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Fuel Performance Annual Report

(Period through December 1978)

M. Dean Houston

**Office of
Nuclear Reactor Regulation**

**U. S. Nuclear Regulatory
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(Period through December 1978)

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M. Dean Houston

**Division of Systems Safety
Office of Nuclear Reactor Regulation
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REA

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Abstract

This annual report, intended to be the first in a series, provides a brief description of fuel performance in commercial nuclear power plants. Brief summaries are given for the reporting period of on-site fuel surveillance programs, fuel performance problems, and changes in commercial fuel designs. The report provides many references to more detailed information and to related NRC evaluations.

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FUEL PERFORMANCE ANNUAL REPORT
(Period through December 1978)

I. INTRODUCTION

The in-reactor performance of nuclear fuel in commercial power reactors provides important feedback to the licensing process. Interest in an available summary of such information has been expressed by members of the public, the Advisory Committee on Reactor Safeguards (ACRS) and the NRC staff. This report, therefore, discusses the current status of fuel performance and is intended to be the first in a series of annual reports on the subject.

Historically, fuel performance details have been requested on operating nuclear power plants by NRC and its predecessor, the AEC, through the reporting requirements of Regulatory Guide 1.16.⁽¹⁾ However, over the years, the subject matter to be covered in these reports has changed. The 1971 version of the guide requested that a summary of fuel performance characteristics be given in the Semiannual Operating Reports, and that special topical reports be prepared to cover fuel inspection details. By 1975, only the abnormal degradation of fuel cladding and the indication of failed fuel were reportable items. This list was further reduced in 1977 to cover only abnormal degradation of fuel cladding when the requirement for an Annual Operating Report was deleted.

The reportable items noted above form the basis for the discussion of fuel performance in the NUREG series entitled "Nuclear Power Plant Operating

Experience," (2,3,4) but that series does not address surveillance results from successful operation, generic problems or design trends. Another performance report⁽⁵⁾ of a more general nature was prepared by NRC, but that report was issued only once (January 1976) and its contents are out of date.

NRC has also had a continuing interest in many of the nonreportable items as added assurance that the operating characteristics of the fuel were satisfactory. Some specific examples of such items are rod bowing, crud behavior, Zircaloy growth, ridging of the cladding due to pellet-cladding interaction (PCI), and fission gas release from the fuel pellets at high burnup. For many of these items, gross abnormal behavior can be detected by non-destructive examination. Individual commitments between the fuel vendors and NRC have been fashioned, usually after the fact, to furnish this type of information.

A uniform and continuous program to furnish NRC with fuel performance information has been a major goal. In 1974, the fuel vendors significantly changed their fuel designs to smaller diameter fuel rods; for example, Westinghouse introduced the 17x17 assembly as an improved fuel design over the previous 15x15 assembly and General Electric the 8x8 assembly to replace the 7x7 assembly. An attempt was made by NRC at that time to produce a regulatory guide that uniformly addressed a surveillance program for new fuel assembly designs. After many iterations, the guide (RG 1.119)⁽⁶⁾ was formally withdrawn due to the difficulties in defining specifics for both PWRs and BWRs and for different vendors. It was decided that such programs would be pursued as plant-specific and handled on a case-by-case basis. Commitments have been secured or are in the process of

being secured for all new fuel designs. The details of these programs will be discussed later in this report. Further, the subject of fuel surveillance has been incorporated into a recent revision of the Standard Review Plan (Section 4.2).

Many fuel surveillance programs are underway and some have furnished first-, second- and third-cycle results. It is the intent of this annual report to identify these surveillance programs, cite their references, and summarize their results during the reporting period. Generic problems also occur from time to time. Such problems for the reporting period will be summarized, and reference will be made to other documents that provide further details on these problems. Finally, some information on current fuel design and future design trends will be given in this annual report as this information is related to fuel performance.

II. STATUS OF FUEL SURVEILLANCE PROGRAMS

A. SUMMARY OF VENDOR FUEL DESIGNS

The various vendor fuel designs are presented in Table 1. It is not the intent of this report to discuss these designs in any detail; rather, the tabulated data are presented to orient the reader. Historically, the older fuel designs have fewer fuel rods per assembly and larger diameter rods. The older designs were operated at higher linear power and with higher resulting fuel temperatures.

