WESTINGHOUSE CLASS 3

WCAP-8183
Rev. 9

OPERATIONAL EXPERIENCE WITH WESTINGHOUSE CORES
(Up to December 31, 1979)

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

SECTION 1

INTRODUCTION

This revision to WCAP-8183 provides the cumulative operating experience of Westinghouse Zircaloy-clad fuel rods and other associated core components up to December 31, 1979. This report, revised annually, is a supporting document to safety analysis reports for licensing purposes. The NRC safety analysis report requirements[1] for evaluating fuel and core component failure and burnup experience are met by this report.

Section 2 summarizes Westinghouse experience with Zircaloy-clad fuel. Section 3 presents a fuel experience overview, including fuel performance for generic problems common to a number of plants, along with solutions to these problems. Section 4 discusses and evaluates other core component experience.

The data on which the overview is based are given in Section 5.

SECTION 2

SUMMARY OF OPERATIONAL EXPERIENCE IN WESTINGHOUSE CORES WITH ZIRCALOY-CLAD FUEL

Westinghouse has had considerable experience with Zircaloy-clad fuel since its introduction in the Jose Cabrera plant in June 1968.

As of December 31, 1979, there were 31 commercial PWRs that have used Westinghouse-supplied Zircaloy-clad fuel. A total of 771,123 fuel rods are operating in these reactors. The addition of previously discharged fuel brings the total number of Westinghouse Zircaloy-clad fuel rods to 1,769,710. This represents 3,697 MTU and a thermal energy production of 73,087 GWD. The average burnup of the discharged fuel is 23,961 MWD/MTU and the average burnup of all fuel is 20,023 MWD/MTU. (The burnup for all fuel is lower because of the new (unburned) fuel included in that number). The region average burnups given in Table 5-1 reflect the total region average burnup, thus active fuel and discharged fuel in the same region are averaged together. This method of burnup averaging does not highlight the individual assemblies with the highest burnups in the region. During this reporting period 16 plants have been refueled.

A number of fuel regions have been discharged at region average burnups in the range of 30,000-37,100 MWD/MTU (see Table 5-6). The peak region average burnup data in Table 5-6 reflects an average of the high burnup assemblies in a region rather than an average of the total number of assemblies in the region. The highest burnups of individual discharged fuel assemblies have been in the range of 38,000-39,900 MWD/MTU. Four high burnup demonstration assemblies, being irradiated in the Zion Unit 2 Cycle 4 core to 47000 MWD/MTU, are planned to have fuel assembly burnups extended to a maximum of about 55,000 MWD/MTU. A significant amount of high burnup experience data has been evaluated for Westinghouse fuel assemblies.(2)

The in-pile performance of Westinghouse fuel has been excellent. In some cases, coolant activity suggests the presence of a small number of cladding defects. No cause has been assigned to the apparent defects, nor is one necessary since low levels of cladding defects are expected to be continuously present. The frequency is low enough that it provides no restriction on plant operation.

Nuclear reactors now operating with Zircaloy-clad fuel have not experienced availability limitations due to fuel defects, thus aiding these plants to continue to exhibit high reactor and plant availability. Problems have been successfully and promptly corrected.

SECTION 3

FUEL EXPERIENCE OVERVIEW

3-1 BURNUP EXPERIENCE

Table 5-1[3] contains a list, in chronological order of plant startup, of the plants in which extensive operating experience has been obtained with Westinghouse Zircaloy-clad rods in open lattice cores. Taken as an aggregate, the table indicates the extent of this experience. On the basis of the total number of fuel rods, and their collective burnup, these data provide excellent assurance of the reliability of Westinghouse fuel.

Figure 5-1 contains a graphic representation of the data in Table 5-1 (also includes burnup performance figures from several previous years). The upper portion of the figure represents the fuel assemblies in service and the lower portion of the graph represents discharged assemblies.

