g}$	0.0304	0.0055
Median Plant Corrected Iodine-131 Activity, $\mu\text{Ci/g}$	0.0181	0.0027

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

Figure 1. Corrected Coolant Activity* Versus Time⁽¹⁾

4.2 General Electric (GE)

A summary of GE fuel performance and fuel rod integrity is provided in Reference 2.

4.2.1 Fuel Performance - Fuel Utilization and Burnup

As of December 31, 1990, over 4.0 million GE 8x8 fuel type production Zircaloy-clad UO_2 rods were in or had completed operation in commercial BWRs. The cumulative number of fuel rods in GE 8x8 bundles loaded as a function of calendar year is illustrated in Figure 2. As of December 31, 1990, over 1.5 million GE fuel rods were in operation. Approximately 1.37 million of these are PCI resistant barrier fuel rods. The cumulative GE core loading by fuel type as of the end of 1990, is presented in Figure 3. All of the fuel that GE now produces is barrier type fuel.⁽²⁾

In 1990, sixteen domestic and six overseas GE BWR plants containing GE fuel had refueling outages with over 3300 new GE 8x8 fuel bundles loaded. Nearly 80% of this new fuel loaded was GE's latest production designs (GE 8X8EB and GE 8X8NB).⁽²⁾

The experience of GE production and developmental BWR Zircaloy-clad UO_2 fuel rods through December 31, 1990 included successful commercial operation of fuel bundles to greater than 45 GWd/MTU bundle average exposure. This equates to approximately 60 GWd/MTU peak pellet exposure.⁽²⁾

4.2.2 Fuel Integrity

Table 12 was taken from Reference 27; it summarizes GE fuel rod reliability as of August 1990.

Table 12. GE 8X8 Fuel Performance (August 1990)⁽²⁷⁾

	All 8x8	Zirconium Liner 8x8
Date Introduced In Manufacturing	1973	1983
Cumulative Fuel Rods Loaded	3,900,000	1,250,000
Fuel Rod Reliability ^a		
- Including CILC Failures	99.981	99.988
- Excluding CILC Failures	99.996	99.998

^a Based on fuel rods completing at least one cycle of operation.

Figure 2. GE 8x8 BWR Fuel Rod Experience⁽²⁾

Figure 3. GE BWR Fuel Rods in Operation on 12/31/90⁽³⁾

4.3 Siemens Nuclear Power Corp. (SNP)

Siemens Nuclear Power Corp. fuel performance and fuel rod integrity for 1990 are described in Reference 3.

4.3.1 Fuel Performance - Fuel Utilization and Burnup

As of December 31, 1990, fuel manufactured by SNP had been loaded into 50 commercial light water reactors (LWRs) in the United States, Europe, and Asia, including 23 BWRs and 27 PWRs. SNP fuel has also been supplied to the Loss of Fluid Test (LOFT) test reactor. Siemens Nuclear Power Corp. fuel comprises a total of 18,412 fuel assemblies containing

2,199,446 fuel rods that had been irradiated. Of these, 64% of the assemblies were irradiated in BWRs and 36% of the assemblies were irradiated in PWRs. Siemens Nuclear Power Corp. fuel experience is summarized in Table 13 and burnup distributions are shown in Figure 4.⁽³⁾

The highest assembly averaged burnups reached by SNP fuel to date is 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. BWR 9x9 and PWR 17x17 fuel assemblies reached new high burnups during 1990. The highest exposures reached by BWR 9x9 and PWR 17x17 fuel are 40.0 GWd/MTU at Gundremmingen-C in Germany and 46.4 GWd/MTU at D.C. Cook-2 in Michigan, respectively.⁽³⁾

Table 13. Summary of Siemens Nuclear Power Corp. Fuel Experience Through December 31, 1990⁽³⁾

A: Fuel Assemblies

Reactor Type	In Core		Discharged		Total Quantity
	Quantity	Max Burnup GWd/MTU	Quantity	Max Burnup GWd/MTU	
BWR	8,147	40.0	3,635	45.1 ^a	11,782
PWR	2,172	52.1	4,458	52.1	6,630
Total	10,319		8,093		18,412

B: Fuel Rods

Reactor Type	In Core	Discharged	Total
BWR	552,294	243,412	795,706
PWR	488,226	915,514	1,403,740
Total	1,040,520	1,158,926	2,199,446

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

Operating Experience

Figure 4. Distribution of Irradiated Siemens Nuclear Power Corporation Fuel By Assembly Averaged Burnup Through the End of 1990⁽³⁾

4.3.2 Fuel Rod Integrity

Through 1990 SNP fuel rod integrity remained better than 99.997%. Failure statistics on SNP fuel rods, through December 31, 1990, are provided in Table 14. Siemens Nuclear Power Corp. uses the FRI described at the beginning of this section. Lower FRI values are indicative of fewer failed rods in the core. The FRI distribution for SNP PWR and BWR fuel is shown in Figure 5. Siemens Nuclear Power Corp. reported no fuel failures attributed to design or manufacturing in 1990. The five-year trend in the SNP FRI indicates continued improvement in fuel performance.⁽³⁾

During 1990, breaches in fuel cladding attributable to causes other than fuel design or manufacturing were determined to be the result of debris fretting.⁽³⁾

Corrosion data was obtained by SNP at eight PWRs and four BWRs in 1990. Beta-quenched cladding reached exposures as high as 39.6 GWd/MTU exhibited good resistance to corrosion in BWRs, particularly in those BWRs which are susceptible to CILC. Siemens Nuclear Power Corp. fuel surveillance programs have been incorporated into Table 8 at the end of Section 3.⁽³⁾ (Note: BWFC has found that in PWRs beta-quenching provides no particular advantage.⁽¹⁷⁾)

4.4 Westinghouse Electric Corporation (W)

The 1990 summary of fuel performance and fuel rod integrity for W is obtained from Reference 4.

4.4.1 Fuel Performance - Fuel Utilization and Burnup

During 1990, 56 domestic commercial nuclear plants operated using W fuel. A total of 73 commercial PWRs have used W supplied Zircaloy-clad fuel. Westinghouse Electric Corp. has manufactured 14x14, 15x15, 16x16, and 17x17 fuel assemblies. Currently, 45 plants operate using 17x17 fuel assemblies.⁽⁴⁾

At the end of 1990, approximately 2.65 million Zircaloy clad fuel rods were in operation, representing 10,760 fuel assemblies. Including discharged fuel, the number of irradiated W Zircaloy-clad fuel rods totals approximately 7.3 million, representing 31,000 fuel assemblies.⁽⁴⁾

The average burnup of all W discharged fuel is about 29 GWd/MTU, and the average burnup of all W fuel (in-core plus discharged) is about 26 GWd/MTU. A summary of burnup through the end of 1990 is given in Table 15. Assembly averaged burnups in excess of 36 GWd/MTU have been achieved with 4,535 assemblies containing approximately 1.1 million rods. Of these, 1,264 assemblies with approximately 295,000 fuel rods reached burnups of over 40 GWd/MTU and four assemblies were irradiated to burnups of 55 GWd/MTU with a peak rod burnup of 60 GWd/MTU. Thirty-one W fueled plants have operated with fuel region average burnups in the range of 36 to 41 GWd/MTU and coolant activities have remained low in these plants. A plant-by-plant status report showing peak region average burnup is given in Table 16.⁽⁴⁾

Table 14. SNP Fuel Rod Failure Statistics Through 1990⁽³⁾

	Number of Irradiated Rods	Failed Rods Burnup Less Than Warranted, Fuel Related		Failed Rods Burnup Less Than Warranted, Core Related		All Other SNP Failures ^a		Total Failures	
		Number	Rate	Number	Rate	Number	Rate	Number	Rate
BWR	795,706	49	0.006%	103	0.013%	14	0.002%	166	0.021%
PWR	1,403,740	9	0.001%	130	0.009%	70	0.005%	209	0.015%
TOTAL	2,199,446	58	0.003%	233	0.011%	84	0.004%	375	0.017%

^a Failures not examined and/or above warranted burnup

Operating Experience

Figure 5. Siemens Nuclear Power Corporation Fuel Reliability Indicator (FRI), Using the INPO Standard Method⁽³⁾

4.4.2 Fuel Rod Integrity

Westinghouse reports its fuel reliability, accounting for all failure mechanisms, to be 99.998%. Uncorrected and corrected coolant activity level distributions for W fueled plants are shown in Table 17. A comparison of coolant activities between 1982 and 1990 is contained in Table 18. The history of uncorrected iodine activity in W fueled plants from 1972-1990 is presented in Figure 6. Westinghouse fueled reactors have continued to exhibit lower activity levels over the last 7-8 years. Presently, 90% of W fueled reactor plants have activity levels less than $0.01 \mu\text{Ci/g}$. The percentage of plants above the $0.05 \mu\text{Ci/g}$ level has dropped from 38% in 1982 to 2% in 1988-1990. The overall average of uncorrected iodine activity has dropped from $0.041 \mu\text{Ci/g}$ at the end of 1982 to $0.0042 \mu\text{Ci/g}$ throughout 1990. There have been no reports of iodine-131 activity greater than $0.1 \mu\text{Ci/g}$ from plants operating with W fuel since 1983.⁽⁴⁾

During 1990, ultrasonic testing (UT) examinations were performed at 21 reactor sites to identify leaking rods. Eighty-two leaking rods were identified in 65 assemblies at nineteen sites. Fuel assembly reconstitution was performed on 48 of the 65 assemblies. Of the fifty rods examined to date, twenty-six rod failures were due to debris induced fretting, fourteen were due to grid-rod fretting, three were due to manufacturing-related causes, and seven have no primary failure mechanism. It is estimated by W that debris caused approximately 70% of the failures in the identified rods.⁽⁴⁾

There was one instance of cladding collapse reported in 1990. As in all the previous cases of cladding collapse in W fuel (observed at 2 plants prior to 1990) it was determined that the bottom 6 inches of the fuel rod had never contained fuel pellets due to localized ovalization of the cladding. The hollow sections then collapsed during operation. Westinghouse Electric Corp. has implemented corrective actions to preclude this problem from recurring, and has received no reports of this problem since implementing these actions.⁽⁴⁾ Weld-related mechanisms during fuel rod fabrication caused two rod failures in 1990. Each failure was the result of a different cause, and it is reported that this type of failure is very infrequent.⁽⁴⁾

Grid-to-rod fretting was observed on the bottom inconel grid at five sites in 1990. Several manufacturing changes have been implemented to

eliminate or reduce the occurrence of cladding damage due to this problem. When the dates of the changes were compared with the dates of manufactured failed fuel it has been determined that the corrective actions have been effective.⁽⁴⁾

4.4.3 Non-Fuel Core Components

Westinghouse Electric Corp. has in operation approximately 3000 rod control cluster assemblies (RCCAs). Operational experience has demonstrated that RCCAs are susceptible to fretting wear against upper internal guide cards while fully withdrawn and stationary, and to hairline cracks at the tips. Hafnium absorber RCCAs are also 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."⁽⁴⁾

Wet Annular Burnable Absorbers have been in use since 1983 (previously borosilicate glass burnable assemblies were used) without any reported operating abnormalities. One borosilicate glass rodlet failed in 1988. The failure mechanism was not determined, and no new incidents have been reported since. Westinghouse thimble plugging devices did not have any operational problems reported in 1990.⁽⁴⁾

Secondary sources were involved in several incidents in 1990. One secondary source assembly was stuck in its host fuel assembly at Ginna. The assembly and its host were reloaded after evaluation. High antimony levels after a hydrogen peroxide addition prompted analysis at Indian Point 2 during a shutdown in February 1990. A leaking secondary source rod was strongly suspected. Westinghouse issued an advisory to all its utility customers to monitor antimony levels in coolant, especially during shutdowns.⁽⁴⁾

4.5 Babcock & Wilcox Fuel Company (BWFC)

BWFC has declined to provide information for this report as per Reference 5 and, therefore, no information is provided in these sections for calendar year 1990.

Operating Experience

4.5.1 Fuel Performance - Fuel Utilization and Burnup

BWFC has declined to provide information for this report as per Reference 5 and, therefore, no information is provided in these sections for calendar year 1990.

report as per Reference 5 and, therefore, no information is provided in these sections for calendar year 1990.

4.5.2 Fuel Rod Integrity

BWFC has declined to provide information for this

Table 15. Zircaloy-Clad Westinghouse Fuel Burnup Status Through 1990⁽⁴⁾

Zircaloy Active Rod Burnup Status

Burnup (GWd/MTU)	14x14 Rods	15x15 Rods	16x16 Rods	17x17 Rods	Total Rods
0-4	10740	40800	0	175560	227100
4-8	14964	8976	3760	185328	213028
8-12	20859	5912	5460	204864	237275
12-16	20716	24376	30785	300432	376309
16-20	15143	58004	15980	291192	380319
20-24	20644	13872	235	193776	228527
24-28	15191	25088	2820	244728	287827
28-32	27828	36687	8460	197208	270183
32-36	31011	51113	8694	149424	240243
36-40	17363	12624	8225	79200	117412
40-44	6444	11016	235	41976	59671
44-48	0	1757	0	3432	5189
48-52	0	200	0	5280	5480
52-56	0	0	0	0	0
56-60	0	0	0	0	0
Totals	200903	290425	84835	2072400	2648563

**Table 15. (cont.)
Zircaloy Discharged Rod Burnup Status**

Assembly Burnup (GWd/MTU)	14x14 Rods	15x15 Rods	16x16 Rods	17x17 Rods	Total Rods
0-4	0	332	0	0	332
4-8	4293	0	0	4488	8781
8-12	23965	6528	6815	56496	93804
12-16	18962	49980	9635	189024	267601
16-20	62292	81267	6110	402336	552005
20-24	53321	108451	7755	187704	357231
24-28	127764	121482	20680	346655	616581
28-32	139196	274584	23735	480216	917731
32-36	213449	252507	20210	519220	1005386
36-40	103426	150291	8695	394015	656427
40-44	24523	45838	1880	107448	179689
44-48	5728	15096	0	21120	41944
48-52	0	0	0	792	792
52-56	0	816	0	528	1344
56-60	0	0	0	1056	1056
Total	776919	1107172	105515	2711098	4700704

Zircaloy Total Rod Burnup Status

Assembly Burnup (GWd/MTU)	14x14 Rods	15x15 Rods	16x16 Rods	17x17 Rods	Total Rods
0-4	10740	41132	0	175560	227432
4-8	19257	8976	3760	189816	221809
8-12	44824	12440	12455	261360	331079
12-16	39678	74356	40420	489456	643910
16-20	77435	139271	22090	693528	932324
20-24	73965	122323	7990	381480	585758
24-28	142955	146570	23500	591383	904408
28-32	167024	311271	32195	677424	1187914
32-36	244460	303620	28905	668644	1245629
36-40	120789	162915	16920	473215	773839
40-44	30967	56854	2115	149424	239360
44-48	5728	16853	0	24552	47133
48-52	0	200	0	6072	6272
52-56	0	816	0	528	1344
56-60	0	0	0	1056	1056
Totals	977822	1397597	190350	4783498	7349267

Table 16. Westinghouse Fuel Performance Status Report Through 1990⁽⁴⁾

Reactor	Location	Owner	Date of First Electrical Power	Nominal MWe Gross	Current Cycle Number	Peak Region Avg. Burnup GWd/MTU	Generation MWh(e) Cumulative
Jose'de Cabrera	Spain	Union Electrica S.A	09/68	160	--	30.8	20,301,080
Beznau 1	Switzerland	Nordostschweizerische Kraftwerke AG	08/69	364	20	36.1	52,991,187
R. E. Ginna	U.S.A.	Rochester Gas & Electric	12/69	517	20	40.0	66,662,474
Point Beach 1	U.S.A.	Wisconsin Electric Power	12/70	524	18	43.3	67,302,380
Point Beach 2	U.S.A.	Wisconsin Electric Power	08/72	524	17	40.6	67,171,260
Surry 1	U.S.A.	Virginia Electric Power	08/72	858	11	38.0	73,576,823
Turkey Point 3	U.S.A.	Florida Power & Light	11/72	728	12	35.1	68,629,568
Surry 2	U.S.A.	Virginia Electric Power	02/73	858	10	36.5	74,230,114
Indian Point 2	U.S.A.	Consolidated Edison	07/73	1022	10	37.5	81,536,756
Turkey Point 4	U.S.A.	Florida Power & Light	06/73	728	12	35.1	65,856,094
Zion 1	U.S.A.	Commonwealth Edison	08/73	1085	12	37.1	94,667,954
Prairie Island 1	U.S.A.	Northern States Power	12/73	561	14	38.4	64,822,370
Zion 2	U.S.A.	Commonwealth Edison	12/73	1085	12	36.7	95,064,554
Kewaunee	U.S.A.	Wisconsin Public Service	03/74	563	--	34.0	64,506,971
Prairie Island 2	U.S.A.	Northern States Power	12/74	561	14	36.8	63,112,770
D. C. Cook 1	U.S.A.	Indiana & Michigan Electric	02/75	1089	11	38.2	95,565,510
Trojan	U.S.A.	Portland General Electric	12/75	1178	13	38.7	81,786,879
Millstone 2 ^a	U.S.A.	Northeast Utilities	11/75	902	--	33.4	77,439,700
Indian Point 3	U.S.A.	Power Authority of the State of New York	05/76	1013	8	36.5	66,940,725
Beaver Valley 1	U.S.A.	Duquesne Light	05/76	897	8	34.3	59,832,729