B. NEW FUEL DESIGN COMMITMENTS

A new fuel design can be introduced into a power reactor in one of two ways: (1) as the initial full core, or (2) regionally as a reload batch. On a smaller scale, lead test assemblies (LTAs) can be the initial application of a new design, but in some instances, especially in PWRs, the LTAs have limited applicability due to non-typical configurations. PWR manufacturers, because of required spacer grid interfacing and unique control rod configurations, have chosen the full-core option while BWR manufacturers, with the individuality of totally enclosed assemblies by channel boxes, have taken the reload route. Another significant difference between the fuel assemblies in the two reactor types, in regard to details of surveillance programs, is that BWR assemblies can be readily disassembled with access to individual fuel rods while PWR assemblies cannot.

For new PWR fuel designs with significant changes, the following paragraphs describe an example of the fuel surveillance program that has been required by NRC:⁽⁷⁾

TABLE 1

TYPICAL FUEL ASSEMBLY PARAMETERS

VENDOR	B&W	B&W	C-E	C-E	WEST.	WEST.	WEST.	EXXON	EXXON	GE	GE	GE
FUEL ROD ARRAY	15X15	17X17	14X14	16X16	14X14	15X15	17X17	15X15	8X8	7X7	8X8	8X8R
REACTOR TYPE	PWR	PWR	PWR	PWR	PWR	PWR	PWR	PWR	BWR	BWR	BWR	BWR
ASSEMBLIES PER CORE	177	205	217	177	121	193	193	193	560	764	560	560
FUEL ROD LOCATIONS PER ASSEMBLY	225	289	196	256	196	225	289	225	64	49	64	64
FUEL RODS PER ASSEMBLY	208	264	176	236	179	204	264	204	60	49	63	62
EMPTY LOCATIONS PER ASSEMBLY	17	25	5	5	17	21	25	21	4	NONE	1	2
ROD PITCH MM (INCH)	14.4 (.568)	12.7 (.501)	14.7 (.580)	12.9 (.5063)	14.1 (.556)	14.3 (.563)	12.6 (.496)	14.3 (.563)	16.3 (.642)	18.7 (.738)	16.3 (.640)	16.3 (.640)
SYSTEM PRESSURE MPa (PSIA)	15.2 (2200)	15.5 (2250)	15.5 (2250)	15.5 (2250)	15.5 (2250)	15.5 (2250)	15.5 (2250)	15.5 (2250)	7.14 (1035)	7.14 (1035)	7.14 (1035)	7.14 (1035)
CORE AVERAGE POWER DENSITY KW/LITER	90.0	101.6	78.5	96.4	95.6	98.1	104.7	98.1	40.57	50.732	50.51	49.15
AVERAGE LHGR KW/M (KW/FT)	20.0 (6.105)	17.8 (5.43)	20.0 (6.09)	17.5 (5.34)	20.3 (6.20)	22.0 (6.70)	17.8 (5.44)	22.0 (6.70)	15.2 (4.63)	23.1 (7.049)	17.9 (5.45)	17.7 (5.38)
AXIAL PEAK IN AN AVERAGE ROD KW/M (KW/FT)	24.00 (7.33)	21.36 (6.52)	24.00 (7.31)	21.00 (6.41)	24.36 (7.44)	26.40 (8.04)	21.36 (6.53)	26.40 (8.04)	18.24 (6.02)	27.72 (9.16)	21.48 (7.09)	21.24 (6.99)
MAX. PEAK LHGR KW/M (KW/FT)	62.4 (19.03)	48.4 (14.74)	53.5 (16.3)	42.7 (13.0)	56.8 (17.3)	61.7 (18.8)	44.6 (13.6)	51.9 (15.83)	47.6 (14.5)	60.2 (18.35)	44.0 (13.4)	44.0 (13.4)
MAX. FUEL TEMPERATURE C (F)	2300 (4170)	2020 (3670)	2140 (3890)	1880 (3420)	2260 (4100)	2340 (4250)	1870 (3400)	2200 (3997)	2040 (3700)	2440 (4430)	1830 (3325)	1890 (3435)
CORE AVERAGE ENRICHMENT WT % U-235	2.57	2.96	2.35	2.36	2.90	2.80	2.60	3.02	2.65	2.19	1.80	1.99

TABLE 1 (Continued)