The practice of evaluating fuel performance in terms of coolant activity level is continued. The iodine-131 activity in the coolant is reported in terms of a percentage of the coolant design basis activity in Table 5-2 for all plants listed in Table 5-1. 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. The activities listed for 1975 and prior years, in which the inferred defect basis had been used, were approximated by multiplying the numbers for the percent defect level reported previously by 100. In all cases, the activity levels are below the coolant activity levels allowed in the technical specifications for the plants mentioned in this report.

Figures 5-2 and 5-3 show the yearly change of coolant activity distributions for Westinghouse-fueled plants. The trend toward lower coolant activities over the periods reported is apparent. Figure 5-2, which employs a logarithmic scale, emphasizes the trend toward lower activities as the number of plants and amount of operating experience increases. Again, these data, combined with burnup experience of Table 5-1 and Figure 5-1, provide substantial proof of the reliability and performance of Westinghouse Zircaloy-clad fuel.

The distinctions among the regions of fuel in Table 5-1 are useful in understanding some of the generic fuel concerns presented in paragraph 3-2.

3-2 GENERIC FUEL CONCERNS

The main concerns investigated here are moisture and hydriding, fuel densification, fretting in peripheral assemblies, ramp rate effects on fuel performance, and fuel rod bowing.

3-3 Moisture and Hydriding

Early in the operation of both the Beznau Unit 1 and the R. E. Ginna units, coolant chemistry indicated the presence of cladding defects. This situation, which occurred after about 1000 effective full-power hours of operation, increased over a period of about 1 month. Following the period of the cladding defect occurrence, coolant activity levels became essentially stabilized, indicating the cessation of the cladding defect process. In both cases, normal reactor operation continued until the scheduled refueling.

At the refueling of each plant, visual examination indicated that defected rods were confined to Region 3 in both cores. The visual appearance of the affected rods indicated that local hydriding had occurred, and led to subsequent cladding defects. The cause of this local hydriding was confirmed to be moisture contained in the as-built fuel pellets and released to the cladding during operation.

Hydride-initiated cladding breach was confined to only two regions, in which the fuel was characterized by excessive moisture associated with low pellet density. However, not all low-density regions showed evidence of hydriding.

Since 1970, more than 30 additional Westinghouse PWR cores, as well as reload regions supplied to the Beznau and R. E. Ginna plants, have accumulated many cycles of operating experience with fuel built to the new moisture specification. Activity levels in all Westinghouse plants have decreased more than an order of magnitude compared to plants operating in 1970, indicating the success of this new specification. All plants including the 17 x 17 plants are currently operating at or well below a coolant activity level of 0.05 μCi/g of iodine-131.

3-4 Fuel Densification

Early in 1972, confirmation was obtained that clad flattening and gaps observed in the columns of fuel pellets in the cores of several pressurized water reactors were the result of densification of the fuel during operation. The implications of these gaps with regard to fuel rod integrity and reactor safety stimulated a substantial effort to understand in-pile densification at low temperatures and to define corrective action[4,5].

By the end of 1972, it was known that low-temperature densification was caused by the fission-induced resolution of fine pores in the fuel. It was determined that densification could be practically eliminated by controlling the pore size distribution and porosity of the fuel. These factors were known to be dependent upon fabrication parameters, of which the sintering conditions were the most important[6].

It should be noted that the clad flattening phenomenon was confined essentially to fuel rods with either low pressurization or no prepressurization,[7] and that its occurrence did not necessarily lead to a loss of cladding integrity.

Since 1972, appropriate controls on procedures and specifications for the product have reduced in-pile densification to insignificant levels, as shown by the reduction in size and number of gaps in the fuel columns. Also fuel rods are prepressurized with helium so that clad flattening does not occur.

3-5 Fretting in Peripheral Assemblies

During refueling of the Jose Cabrera plant in 1971, visual examination disclosed two broken fuel rods and one partly damaged rod in a peripheral assembly. In a related case, during the Point Beach Unit 1 Cycle 3/Cycle 4 refueling, a section of broken fuel rod was found on the lower core plate and traced to a discharged assembly which had been in service close to a joint in the core baffle. This assembly also had other damaged rods adjacent to the broken rod.