Table 16 (cont.) Westinghouse Fuel Performance Status Report Through 1990⁴

Reactor	Location	Owner	Date of First Electrical Power	Nominal MWe Gross	Current Cycle Number	Peak Region Avg. Burnup GWd/MTU	Generation MWh(e) Cumulative
Byron 1	U.S.A.	Commonwealth Edison	02/85	1175	4	36.8	40,109,841
Maanshan 2	Taiwan	Taiwan Power Company	02/85	951	5	36.3	31,661,508
Wolf Creek	U.S.A.	Kansas Gas & Electric	06/85	1192	5	36.6	43,482,425
Diablo Canyon 2	U.S.A.	Pacific Gas & Electric	10/85	1164	4	36.1	37,662,300
Millstone 3	U.S.A.	Northeast Utilities	02/86	1209	3	34.2	37,873,631
Kori 4 (was KNU 6)	Korea	Korea Electric Power	04/86	950	4	33.8	31,621,724
Catawba 2	U.S.A.	Duke Power Company	05/86	1205	4	36.8	30,003,189
Yonggwang 1 (was KNU 7)	Korea	Korea Electric Power	08/86	996	4	35.7	30,285,768
Yonggwang 2 (was KNU 8)	Korea	Korea Electric Power	11/86	996	3	33.2	24,915,109
Shearon Harris	U.S.A.	Carolina Power & Light	01/87	960	3	30.7	22,917,655
Byron 2	U.S.A.	Commonwealth Edison	02/87	1175	3	32.4	24,318,160
Vogtle 1	U.S.A.	Georgia Power	03/87	1210	3	28.7	28,336,532
Braidwood 1	U.S.A.	Commonwealth Edison	07/87	1175	2	25.9	20,662,650
Beaver Valley 2	U.S.A.	Duquesne Light	08/87	888	3	27.9	17,605,600
South Texas 1	U.S.A.	Houston Lighting	04/88	1315	3	20.9	17,140,343
Braidwood 2	U.S.A.	Commonwealth Edison	05/88	1175	2	24.3	17,308,341
Vogtle 2	U.S.A.	Georgia Power	04/89	1210	2	24.3	13,058,438
South Texas 2	U.S.A.	Houston Lighting	05/89	1315	2	13.9	10,785,163
Seabrook	U.S.A.	Public Service Company of New Hampshire	05/89	1194	1	5.8	4,268,765

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Table 16 (cont.) Westinghouse Fuel Performance Status Report Through 1990⁴

Reactor	Location	Owner	Date of First Electrical Power	Nominal MWe Gross	Current Cycle Number	Peak Region Avg. Burnup GWd/MTU	Generation MWh(e) Cumulative
Salem 1	U.S.A.	Public Service Electric & Gas	12/76	1170	9	37.3	79,207,820
Kori 1	Korea	Korea Electric Power	07/77	595	10	33.7	43,053,158
Farley 1	U.S.A.	Alabama Electric Power	08/77	873	10	45.8	72,050,338
D. C. Cook 2	U.S.A.	Indiana & Michigan Electric	03/78	1133	--	35.3	73,544,720
North Anna 1	U.S.A.	Virginia Electric Power	04/78	990	8	38.7	68,299,478
North Anna 2	U.S.A.	Virginia Electric Power	08/80	990	8	41.2	62,017,412
Sequoyah 1	U.S.A.	Tennessee Valley Authority	10/80	1183	5	37.9	45,964,147
Salem 2	U.S.A.	Public Service Electric & Gas	05/81	1170	6	36.6	51,731,927
Farley 2	U.S.A.	Alabama Power Company	05/81	872	7	39.7	58,205,838
Mcguire 1	U.S.A.	Duke Power Company	09/81	1225	7	38.4	57,776,299
Krsko	Yugoslavia	Nuklearna Elektrarne, Krsko	10/81	664	8	35.0	37,171,478
Sequoyah 2	U.S.A.	Tennessee Valley Authority	12/81	1183	5	35.3	42,115,010
V. C. Summer 1	U.S.A.	South Carolina Electric & Gas	11/82	954	6	46.1	45,026,370
Kori 2	Korea	Korea Electric Power	04/83	650	6	33.0	34,377,500
Mcguire 2	U.S.A.	Duke Power Company	05/83	1225	7	38.8	53,317,390
Diablo Canyon 1	U.S.A.	Pacific Gas & Electric	11/84	1137	4	39.5	43,272,400
Maanshan 1	Taiwan	Taiwan Power Company	05/84	951	5	34.4	30,371,412
Callaway 1	U.S.A.	Union Electric Company	10/84	1219	5	38.6	50,285,993
Kori 3 (was KNU 5)	Korea	Korea Electric Power	01/85	950	5	35.1	35,969,686
Catawba 1	U.S.A.	Duke Power Company	01/85	1205	5	35.3	40,664,034

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Table 16 (cont.) Westinghouse Fuel Performance Status Report Through 1990⁴

Reactor	Location	Owner	Date of First Electrical Power	Nominal MWe Gross	Current Cycle Number	Peak Region Avg. Burnup GWd/MTU	Generation MWh(e) Cumulative
Comanche Peak 1	U.S.A.	Texas Utilities Generating Company	04/90	1130	1	6.6	

(a) Non-Westinghouse plant with Westinghouse fuel

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Table 17. Summary of Coolant Activity Through 1990^{a,(4)}

Activity Range Iodine-131 ^b $\mu\text{Ci/g}$	Uncorrected Iodine-131		Corrected Iodine-131	
	Number of Plants in Range	Percentage of Plants in Range	Number of Plants in Range	Percentage of Plants in Range
0.030-0.100	1	2%	0	0%
0.010-0.030	5	8%	3	5%
0.003-0.010	15	25%	9	15%
0.001-0.003	17	29%	6	10%
Below 0.001	21	36%	41	70%

^a Iodine-131 values are given as of the end of 1990 (December basis)

^b All data have been normalized to 100% power and the same cleanup rate

Uncorrected: Normalized Measured data

Corrected: Normalized Measured data corrected for tramp uranium

Table 18. Comparison of Coolant Activity,^c 1982-1990⁽⁴⁾

Activity Range (Iodine-131 $\mu\text{Ci/g}$)	1990 Number of Plants	1990 Percentage of Plants	1982 Number of Plants	1982 Percentage of Plants
0.100-0.300	--	--	1	4%
0.030-0.100	1	2%	10	38%
0.010-0.003	5	8%	3	11%
0.003-0.010	15	25%	8	31%
0.001-0.003	17	29%	2	8%
Below 0.001	21	36%	2	8%

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

* Data points show the number of plants (■) in a given activity range

Figure 6. Uncorrected Reactor Coolant Activity Distributions for Westinghouse-Fueled Plants^(4,17)

5.0 Problem Areas Observed During 1990

This section contains information on events/items that involve fuel failure or damage. It also contains information on other events/items that are of concern or interest to the fuel systems. Event listings by reactor are found in Tables C.1 and C.2 for BWRs and PWRs, respectively.

5.1 Fuel Oriented

5.1.1 Fuel - Failure, Damage, Potential for Damage

One event in the U.S and three events or items of interest in foreign countries involved failure, damage or potential for damage to fuel. Those events or items of interest are described below.

5.1.1.1 Atucha-1

A fuel failure in August 1988 forced Argentina's Atucha-1 reactor into a 17-month outage, lasting until January 1990.^(51,52) Neutron radiation is suspected to have caused cobalt in a vertical thimble to swell up.⁽⁵³⁾ Engineers believe this caused local turbulence and further oscillations of the guide tube, which then broke and began knocking against the pressure tube. Repair work included removal of damaged parts, replacing damaged fuel channels, and cleaning the lower plenum. Information on this 1988 event was reported in 1990.

5.1.1.2 Darlington-2

Darlington-2, a Canadian CANDU (Canadian Deuterium-Uranium Reactor) PHWR, was refueling at power on December 5, 1990, when a fuel failure was detected.⁽⁵⁴⁾ Power was reduced and the defective bundle was removed.⁽⁵⁵⁾ Later examination of the fueling machine revealed pieces from a couple of fuel pencils.⁽⁵⁴⁾ Turbulence in the fuel channel or metallurgical problems with the end plate were cited as possible causes.⁽⁵⁶⁾

5.1.1.3 Hamaoka-1

In October 1990, sipping tests detected leaking fuel elements at Japan's Hamaoka-1 BWR. Detailed inspection showed that 78 fuel assemblies had cladding that either flaked off or failed. Chubu Electric Power Co. determined that a combination of the effects of

reactor coolant water chemistry and fuel cladding material was responsible for the failures.⁽⁵⁷⁾

5.1.1.4 Vermont Yankee

During a refueling outage at the Vermont Yankee BWR in September 1990, a broken fuel rod was discovered. The broken rod resulted in uranium fuel pellets being scattered in the reactor cooling system. Examination showed that the "the bottom 8 inches of a 12-foot fuel rod" was missing due to a manufacturing defect. A failure was expected as "slightly elevated off-gas levels" were detected in June 1989.⁽⁵⁸⁾

5.1.2 Pellet-Cladding Interaction

There was one article published in 1990 describing pellet-cladding interaction failures.

5.1.2.1 Review of PCI Failures in Zirconium Alloy Fuel Cladding

An article⁽⁵⁹⁾ discusses the history of PCI failures, especially with Zircaloy-clad UO_2 fuel. The article discusses various claddings which inhibit PCI, namely CANLUB-graphite cladding for CANDU reactors and Zr in BWRs. The article reports that PCI failures for PWRs have not been numerous enough to warrant investigation.

5.1.3 Swelling, Wear, Oxidation, Other Corrosion

One event or item of interest in Canada has been identified that involved swelling, wear, oxidation, or other corrosion.

5.1.3.1 Bruce; Pickering

"Ontario Hydro will accelerate its long-term schedule to replace fuel channels in the first six reactors at Pickering and Bruce but has dropped earlier plans to retube subsequent units at Pickering B and Bruce B stations. ...The pressure tube design...appears to have eliminated the conditions which created the zirconium hydride problem in Pickering-1 to -4 and Bruce-1 and -2..."⁽⁶⁰⁾

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5.1.4 Guide Pin or Alignment Pin Problems

There was one item of interest in Germany that involved guide/alignment pin problems.

5.1.4.1 Biblis B

"Ultrasonic examination of 386 fuel alignment pins at the Biblis B PWR, followed by replacement of 67 of the pins, has been completed by ABB/Voest."⁽⁶¹⁾

5.1.5 Iodine Spiking

There was one article discussing iodine spiking in PWRs. The article is summarized below.

5.1.5.1 Tramp Uranium in PWRs

A recent article⁽⁶²⁾ discusses tramp uranium in PWRs. Besides fuel failures, there are two other sources of tramp uranium, which could give incorrect impressions about the number of failed fuel assemblies, since tramp uranium can cause an increased iodine-131 spike after a reactor trip. As other causes of fuel failures are eliminated, tramp uranium becomes more important to the "no-leaker" fuel concept.

5.1.6 Miscellaneous - Fuel Related

There were two events in the U.S. and two events or items of interest in foreign countries that involved miscellaneous fuel oriented topics. Those events or items of interest are summarized below.

5.1.6.1 SNP's Fuel Rod Clips

"ANF's fuel rod clips are small structural members which reinforce fuel rods at the exterior of a nuclear fuel assembly to prevent flow-induced vibration of the rods due to pressure baffle jetting which occurs in certain pressurized water reactors. If left unchecked, the flow-induced vibration can cause fuel rod failure. The fuel rod clips have proven to be extremely effective and afford reactor operators a simple, economic and reliable way to protect expensive fuel assemblies from damage caused by baffle jetting."⁽⁶³⁾

5.1.6.2 Fragma Advanced Fuel Assembly Upgrades

An article⁽⁶⁴⁾ describes Fragma's upgrades for its 17x17 12ft Advanced Fuel Assembly. The assembly incorporates features such as an improved spacer grid, an anti-debris filter, a Zry-4 cladding with reduced tin content to reduce corrosion, an improved internal surface finishing technique (to reduce pellet-cladding interaction), and improved inspection techniques.

5.1.6.3 Goesgen

At Switzerland's Goesgen PWR, longer fuel rods were to replace standard fuel rods in the summer of 1990. The longer rods, fabricated by Siemens AG, were designed to have increased resistance to oxide corrosion. The improvement was sought because corrosion problems were limiting a planned increase in thermal output.⁽⁶⁵⁾

5.1.6.4 Sequoyah-1

A May 1990 amendment modified the Sequoyah-1 technical specifications "to permit the use of the VANTAGE 5 Hybrid fuel at Unit 1 in the upcoming Operating Cycle 5."⁽⁶⁶⁾

5.2 Fuel Handling Oriented

5.2.1 Fuel Handling

There were four events or items of interest in the U.S. and no events or items of interest in foreign countries that involved fuel handling incidents. Those events or items of interest are described below.

5.2.1.1 Nine Mile Point-1

On January 15, 1990, at Nine Mile Point-1, a source range monitor was found to be bypassed during fuel loading resulted in a technical specification violation. The root cause was determined to be personnel error due to a breakdown in communication, inadequate procedures, and poor training. Appropriate procedures and training practices have been revised.⁽⁶⁷⁾

5.2.1.2 San Onofre-3

A technical specification violation occurred at San Onofre-3 on May 1, 1990. Movement of irradiated fuel in the spent fuel pool took place while salt water

cooling and Component Cooling Water systems were being repaired. This rendered the Post Accident Cleanup Unit (BINS) train "A" inoperable. Fuel movement is still possible provided that the train "B" BINS is in operation. The requirement to place the train "B" BINS in operation during movement of fuel in the spent fuel pool was ignored. This was overlooked by both the control room operators and the control room coordinator due to inadequate attention to detail. Appropriate disciplinary actions have been taken.⁽⁶⁸⁾

5.2.1.3 Susquehanna-1

On October 18, 1990, a ninth fuel bundle was loaded into the core prior to verifying that the source range monitor downscale functions were operable. This is a violation of technical specification 3.9.2. This event is attributed to inadequate administrative procedural controls and cognitive personnel error. Appropriate procedures are being revised to ensure that this does not reoccur.⁽⁶⁹⁾

5.2.1.4 Zion-2

On May 7, 1990, at Zion 2, personnel had difficulty inserting a bowed and twisted assembly into the core. The assembly was partially inserted but then removed to install guides. A piece of grid strap fell when the assembly cleared the top of adjacent assemblies. No procedural or personnel errors were found. The event was attributed to the difficulty of the loading task.⁽⁷⁰⁾

5.2.2 Dropped, Broken, Damaged, Potential for Damage

There were three events or items of interest in the U.S. and three events or items of interest in foreign countries that involved dropped, broken, damaged, or potentially damaged fuel. Those events or items of interest are described below.

5.2.2.1 Bohunice A-1

Heavy corrosion caused a spent fuel assembly to be "ripped apart 60 centimeters below the fuel assembly head" during the decommissioning of the Bohunice A-1 reactor in Czechoslovakia. The event occurred in March 1988, but was reported in 1990.⁽⁷¹⁾

5.2.2.2 Bruce-4

On January 23, 1990, at Canada's Bruce-4 a software error caused the fueling machine to be driven 20 cm into a fuel channel.^(72,73,74,75,76,77) The event started with an erroneous computer instruction to release the brake, which allowed the fueling machine cylinder to drop 16 inches.⁽⁷⁶⁾ This damaged the end fitting and allowed 2600 gallons of heavy water moderator to spill out.^(73,74,77)

5.2.2.3 Byron-2

A fuel assembly was dropped at Byron-2 on September 29, 1990. Fuel handling personnel were performing bottom nozzle fuel reconstitution activities in the spent fuel pool in preparation for fuel reload. The reconstitution basket lid was closed and secured. During rotation the fuel assembly slipped out and came to rest on top of an empty fuel rack. The assembly was transferred to a failed fuel canister in the spent fuel pool. Procedural and work activity changes have been made.^(78,79)

5.2.2.4 Grand Gulf-1

On October 24, 1990, at Grand Gulf-1 a fully grappled fuel bundle was lowered into the vessel in an uncontrolled manner due to independent failures of redundant refueling equipment brake systems. The bundle entered its targeted core location striking no adjacent structures. The normal safety brake failed due to the manual disengage lever binding in the disengaged position due to close proximity with the brake housing cover, and a deformed spring caused the emergency brake to fail also. The brake was fixed and the fuel bundle replaced.⁽⁸⁰⁾

5.2.2.5 Gundremmingen-C

"At the Gundremmingen-C BWR on September 9 (1990), during refuelling, the fuel assembly hoist mistakenly grabbed a fuel assembly, which was in [the] stripping machine. The assembly, incorrectly mounted in the machine, fell onto the floor of the spent fuel pool. No radiation was released and the storage pool was not damaged."⁽⁸¹⁾

5.2.2.6 Indian Point-3

On October 4, 1990, at Indian Point-3, two fuel assemblies were inadvertently withdrawn from the core. While removing the upper reactor internals, the guide pins were snagged.^(82,83,84,85) On October 16 and 17, personnel removed the fuel

Problem Areas

assemblies from the upper internals. While placing the assemblies in storage baskets, one assembly was dislodged and fell ten feet.^(85,86,87,88) All fuel movement occurred underwater and no radiation was released.⁽⁸⁷⁾

5.2.2.7 North Anna-1

Operations personnel discovered that the fuel building ventilation was not aligned through the charcoal filters during fuel movement on January 27, 1990, at North Anna-1. Because a vendor was performing fuel constitution work, technical specification 3.9.12 was violated. During this work, a mechanical failure caused a fuel rod to be placed in its storage location in a uncontrolled manner. There was no damage to the fuel rod nor was there any release of radioactive material. The cause of the event was personnel error and procedural inadequacy. Appropriate operating procedures were revised.⁽⁸⁹⁾

5.2.3 Fuel Handling Procedural Violations

There was one event or item of interest in the U.S. and no events or items of interest in foreign countries that involved fuel handling procedural violations. That event is described below.