VENDOR	B&W	B&W	C-E	C-E	WEST.	WEST.	WEST.	EXXON	EXXON	GE	GE	GE
FUEL ROD ARRAY	15X15	17X17	14X14	16X16	14X14	15X15	17X17	15X15	8X8	7X7	8X8	8X8R
MAX. LOCAL EXPOSURE RMD/KG-METAL	55	55	50	55	50	50	50	47.5	35	40	40	45
CLADDING MATERIAL	ZIRC-4	ZIRC-4	ZIRC-4	ZIRC-4	ZIRC-4	ZIRC-4	ZIRC-4	ZIRC-4	ZIRC-2	ZIRC-2	ZIRC-2	ZIRC-2
FUEL ROD LENGTH M (INCH)	3.89 (153.13)	3.86 (152.13)	3.71 (145.9)	4.09 (161.02)	3.87 (152.36)	3.80 (149.7)	3.85 (151.6)	3.86 (152.0)	3.99 (156.92)	4.09 (161.1)	4.09 (161.1)	4.20 (165.4)
ACTIVE FUEL HEIGHT M (INCH)	3.66 (144)	3.63 (143)	3.47 (136.7)	3.81 (150)	3.66 (144)	3.66 (144)	3.65 (143.7)	3.66 (144)	3.66 (144)	3.66 (144)	3.71 (146)	3.81 (150)
PLENUM LENGTH M (INCH)	0.29 (11.27)	0.24 (9.62)	0.22 (8.6)	0.25 (10.0)	0.18 (6.99)	0.21 (8.2)	0.16 (6.3)	0.17 (6.8)	0.27 (10.63)	0.41 (16.0)	0.36 (14.0)	0.25 (10.0)
FUEL ROD O.D. MM (INCH)	10.92 (.430)	9.63 (.379)	11.18 (.440)	9.70 (.382)	10.72 (.422)	10.72 (.422)	9.50 (.374)	10.77 (.424)	12.74 (.5015)	14.30 (.563)	12.52 (.493)	12.27 (.483)
CLADDING I.D. MM (INCH)	9.58 (.377)	8.43 (.332)	9.86 (.388)	8.43 (.332)	9.48 (.3734)	9.48 (.3734)	8.36 (.329)	9.25 (.364)	10.91 (.4295)	12.68 (.499)	10.80 (.425)	10.64 (.419)
CLADDING THICKNESS MM (INCH)	.673 (.0265)	.597 (.0235)	.660 (.026)	.635 (.025)	.617 (.0243)	.617 (.0243)	.572 (.0225)	.762 (.030)	.914 (.036)	.813 (.032)	.864 (.034)	.813 (.032)
DIAMETRAL GAP MICRON (MIL)	178 (7.0)	203 (8.0)	216 (8.5)	178 (7.0)	190 (7.5)	190 (7.5)	165 (6.5)	190 (7.5)	254 (10.0)	305 (12.0)	229 (9.0)	229 (9.0)
FUEL PELLET DIAMETER MM (INCH)	9.40 (.370)	8.23 (.324)	9.64 (.3795)	8.26 (.325)	9.29 (.3659)	9.29 (.3659)	8.19 (.3225)	9.06 (.3565)	10.66 (.4195)	12.37 (.487)	10.57 (.416)	10.41 (.410)
FUEL PELLET LENGTH MM (INCH)	17.78 (.700)	9.53 (.375)	16.51 (.650)	9.91 (.390)	15.24 (.600)	15.24 (.600)	13.46 (.530)	6.93 (.273)	8.13 (.320)	12.70 (.500)	10.67 (.420)	10.41 (.410)
FUEL PELLET DENSITY % T.D.	94	94	94.75	95	94	95	95	94	95	95	95	95

17X17 Fuel Surveillance Program

To provide verification of the reliable performance of the 17x17 fuel assemblies, a supplemental fuel surveillance program will be conducted at Plant A and B. The program will consist of a visual inspection of all the peripheral rods in the initial core fuel assemblies as they are discharged into the spent fuel pool. Approximately one-third of the initial core fuel assemblies will be inspected during each of the first three refueling periods. The visual inspection will include observations for cladding defects, fretting, rod bowing, corrosion, crud deposition and geometric distortion.