The cause of the damage in the case of Jose Cabrera has been identified as the leakage of high-velocity coolant cross-flow through gaps in the corner joints in the core baffle. The cross-flow caused excessive rod vibration and eventual fretting through the cladding in the grid support areas. In the case of Point Beach Unit 1, the rod was believed to have been thinned by wear caused by cross-flow-induced vibration when in a peripheral location and severed when hit by an adjacent fuel assembly during refueling operations. The baffle joints on both plants were repaired to eliminate the leakage. Inspection and repair recommendations were made to other affected plants.

During refueling of the KORI Unit 1 and Ringhals Unit 2 reactors in 1979, fuel rod failure attributable to high velocity baffle leakage were identified. These failures occurred at a different baffle joint configuration compared to the previously identified incidents at the Cabrera and Point Beach plants. KORI Unit 1 had two fuel assemblies affected, each with two rods having cladding failure. Ringhals Unit 2 showed indications of a number of cladding failures in three Westinghouse fuel assemblies and five non-Westinghouse assemblies. For operating plants peening of baffle joints may be performed during refueling, if the routine fuel inspection and operating coolant activity levels justify such action. A design modification at the baffle joints is being implemented to eliminate this type of fuel failure for plants not yet in operation.

3-6 Ramp Rate Effects on Fuel Performance

A number of indications of defects were observed during the Cycle 3 startup of Point Beach Unit 1 after the refueling shutdown. These defects have been attributed to pellet/clad interaction due to a rapid rate of reactor power increase during the startup. After an initial increase, the primary coolant activity decreased significantly during the cycle. During the entire cycle, coolant activity was well below technical specification limits and plant operation was not affected. As a result of these observations, modest startup limits were implemented in terms of rate of reactor power increase following refueling or extended (approximately 30 days) reduced power operation. These restrictions apply only during the initial startup of a reload cycle; plant operation may continue during the remainder of the cycle without any ramp rate restrictions. Therefore, load follow operation may be conducted without any limitations on ramp rate or frequency of load cycles.

Since these recommendations were implemented in January 1975, there have been 76 refuelings through 1979 without any indications of coolant activity increases. There was one exception in which startup defects did occur; however, leak testing at the end of this cycle indicated that the leaking fuel assemblies were largely non-Westinghouse demonstration assemblies.

3-7 Fuel Rod Bowing

Rod bow has been observed in a large number of fuel assemblies over the past several years, and considerable attention has been applied toward understanding both the causes and possible effects of this phenomenon.[8,9,10]

The concern associated with the bowing of fuel rods during reactor operation is that partial or complete closure of the channel between fuel rods potentially degrades the thermal-hydraulic conditions in that channel. An empirical model has been developed for predicting the extent of rod bow that will be experienced during operation. Rod bow observations have been evaluated for over 1600 fuel assemblies from approximately 55 regions of fuel at assembly burnups up to about 40,000 MWD/MTU. This base of experience represents an acceptable statistical sample around which future operating behavior may be assessed.

Departure from nucleate boiling (DNB) on fuel performance must be addressed in light of the occurrence of rod bow; this concern has received attention throughout the history of rod bow. The NRC has reviewed the data base and evaluations presented by Westinghouse and addressed the subject of rod bow via several safety evaluation reports (SERs) issued to date. The current licensing basis for operating plants accounting for the effects of rod bow are defined by an NRC interim SER(11) and additional NRC-Westinghouse correspondence(12,13) on this subject.