5.2.3.1 Perry-1

On October 11, 1990, at Perry-1 it was discovered that fuel movement had been allowed to proceed on October 10, 1990, in violation of technical specification 3.9.1.B. Previously, rods had been withdrawn from the core and replaced. The one-rod-out interlock was not functionally tested prior to fuel movement. The cause was misinterpretation of technical specifications. Corrective action has been taken.⁽⁹⁰⁾

5.2.4 Spent Fuel Pool Problems

There was one event in the U.S. and no events or items of interest in foreign countries that involved spent fuel pool problems that are not included in the categories of heat removal or ventilation. The event is described below.

5.2.4.1 Turkey Point-3

With Turkey Point-3 in mode 6 (refuelling), approximately three inches of borated contaminated

water had accumulated in the spent fuel pool (SFP) heat exchanger room and cask wash area on February 20, 1990. The event occurred when the 3B SFP cooling pump shaft sheared due to fatigue failure. The impeller and the portion of the pump shaft up to the fracture point continued to rotate and resulting in a mechanical seal failure which caused an approximately 18 gpm leak. Partially clogged drain lines caused the accumulation. The 3A SFP pump has been aligned as the primary spent fuel pool cooling pump.⁽⁹¹⁾

5.2.5 Heat Removal

There were two events or items of interest in the U.S. and no events or items of interest in foreign countries that involved heat removal in spent fuel pools. Those events or items of interest are described below.

5.2.5.1 Connecticut Yankee

While Connecticut Yankee was in mode 6 (refueling) on June 8, 1990, the spent fuel pool cooling pumps were temporarily de-energized while restoring circuit breaker alignments to normal following the replacement of all breakers on a 125 V D.C. distribution panel. The event occurred because of a procedural inadequacy in that the restoration procedure did not specify the sequence in which electrical loads were to be re-energized, resulting in a bus undervoltage protection circuit being energized before the bus itself. The procedure has been revised.⁽⁹²⁾

5.2.5.2 San Onofre-1

At San Onofre-1 on April 24, 1990, it was determined that the spent fuel pool cooling maximum heat load was greater than the design was capable of handling. The actual loads on the spent fuel pool cooling systems will be greater than those in the updated final safety analysis report (UFSAR) because the decay heat was calculated using the plant electrical rated capacity and not the thermal rate. The problem will be remedied by permanently installing a spare pump that was originally intended for backup in case of primary failure.⁽⁹³⁾

5.2.6 Ventilation

There were eight events or items of interest in the U.S. and no events or items of interest in foreign countries that involved spent fuel pool or fuel building ventilation. Those events or items of interest are described below.

5.2.6.1 Arkansas Nuclear-1

On October 26, 1990, at Arkansas Nuclear-1, it was identified by operations personnel that the total operating hours since the last surveillance of the spent fuel ventilation system had been greater than 720 hours at the time of refueling operations were begun. This was in violation of ANO-1 technical specifications. Fuel handling operations were halted and surveillance of the ventilation system was performed later that day. Inadequate procedures to ensure that the spent fuel ventilation surveillance was performed at the required frequency was determined to be the root cause. The procedure was being revised.⁽⁹⁴⁾

5.2.6.2 Davis-Besse-1

A fuel assembly was moved in the spent fuel pool while the emergency ventilation systems were inoperable at Davis-Besse-1 on February 2, 1990. This violated technical specification 3.9.12 action statement B. The cause was personnel error. Additional training will be required in reference to this event.⁽⁹⁵⁾

5.2.6.3 Diablo Canyon-2

On March 12, 1990, an NRC resident inspector at Diablo Canyon-2 questioned the Fuel Handling building ventilation requirements when he noticed personnel doors were blocked open because of temporary hoses. Adequate negative pressure was not maintained while fuel movement was occurring. Personnel failed to recognize the door as part of the ventilation system boundary. The doors will have signs posted and training will be conducted.⁽⁹⁶⁾

5.2.6.4 Fort Calhoun-1

On February 23, 1990, plant management determined that the spent fuel area charcoal filtration unit, was outside its design basis due to insufficient air flow into the unit from the spent fuel pool area. The Updated

Safety Analysis Report (USAR) assumes that all activity from a damaged fuel assembly would be filtered through the filtration unit in the event of an accident. The USAR will be updated and submitted to the NRC.⁽⁹⁷⁾

5.2.6.5 McGuire-1

On March 29, 1990, at McGuire-1 a problem incident report was submitted to design engineering by performance personnel identifying flow discrepancies on unit 2 fuel pool ventilation system (VF). Unit 1 VF system was similarly declared inoperable on April 10, 1990. Improper installation and a contributory cause of design selection deficiency have been cited as causes for this event.⁽⁹⁸⁾

5.2.6.6 North Anna-1

Operations personnel discovered that the fuel building ventilation was not aligned through the charcoal filters during fuel movement on January 27, 1990, at North Anna-1. Because a vendor was performing fuel constitution work, technical specification 3.9.12 was violated. During this work, a mechanical failure caused a fuel rod to be placed in its storage location in a uncontrolled manner. There was no damage to the fuel rod nor was there any release of radioactive material. The cause of the event was personnel error and procedural inadequacy. Appropriate operating procedures were revised.⁽⁸⁹⁾

5.2.6.7 Prairie Island-1

At Prairie Island on May 4, 1990, a test of a monitor activated the spent fuel pool special ventilation system. The monitor being tested was placed in reset to prevent actuation of the spent fuel pool special ventilation system. During the test, the test source caused a nearby backup monitor to respond, which activated the system. The event is attributed to a procedural inadequacy. The procedure considered the effect of the monitor bugging operation, but the caution was written as a note instead of a procedural signoff step.⁽⁹⁹⁾

5.2.6.8 Prairie Island-1

On May 5 and 14, 1990, at Prairie Island-1 the control room received high radiation alarms and indications of an automatic start of the spent fuel pool special exhaust fan. Both events occurred on the R25 unit and were caused by electrical spikes. One spike was

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caused by another monitor being shut off and the other spike appeared to be random. Radiation monitor R-25 was found to be in alarm with a normal response indicated by the meter located on the monitor, while in fact no high radiation condition existed. The unit was to be upgraded.⁽¹⁰⁰⁾

5.2.7 Dry Storage

There were three events or items of interest in the U.S. and two events or items of interest in foreign countries that involved dry storage of spent fuel. Those events or items of interest are described below.

5.2.7.1 Dry storage for Pickering Spent Fuel

Canada's Ontario Hydro is testing dry storage for fuel from the Pickering spent fuel pools. After the fuel has been cooled for six years, it can be loaded directly from the pool to the container. According to the utility, the system can also handle LWR fuel.⁽¹⁰¹⁾

5.2.7.2 Dry Storage at Fort St. Vrain

An article⁽¹⁰²⁾ describes the GEC-Alsthom Modular Vault Dry Store (MVDS) that is to be commissioned in 1992 at the Fort St. Vrain HGTR. The MVDS supplements existing onsite spent fuel pool storage, and will store 1488 spent fuel blocks. The MVDS is designed for a minimum lifetime of 40 years.

5.2.7.3 Prairie Island-1 and -2

"Independent spent fuel storage installation (ISFSI) license application received: on Oct. 12 by the NRC staff from Northern States Power Company for its two-unit Prairie Island plant. The utility wants to build an independent spent fuel storage installation to contain dry casks full of spent fuel that has undergone enough pool storage to be thermally cool. The deadline for petitions to intervene was November 19."⁽¹⁰³⁾

5.2.7.4 Spent Fuel Dry Storage Temperature Limits

An article⁽¹⁰⁴⁾ discusses a method that was developed to determine the temperature limits allowable for spent fuel after an accident causing higher temperatures. The method uses an empirical creep equation that is more "practical and realistic"

than others that are based on the life fraction rule or creep rupture criterion.

5.2.7.5 Wylfa Magnox Dry Storage Leak

After noticing a water leak in July 1990, a "special remote camera" was developed and used at the Wylfa Magnox Dry Store. The leak caused corrosion of at least 46 fuel elements.⁽¹⁰⁵⁾

5.2.8 Spent Fuel Consolidation

There were six events or items of interest in the U.S. and one item of interest in foreign countries that involved spent fuel consolidation. Those events or items of interest are described below.

5.2.8.1 Advances in LWR Spent Fuel Storage Technology

An article⁽¹⁰⁶⁾ discusses various advances in technology for storing LWR spent fuel. Utilities are looking at other methods to supplement or increase on-site storage capability as capacity limits are being reached on spent fuel pools. These methods include rod consolidation, reracking, and dry storage. The article also estimates that there are at least 51,000 fuel assemblies currently in storage, and that roughly 3100 are defective in some manner.

5.2.8.2 Department of Energy Spent Fuel Consolidation Research

Work on dry consolidation research of spent fuel assemblies performed for DOE is described in a recent article.⁽¹⁰⁷⁾ The system is designed to achieve a 2:1 consolidation and volume reduction without compromising the integrity of individual fuel rods.

5.2.8.3 Fuel Assembly Hardware Melter

A radioactive waste melter for the compaction of spent fuel assembly hardware was developed by Battelle Pacific Northwest Laboratories. The first step is to remove the fuel rods from the assemblies and place them into a container that holds two assemblies worth of fuel rods in the space of one assembly. The leftover hardware then is compacted using the melter. Battelle claims that fuel from 10 years of operation could be consolidated in 3-4 months.⁽¹⁰⁸⁾

5.2.8.4 Fuel Master rod consolidation machine

A second generation fuel consolidation machine has been developed by SGN and BWFC. A potential consolidation ratio of 10:1 could be achieved for non-B&W fuel. The system is known as Fuel Master and can be operated with three people.⁽¹⁰⁹⁾

5.2.8.5 KNK-II Spent Fuel Reprocessing

A contract to reprocess spent fuel from the KNK-II experimental breeder reactor was signed by the Commissariat à l'Energie Atomique (CEA) and the Karlsruhe Research Center. The chopping and condition will take place at the CEA's Cadarache research center and Germany will be the recipient of waste and reprocessed fuel.⁽¹¹⁰⁾

5.2.8.6 Spent Fuel Consolidation

The Electric Power Research Institute reports that utilities can substantially increase capacity of spent fuel pools by using fuel consolidation systems.⁽¹¹¹⁾

5.2.8.7 Spent Fuel Consolidation

Utilities can substantially increase the capacity of spent-fuel storage pools by using fuel consolidation systems. A demonstration at the Northeast Utilities Service Co. Millstone-2 provided valuable information.⁽¹¹²⁾

5.2.9 Reracking/Storage Rack Issues

There were two events or items of interest in the U.S. and two events or items of interest in foreign countries that involved storage rack issues. Those events or items of interest are described below.

5.2.9.1 Almaraz-1 and -2; Asco-1 and -2

"The KWU Group of Siemens with its Spanish partner Equipos Nucleares, is to backfit compact storage racks in the fuel pools at Spain's Almaraz 1 and 2 and Asco 1 and 2 nuclear plants. The racks will increase the lifetime of the fuel pools to over 30 years."⁽¹¹³⁾

5.2.9.2 Chinshan

A 3-D seismic analysis was performed on high-density spent fuel storage racks at Taiwan's Chinshan-1 and -2 by Holtec International. The analysis, which was

based on results from a shake table, predicted that the storage racks wouldn't hit each other in an earthquake.⁽¹¹⁴⁾

5.2.9.3 Nine Mile Point-2

On July 17, 1990, Nine Mile Point-2 received an amendment revising technical specifications to allow "use of a single-failure-proof handling system to handle and transport loads in excess of 1000 pounds over fuel in spent fuel storage pool racks."⁽¹¹⁵⁾

5.2.9.4 Siemens's Storage Racks

An article describing the history of Siemens'⁽¹¹⁶⁾ high-density storage racks, which "have been installed in over 30,000 fuel assemblies in more than 20 nuclear stations across Europe." The article also describes some of the reasoning behind the design.⁽¹¹⁷⁾

5.2.10 Spent Fuel Storage Issues

There were four items of interest in the U.S. and one generic item of interest in foreign countries that involved spent fuel storage issues. Those items of interest are described below.

5.2.10.1 Crystal River-3

"EA/FONSI issued: on August 23 (1990) by the NRC staff on Florida Power Corporation's request to increase spent-fuel capacity at Crystal River-3 from 1153 to 1357 assemblies. The environmental assessment and finding of no significant impact clears the way for issuance of a license amendment, but does not in itself constitute approval of the amendment."⁽¹¹⁸⁾

5.2.10.2 Hope Creek

"License amended: on June 21 by the NRC staff on Public Service Electric and Gas Company's Hope Creek unit. The amendment allows an increase in spent-fuel storage at the unit for up to 4006 fuel assemblies."⁽¹¹⁹⁾

5.2.10.3 Nuclear Regulatory Commission and Spent Fuel Storage

Long-term on-site storage of spent fuel was endorsed by the NRC. The NRC believes that "spent fuel can be stored safely for at least 30 years beyond the

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reactor's operating life, including a renewal or extension." The NRC also predicts a repository will be available by 2025.⁽¹²⁰⁾

5.2.10.4 Prairie Island

Dry storage at Northern States Power's Prairie Island plant is based on a cask design by Transnuclear. The cask design is being modified to handle more spent fuel assemblies (24 assemblies). The casks will be placed on a concrete pad where the fuel will be vertical.⁽¹²¹⁾

5.2.10.5 Spent Fuel Oxidation Analysis

To support dry storage of spent LWR fuel, samples of BWR and PWR fuel were placed in controlled atmosphere baths. Oxidation rates were measured for several temperatures in moist and dry air. Results indicated BWR spent fuel oxidized faster than the PWR spent fuel.⁽¹²²⁾

5.2.11 Fuel Production Plants

There was one event in the U.S. and three events or items of interest in foreign countries that involved fuel production plants. Those events or items of interest are described below.

5.2.11.1 ABB Fuel Fabrication Plant Accident

One worker was injured from a release of uranium hexafluoride (UF₆) on December 18, 1990, at ABB Combustion Engineering Nuclear Fuel's Hematite, Missouri fuel fabrication plant. The worker disconnected sampling equipment before turning off a valve on a cylinder of uranium hexafluoride, but no uranium hexafluoride was released from the building.⁽¹²³⁾

5.2.11.2 Hanau Fuel Fabrication Plant Accident

On December 12, 1990, three workers were injured in an explosion at the Hanau Fuel Fabrication plant in Germany. The explosion is thought to have occurred when "the metallic section of an off-gas washing vessel exploded."⁽¹²⁴⁾

5.2.11.3 MELOX Fuel Fabrication Plant

In an article, France's MELOX fabrication plant construction, design and safety issues are discussed.

The plant is expected to start up in the mid-1990s.⁽¹²⁵⁾

5.2.11.4 Siemens' Hanau MOX Fuel Production Plant

Siemens'⁽¹²⁶⁾ new MOX fuel plant, located in Hanau, Germany, is expected to begin production in 1992, as discussed in a recent article.⁽¹²⁷⁾ Also described are the plant protection, automation and the future of the MOX cycle.

5.2.12 Miscellaneous - Fuel Handling Related

There was one event in the U.S. that falls under the category of miscellaneous. The event is listed below.

5.2.12.1 Calvert Cliffs-1

At Calvert Cliffs-1 it was determined on January 16, 1990, that a condition existed as to be reportable as a condition outside of the plant design basis as described by the Updated Final Safety Analysis Report (UFSAR). The UFSAR assumed that only 1 fuel element can be damaged at a time but procedures allow for more than 1 assembly to be in a designated location, thus 2 elements could be involved. This procedure has been revised to assure that the UFSAR assumption isn't violated.⁽¹²⁸⁾

5.3 Control Rod Oriented

5.3.1 Control Rod System Problems

There were 11 events or items of interest in the U.S. and two events or items of interest in foreign countries that involved control rod system problems. Those events or items of interest are described below.