The initial core loading will contain two precharacterized fuel assemblies in each of the three fuel zones of the core. Precharacterization will establish baseline data that could be used to facilitate the evaluation of fuel performance, dimensional changes, or any anomalies that might be evident during the visual examination. If any anomalies are detected during the visual examination, further investigation will be performed. Depending on the nature of the observed condition, this further examination could include appropriate surface, dimensional, or gamma inspections of the fuel assemblies. If the fuel assembly design enables reconstitution, individual fuel rods may also be examined. Under unusual circumstances, destructive examination of a fuel assembly or individual fuel rods may be required, but this would not be accomplished on site or within the time of the refueling outage. The NRC will be advised of the normal refueling schedule, and will be notified at least ten days in advance of any planned supplemental fuel surveillance inspections so that it may observe the inspections and the resultant evaluations of the fuel assembly performance. Following the inspection of a significant number of assemblies and prior to reactor re-startup, an oral report of the results will be made to NRC. Within 30 days of completion of the total inspection, a written report will be submitted to NRC.

The detailed program above has been discussed with the utilities and PWR vendors. Definite commitments for this program have been secured from Westinghouse on two of the following plants (Trojan, Beaver Valley, Salem 1, Farley) with 17x17 fuel and Combustion Engineering for one plant (Arkansas 2) with 16x16 arrays. The next C-E 16x16 plant (San Onofre 2) is scheduled for late 1980. Plants that use the Babcock & Wilcox 17x17 design are not scheduled to operate until 1982.

For the BWRs, the new fuel designs have been applied in reload batches. Surveillance programs involve a small number of fuel assemblies (typically 2 to 4) that contain highly precharacterized fuel rods in each of 3 or 4 reactor cores. The programs include assembly and individual rod examination.

C. FUEL VENDOR PROGRAMS

The major fuel surveillance programs for each individual fuel vendor are listed in Table 2 and their results to date are briefly discussed below. Most of these programs utilize both non-destructive and destructive techniques. The non-destructive tests usually include visual examination, gross gamma scanning, crud sampling and dimensional measurements. The destructive tests include burnup and fission gas analysis, pellet density determination, metallography and cladding mechanical properties. For the specific details of these programs, the reader is directed to the noted references.

Many of these programs are jointly sponsored with participation by the fuel vendor, a utility, groups of utilities and the Electric Power Research Institute (EPRI). Many contractual arrangements have not been finalized for the complete program. The listing of a program under a vendor's heading should not be construed to imply that the individual organization is the sole sponsor as EPRI is heavily involved in many of these programs. The projected schedules or other details as listed reflect our current understanding of the vendor's effort and are subject to change for many reasons. The unanticipated and prolonged shutdown of the Trojan plant after just one cycle of operation is an appropriate example of how program schedules are changed. The vendors are listed in alphabetical order.

Table 2

Major Fuel Performance Programs
Status Through 1978

Vendor	Fuel Type	Power Plant	Planned Burnup or Operating Cycle	Scheduled Completion	Interim Inspections to Date
Babcock and Wilcox	15x15	Oconee 1	3 cycles	1978	Completed
	15x15	Oconee 2	3 cycles	1979	2
	17x17(LTA)	Oconee 2	3 cycles	1980	1
Combustion Engineering	14x14	Calvert Cliffs 1	3 cycles	1979	2
	16x16	ANO-2	3 cycles	1981	None
Exxon	14x14	Ginna	3 cycles	1978	2
	15x15	H.B. Robinson	3 cycles	1979	1
	7x7	Oyster Creek	25,000 MWd/tU	1977	Completed
	8x8	Oyster Creek	3 cycles	1980	1
General Electric	8x8	Monticello	5 cycles	1980	3
	8x8	Quad Cities 1	5 cycles	1981	2
	8x8(R)	Vermont Yankee	5 cycles	1982	1
	8x8(R)	Peach Bottom 2	5 cycles	1982	2
	8x8(I)	Peach Bottom 2	5 cycles	1983	1
	8x8(P)	Peach Bottom 3	5 cycles	1983	None
Westinghouse	14x14	Point Beach 1	3 cycles	1978	Completed
	15x15	Zion 1	3 cycles	1978	2
	17x17(LTA)	Surry 1	2 cycles	1977	Completed
	17x17(LTA)	Surry 2	3 cycles	1979	2

1. Babcock and Wilcox

Fuel performance results for the B&W Mark-B (15x15) fuel design have been presented⁽⁸⁾ after two cycles of reactor operation in Oconee with peak assembly burnups of 26,000 MWd/tU. A low defect level (0.01 to 0.1%) has been indicated based on primary coolant activity. Only two actual fuel rod defects have been visually observed. Based on these results, the fuel appears to be capable of operating for its intended lifetime in a satisfactory manner.