5.3.1.1 Arnold

A video probe inspection revealed a control rod drive (CRD) withdraw line had a small leak on May 19, 1990. Although the root cause is unknown, an unknown forcing function causing high cycle fatigue is suspected, based on ultrasonic testing and metallurgical analysis. Investigation was continuing and the CRD withdraw line was being replaced using a new weld design.⁽¹²⁹⁾

5.3.1.2 Braidwood-2

On May 17, 1990, at Braidwood-2, reactor trip and gripper coil time measurements were being tested during cold shutdown. Switch S501 of the solid state protection system was rotated to off in a clockwise direction. A reactor trip signal was generated. The rod drive system was not capable of rod withdrawal as the disconnect switches for the rod lift circuits were open. The cause was found to be procedural deficiency which did not specify that the switch needs to be rotated counter-clockwise. The procedure was to be revised.⁽¹³⁰⁾

5.3.1.3 Callaway-1

At Callaway-1 on February 2, 1990, a control rod urgent failure alarm was received during the performance of a monthly surveillance on movable control rod assemblies. At this time the reactor was in mode 1, power operations at 100% power. The problem was traced to a defective chip in a slave cyclor counter card. The defective card was replaced.⁽¹³¹⁾

5.3.1.4 Chernobyl-3

On August 27, 1990, a failed power supply for the control system forced the Chernobyl-3 (light water cooled, graphite moderated) reactor in the Ukraine to shut down. After noting the failed power supply, thirty minutes are allowed to restore it before the reactor must be shut down. At the end of this period the power supply was still inoperable and the reactor was shut down.⁽¹³²⁾

5.3.1.5 Grand Gulf-1

A manual scram was inserted following a lockup of the rod pattern control system (RPCS) on November 24, 1990, at Grand Gulf-1. The RPCS locked up due to multiple control rod drift. The rod drift was caused by excessive differential pressure of the CRD cooling water. This condition was a result of the operator's effort to increase reactor water level due to open drain valves in the main steam lines. This event was attributed to deficient procedures as the operators were fully aware of all the conditions present and no equipment failure occurred. The startup procedure has been amended to specify main steam line drain valve lineups and the minimum hotwell temperature requirement.⁽¹³³⁾

5.3.1.6 Mutsu, Nuclear Powered Ship

Japan's experimental nuclear-powered ship Mutsu experienced a reactor scram on July 30, 1990, due to "noise" in a control rod mechanism.⁽¹³⁴⁾

5.3.1.7 Oyster Creek

At Oyster Creek on February 20, 1990, a manual reactor scram was initiated due to spurious actuation of the Alternate Rod Injection (ARI) System. The event occurred when a technician keyed a hand-held radio near analog trip units for the ARI system causing it to actuate due to radio frequency interference. Activation caused control rods to drift into the core and the operators initiated a manual scram within 5 seconds. The occurrence was attributed to personnel error as the area was clearly marked with the restriction on radio usage. Minimizing the ARI sensitivity to interference is being investigated and appropriate people have received training on the incident.⁽¹³⁵⁾

5.3.1.8 Perry-1

On May 21, 1990, Perry-1 had a technical specification violation when a failure of the rod control and information system was not followed up by appropriate operator responses. With more than one control rod scram accumulator inoperable, technical specifications require that the associated control rods be declared inoperable, and immediate verification of control rod drive pump operation. The event occurred due to lack of procedural guidance for operator response. The procedures have been revised and a possible technical specification change is being pursued for improved clarification.⁽¹³⁶⁾

5.3.1.9 Perry-1

Surveillance testing on December 7, 1990, revealed the failure of 54 (out of 177) control rod scram accumulator level switches at Perry-1. Improper servicing of the control rod scram accumulator level switches was determined to be responsible. Improper charging of accumulators could have rendered the potentially inoperable during or before the previous fuel cycle. Additional tests were performed and a system operating instruction is being revised to ensure proper operational status after servicing.⁽¹³⁷⁾

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5.3.1.10 Riverbend-1

On October 27, 1990, while in refueling mode, the Riverbend-1 reactor protection system (RPS) actuated on a high neutron flux signals from the Intermediate Range Monitors (IRM). No rod movement resulted from this RPS actuation. The root cause was shorting in two IRM detector connectors due to the introduction of water into the connectors. Control rod drive maintenance procedures will be revised.⁽¹³⁸⁾

5.3.1.11 Surry-2

At Surry-2 on November 20, 1990, rod control cluster assembly M-12 was declared inoperable due to a mismatch between group and rod position indication. This was within technical specifications but a subsequent startup generated a quadrant power tilt greater than 2% for more than 24 hours until the excore detectors were recalibrated. Resulting hot channel factors from flux maps were within technical specification limits.⁽¹³⁹⁾

5.3.1.12 Vogtle-1

On December 2, 1990, Georgia Power Company failed to comply with a technical specification 4.1.3.2 special condition surveillance which is applicable when the rod position deviation monitor is inoperable. The cause of this event was the failure of the unit shift supervisor to follow procedure when recentering the rod position values into the Proteus (plant status) computer which disabled the monitor. Disciplinary action was taken.⁽¹⁴⁰⁾

5.3.1.13 Vogtle-2

On November 10, 1990, at Vogtle-2 the group 2 step demand counter for control bank D stopped counting. Fourteen minutes later the group 1 step demand counter also stopped. Technical specification 3.0.3 was entered because of both counters were inoperative. The event was caused by the cover for the group 1 counter being removed and disengaging the counter. Inadequate operator technical knowledge regarding the effect of opening the cover on the operation of the step counter. The cover had been apparently opened during investigation of the problem with the group 2 step demand counter. Training will be provided in upcoming operator requalification.⁽¹⁴¹⁾

5.3.2 Control Rod Operation

There were two events or items of interest in the U.S. and one event in foreign countries that involved control rod operation. Those events or items of interest are described below.

5.3.2.1 Chinon B2

"Chinon B2 was at 300 MW and rising to power after an outage August 2 (1990) when certain control rod clusters were left too far inserted in the core for an hour and 20 minutes. Operators then noticed it, introduced borated water, and lowered reactor power."⁽¹⁴²⁾

5.3.2.2 Clinton-1

On April 8, 1990, at Clinton-1 the control room operator made 14 control rod withdrawals while the main turbine bypass valves were open and the reactor power was greater than the low power setpoint of the RPCS. Under these conditions, technical specification 3.1.4.1 was violated because of the control rod withdrawal and technical specification 4.1.4.1 was violated because a second individual didn't prevent the withdrawal. Numerous procedural and disciplinary actions have been carried out to ensure that this type of event is not repeated.⁽¹⁴³⁾

5.3.2.3 Three Mile Island-1

On March 4, 1990, the operator at Three Mile Island-1 inadvertently withdrew rod groups 5 and 6 simultaneously. The event occurred during zero power physics testing. The event is attributed to human error.⁽¹⁴⁴⁾

5.3.3 Control Rod Position Indicator

There were six events in the U.S. and no events or items of interest in foreign countries that involved control rod position indicators. Those events are described below.

5.3.3.1 Callaway-1

Technical specification 3.0.3 of Callaway-1 was entered on February 8, 1990, when the digital rod position indication system (DRPI) failed due to a faulty central control card. Trouble shooting revealed that the DRPI +15 VDC power supply was actually supplying

+20 VDC with a ripple of 200 millivolts RMS (root-mean-squared).⁽¹⁴⁵⁾

5.3.3.2 Catawba-1

The functionality of the digital rod position indication system was lost on September 26, 1990, at Catawba-1 for six control rods. Annunciators indicating rod position indication failure were received in the control room. Technical specification 3.0.3 was entered because rod position indication for more than one rod in a bank was inoperable. After a high priority work request was initiated, personnel found the cause of the failure was a malfunctioning display card. After the card was replaced, displays returned to normal. Throughout the event, neither control rod positions or controls were affected.⁽¹⁴⁶⁾

5.3.3.3 Quad Cities-2

While at 100% core rated thermal power, a partial loss of the rod position indication system occurred on May 31, 1990, at Quad Cities-2. Several alarms were received and investigation found that the event was due to a blown power supply fuse. The fuse was replaced. A technical specification and procedure change are being pursued.⁽¹⁴⁷⁾

5.3.3.4 Robinson-2

Loss of the control rod position indication (RPI) system occurred on January 10, 1990, at Robinson-2. The failure of a two pole, single phase, 120 volt AC circuit breaker which supplies the RPI system caused the event. The breaker was repaired and the RPI system returned to normal.⁽¹⁴⁸⁾

5.3.3.5 Salem-1

An operator at Salem-1 found on August 14, 1990, that more than one analog rod position indicator (ARPI) per bank was inoperable. This necessitated that the unit go to hot shutdown. Due to a design problem, the ARPI system electronics settings have some drift due to the nature of the analog stack coils susceptibility to temperature changes associated with a mode change. The settings were successfully recalibrated.⁽¹⁴⁹⁾

5.3.3.6 Surry-2

A Surry-2 control room operator observed on February 5, 1990, that the Individual Rod Position Indicators (IRPIS) differed from the rod group demand counter by greater than 12 steps. The event occurred during a rampdown and inadequate procedures were available for stopping the rampdown when IRPIS were drifting. Procedures are being revised and the IRPIS were adjusted properly.⁽¹⁵⁰⁾

5.3.4 Control Element Assembly Problems

There were twelve events in the U.S. and two events in foreign countries that involved control element assembly problems. Those events are described below.

5.3.4.1 Arkansas Nuclear-2

On April 1, 1990, at Arkansas Nuclear-2, control room operators failed to complete a control element assembly (CEA) position log. Technical specifications required that CEA positions be determined and logged at least once every 12 hours, either manually or by using CEA position indications. Procedural changes have been made and a surveillance group was established to oversee the surveillance program.⁽¹⁵¹⁾

5.3.4.2 Asco-1

Spain's Asco-1 PWR experienced a control rod drop, "which in turn affected a pump in the primary cooling circuit." The Spanish nuclear agency reported this incident as one of the most significant events in 1990.⁽¹⁵²⁾

5.3.4.3 Byron-1

On August 19, 1990, at Byron-1, a lightning strike induced a voltage surge that activated nine over-voltage protection devices installed on power supplies in the rod drive power cabinets, causing twelve of fifteen control rod cluster assembly groups to be released into the core. A high negative flux rate reactor trip occurred because of the rod drops. Enhancements have been made and more are being pursued to further prevent this type of problem.⁽¹⁵³⁾

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5.3.4.4 Comanche-1

Comanche-1 was automatically tripped from 38 percent power on September 8, 1990, when a lightning strike is believed to have caused a surge in the input power resulting in the rods controlled by one rod control cabinet to drop into the core. This resulted in a high negative flux rate reactor trip. Surge suppressors have been installed to mitigate this problem in the future.⁽¹⁵⁴⁾

5.3.4.5 Oconee-3

Oconee-3 tripped from 49% full power on January 19, 1990, during a test to verify proper operation of the control rod power supplies. The trip occurred when control rod group 6 dropped into the core. Although the root cause was unknown, procedures were revised to prevent this incident from reoccurring.⁽¹⁵⁵⁾

5.3.4.6 Oconee-3

On November 13, 1990, Oconee-3 control rod group 7 dropped into the core. From control room indicators operations personnel recognized that the rod group dropped and were able to trip the reactor from 60% full power before the RPS could automatically trip the reactor. A failed solid state programmer, which controls power to the control rod drive stators, caused the event. The programmer was replaced and the unit was returned to critical.⁽¹⁵⁶⁾

5.3.4.7 Palo Verde-3

While at approximately 100% full power in mode 1 (power operation), a regulating CEA slipped into the core on August 5, 1990. The misaligned CEA could not be withdrawn in one hour and the plant was shutdown, in accordance with the technical specifications limiting conditions for operation. The cause of the CEA dropping into the core was a coil driver actuating logic card malfunction. The card was replaced and successfully tested. Similar events have been reported in past LER's for both Unit 1 and Unit 3.⁽¹⁵⁷⁾

5.3.4.8 Palo Verde-3

Palo Verde-3 was at approximately 81% full power on April 14, 1990, when a reactor trip occurred which resulted from a dropped shutdown group CEA. A random failure of a microchip on the optical isolator card was determined to be the cause and was replaced.

A similar event has been reported for Unit 1 in a past LER.⁽¹⁵⁸⁾

5.3.4.9 Prairie Island-2

Prairie Island-2 had a reactor trip while troubleshooting rod control system on March 16, 1990, while operating at 100% full power. The trip was caused by a technician attempting to trouble shoot an electronic noise problem on the rod control system. The "V-Ref" control signal was forced to a low value by the low impedance of an oscilloscope. This signal caused two rods to drop 10 steps into the core before automatic corrective action was taken. When the signal was reset the rods continued to drop and caused a high negative flux rate reactor trip.⁽¹⁵⁹⁾

5.3.4.10 San Onofre-3

Unit three was shutdown for 535 hours to investigate the reason why a control element assembly failed to insert. The down time exceeded the originally scheduled 358 hours.⁽¹⁶⁰⁾

5.3.4.11 St. Laurent-B1

On January 17, 1990, at St. Laurent-B1, "a routine inspection revealed that one of 57 control rod clusters was sticking momentarily. During subsequent tests and operations, the cluster worked satisfactorily."⁽¹⁶¹⁾

5.3.4.12 St. Lucie-1

At St. Lucie-1 on April 4, 1990, a quality control supervisor found a deficient procedure which allowed surveillance on control element assemblies to be missed each time the reactor was started up. The technical specification violation was followed up with procedural changes to ensure the surveillance is completed within the required time period. A review of St. Lucie-2 found a similar problem.⁽¹⁶²⁾

5.3.4.13 St. Lucie-1

While operating at 100% full power on June 14, 1990, at St. Lucie-1 a CEA dropped into the core repeatedly after being withdrawn. An unusual event was declared and the reactor was shutdown. The cause was determined to be personnel error in the installation of a power supply fuse in the CEA +12V DC logic circuit which was not locked in place and caused intermittent power losses.⁽¹⁶³⁾

5.3.4.14 Waterford-3

On March 22, 1990, at Waterford-3 an automatic reactor trip from 100% power occurred when two CEAs dropped into the core while their drive mechanisms were being transferred to the Control Element Drive Mechanism System (CEDMCS) hold bus. The event was caused by several electrical connectors used to transmit power from the CEDMCS panels to the CEA drive mechanism which were damaged from misalignment during previously performed maintenance. The damaged components were replaced and checked to be operational.⁽¹⁶⁴⁾

5.3.5 Control Rod Swelling, Wear, Corrosion, Cracking

There were three events or items of interest in the U.S. and two events or items of interest in foreign countries that involved control rod swelling, wear, corrosion, or cracking. Those events or items of interest are described below.

5.3.5.1 French PWR Internals: Vibration and Wear

According to an article⁽¹⁶⁵⁾, flow-induced fretting has caused wear in bottom mounted instrumentation thimbles and control rods in French PWRs. The problem has been analyzed and solved.

5.3.5.2 Maine Yankee

On June 7, 1990, during CEA cold functional testing prior to reactor heatup at Maine Yankee, a dual CEA would not fully reinsert into the core. One of the CEA's in the dual unit had lost an end cap, spacer and boron carbide pellets. The pellets were in the guide tube and prevented full insertion of the CEA. Eddy current testing revealed 2 more CEA's were missing end caps and 6 CEA's showed cracking. All of the 23 original design CEA's were replaced. Fuel assemblies holding the CEA's were also inspected and any debris was removed.⁽¹⁶⁶⁾

5.3.5.3 Millstone-3

A negative flux rate signal occurred due to a dropped control rod on June 6, 1990, at Millstone-3. The drop was caused by a broken connection in the stationary gripper coil power cable for rod G13 due to corrosion at the conductor/pin interface. The connection was

repaired. Connectors will be inspected during the next refuel and replaced as needed.⁽¹⁶⁷⁾

5.3.5.4 Point Beach-1

There was a reactor coolant system leakage at Point Beach-1 on July 20, 1990. The canopy seal weld on control drive mechanism and upstream weld on B steam generator channel head drain were leaking. They were repaired by welding.^(160,168)

5.3.5.5 PWR Control Rod Wear

Laborlec (Belgium) has developed a new eddy current system to analyze wear in PWR control rod cluster assemblies. The system can withstand the presence of boron and high levels of radioactivity. The system has detected four main types of degradation: sliding wear, fretting, hairline cracks, and bulging.⁽¹⁶⁹⁾

5.3.6 Guide Tube Problem

There was one item of interest in foreign countries that involved guide tube problems. The item is listed below.

5.3.6.1 Koeberg-1

"Framatome replaced all 114 control rod guide tube split pins at the Koeberg-1 PWR during the unit's recent refueling and maintenance outage. The pins are of Framatome's new third-generation design, identical to those installed in 1989 and 1990 in nine French 900-MW-class PWRs after second-generation split pins...began to crack."⁽¹⁷⁰⁾

5.3.7 Miscellaneous - Control Rod Related

There were three events or items of interest in the U.S. and one item of interest in foreign countries that involved miscellaneous control rod topics. Those events or items of interest are described below.

5.3.7.1 Canopy Seal Leakage Repair

ABB Combustion Engineering Nuclear Fuel announced its development of a seal clamp assembly for control rod mechanisms. The company claims its seal clamp assembly has increased resistance to weld leaks and reduces worker doses during installation.⁽¹⁷¹⁾

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5.3.7.2 Catawba-1 and -2

On July 13, 1990, Catawba-1 and -2 received amendments to technical specifications that "provides the flexibility to withdraw the inconel clad rod cluster control assembly and replace it with a W 17x17 RCCA should unexpected wear be discovered during future inspections."⁽¹⁷²⁾

5.3.7.3 Maanshan-1 and -2

"The hafnium control rods at Maanshan (2 x 951 MWe PWRs) were replaced with rods made from a silver/indium/cadmium mixture."⁽¹⁷³⁾

5.3.7.4 Perry-1

At Perry-1 on October 22, 1990, a control rod was withdrawn without first demonstrating by channel check that the scram discharge column (SDV) level instrumentation was operable. Technical specification 3.3.1 was violated because a channel check of the SDV instrumentation is required at least once every 12 hours when any control rod is withdrawn. Lack of communication between operator and shift supervisor as well as inattention to detail were cited as the causes.⁽¹⁷⁴⁾

5.4 Core/Coolant Oriented

5.4.1 100% Power Exceeded

There were eight events in the U.S. and no events or items of interest in foreign countries that involved exceeding 100% power. Those events are described below.