Preliminary results for the lead test assemblies of the Mark-C (17x17) fuel design have been presented⁽⁹⁾ after one cycle of reactor operation. Visual examination showed no evidence of defects. Other observations indicated that the fuel was performing in an expected manner.

2. Combustion Engineering

The C-E 14x14 fuel design has been intensively examined. Early failures in the fuel rods (Maine-Yankee, 1974)⁽¹⁰⁾ and poison rods (St. Lucie 1, 1976)⁽¹¹⁾ required a thorough analysis of the causes so that effective measures could be applied to overcome these defects in current and future core loadings. The Calvert Cliffs 1 program utilizes the improved C-E fuel and poison pellets and thus should establish the basis for the evaluation of their 14x14 fuel performance. The inspection program at Calvert Cliffs has shown⁽¹²⁾ that overall fuel rod performance to maximum burnups of 29,000 MWd/tU was fault-free and satisfactory and that anomolous behavior was not observed after two operating cycles.

By mid-1978, the C-E irradiation program for the new 16x16 fuel at ANO-2 was just underway. The utility has committed to perform a 100% visual examination of this fuel according to the program outlined in Section II.B above⁽¹³⁾.

3. Exxon

Exxon Nuclear (ENC) is the sole domestic supplier of both BWR and PWR fuel. Since they do not design and build nuclear power plants, ENC fuel has always been introduced in reload batches. The fuel design of the reloads has been very similar to the original fuel design that is being replaced. However, Exxon does not merely copy the previous design but instead, provides their own design and performs the analysis to meet ENC criteria.

As of November 1978, Exxon had a total of 1,342 fuel assemblies operating in nine domestic reactors -- four BWRs and five PWRs.* Also, a total of 204 fuel assemblies in BWRs and PWRs have achieved design burnups and have been discharged to spent fuel pools. Peak assembly burnups on the order of 31,500 MWd/tU have been achieved. A few fuel rod failures (7) have been observed in the old 7x7 BWR fuel. There is some likelihood that a few fuel rods (1 to 2) have failed in PWRs based on coolant activity levels. A summary of ENC fuel performance has been reported in annual reports.^(14,15,16)

4. General Electric

The current fuel surveillance programs at General Electric have been concerned mainly with the 8x8 and 8x8(R) fuel designs since the previous 7x7 designs

* - Private communication from G. A. Sofer (ENC), December 6, 1978.

will be completely phased out toward the end of 1980. The 8x8(R) design is an improved version of the 8x8 and differs only slightly from the original 8x8 design. The differences are mainly in the fuel rod diameter (0.483" vs 0.493"), cladding thickness (32 vs 34 mils) and number of non-fueled water rods (2 vs 1). The 8x8(I) and 8x8(P) designs are slight variations of the 8x8(R) design. In the 8x8(I), natural UO_2 segments are placed on both ends of the fuel column. The 8x8(P) is a prepressurized design with a helium pressure of 3 atmospheres.

Fuel performance results have been obtained on highly precharacterized individual fuel rods (11 per assembly) and fuel assemblies. These results obtained after two reactor cycles in Monticello^(17,18) and Quad Cities 1^(19,20) (average assembly burnup up to 12,500 MWd/tU) show that the 8x8 fuel is performing in a fault-free and satisfactory manner. The first cycle results for 8x8(R) fuel in Peach Bottom 2 are also fault-free.⁽²¹⁾ Other fuel performance data for General Electric have been presented in operating experience reports^(22,23) that have been issued on two year intervals.

5. Westinghouse

Fuel performance for Westinghouse Zircaloy-clad fuel in both domestic and foreign reactors is discussed in their annual report⁽²⁴⁾ on operating experience. The defect level for 14x14, 15x15 and 17x17 production fuel is inferred from the coolant activity level. For the year 1977, the inferred defect levels ranged from 0.0001 to 0.062 percent, well below the 1.0 percent design basis value used by Westinghouse to design the primary coolant purification and waste handling systems. Region average burnups to 37,760 MWd/tU are reported while the average burnup of discharged fuel is 22,120 MWd/tU.

In addition to their annual report, detailed fuel surveillance programs are in progress or have been reported for all configurations of Westinghouse standard fuel plus 17x17 lead test assemblies (LTA). Forty-three individual 14x14 fuel rods (rod average burnups from 15,500 to 29,400 Mwd/tU) in one fuel assembly in Point Beach 1 were examined⁽²⁵⁾ for incipient cladding defects. Although these rods had been subjected to a severe power ramp and were suspected of being leakers, exhaustive non-destructive and destructive examinations showed no incipient or real failures. The performance data base for 15x15 fuel is being developed in a program at Zion 1. The reported results⁽²⁶⁾ after one cycle of reactor operation showed that the assemblies and fuel rods were in excellent condition and that no anomalies were observed except for a high frequency of small circular crud deposits on some peripheral rods. These were stated to have no adverse effect upon the fuel rod performance.

Westinghouse introduced the 17x17 fuel design⁽²⁷⁾ in lead test assemblies (LTAs) in Surry 1 and 2. The LTAs were slightly different than the intended design for full core application. They had a longer span length between spacer grids (26.2" vs 20.5") and fewer spacer grids (7 vs 8). Their performance to date, after one and two cycles of reactor operation with assembly burnups to 16,330 Mwd/tU, has been satisfactorily with no observed failures or anomalies.^(24,28,29,30) The first full core of 17x17 fuel began operation in the Trojan plant in December, 1975 and was inspected after one cycle of operation (assembly average burnups of 7,200 to 17,800 Mwd/tU) in April, 1978. The preliminary results⁽³¹⁾ from the visual inspection showed the fuel assemblies to be in excellent overall physical condition with negligible crud deposits.

6. Electric Power Research Institute (EPRI)

As stated previously, the Electric Power Research Institute (EPRI) is heavily involved in the fuel surveillance programs that are based on standard product line fuel. These surveillance projects are but a portion of a larger, overall program directed by EPRI entitled, "The Light Water Reactor (LWR) Fuel Performance Program."^(32,33) The stated goals of this program are to develop a comprehensive fuel performance data base with verified predictive models and codes to improve fuel rod reliability and, hence, increase plant availability. The LWR program has twenty-one active projects that range in scope from laboratory tests of Zircaloy cladding behavior to large-scale testing of prototypic fuel assemblies. The reference planning support document fully describes the details of these projects.⁽³²⁾

III. PROBLEM AREAS OBSERVED DURING 1977 AND 1978

Fuel problems in the past have been caused by the following factors: fuel densification, hydriding of the cladding, fretting and wear of the cladding due to waterjetting, abnormal in-reactor growth of Zircaloy cladding, and pellet/cladding interaction (PCI). Each problem has received attention after the observance of failures and each has been remedied to eliminate or reduce fuel failures in current operating cycles. Although current fuel designs and operating procedures result in fewer PCI failures, PCI remains a concern for economic and safety reasons and PCI programs are continuing. The subject of prior failures and the actions by the industry to deal with them have been discussed in the vendor operating experience reports and other topical reports and will not be repeated here. The effectiveness of these actions to reduce fuel rod failures is seen in the low failure rates given in Section II.C above and an overall LWR fuel rod failure rate in the range from 0.05 to 0.1% reported in 1977⁽³⁴⁾. With the exception of the unique failure of stainless steel cladding in the La Crosse BWR, as will be discussed later, no other events involving fuel failures of a significant nature were reported during the period. However, failures and problem areas associated with other components in the fuel assembly have been experienced and these will be discussed in the following sections.

A. BURNABLE POISON ROD FAILURES

Failure of Combustion Engineering burnable poison rods was first noted in St. Lucie 1 in mid-1976.⁽¹¹⁾ The burnable poison rods were fabricated from $\text{Al}_2\text{O}_3\text{-B}_4\text{C}$ pellets and Zircaloy cladding. The failure mechanism was

hydriding of the cladding caused by excessive moisture in the pellets. Manufacturing procedures were modified to lower the moisture content and the failures have been effectively eliminated.