5.4.1.1 FitzPatrick

Two nonconservative errors in feedwater flow measurement led to power in excess of licensed thermal power limit at FitzPatrick. The feedwater flow transmitters were replaced on October 3, 1988, but weren't calibrated properly. The calibration was completed on November 14, 1989. When more accurate transmitters were placed in service on January 29, 1990, the power level was found to exceed the licensed core thermal power limit. Power was immediately reduced. Flow element vendor input errors have since been identified and corrected.^(175,176)

5.4.1.2 Oyster Creek

Operation in excess of thermal power limit occurred at Oyster Creek on May 11, 1990. The event occurred because of a miscalculation in the plant heat balance equation. A miscalculation in the plant heat balance equation, the omission of 7.84 MW due to the clean up system led to a 1.64 MW over power (average of 100.08% full power). Procedures did not cover the clean up system activities but changes will be made to ensure that the heat balance inputs are correct.⁽¹⁷⁷⁾

5.4.1.3 Oyster Creek

It was noted on August 1, 1990, at Oyster Creek that a revision to the feedwater flow calibration calculation procedure had regulated in a 2% correction to the indicated feedwater flow. This caused a decrease in the allowed reactor plant power. It was determined that 100% power had been exceeded. Loss of coolant accident studies have revealed that the previous analyses were extremely conservative in terms of fuel bundle heatup and plant response. So a larger safety margin to the maximum average planar linear heat generation rate limit was available. Therefore this event had minimal safety significance.⁽¹⁷⁸⁾

5.4.1.4 Palo Verde-1

According to the NRC, on December 6, 1990, the "licensee procedure...was not appropriate to the circumstances in that the procedure did not provide adequate instructions to preclude an inadvertent dilution of the reactor coolant system boron concentration, which resulted in the reactor exceeding 100 percent power during a period of about 14 to 26 minutes." "This is a Severity Level IV violation (Supplement I) applicable to Unit 1."⁽¹⁷⁹⁾

5.4.1.5 Perry-1

At Perry-1 on June 14, 1990, reactor thermal power level exceeded 102% of the maximum power level authorized by the facility operating license. Failure of the feedwater heater level control valve, resulted in a decrease in reactor temperature and then a power transient. The root cause of this event was component failure with a contributing procedural deficiency. Repairs will be made during the next outage and procedural enhancements were made to the appropriate instructions.⁽¹⁸⁰⁾

5.4.1.6 Susquehanna-2

On May 15, 1990, Unit 1 at Susquehanna-2 was at 100% power when an inadvertent actuation of the Reactor Core Isolation Cooling System (RCIC) led to a power increase to 102%. The event was caused when an I&C technician improperly connected test leads in such a way as to cause a short circuit and energize the RCIC initiation logic. The power increase lasted for only 16 seconds and did not pose a safety problem.⁽¹⁸¹⁾

5.4.1.7 Vogtle-2

At Vogtle-2 on April 1, 1990, a power excursion resulted in exceeding the maximum power level (3411 MWth) specified in the plant operating license. A heater drain pump actuation caused cooler feedwater flow to the steam generators leading to a power excursion. The reactor peaked at 105.2% of rated thermal power. The pump actuated because the heat drain tank dump valve failed to open. The valve failed to open a manual actuation pin for the valve was inappropriately inserted into the manual position.⁽¹⁸²⁾

5.4.1.8 Vogtle-2

On April 27, 1990, Proteus (plant) computer point F0-424A was discovered to be reading lower than control board indications. A later review of computer data revealed that the actual increase of indicated power was 100.5% RTP. A computer input card for the F0-424A was replaced and the computer point was verified to be indicated correctly.⁽¹⁸³⁾

5.4.2 Unexpected Power Fluctuation

There were two events in the U.S. and no events or items of interest in foreign countries that involved unexpected power fluctuations. Those event are described below.

5.4.2.1 Millstone-3

On January 15, 1990, at Millstone-3, the axial flux difference (AFD) monitor alarm failed. This caused a technical specification violation because manual logging of the AFD did not begin until 5 hours later, technical specification 4.2.1.1.13 sets a one hour time limit between readings. The inadequate procedures have been revised and the AFD monitor alarm was to

be modified. Inadequate design and procedural deficiency were cited as causes.⁽¹⁸⁴⁾

5.4.2.2 Point Beach-1

Point Beach-1 had an axial flux distribution outside the proscribed limits on August 16, 1990. A turbine electro-hydraulic governor control malfunctioned, causing the event. The event lasted for 17 minutes according to the Plant Processing Computing System (PPCS) but less than 15 minutes on the Control Board indication. Technical Specifications require power to be reduced after 15 minutes of a "Delta flux-outside envelope" alarm on the PPCS and power was not reduced.⁽¹⁸⁵⁾

5.4.3 Axial Shape Index Related

There were two events in the U.S. and no events or items of interest in foreign countries that involved axial shape index problems. Those events are described below.

5.4.3.1 Diablo Canyon-1

At Diablo Canyon-1 on October 19, 1990, the time interval requirement of technical specification 4.2.1.1.A.2, including the allowed extension of technical specification 4.0.2.A, was exceeded. When the axial flux difference monitor alarm is inoperable, technical specification 4.2.1.1.A.2 requires monitoring at least once per hour. Plant process computer P-250 was not performing the calculations and the monitor was declared inoperable. The root cause for the computer problem was not identifiable. The computer was to be replaced during the next outage and procedures were to be written to verify proper computer operations.⁽¹⁸⁶⁾

5.4.3.2 Sequoyah-1

On August 28, 1990, at Sequoyah-1, it was determined that the unit had operated in noncompliance with surveillance requirement 4.2.1.1., due to the AFD monitor alarm having been determined to be inoperable. The P-250 plant computer AFD monitor alarm constants had not been updated in a timely manner to reflect the most recent calibration data. The update did not occur because of procedural inadequacy and inadequate personnel actions. It was also found that two previous examples of not updating the P-250 computer constants had occurred.⁽¹⁸⁷⁾

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5.4.4 Other Power Limit Exceeded

There was one event in the U.S. and two events in foreign countries that involved exceeding a power limit other than 100% power. Those events are described below.

5.4.4.1 Bugey-4

On December 24, 1990, the primary coolant temperature at France's Bugey-4 PWR exceeded the technical specification limit. The coolant temperature was 20°C higher than allowed for approximately an hour. EdF, the French nuclear utility, said that the attention of the operators was diverted from the coolant temperature by tests in support of backfits from a recent outage.⁽¹⁸⁸⁾

5.4.4.2 Perry-1 or -2

A Severity Level IV Violation occurred on November 19, 1990, when "plant operators allowed the reactor coolant temperature to increase to 121 degrees F, which was outside of the specified 75 to 85 degrees F range. At the time of this event, the plant was in Operational Condition 5 (refueling)."⁽¹⁸⁹⁾

5.4.4.3 Tricastin-3

On June 3, 1990, the coolant temperature at Tricastin-3, a French PWR, was 25°C greater than allowed by technical specifications. The temperature rise was caused by failing to stabilize the reactor coolant before isolating the pressure at 32 bar. Normal coolant temperature was achieved 1.5 hours later.⁽¹⁹⁰⁾

5.4.5 Lowering of Water Level

There was one event in the U.S. and one event in foreign countries that involved lowering of the water level. Those events are described below.

5.4.5.1 Arnold

On October 19, 1990, two night shift electricians assigned to the task of troubleshooting the source of an apparent fault on the 'B' recirculation MG system drive motor pulled what they perceived to be the potential transformer for the circuit. This triggered a series of events which caused 3 of 4 inservice demineralizer beds to isolate. A loss of feedwater resulted and the reactor scrambled on low reactor

vessel level. The initiating event, removal of the potential transformer, was caused by personnel error. The root cause of the scram was an undesirable design feature of the condensate demineralizer control logic.⁽¹⁹¹⁾

5.4.5.2 Mutsu, Nuclear Powered Ship

"The 36-MW (thermal) reactor of Japan's experimental nuclear ship Mutsu scrambled automatically after an unscheduled drop in primary coolant volume on May 28 (1990), once again postponing the ship's long-planned off-shore test"⁽¹⁹²⁾

5.4.6 Water Chemistry

There were four events or items of interest in the U.S. and one item of interest in foreign countries that involved water chemistry. Those events or items of interest are described below.

5.4.6.1 Integrated Water Chemistry Monitoring System (IWCMS)

Babcock and Wilcox has developed an on-line water chemistry monitoring system, the Integrated Water Chemistry Monitoring System (IWCMS). The system is capable of trend analysis and is PC-based. IWCMS has been installed at Three Mile Island-1 and Davis-Besse.⁽¹⁹³⁾

5.4.6.2 Occupational Doses and PWR pH

An article on occupational doses at French PWRs shows that EdF is considering increasing primary water circuit pH from 6.9 to 7.1. Preliminary tests at Cruas-1 showed that reductions of up to 30% in dose rates are possible with the new water chemistry. The article also notes that higher pH levels (up to 7.4) have been tested in Sweden, but that primary water stress corrosion cracking problems occurred.⁽¹⁹⁴⁾

5.4.6.3 Optimum pH for PWRs

Dose reduction benefits from increasing coolant pH while avoiding stress corrosion cracking (SCC) is discussed in an article.⁽¹⁹⁵⁾ To avoid SCC and Zircaloy corrosion, system pH should be "gradually increased throughout the cycle." The article notes lithium concentration is also an important factor in reducing the effects of SCC.

5.4.6.4 Shearon-Harris

"Carolina Power & Light's Shearon-Harris was down for 195 forced hours due to secondary water chemical levels."⁽¹⁸⁸⁾

5.4.6.5 Water Chemistry effects on Dose Rates

An article⁽¹⁹⁶⁾ discussed changes being made in water chemistry for BWRs and PWRs to improve radiation control. Increases in water pH can reduce direct dose rates and pH levels above 6.9 (the industry standard) can reduce corrosion and crud deposits. Corrosion and crud can lead to increased doses for workers.

5.4.7 Boron Related Problems

There were no events or items of interest in the U.S. and two events in foreign countries that involved boron related problems. Those events are described below.

5.4.7.1 Blayais 4

Non-borated water was leaked into the primary coolant of France's Blayais-4 PWR. The 30 cubic meter leak was due to a maintenance error in which an opening was left in a steam generator tube after a sample was taken for destructive analysis. An immediate analysis "showed the reactor's antireactivity reserve remained sufficient."⁽¹⁹⁷⁾

5.4.7.2 Dampierre-1

A boron concentration greater than allowed by technical specifications was detected at the French PWR Dampierre-1 on November 17, 1990. The concentration was detected in a "primary circuit borated water backup tank" and operators exceeded the time limits allowed to reduce the boron concentration. The cause was an incorrectly calibrated boron measuring device, which led to higher concentrations in all tanks.⁽¹⁹⁸⁾

5.4.8 Debris in Coolant

There were no events or items of interest in the U.S. and one item of interest in foreign countries that involved debris in the reactor coolant. The item of interest is described below.

5.4.8.1 Olkiluoto-1

"It was thought by Teollisuuden Voima Oy (TVO) that the metal powder found lodged in 15 of the 121 control rod drives at Olkiluoto-1 (735 MWe BWR) might have been in the plant since it was commissioned up in 1978."⁽¹⁹⁹⁾

5.4.9 Miscellaneous Core Problems

There were three events in the U.S. and no events or items of interest in foreign countries that involved other core problems not listed above. Those events are described below.

5.4.9.1 Callaway-1

On October 3, 1990, at Callaway-1 a crew began to remove an irradiation specimen from the reactor vessel. A Source Range Nuclear Instrument (SRNI) was declared inoperable to allow preventive maintenance. According to technical specification 3.9.2 no core alterations are allowable with an inoperable SRNI and personnel failed to recognize that the specimen movement was considered a core alteration, although procedure OTS-KE-00009 (removing specimen) did not identify it as such. During this period the reactor was in mode 6, refuelling. Procedures have been revised and training will emphasize this area. This event was attributed to cognitive personnel error.⁽²⁰⁰⁾

5.4.9.2 Limerick-1

On July 9, 1990, GE notified Philadelphia Electric Company (PECo) an error had been made by GE concerning data supplied in the databank for the Limerick-1 third operating cycle. This databank provides the cycle specific data necessary to calculate and analyze reactor core performance. This may have caused violations of technical specification 3.2.2. The cause of this problem is attributed to personnel error, procedural deficiency, lack of verification by GE and PECO, and inadequate communications.⁽²⁰¹⁾

5.4.9.3 San Onofre-3

At San Onofre-3 on May 10, 1990, core alterations were initiated with one source range detector inoperable, contrary to technical specifications 3.9.2. A control room operator observed that acceptance criteria for signal-to-noise ratio for source range

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detector was not met on excore channel "C". At this time, fuel movement was secured until verification that excore channel "A" was acceptable. The root cause of this event was the failure to determine the inoperability of excore channel "C" prior to core reload due to personnel error.⁽²⁰²⁾

5.5 Personnel or Equipment Oriented

This section is primarily composed of personnel error events that resulted in other problems. For this reason most of the personnel events are listed in the section under the resulting problem. For example, if a fuel handling incident is caused because of a procedural

error, the event is listed in the fuel handling section. Provided in Table 19 is a compilation of cross-references that shows how many problems there were in each category that were caused by personnel or equipment problems. The table also shows where the events are listed. In the previous example, the fuel handling incident would have its section number cross-referenced under procedural deficiency in the fuel handling column.

Events that are personnel or equipment oriented that are not related to other categories are listed after the table.

Table 19. Personnel or Equipment Oriented Cross-References

	Fuel Oriented	Fuel Handling Oriented	Control Rod Oriented	Core/Coolant Oriented	Personnel or Equipment Oriented	Miscellaneous	Total
							14
		5.2.1.1, 5.2.1.3, 5.2.5.1, 5.2.6.1, 5.2.6.7	5.3.1.2, 5.3.1.5, 5.3.1.8, 5.3.3.6, 5.3.4.12	5.4.1.4, 5.4.2.1, 5.4.3.2, 5.4.9.2			0
							17
		5.2.2.4, 5.2.2.7, 5.2.4.1	5.3.1.3, 5.3.1.4, 5.3.1.6, 5.3.3.1, 5.3.3.2, 5.3.3.3, 5.3.3.4, 5.3.4.6, 5.3.4.7, 5.3.4.8	5.4.1.5, 5.4.1.8, 5.4.2.1, 5.4.2.2			3
		5.2.2.2		5.4.3.1, 5.6.1.9			8
		5.2.6.5	5.3.1.9, 5.3.1.10, 5.3.4.13, 5.3.4.14	5.4.1.1, 5.4.1.7, 5.4.7.2			0
							4
		5.2.6.5	5.3.1.1, 5.3.3.5	5.4.5.1			1
		5.2.12.1					2
		5.2.5.2		5.4.1.2			18
		5.2.1.2, 5.2.1.3, 5.2.6.2, 5.2.6.3, 5.2.6.6	5.3.1.7, 5.3.1.12, 5.3.2.2, 5.3.2.3, 5.3.4.1	5.4.1.6, 5.4.4.1, 5.4.5.1, 5.4.7.1, 5.4.9.1, 5.4.9.2, 5.4.9.3	5.5.1.1		

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Table 19 (cont.)

	Fuel Oriented	Fuel Handling Oriented	Control Rod Oriented	Core/Coolant Oriented	Personnel or Equipment Oriented	Miscellaneous	Total
Human Action (not in error)			5.3.4.9				1
Fatigue							0
Training		5.2.1.1	5.3.1.13				2
Communication Problem			5.3.7.4	5.4.9.2			2
Acts of Nature			5.3.4.3, 5.3.4.4				2
Total	0	19	31	23	1	0	74

5.5.1 Equipment Failure - Loss of Power

There were two events in the U.S. and one event in foreign countries that involved loss of power. The events are listed below.

5.5.1.1 Biblis-A

"On June 6, during a repair outage, a human error at the Biblis-A PWR led to loss of power in a 24-volt direct current power source. This led to interruption of power to important (I&C) equipment. Operators immediately recognized the interruption and restored power supply to I&C systems within three minutes."⁽⁸¹⁾

5.5.1.2 Nine Mile Point-1

On November 12, 1990, while at 96% power, Nine Mile Point-1 experienced a loss of offsite power which resulted in the automatic starting of emergency diesel generators. One power board did not recover automatically or by procedure, resulting in the loss of a reactor recirculation pump. Power was reduced to 77% and electricity loads were stabilized. The immediate cause of the event was a phase imbalance detected on phases two and three of reserve transformer 101.⁽²⁰³⁾

5.5.1.3 Vogtle-1

On March 20, 1990, a truck backed into a power pole in the low voltage switchyard at Vogtle-1, causing a loss of offsite power.^(204,205,206,207,208)

At the time, Vogtle-1 was down for refueling and one of the two backup diesel generators was down for maintenance.^(206,209) Operators had trouble keeping the other generator running continuously, but stabilized it after about 40 minutes. Between the interruption and restoration of electrical power, the coolant temperature rose from 90°F to 136°F.^(206,207) The attempts to start the diesel generator caused a turbine and reactor trip at nearby Vogtle-2, which was at 100% power.⁽²⁰⁶⁾

5.5.2 Nonconservative/Incorrect Assumptions

There was one event involving nonconservative or incorrect assumptions in foreign countries. The event is listed below.

5.5.2.1 St. Alban-1

The criticality monitoring system at St. Alban-1 was found to be inadequate during a July 1990 refueling. Later analysis of procedures determined that operational neutron flux detectors were not close enough to fuel. Procedures and surveillances were modified.⁽¹⁴²⁾

5.5.3 Acts of Nature

There was one item of interest involving acts of nature. The item is listed below.

5.5.3.1 Control Rod Behavior and Earthquakes

An article⁽²¹⁰⁾ discussed the potential of earthquakes to slow control rod drop times. The Tadaru Engineering Laboratory in Japan used a full-seized PWR model on a large vibration table to simulate an earthquake. The tests showed that "even for the most severe earthquake, the drop time is within the specified time used in safety analyses." The simulated earthquakes produced elastic deformations in the grid on some fuel assemblies and showed that the stresses on the control rod mechanisms would be small.

5.6 Miscellaneous

5.6.1 Generic Issues/General Interest

Selected items of general interest are listed below.

5.6.1.1 Beznau

"Westinghouse is installing a state-of-the-art, distributed I&C system at Beznau in Switzerland, in a three-phase program that is due to be completed in the autumn of 1994. The system will collect and process data from 1400 analog and 3200 digital signals throughout the plant."⁽²¹¹⁾

5.6.1.2 Bohunice A-1

In May 1979, Czechoslovakia's Bohunice A-1 carbon dioxide cooled heavy water moderated reactor was shut down. Previously, in 1977, an accident damaged a fuel channel and partially melted a fuel assembly. Currently, "the step-by-step decommissioning of the first Czechoslovak demonstration plant is going ahead

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and the resulting radioactive wastes (are) being processed and disposed of."⁽²¹²⁾

5.6.1.3 Higher Burnups

An article⁽²¹³⁾ discussed a report by the OPEN group on increased fuel burnup. The report states that higher burnups "can be achieved without compromising fuel reliability and performance." The benefits for higher burnups are decreases in fuel costs due to fewer fuel assemblies required over a period of time. The report states that burnups of 45 GWd/MTU are possible "without excessive risk of fuel failure," and potential burnups in the mid-1990s could reach 60 GWd/MTU.

5.6.1.4 BWR Instability

"Recent instability events at boiling water reactors prompted NEA/OECD and the NRC to organize a workshop on the problem at Brookhaven National Laboratory, New York, 17-19 October 1990."⁽²¹⁴⁾

5.6.1.5 Channel Boxes and the NRC

The NRC is asking utilities for information on the use of channel boxes in BWRs. In BWRs, channel boxes contain bundles of fuel rods, and information from a fuel failure at a foreign BWR has led the NRC to believe that the potential exists for fuel failures if channel boxes are reused. The NRC believes that bowing is not a problem during the first use of a channel box but that future reuse may exceed the thermal limits and critical power ratio. If these possibilities are ignored, the result could be fuel failure.^(215,216)

5.6.1.6 CHIRON Computer Program

Estimates of the number of failed fuel pins can be obtained using the computer code CHIRON for both BWRs and PWRs. The program is written for IBM PC's and use data from 30 different reactors during 39 cycles. The program can "predict the number of failures within a factor of 2 in 85% of the cases."⁽²¹⁷⁾

5.6.1.7 Cook-1 and -2

"These amendments modify Technical Specifications so that Westinghouse fuel assemblies with enrichments of up to 4.95 weight percent U-235 may be received."⁽²¹⁸⁾

5.6.1.8 Cook-1 and -2

Indiana & Michigan Electric Co. purchased a TENERA maintenance work software package for Cook-1 and -2." The utility "will install the plant information management system (PIMS) in July. PIMS identifies work to be done, estimates its scope, tracks its progress, and maintains a history of it..."⁽²¹⁹⁾

5.6.1.9 Darlington

Problems with a computerized shutdown system that delay the licensing process at Darlington will be rewritten to incorporate changes easier. The rewrite will take two to three years.⁽²²⁰⁾

5.6.1.10 Fort St. Vrain

Public Service Company of Colorado has begun the process of early decommissioning its Fort St. Vrain HTGR. The system will be reconfigured as a gas fired facility." Originally the 60 year SAFSTOR method was to be used, but practicality dictates an early decommissioning.⁽²²¹⁾

5.6.1.11 French Experience with MOX Fuel

An article⁽²²²⁾ discusses recent French experience with mixed-oxide (MOX) fuel. The article discusses the use of plutonium in the French nuclear program in general, and states that "Experience gained during the last three years has been satisfactory, with no clad failures observed"

5.6.1.12 Indian Point-2

By using three LOTUS 1-2-3 Version 2.01 macros, engineers at Indian Point-2 find that analysis of reactor trips can be completed faster. The routines use a relatively simple artificial intelligence method to review all plant parameter values, not just a selected few.⁽²²³⁾

5.6.1.13 Longer Cycles

A recent article⁽²²⁴⁾ describes transitions from 12-month to 18- and 24-month refuelling cycles many PWR operators in the US and Europe have made and are considering.

5.6.1.14 Loss of Fluid Test Facility (LOFT)

An article⁽²²⁵⁾ describing the international joint-venture that sponsored OECD's LOFT Reactor. The small nuclear reactor was "designed to model a large four-loop PWR retaining the same volume/power ratio on a 1:50 volume scale." Six thermohydraulic (including loss of coolant accident) and two fission product experiments were carried out.

5.6.1.15 Mixed Oxide (MOX) Fuel

The fabrication, future use, economics and effects of increasing burnup of MOX fuel are discussed in an article.⁽²²⁶⁾

Another article⁽²²⁷⁾ states that "Mixed oxide fuel is now commercially well established," and continues on to discuss various factors in MOX fuel production including productivity, occupational doses, fuel solubility, waste management and transportation.

5.6.1.16 North Anna-1 and -2

On June 6, 1990, Virginia Electric and Power Co. was issued an amendment to the operating licenses of North Anna-1 and -2 to allow an enrichment increase to 4.3 weight percent U-235.⁽²²⁸⁾

5.6.1.17 NRC Proposal for Computer Monitoring of Nuclear Emergencies

An NRC proposal would create a computer network connecting to all nuclear plants in the United States that would monitor and report plant conditions in the event of an emergency. The computer link would be used to complement the voice-telephone method in used today. The system would be activated by the utility and be entirely automatic.⁽²²⁹⁾

5.6.1.18 Onagawa-1

An expert system that optimizes refueling time has been developed at Japan's Onagawa-1 BWR. The program replaces the conventional trial-and-error procedure to plan refueling. The refueling operation is a critical path in an outage, so there is an incentive to reduce the amount of time it takes to be completed. The program is run on a Sun Workstation, and takes about 30 minutes to complete. During a recent refueling, the conventional method required 818 movements, while the expert system needed only 784.⁽²³⁰⁾

5.6.1.19 Pickering-3 and -4

"In Canada, Ontario Hydro's Pickering-3 and -4 were down all month (November 1990) for large-scale fuel channel replacement and fuel channel inspection, respectively."⁽¹⁹⁸⁾

5.6.1.20 Robinson-2

"The amendment changes the Technical Specifications to increase allowable fuel enrichment in the reactor, the new fuel storage racks and the spent fuel storage pit from 3.9 weight percent to 4.2 plus 0.05 (nominal 4.3 w/o)."⁽²³¹⁾

5.6.1.21 Sequoyah-1 and -2

Amendments to the Sequoyah-1 and -2 technical specifications allows the Tennessee Valley Authority to "increase the maximum enrichment of fuel allowed on the site from 4.0 to 5.0 percent Uranium 235" and that "New fuel with an enrichment greater than 4.5 weight percent may not be stored in the new fuel pit storage racks. This fuel may be stored in the spent fuel storage pool."⁽²³²⁾

5.6.1.22 Shuffleworks™ Computer Program

A computer program for creating fuel movement sequences (shuffle plans) has been developed by ABB Combustion Engineering Nuclear Fuel. The program, called ShuffleWorks™, can produce complete shuffle plans and can adapt to problems encountered during refueling. The main benefit of the program is optimization of fuel movement plans for better use of outage time.⁽²³³⁾

5.6.1.23 Soviets Processing Uranium for US Utilities

Four northeastern U.S. utilities have contracted with a company in the former Soviet Union to enrich uranium. The price is "reportedly 25% less than the U.S. Department of Energy services."⁽²³⁴⁾

5.6.1.24 Windscale Piles

"The Windscale Piles have been largely undisturbed since 1957. Now AEA Technology is undertaking work that will enable the Piles eventually to be decommissioned."⁽²³⁵⁾

5.6.2 Inspection Technology

Selected items on inspection technology are listed below.

5.6.2.1 Comprehensive Corrosion-Erosion Monitoring System

Westinghouse has developed a PC-based system to evaluate the data from non-destructive examinations. Non-destructive evaluation data is used to create 2D and 3D displays of components. The Corrosion-Erosion Monitoring System can be used to predict component life and structural integrity.⁽²³⁶⁾

5.6.2.2 Destructive Gamma-Ray Analysis of Spent Fuel Rods

The Institute of Nuclear Energy conducted destructive gamma-ray analysis of spent fuel rods to compare actual radioactivities and burnups with those predicted by the ORIGEN-II computer code. The library in the program is based on CANDU reactors, and may account for the 15% underestimation of radioactivities. The spent fuel rods were irradiated in the Taiwan Research Reactor.⁽²³⁷⁾

5.6.2.3 Failed Fuel Detection at Calvert Cliffs

The experience in detecting failed fuel at Calvert Cliffs is discussed in a recent article.⁽²³⁸⁾ The article describes failure detection from 1984 on.

5.6.2.4 Failed Fuel Detection at Hatch

An article⁽²³⁹⁾ discusses the detection of failed fuel at the Hatch BWRs. The article states that approximately 108 cladding failures were detected in fuel assemblies since 1980, and most were due to "localized, accelerated corrosion."

5.6.2.5 FITEL Fiberscopes

An article⁽²⁴⁰⁾ describes Furukawa Electric's FITEL fiberscope, designed to withstand radioactive environments. The fiberscope has a semi-flexible and flexible version, and can be used to monitor refueling operations, examine welds, and inspect narrow sections.

5.6.2.6 Fuel Assembly Repair and Inspection Station (FARIS)

BWFC's portable cleanup station, FARIS, is described in a May 1990 article.⁽²⁴¹⁾ It has been used in three nuclear plants and can be installed in about 12 hours. The system has the capability to clean fuel assemblies and remove debris, as well as "carrying out visual inspections on fuel assemblies, spacer grid repair, fuel rod oxide thickness measurements and for fuel rod water channel inspections."

5.6.2.7 In-core Housing Inspection System

GE has developed an In-core Housing Inspection System that can detect fabrication defects as well as service-related flaws in BWRs. "The system uses ultrasonic radio-frequency waveform A-scan recording and imaging, eddy current and visual inspection techniques."⁽²⁴²⁾

5.6.2.8 Light Water Reactor Failed Fuel Detection (Historical)

A recent article⁽²⁴³⁾ discusses the history of fuel failure detection in LWRs. Experience has shown that cladding failures normally do not cause unscheduled shutdowns. This caused detection methods to shift towards continuous coolant activity surveillance and effluent release rates and away from failed fuel rod location. During an outage the exact location of failed fuel is found using ultrasonic detection in PWRs, while BWRs use assembly sipping followed up by eddy current tests to determine failure location. "The two most widely used ultrasonic detection methods...have established accuracies of 99.36 and 99.998%"

5.6.2.9 Photogrammetric Techniques of Fuel Assembly Inspection

An article discusses the use of photogrammetric techniques to inspect fuel assemblies. These methods allow the "observation of oxidation processes and length changes" in fuel assemblies. The article describes techniques used and states "special advantages of photogrammetric techniques like non-contact measuring, short duration while taking the exposures, photographic object documentation, etc. are important arguments to use photogrammetry in nuclear plants."⁽²⁴⁴⁾

5.6.2.10 Three Dimensional Television

AEA Technology has developed a 3D-television system that is radiation tolerant. The system can be used to observe processes or inspect items.⁽²⁴⁵⁾

5.6.2.11 Ultrasonic Detection of Failed Fuel

The history and current use of ultrasonics to locate fuel failures is described in a recent article⁽²⁴⁶⁾. Three systems, the Failed Fuel Rod Detection System by ITI/Brown, Beveri, the Echo-330™ by Babcock and Wilcox, and the ULTRATEST™ system by ANF, are discussed.

5.6.3 Hot Particles

There were two hot particle incidents in the U.S. and two items of interest. Those events or items of interest are described below.

5.6.3.1 ANS Winter Meeting Special Session

An article⁽²⁴⁷⁾ summarized the special session on hot particles that was held at the November 1989 ANS Winter Meeting. The article discusses hot particles in general and quotes a survey that suggests "70 percent of the plants had some problem with hot particles." The article also discussed the historical perspective of hot particles, dosimetry, and the experience at the Three Mile Island-1 and -2 and Oyster Creek plants.

5.6.3.2 FitzPatrick Hot Particle Incident

An NRC Notice of Violation was issued to New York Power Authority for an incident on June 12, 1989. The violation states that "the surveys provided to support on-going Spent Fuel Pool work...were inadequate to readily identify the presence of a highly radioactive object measuring up to 1000 R/hr on contact which appeared in the work area." "As a result, several workers were exposed to radiation fields emanating from the object, and received unplanned exposures."⁽²⁴⁸⁾

5.6.3.3 Hot Particle Shipping Incident

An NRC Notice of Violation was issued to Northeast Utilities (NNECO) on April 30, 1990, for an incident involving an underwater shear cutter that was shipped from Millstone to Duane Arnold. According to NNECO, the package left Millstone-1 with surface

radiation measured at less than 200 millirem per four. At Duane Arnold, surface radiation levels between 200 and 300 millirems per hour were measured on the bottom of the shipping package. A hot particle is thought to have dislodged during shipping.⁽²⁴⁹⁾

5.6.3.4 Hot Particles and Health Hazards

An article⁽²⁵⁰⁾ states that hot particles may not be as big issue as once thought, but that other concerns about them still need to be addressed. Skin contact may result in "some reddening, flaking, or a small blister," but the effect of hot particles on other organs of the body (ie., the larynx, eyes, ears, etc.) is not understood very well and needs to be investigated further.

5.6.3.5 Hot Particles and Skin Contact

An article⁽²⁵¹⁾ announced a report on the "Limit for Exposure to Hot Particles on the Skin" has been released by the National Council on Radiation Protection and Measurements. The article describes hot particles and the potential doses they can cause.

5.6.4 Trends

A selected item dealing with future market trends is listed below.

5.6.4.1 NSSS Vendor Views on Future Market Trends

In response to a questionnaire concerning "New Orders", "Restructuring", "Meeting Utility Requirements", "Design Developments", "Small Reactors", "Improving Performance", and "Vendor Ownership", eight vendors gave their opinions on the topics. The vendors were ABB CENF, AECL, BWFC, Framatome, GE, Mitsubishi, SNP, and W.⁽²⁵²⁾

6.0 Trends

This section discusses the trends of fuel failures and reportable events that impact the fuel in U.S. commercial reactors. Generic issues are important to discuss because the problems of one plant can often be analyzed to prevent problems at other plants.

Over the past several years, considerable attention has been given to reducing the number of fuel failures in commercial reactors. Presently, 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, fuel vendors in most cases are not reporting the number or the cause of fuel failures in their fuel operating experience reports to the NRC.

Because neither the utilities or the vendors are reporting the 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 for some failure mechanisms. Failures may even be anticipated because of the large amount of failure data at a given reactor or reactors. This causes the issue to fade away even though failures may still be occurring. This makes it difficult to determine what effect, if any, the various corrective solutions that are initiated have on the problem.

Table 20. Failure Mechanisms Over the 1986-1990 Period⁽³⁵⁾

Cause	1986-1987 Percentage	1988-1989 Percentage	1990 Percentage
PWRs			
Handling Damage	--	--	2
Debris	9	54	14
Baffle-Jetting	1	1	--
Grid Fretting	1	4	17
Primary Hydriding	10	2	
Other Fabrication	1	10	19
Other Hydraulic	1	1	--
Unknown	76	28	48
Total	100	100	100
BWRs			
CILC	64	71	32
Fabrication	4	4	10
PCI	0	1	--
Debris	--	--	10
Unknown	32	24	48
Total	100	100	100

6.1 Fuel Failure Trends

A recent EPRI paper⁽³⁵⁾ points out the failure percentages attributed to debris fretting, CILC, and PCI over the past five years. It also shows an increased percentage of fabrication oriented problems, grid fretting problems (PWRs), and unknown problems; these trends are illustrated in Table 20.

The largest percentage of fuel failures (48%) reported in 1990 were classified as unknown. This is of particular concern since new, previously unidentified, fuel failure mechanisms may be grouped into this category. Problems labeled as "unknown cause" are significant and need to be addressed if fuel performance is to be improved further. Additional monitoring, inspection, data gathering, and studies are needed to correctly identify, model, and develop solutions for pin failure phenomena. Further studies may also improve the utilities ability to better identify fuel failure causes and then determine if modifications to fuel designs or plant operations and maintenance practices are needed to further reduce fuel failures.