Burnable poison rods with the higher moisture content were already in a number of operating plants when the St. Lucie 1 failures occurred. C-E performed a visual inspection of the fuel at these plants during 1977 at their normal refueling outage. Burnable poison rod failures were noted at all of the plants, but the frequency of failure was less than that for St. Lucie 1. Circumstances surrounding these failures and modifications to eliminate the problem have been reviewed by NRC.⁽³⁵⁾ Modification to the fabrication process (principally tightening of moisture specifications) should eliminate this problem from current and future core loadings.

B. GUIDE TUBE WEAR

During the visual inspection of C-E fuel assemblies for burnable poison rod failures, significant indications of wear damage in the control element (CEA) guide tubes were discovered.⁽³⁶⁾ The wear occurred by the rubbing of the Inconel 625 control rod against the softer Zircaloy-4 guide tube and at the elevation where the control rods were in their fully withdrawn "parked" position. The cause for the problem was flow induced vibration of the control rods.

For plants in an operating cycle, an interim solution for continued operation was to insert the CEAs by a small amount, 1 to 3 inches, thereby moving them off the worn "parked" location. At refueling outages for all C-E plants with

that guide tube design (Palisades the only exception), guide tubes were inspected by visual and eddy current techniques and the results reported to NRC.^(37,38,39,40) To date, significant wear in CEA guide tubes has been observed at Millstone 2, Calvert Cliffs 1, St. Lucie 1 and Maine-Yankee.

Combustion Engineering proposed^(38,39) a sleeving process to modify the guide tubes in the partially irradiated fuel assemblies and in new reload or initial core assemblies for operating plants. NRC approved sleeving⁽⁴¹⁾ as an interim solution to the problem while C-E pursued other modifications for a long term or permanent solution. C-E will proof test these latter modifications in operating plants, currently in Arkansas Nuclear One-2 and planned for Calvert Cliffs 2.^(42,43) NRC has conducted a preliminary review⁽⁴⁴⁾ of the potential for wear of the guide tubes in Westinghouse and Babcock & Wilcox designs. While these designs are susceptible to wear, it was tentatively concluded that the magnitude of wear may be insufficient to cause any problem. NRC will continue to monitor these programs.

C. STAINLESS STEEL CLADDING DEGRADATION

The LaCrosse Boiling Water Reactor (LACBWR) is a one-of-a-kind nuclear plant built by Allis-Chalmers and has a thermal rating of 165 MW. The LACBWR core contains 72 fuel assemblies, each with a 10x10 fuel rod configuration. Each fuel rod consists of UO₂ fuel pellets and Type 348 SS cladding. LACBWR is the only domestic BWR that operates with stainless steel cladding.

During Cycle 4 operation, the off-gas activity increased significantly and the reactor power had to be reduced to limit the level of activity. The operating cycle was terminated in May, 1977 and the fuel was visibly inspected.⁽⁴⁵⁾

Sections of the fuel rods, the longest being 6.5 inches, were observed to be loose in or missing from the fuel assemblies. A total of 53.9 equivalent fuel rod inches of fuel rod material was displaced from three fuel assemblies. Approximately 30% of the displaced material is unaccounted for after an intensive recovery program.

The mechanism for initial fuel rod failure is believed to be caused by oxygen-assisted intergranular stress corrosion cracking of the stainless steel cladding that was accelerated by relatively high pellet/cladding interaction during power escalation. The failed rods had both circumferential and longitudinal cracks and were subjected to flow-induced vibration. This vibration aggravated the defects, and loose section of the fuel rods resulted. NRC has performed a detailed analysis of these failures and imposed lowered burnup limits to keep the number of future failures low.⁽⁴⁶⁾

The problem discussed above is unique to LACBWR and remains as a continuing concern for that reactor.

D. LOOSE BURNABLE POISON AND ORIFICE ROD ASSEMBLIES

In April 1978, Florida Power Corporation⁽⁴⁷⁾ informed NRC of a fuel assembly related problem at the Crystal River plant (B&W design) that was caused by the ejection of two burnable poison rod assemblies (BPRAs) from the core while operating at power. The BPRAs are similar to control rod assemblies with individual burnable poison rods attached to a spider hub and with the rods occupying control rod guide tube positions in the fuel assemblies. Unlike the control rod assemblies that are moveable and attached to the end of a drive mechanism, the BPRAs are held in a fixed position by means of a latching

mechanism located on the upper end fitting of the fuel assembly. The ejected BPRAs were carried out of the reactor vessel through the hot leg and into the steam generator. Here, the vibratory motion of the spider hub (~10 lbs.) caused significant damage to the steam generator tube ends. B&W determined⁽⁴⁸⁾ that the ejection of BPRAs from the core resulted from fretting wear in the holddown latching mechanism. The fretting wear occurred when the BPRAs were levitated during four-pump operation. As an interim solution, B&W recommended continued operation of their plants at reduced flow on three pumps. For the long term solution, B&W is developing additional hardware to positively lock and hold these BPRAs in place.