6.2 Other Reported Event Trends

Trends that may be inferred from the reported events in Section 5 are described below. The total number of events that occurred in each category in the U.S. in 1990 are provided in Table 21. This table is developed from information presented in Appendix C that specifically cross references the events in Section 5 by reactor plant.

There was one fuel oriented event reported in the United States in 1990; it involved a failure due to a manufacturing defect at Vermont Yankee.

There were 21 fuel handling oriented events reported in the United States in 1990. One event was a condition reported as outside of the design basis of the plant. Eight events dealt with fuel handling of which half were attributed to personnel error, two were equipment failures, one was a procedural deficiency, and one was a fuel loading problem which was attributed to the fact that the assembly was bowed and difficult to load. Twelve events dealt with fuel storage related problems of which four were personnel related, three were procedural problems, three were equipment problems, and two were design inadequacies. Eight of the fuel storage problems were ventilation system oriented.

There were 35 control rod system related events reported in the U.S. in 1990. Ten events were associated with defective electronics, ten with personnel errors, six were procedural problems, four were equipment oriented, two were caused by lightning strikes, two were caused by unknown mechanisms, and one event related to control rod damage.

There were eighteen core/coolant oriented events reported in the U.S. in 1990. Eight of the events were violations of technical specifications 100% power limits, four were personnel errors, three were equipment failures, two were procedural problems, and one was a water chemistry problem. There were two events that were personnel or equipment oriented reported in the U.S. in 1990 that did not affect the previously mentioned categories. One event was an equipment failure and one was a personnel error.

The miscellaneous category had two events reported in the U.S. in 1990 and they were both related to hot particle incidents.

Table 21. Total Number of Reported Domestic Events by Category in 1990

Year	Fuel Oriented ^(a)	Fuel Handling Oriented ^(b)	Control Rod Oriented ^(c)	Core/Coolant Oriented ^(d)	Personnel/Equipment Oriented ^(e)	Misc. ^(f)	Total
BWRs							
1990	1	4	9	8	1	2	25
PWRs							
1990	0	17	26	10	1	0	54
Total Events							
1990	1	21	35	18	2	2	79

a Fuel Oriented Category (section 5.1) Includes: All fuel failure mechanisms; Swelling, wear, oxidation/ corrosion of fuel cladding; Thinning of in-core instrumentation; Iodine spiking incidents; Miscellaneous or unknown fuel problems.

b Fuel Handling Category (section 5.2) Includes: Fuel handling events with dropped, broken, damaged or potential for damage to fuel; Fuel in incorrect position; Crane operations; Fuel handling procedural violations; Spent fuel pool problems (heat removal, ventilation); Dry storage; Spent fuel consolidation; Spent fuel storage issues; Fuel production plants; Miscellaneous or unknown fuel problems.

c Control Rod Category (section 5.3) Includes: Control rod system problems; Control rod operation; Control rod position indications; Control element assembly problems. Control rod swelling, wear, corrosion, cracking; Guide tube problems; Miscellaneous or unknown control rod problems.

d Core/Coolant Category (section 5.4) 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.

e Personnel/Equipment Category (section 5.5) Includes: Procedural, administrative, or communication problems; Equipment failure, installation, maintenance, manufacturing, or design problems; Human error, actions (not in error), fatigue, or training inadequacy; Unanalyzed conditions; Incorrect or non-conservative assumptions; Equipment problems due to acts of nature.

f Miscellaneous Category (section 5.5.3.1) Includes: Generic interests; Inspection technology; Hot particles; Airborne contamination.

7.0 Summary of High-Burnup Fuel Experience

In 1978, DOE established an extended burnup program. The program's goal was to demonstrate the technology necessary to extend discharge burnup levels to 45 GWd/MTU for BWRs and 50 GWd/MTU for PWRs.⁽²⁵³⁾ Although reduction in fuel costs to utilities was the original objective of the program, the industry's needs have naturally resulted in other reasons for extending fuel burnup, such as, the reduction of generated spent fuel by LWRs (i.e., waste minimization) and improved capacity/availability factors.

DOE 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%.⁽²⁵⁴⁾ It is estimated that 26 commercial reactors will be required to expand their spent fuel storage capacity by the year 2000.⁽²⁵⁵⁾ An industry wide trend towards 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.

Longer fuel cycles are beginning to be used in many plants today. The longer fuel cycles result in higher capacity/availability factors because the plant has less outage time per year of operation. In most cases, the longer fuel cycles result in extended fuel burnups when compared to a standard 12 month cycle.

The 1978 extended burnup program resulted in newer fuel designs and improved manufacturing processes, reducing many fuel failure mechanisms which limited fuel performance. 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.⁽²⁵⁵⁾ 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, methodology and tests.

Table 22. Highest Burnup Fuel Experience by Vendor

Vendor	Plant or Test	Type	Burnup (GWd/MTU)	Comment
ABB CENF ⁽¹⁾	ANO-2	PWR	44.8	Batch Average ^(a)
	St. Lucie-2	PWR	44.0	Batch Average ^(b)
	702 rods discharged		56-59.9	Batch Average
GE ⁽²⁾		BWR	>45.0	Bundle Average
		BWR	60.0	Peak Pellet Exposure
SNP ⁽³⁾	R. E. Ginna	PWR	52.1	Assembly Average
	D. C. Cook-2	PWR	46.4	Assembly Average ^(c)
	Big Rock Point	BWR	45.1	Assembly Average ^(d)
	Gundremmingen-C	BWR	40.0	Assembly Average ^(e)
W ⁽⁴⁾	Zion-1 & 2	PWR	55.0	4 Assemblies Average
			60.0	Peak Rod Burnup
	North Anna-1	PWR	58.4	Lead Assembly Average
			>60.0	Lead Fuel Rod Average
BWFC	Information was not provided by BWFC			

(a) In Core

(b) Discharged in 1990

(c) 17x17 assembly

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

(e) 9x9 assembly

High-Burnup

While several utilities have achieved some extended burnup experience, the industry, on average, has yet to achieve high discharge burnups. The highest annual average burnups for all BWR and PWR discharged assemblies were reached in 1990 (25.0 and 33.8 GWd/MTU respectively). However, the number of utilities that have used, or are planning to use, extended burnup fuel in current and future cores is rapidly increasing. Since a single batch of extended burnup fuel will typically have a useful life of 5 to 7 years, average discharge burnups will increase by approximately 50% over the next seven years.

The highest burnups achieved through 1990 by the four reporting fuel vendors are summarized in Table 22. A historic perspective of domestic BWR and PWR fuel burnup experience is given in Figures 7 and 8 respectively. (Most of the data for Figures 7 and 8 were obtained from Reference 256). Discharged fuel assembly burnup statistics for domestic BWRs and PWRs are summarized in Table 23 and illustrated in Figure 9.⁽²⁵⁷⁾

Table 23. Spent Fuel Burnup (Assemblies)⁽²⁵⁷⁾

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	Avg.
Bolling Water Reactors														
1990 Alone	#	0.0	0	184	369	582	1,357	893	9	0	0	0	3,486	
	%	0.0	0.0	5.3	10.6	16.7	38.9	25.6	0.3	0.0	0.0	0.0	100.0	25.0
Since 1968	#	2,408	2,350	4,011	7,218	11,837	14,314	3,108	10	7	0	0	45,293	
	%	5.3	5.2	8.9	15.9	26.1	31.6	6.9	0.0	0.0	0.0	0.0	100.0	21.3
Pressurized Water Reactors														
1990 Alone	#	0	0	44	183	86	291	940	1,464	572	13	1	3,594	
	%	0.0	0.0	1.2	5.1	2.4	8.1	26.2	40.7	15.9	0.4	0.0	100.0	33.8
Since 1968	#	49	140	1,167	3,279	2,206	6,691	10,124	6,716	1,613	66	17	32,105	
	%	0.2	0.4	3.6	10.2	6.9	20.8	31.5	20.9	5.0	0.2	0.1	100.0	29.7

In 1987 the Electric Power Research Institute set a goal of 60 GWd/MTU assembly-average burnup to be reached by 1997.⁽²⁵⁸⁾ These goals are being spurred by the industries desire to achieve longer fuel cycles (18-24 months) thereby reducing operating and fuel costs. Extending fuel burnup further will also continue to provide the following benefits: reduced costs associated with expanding spent fuel storage capacities, and reduced uranium resource requirements. 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 will be a significant safety concern to the NRC as the industry seeks higher burnups.

For further discussions of extended burnup experience for prior years refer to the Fuel Performance Annual Report for calendar year 1989 (Reference 17).

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

Figure 7. Domestic BWR Fuel Burnup Experience

Figure 8. Domestic PWR Fuel Burnup Experience

High-Burnup

Figure 9. Spent Fuel Burnup - Comparison of All Fuel Discharged Since 1968 with Fuel Discharged in 1990⁽²⁵⁷⁾

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Appendix A

Typical Fuel Assembly Parameters

Table A.1 Typical Fuel Assembly Parameters

Vendor	BWFC (B&W Reactor System)		BWFC (W Reactor System)		ABB CENF		W		SNP			GE					
Fuel Rod Array	15x15	17x17	15x15	15x15	17x17	14x14	16x16	14x14	15x15	17x17	15x15	17x17	8x8	9x9	7x7	8x8	8x8
Reactor Type	PWR	PWR	PWR	PWR	PWR	PWR	PWR	PWR	PWR	PWR	PWR	PWR	BWR	BWR	BWR	BWR	BWR
Assemblies Per Core	177	205	157	157	193	217	217	121	193	193	193	193	560	724	764	560	560
Fuel Rods Per Assembly	208	264	204	204	264	176	236 ^(a)	179	204	264	204	264	60	80	49	63	62
Empty Locations Per Assembly	17	25	21	21	25	20	20	17	21	25	21	25	4	1	None	1	2
Rod Pitch (in.)	0.568	0.502	0.563	0.563	0.496	0.580	0.5063	0.556	0.563	0.496	0.563	0.496	0.842	0.572	0.738	0.640	0.640
Rod Pitch (mm)	14.4	12.8	14.3	14.3	12.6	14.7	12.9	14.1	14.3	12.6	14.3	12.6	16.3	14.5	18.7	16.3	16.3
System Pressure (MPa)	15.2	15.5	13.9	13.9	15.5	15.5	15.5	15.5	15.5	15.5	15.5	15.5	7.14	7.07	7.14	7.14	7.14
System Pressure (psia)	2200	2250	2015	2015	2250	2250	2250	2250	2250	2250	2250	2250	1035	1026	1035	1035	1035
Core Average Power Density (kW/liter)	91.4	107.3	82.25	82.25	82.25	78.5	96.4	95.6	98.1	104.7	98.1	104.7	40.57	46	50.732	50.51	49.15
Average LHGR (kW/m) ^(b)	20.3	18.8	18.1	18.4	17.8	20.0	18.2	20.3	22.0	17.8	22.0	17.8	15.2	12.1	23.1	17.9	17.7
Average LHGR (kW/ft)	6.20	5.73	5.53	5.60	5.43	6.09	5.54	6.20	6.70	5.44	6.70	5.44	4.63	3.68	7.049	5.45	5.38
Axial Peak LHGR in an Average Rod (kW/m)	24.4	22.6	25.1	25.5	27.6	24.00	21.00	24.36	26.40	21.36	26.40	21.4	18.24	17.5	27.72	21.48	21.24
Axial Peak LHGR in an Average Rod (kW/ft)	7.44	6.88	7.66	7.76	8.42	7.31	6.41	7.44	8.04	6.53	8.04	6.53	6.02	5.34	9.16	7.09	6.99
Max. Peak LHGR (kW/m)	53.0	49.9	47.6	47.6	42.7	53.5	42.7	56.8	61.7	44.6	51.9	54.5	47.6	37.7	60.2	44.0	44.0
Max. Peak LHGR (kW/ft)	16.16	15.20	14.5	14.5	13.0	16.3	13.0	17.3	18.8	13.6	15.83	16.6	14.5	11.5	18.35	13.4	13.4
Max. Fuel Temp °C	2340	2290	2149	2149	1927	2140	1880	2260	2340	1870	2200	1747	2040	2040	2440	1830	1890
Max. Fuel Temp °F	3650	4155	3900	3900	3500	3890	3420	4100	4250	3400	3997	3177	3700	3705	4430	3325	3435
Core Average Enrichment wt% ²³⁵ U	3.30 ^(c)	3.15 ^(c)	4.00 ^(c)	3.41 ^(c)	3.40 ^(c)	3.89 ^(c)	2.36	2.90	2.80	2.60	3.02	3.65	2.65	2.8	2.19	1.80	1.99
Max. Local Exposure GWd/MTU ^(d)	55	55	55	55	55	50	55	50	50	50	47.5	52	35	55	40	40	45
Cladding Material ^(e)	Zry-4	Zry-4	304SS	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-4	Zry-2	Zry-2	Zry-2	Zry-2	Zry-2
Fuel Rod Length (m)	3.904	3.878	3.218	3.197	3.848	3.71	4.09	3.87	3.80	3.85	3.86	3.86	3.99	3.99	4.09	4.09	4.20
Fuel Rod Length (in.)	153.7	152.7	126.7	125.9	151.5	145.9	161.0	152.4	149.7	151.6	152.0	152.0	156.9	157.2	161.1	161.1	165.4
Active Fuel Height (m)	3.602	3.632	3.061	3.012	3.658	3.47	3.81	3.66	3.66	3.65	3.66	3.66	3.66	3.69	3.66	3.71	3.81

A.1

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Appendix A

Table A.1 (cont.)

Vendor	BWFC (B&W Reactor System)		BWFC (W Reactor System)		ABB CENF		W		SNP		GE						
Active Fuel Height (in.)	141.8	143.0	120.5	118.6	144.0	136.7	150	144	144	143.7	144	144.00	144	145.24	144	146	150
Plenum Length (m)	0.298	0.242	0.122	0.159	0.164	0.22	0.25	0.18	0.21	0.16	0.17	0.18	0.27	0.24	0.41	0.36	0.25
Plenum Length (in.)	11.7	9.5	4.8	6.3	6.4	8.6	10.00	6.99	8.2	6.3	6.8	7.20	10.63	9.37	16.0	14.0	10.0
Fuel Rod OD (mm)	10.92	9.63	10.72	10.72	9.50	11.18	9.70	10.72	10.72	9.50	10.77	9.14	12.74	10.77	14.30	12.52	12.27
Fuel Rod OD (in.)	0.430	0.379	0.422	0.422	0.374	0.440	0.382	0.422	0.422	0.374	0.424	0.360	0.5015	0.424	0.563	0.493	0.483
Cladding ID (mm)	9.58	8.41	9.88	9.35	8.28	9.75	8.43	9.48	9.48	8.36	9.25	7.87	10.91	9.25	12.68	10.80	10.64
Cladding ID (in.)	0.377	0.331	0.389	0.368	0.326	0.384	0.332	0.3734	0.3734	0.329	0.364	0.310	0.4295	0.364	0.499	0.425	0.419
Cladding Thickness (mm)	0.673	0.610	0.419	0.686	0.610	0.711	0.635	0.617	0.617	0.572	0.762	0.64	0.914	0.762	0.813	0.864	0.813
Cladding Thickness (in.)	0.0265	0.024	0.0165	0.027	0.024	0.028	0.025	0.0243	0.0243	0.0225	0.030	0.025	0.036	0.030	0.032	0.034	0.032
Diametral Gap ^(f) (micron)	213.4	198.1	165	178	165	190.5	178	190	190	165	190	177.8	254	190	305	229	229
Diametral Gap ^(f) (mil)	8.4	7.8	6.5	7.0	6.5	7.5	7.0	7.5	7.5	6.5	7.5	7.0	10.0	7.5	12.0	9.0	9.0
Fuel Pellet Diameter (mm)	9.362	8.209	9.715	9.17	8.115	9.56	8.26	9.29	9.29	8.19	9.06	7.70	10.66	9.06	12.37	10.57	10.41
Fuel Pellet Diameter (in.)	0.3686	0.3232	0.3825	0.361	0.3195	0.3765	0.325	0.3659	0.3659	0.3225	0.3565	0.3030	0.4195	0.3565	0.487	0.416	0.410
Fuel Pellet Length (mm)	11.05	9.53	11.63	10.80	10.16	11.43	9.91	15.24	15.24	13.46	6.93	8.84	8.13	10.41	12.70	10.67	10.41
Fuel Pellet Length (in.)	0.435	0.375	0.458	0.425	0.400	0.450	0.390	0.600	0.600	0.530	0.273	0.348	0.320	0.410	0.500	0.420	0.410
Fuel Pellet Density %TD ^(g)	95	95	95	95	96/95 ^(h)	95	95	94	95	95	94	94.0	95	94.5	95	95	95

(a) Unshimmed assemblies.

(b) LHGR = linear heat generation rate.

(c) Reload batch average enrichment.

(d) 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., U=uranium).

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

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

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

(h) Design may use either density.

Appendix B

Historical Background On Fuel Reliability

Appendix B

Historical Background On Fuel Reliability

This appendix consists of several tables detailing historical information on the reliability of LWR

Zircaloy-clad and stainless steel-clad fuel for each of the various vendors.

Table B.1 BWFC Fuel Reliability

Year	Annual Fuel Rod Reliability, %	Fuel Failure Index	Reference Number
1990	Report not submitted	Report not submitted	5
1989	99.997, 98.6 ^(a)		260
1988	99.99, 99.98 ^(a)		261
1987	99.98, 99.98 ^(a)		262
1986	99.999, 99.998 ^(a)		263
1985	99.995, 99.997 ^(a)	See Figure 1 in Ref. 264	13
1984	99.990, 100 ^(a)	"	12
1983	99.991, 100 ^(a)	"	11
1982	99.994	"	10
1981	99.992	"	9
1980	99.997	"	8
1979	~99.97	"	7
1978	99.9-99.99	"	6
1977	--	"	--
1976	--	"	--
1975	--	"	--
1974	--	"	--

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Table B.2 ABB Combustion Engineering Nuclear Fuel (ABB CENF) Fuel Reliability

Year	Annual Fuel Reliability, %	Defect Level, %	Reference Number
1990	99.998 ^(b)	0.002	1
1989	99.997 ^(b)	0.003	265
1988	(c)	--	266
1987	(c)	--	267
1986	(c)	--	268
1985	(c)	--	13
1984	99.98	0.02	12
1983	99.98	0.02	11
1982	99.98	0.02	10
1981	(h)	--	9
1980	(h)	--	8
1979	>99.99	<0.01 ^(d,e)	7,269
1978	99.99	0.01 ^(e)	6,268
1977	99.98	0.02 ^(f)	268,270
1976	99.98	0.02 ^(f)	268,269
1975	99.97	0.03 ^(f)	268,269
1974	>99.75	<0.25 ^(f)	268,269
1973	99.96	0.04 ^(f)	268,269
1972	99.99	0.01 ^(f)	268,269
1971	99.99	0.01 ^(f)	268,269

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Table B.3 General Electric Company (GE) Fuel Reliability

Year	Annual Fuel Rod Reliability		Reference Number
	Percentage	Comments	
1990	>99.98	All Fuel Types, 1974-1990	2
1989	99.98	All Fuel Types, 1974-1989	271
1988	>99.97	All Fuel Types, 1974-1988	272
1987	>99.99	All Fuel Types, 1974-1987	273
	>99.999	Barrier Types, 1987	272
1986	>99.99	All Fuel Types, 1974-1986	274
	99.994	All Fuel Types, 1986	275
	>99.999	Barrier Types, 1986	274
1985	>99.99	All Fuel Types, 1974-1985	13
1984	>99.99	All Fuel Types, 1974-1984	12
	100.00	Barrier Types, 1984 ^(g)	12
1983	99.993	All Fuel Types, 1974-1983	11
	99.998	All Fuel Types, 1974-1983 ^(h)	11
1982	>99.98	All Fuel Types, 1974-1982	10
1981	>99.98	All Fuel Types, 1974-1981	9
1980	>99.98	All Fuel Types, 1974-1980	8
1979	99.984	All Fuel Types, 1974-1979	7
	99.998	For 8x8R plus 8x8R(PP) Types ⁽ⁱ⁾	7
1971-1978	--	See Table A.1 in Reference 10	6,10

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Table B.4 Siemens Nuclear Power Corporation (SNP) Fuel Reliability

Year	Annual Fuel Rod Reliability, %^(j)	Reference Number
1990	99.997	3
1989	99.997	276
1988	>99.994	277
1987	≥99.995 ^(k,l)	278
1986	99.995 ^(k,l)	279
1985	99.994 ^(k,l)	13
1984	99.995 ^(k,l)	12
1983	99.998, ^(k,l) 99.87 ^(k,m)	11
1982	99.998 ^(k,l)	10
1981	99.998, ^(k,l) 99.987 ^(k,m)	9
1980	100	8
1979	(e)	7
1978	(f)	6

Notes at end of Appendix

Table B.5 Westinghouse Electric Corporation (W) Fuel Reliability

Year	Annual Fuel Rod Reliability, %	Cladding Defect Level, % of Rods	Average Coolant Activity, ^(p) $\mu\text{Ci/g}$	Fourth Quarter % of Design Basis Activity Release Rate ^(p,q) in W-Fueled Reactors	Range of Maximum Iodine-131 Activity in Primary Coolant in W-Fueled Reactors, $\mu\text{Ci/g}$	Ref No
1990	99.998 ^(m)	0.002	0.0042	--	See Table 17 and Figure 6 in this report	4
1989	99.994 ^(m)	0.006	0.0047	--	See Table 16 and Figure 6 in Ref 17	280
1988	99.994 ^(m)	0.006	0.0049	--	See Table 17 and Figure 6 in Ref 16	281
1987	99.994 ^(m)	0.006	0.0045	--	See Table 17 and Figure 6 in Ref 282	282
1986	(r)	--	0.0060, ^(s) 0.0070 ^(t)	--	<0.001- $\leq$ 0.100	283
1985	(r)	--	0.0086, ^(s) 0.0092 ^(t)	--	<0.001-0.100 ^(v)	13
1984	(r)	--	0.008	--	0.008-0.121 <0.001-0.1 ^(v)	12
1983	(r)	--	0.030	--	0.0001-0.102 <0.001-3 ^(v)	11
1982	(r)	--	0.0296, ^(s) 0.0041 ^(t)	--	0.0005-0.105 <0.001-0.3 ^(v)	10
1981	(r)	--	--	<0.001-6.38	<0.001-0.3 ^(v)	9
1980	(r)	--	--	<0.01-4.2	<0.001-0.1 ^(v)	8
1979	~99.983	~0.017	--	--	--	7
1978	(r)	--	--	--	<0.001-0.1 ^(v)	6
1977	99.938-99.9999	0.0001-0.062	--	--	--	6
1976	(r)	--	--	0.05-5.2 ^(w)	<0.001-0.3 ^(v)	--
1975	99.75-100.00 ^(w)	0.00-0.25 ^(w)	--	0.0-15 ^(w)	--	--
1974	99.790-99.999 ^(w)	0.001-0.210 ^(w)	--	0.1-21 ^(w)	<0.001-1.0 ^(v)	--
1973	99.91-99.999 ^(w)	0.001-0.09 ^(w)	--	0.1-2.8 ^(w)	--	--
1972	99.74-100 ^(w)	0-0.26 ^(w)	--	0.1-6.0 ^(w)	<0.001-0.3 ^(v)	--
1971	99.23-100 ^(w)	0-0.77 ^(w)	--	0.1-22 ^(w)	--	--
1970	99.24-99.999 ^(w)	0.001-0.76 ^(w)	--	0.1-76 ^(w)	--	--
1969	99.64-100 ^(w)	0-0.36 ^(w)	--	0.0-36 ^(w)	--	--
1968	--	--	--	--	--	--

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- (a) Reliability of stainless steel-clad fuel.
- (b) Excluding failures caused by debris-induced fretting wear.
- (c) Annual fuel rod reliability of fuel rods not stated by C-E, but they provided data on coolant activity. In their input (Ref. 284) for the annual report for 1988, ABB CENF indicates that the overall fuel rod reliability of their fuel fabricated since 1984 is estimated to be 99.997%, excluding failures caused by debris-induced fretting wear and by baffle jetting in the Yankee Rowe plant (an older Westinghouse plant).
- (d) As of February 1, 1979.
- (e) See Figure 1 in Ref. 268.
- (f) See Figure 1 in Ref. 268 and Figure 1 in Ref. 269.
- (g) Based on 1983 data (Ref. 12).
- (h) Reliability of 8 x 8 fuel if fuel failures involving CILC are excluded.
- (i) R = retrofit design, PP = prepressurized.
- (j) See references for reliability of BWR and PWR fuel rods, respectively.
- (k) On a cumulative basis.
- (l) 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).
- (m) The fuel reliability value is based on fuel failures from all causes.
- (n) Annual fuel reliability not stated (9 BWR fuel rods and 4 PWR fuel rods were reported as failed). As of December 1979, SNP (previously ANF) had 2190 fuel assemblies in domestic and foreign plants.
- (o) Annual fuel reliability not stated (7 BWR fuel rods failed and 1 or 2 PWR fuel rods may have failed). As of November 1978, SNP (previously ANF) had 1342 fuel assemblies in domestic plants.
- (p) In Revision 5 of WCAP-8183 (Ref. 285), Westinghouse reported that, starting June 30, 1976, they were reporting fuel performance in terms of coolant activity level. Westinghouse indicated that the prior concept of a "cladding defect level" implies that all defects introduce activity into the coolant at the same rate; however, leak rates of defected rods can decrease (or increase) as a function of time. Hence, Westinghouse decided to abandon reporting of reactor core condition in terms of a number of defects and started reporting activity of iodine-131 in the coolant as a percentage of the coolant design basis activity. In Revision 9 of WCAP-8183 (Ref. 286), Westinghouse states that "the coolant design basis activity varies somewhat from plant to plant depending upon such factors as reactor power and coolant purification flow rate; however, a value of approximately 2 μCi of iodine-131 per gram of coolant water can be used for purposes of comparison. Since the coolant design basis activity was based on an inferred 1-percent defect level, the new basis of reporting (activity) produces a number approximately 100 times larger than the previous basis (inferred defects). That is, 1 percent of design basis activity would previously have been reported as 0.01 percent defected rods." Starting in 1982, Westinghouse provided data on average coolant activity level (also maximum iodine-131 activity in the primary coolant for each Westinghouse-fueled reactor) in terms of $\mu\text{Ci/g}$. Activity release rate calculated from coolant activity averaged over the quarter and presented as percent of that iodine-131 release rate which establishes the basis for design of plant shielding and coolant cleanup system equipment.
- (q) Westinghouse did not state a fuel rod reliability (integrity) value. Westinghouse continues to evaluate fuel performance in terms of coolant activity level.
- (r) Excludes fuel failures due to baffle jetting.
- (s) Includes fuel failures due to baffle jetting.
- (t) See Figures 8 and 9 in Ref. 13, Figure A.3 in Ref. 10, and Ref. 287.
- (u) See Figure 9 in Ref. 13.
- (v) The range of values noted in WCAP-8183, Revisions 1-6, for individual plants in all four quarters of the given year is shown. For an idea of the average annual fuel reliability (or defect level), see Figure 9 in Ref. 13.
- (w)

Appendix C

List of Domestic Events by Reactor

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

Reactor	Fuel Oriented	Fuel Handling Oriented	Control Rod Oriented	Core/Coolant Oriented	Personnel or Equipment Oriented	Miscellaneous	Total
Arnold			5.3.1.1	5.4.5.1			2
Big Rock Point							0
Browns Ferry-1							0
Browns Ferry-2							0
Browns Ferry-3							0
Brunswick-1							0
Brunswick-2							1
Clinton-1			5.3.2.2				0
Cooper Station							0
Dresden-2							0
Dresden-3							0
Enrico Fermi-2						5.6.3.2	2
FitzPatrick				5.4.1.1			2
Grand Gulf-1		5.2.2.4	5.3.1.5				0
Hatch-1							0
Hatch-2							0
Hope Creek							0
LaSalle County-1							0
LaSalle County-2							1
Limerick-1				5.4.9.2			

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Table C.1 (cont.)

Reactor	Fuel Oriented	Fuel Handling Oriented	Control Rod Oriented	Core/Coolant Oriented	Personnel or Equipment Oriented	Miscellaneous	Total
Limerick-2							0
Millstone-1						5.6.3.3	1
Monticello							0
Nine Mile Point-1		5.2.1.1			5.5.1.2		2
Nine Mile Point-2							0
Oyster Creek			5.3.1.7	5.4.1.2, 5.4.1.3			3
Peach Bottom-2							0
Peach Bottom-3							0
Perry-1		5.2.3.1	5.3.1.8, 5.3.1.9, 5.3.7.4	5.4.1.5, 5.4.4.2			6
Pilgrim-1							0
Quad Cities-1							0
Quad Cities-2			5.3.3.3				1
Riverbend-1			5.3.1.10				1
Susquehanna-1		5.2.1.3					1
Susquehanna-2				5.4.1.6			1
Vermont Yankee	5.1.1.4						1
Washington Nuclear-2							0
Total	1	4	9	8	1	2	25

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

	Reactor	Fuel Oriented	Fuel Handling Oriented	Control Rod Oriented	Core/Coolant Oriented	Personnel or Equipment Oriented	Miscellaneous	Total
C3 NUREG/CR-3950	Arkansas Nuclear-1		5.2.6.1					1
	Arkansas Nuclear-2			5.3.4.1				1
	Beaver Valley-1							0
	Beaver Valley-2							0
	Braidwood-1							1
	Braidwood-2			5.3.1.2				1
	Byron-1			5.3.4.3				1
	Byron-2		5.2.2.3					3
	Callaway-1			5.3.1.3, 5.3.3.1	5.4.9.1			1
	Calvert Cliffs-1		5.2.12.1					0
	Calvert Cliffs-2							1
	Catawba-1			5.3.3.2				0
	Catawba-2							1
	Comanche-1			5.3.4.4				1
	Connecticut Yankee		5.2.5.1					0
	Cook-1							0
	Cook-2							0
	Crystal River-3							1
	Davis-Besse-1		5.2.6.2					1
	Diablo Canyon-1				5.4.3.1			

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

Reactor	Fuel Oriented	Fuel Handling Oriented	Control Rod Oriented	Core/Coolant Oriented	Personnel or Equipment Oriented	Miscellaneous	Total
Diablo Canyon-2		5.2.6.3					1
Farley-1							0
Farley-2							0
Fort Calhoun-1		5.2.6.4					1
Ginna							0
Harris-1				5.4.6.4			1
Indian Point-2							0
Indian Point-3		5.1.4.2					1
Kewaunee							0
Maine Yankee			5.3.5.2				1
McGuire-1		5.2.6.5					1
McGuire-2							0
Millstone-2							0
Millstone-3			5.3.5.3	5.4.2.1			2
North Anna-1		5.2.2.7, 5.2.6.6					2
North Anna-2							0
Oconee-1							0
Oconee-2							0
Oconee-3			5.3.4.5, 5.3.4.6				2
Palisades							0

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

Reactor	Fuel Oriented	Fuel Handling Oriented	Control Rod Oriented	Core/Coolant Oriented	Personnel or Equipment Oriented	Miscellaneous	Total
Palo Verde-1				5.4.1.4			1
Palo Verde-2							0
Palo Verde-3			5.3.4.7, 5.3.4.8				2
Point Beach-1			5.3.5.4	5.4.2.2			2
Point Beach-2							0
Prairie Island-1		5.2.6.7, 5.2.6.8					2
Prairie Island-2			5.3.4.9				1
Robinson-2			5.3.3.4				1
Salem-1			5.3.3.5				1
Salem-2							0
San Onofre-1		5.2.5.2					1
San Onofre-2							0
San Onofre-3		5.2.1.2	5.3.4.10	5.4.9.3			3
Sequoyah-1				5.4.3.2			1
Sequoyah-2							0
Seabrook							0
South Texas-1							0
South Texas-2							0
St. Lucie-1			5.3.4.12, 5.3.4.13				2

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

Reactor	Fuel Oriented	Fuel Handling Oriented	Control Rod Oriented	Core/Coolant Oriented	Personnel or Equipment Oriented	Miscellaneous	Total
St. Lucie-2							0
Summer							0
Surry-1							0
Surry-2			5.3.1.11, 5.3.3.6				2
Three Mile Island-1			5.3.2.3				1
Trojan							0
Turkey Point-3		5.2.4.1					1
Turkey Point-4							0
Vogtle-1			5.3.1.12		5.5.1.3		2
Vogtle-2			5.3.1.13	5.4.1.7, 5.4.1.8			3
Waterford-3			5.3.4.14				1
Wolf Creek-1							0
Yankee Rowe							0
Zion-1		5.2.1.4					1
Zion-2							0
Total	0	17	26	10	1	0	54

Table C.3 (Table 22 reprint). Total Number of Reported Domestic Events by Category in 1990

Year	Fuel Oriented ^(a)	Fuel Handling Oriented ^(b)	Control Rod Oriented ^(c)	Core/Coolant Oriented ^(d)	Personnel/Equipment Oriented ^(e)	Misc. ^(f)	Total
BWRs							
1990	1	4	9	8	1	2	25
PWRs							
1990	0	17	26	10	1	0	54
Total Events							
1990	1	21	35	18	2	2	79

a Fuel Oriented Category (section 5.1) Includes: All fuel failure mechanisms; Swelling, wear, oxidation/ corrosion of fuel cladding; Thinning of in-core instrumentation; Iodine spiking incidents; Miscellaneous or unknown fuel problems.

b Fuel Handling Category (section 5.2) Includes: Fuel handling events with dropped, broken, damaged or potential for damage to fuel; Fuel in incorrect position; Crane operations; Fuel handling procedural violations; Spent fuel pool problems (heat removal, ventilation); Dry storage; Spent fuel consolidation; Spent fuel storage issues; Fuel production plants; Miscellaneous or unknown fuel problems.

c Control Rod Category (section 5.3) Includes: Control rod system problems; Control rod operation; Control rod position indications; Control element assembly problems; Control rod swelling, wear, corrosion, cracking; Guide tube problems; Miscellaneous or unknown control rod problems.

d Core/Coolant Category (section 5.4) 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.

e Personnel/Equipment Category (section 5.5) Includes: Procedural, administrative, or communication problems; Equipment failure, installation, maintenance, manufacturing, or design problems; Human error, actions (not in error), fatigue, or training inadequacy; Unanalyzed conditions; Incorrect or non-conservative assumptions; Equipment problems due to acts of nature.

f Miscellaneous Category (section 5.5.3.1) Includes: Generic interests; Inspection technology; Hot particles; Airborne contamination.

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