A similar component in the core is the Orifice Rod Assembly (ORA). The ORAs consist of short stubby rods attached to a spider hub and are considerably lighter than BPRAs (15 lbs. vs 45 lbs.). ORAs are installed in those fuel assemblies that do not have BPRAs or control rods and serve to reduce the bypass coolant flow through the empty guide tubes. They are held in the B&W fuel assemblies by means of the same latching mechanism as the BPRAs, but, because of the weight difference, the ORAs are firmly held in place during full-flow conditions. At reduced flow (two pump operation), however, the ORAs may be levitated. Although no instances of loose or ejected ORAs had been observed, the similarities to the BPRAs justified an examination of their latching mechanisms after reactor operation. Some wear was observed in that mechanism. B&W recommended⁽⁴⁸⁾ that two-pump operation be minimized. Alternatively, the ORAs are being removed from some reactor cores or are continuing in operation with additional hardware designed to hold them in place. NRC has approved⁽⁴⁹⁾ these changes and will continue to monitor plant operation with these modifications.

E. SPACER GRID DEFORMATION

In May 1977, Westinghouse reported^(50,51) that spacer grid deformations had been observed in some 17x17 fuel assemblies at one foreign reactor and Farley upon receipt of these assemblies at the reactor site. Other domestic plants that had received 17x17 assemblies were North Anna 1 and Diablo Canyon 1 and 2. NRC's Office of Inspection and Enforcement participated in the Westinghouse special fuel inspections at these plants.^(52,53) In these fuel inspections, two types of grid deformation were observed:

- (1) parallelogramming of the outer two rows of cells and
- (2) buckling of grid straps in the outer row.

Most of the grid deformation was so slight that the fuel assembly dimensional characteristics were still within the original specifications. Where significant deformation occurred, the fuel bundles were returned to the fabrication plant and the damaged spacer grids were replaced. The grid deformation was traceable to loads imposed during loading and shipping by the shipping container clamping frame, originally built for 15x15 fuel.

Westinghouse performed⁽⁵¹⁾ a safety analysis for those fuel assemblies with the slightly deformed grids. Their analysis included the deformation effect upon rod bow, DNBR, local power peaking and LOCA. NRC agreed⁽⁵⁴⁾ with the Westinghouse conclusions that the slight deformations raised no safety concerns.

IV. FUEL DESIGN CHANGES

Fuel design concepts for both PWRs and BWRs remained fairly stable for the period January 1977 through December 1978. The only design change proposed during the period was associated with prepressurization of BWR fuel rods. The concept of fuel rod prepressurization is not new as PWR fuel rods have for years been pressurized to levels in excess of 300 psi. General Electric⁽⁵⁵⁾ and Exxon⁽⁵⁶⁾ have proposed a slight prepressurization of BWR fuel rods (in the range from 45 to 65 psi).

Prepressurization of BWR fuel is desirable because it improves pellet-to-cladding gap conductance and thus results in lower fuel temperatures, reduced UO_2 thermal expansion and reduced fission gas release. This reduction in fuel temperatures is equivalent to a reduction in linear power (kW/ft) and thus expected to result in fuel rods that are more resistant to the pellet/cladding interaction (PCI) failure mechanism. Having a more PCI resistant fuel rod design will yield improved fuel reliability or alternatively increased operating flexibility. The concept of BWR fuel rod prepressurization has been reviewed and approved by NRC.⁽⁵⁷⁾

Lead test assemblies with prepressurized BWR fuel have been proposed by both GE and Exxon. The GE program with one prepressurized test assembly (PTA) began operation in Peach Bottom-3⁽⁵⁵⁾ in April 1977. The Exxon program will begin at some future date. NRC will continue to monitor these programs.

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