In October 1977 a request for a partial rod bow penalty reduction and supporting test results were submitted for NRC review. An NRC acceptance letter(13) was issued which essentially eliminated rod bow penalties for 15 x 15 (14 x 14) fuel and significantly reduced penalties for 17 x 17 fuel. A revised fuel rod bow report was submitted for NRC review in September 1979. The report develops revised rod bow correlations for current 15x15 (14x14) and 17x17 (16x16) fuel. The bow correlations are used with the NRC approved partial rod bow function in a statistical manner to evaluate rod bow DNBR effects. When approved by the NRC, additional DNBR margins would be available for Westinghouse fuel to increase operating flexibility.

In April 1979 it was observed that some Region 4 fuel assemblies discharged from the Prairie Island Unit 1 reactor exhibited a higher than normal degree of fuel rod bowing. An investigation concluded that most of the abnormal rod bow was associated with as-fabricated tubing characteristics in tubing lots made from a specific ingot of Zircaloy 4.

The Region 4 bow data was confirmed to be within the current Westinghouse licensing basis for rod bow. This data is included in the data base for the September 1979 revised rod bow report undergoing NRC review. The specifications for tubing characteristics suspected to contribute to rod bow have been tightened.

3-8 17 x 17 FUEL ASSEMBLY EXPERIENCE

Four demonstration assemblies of the 17 x 17 design were placed in Surry Units 1 and 2, two assemblies in each unit. Assembly and fuel rod visual examinations were performed for all the assemblies after one and two cycles of operation. The two assemblies in Unit 1 were discharged in October 1976 after two cycles of irradiation as planned. The two assemblies in Surry Unit 2 completed their third cycle of irradiation, and one of these assemblies is to be included for a fourth cycle of irradiation. Results of all examinations to date verify that the 17 x 17 design is performing well; the assemblies were in excellent condition and no anomalies were observed.

The Trojan Nuclear Power Plant was the first plant to achieve one cycle of operation with Westinghouse standard 17 x 17, eight grid fuel assemblies. Reactor coolant activities were normal, indicative of excellent overall fuel integrity. After one cycle of operation on-site TV visual fuel examinations and channel spacing measurements were performed. The Trojan data was generally consistent with data obtained at Surry on the 17 x 17 demonstration assemblies.

Six plants of the 17x17 design are now in commercial operation in the United States. Eight additional foreign plants of the 17x17 design built by W and W licensees are in operation in France and Japan. The coolant activities in these plants fall within the range of activities found in all 14x14 and 15x15 plants. The six domestic plants and two plants in Japan which contain W built cores are included in Figure 5.2. Trojan and Fessenheim 1 (France) should complete their second cycle of operation early in 1980. Most of the other 17x17 reactors should be in their second cycles at some time in 1980.

Two demonstration 17x17 Optimized Fuel Assemblies (OFA) were placed into the Farley Unit 1 Cycle 2 core and two into the Salem Unit 1 Cycle 2 core. The 17x17 OFA employs a slightly reduced fuel rod diameter compared to the standard 17x17 fuel rod while retaining the same fuel rod pitch. Also, for the middle six grids, the OFAs use Zircaloy grid design compared to Inconel Used for standard 17x17 fuel. These design changes result in a significant improvement in fuel efficiency by improving neutron moderation and reducing parasitic capture. In the last quarter of 1979, Farley Unit 1 reached rated power, and Salem Unit 1 was in a power escalation.

3-9 FUEL PERFORMANCE AT HIGH BURNUP

A number of fuel assemblies are now approaching burnups in excess of 40,000 MWD/MTU. Point Beach Unit 2 - Cycle 6, containing one assembly that will be discharged in March, 1980 at a burnup of about 44,000 MWD/MTU, is currently operating at a coolant activity of 0.0010 μCi/g of iodine-131. A second plant, Zion Unit 2 - Cycle 4, contains four assemblies that will attain a burnup of about 47,000 MWD/MTU in March, 1980, and this plant is currently operating at a coolant activity of 0.0015 μCi/g of iodine-131.

The activity levels in both plants have remained constant throughout their current cycles indicating that no new defects are forming.

3-10 FUEL PERFORMANCE AT HIGH POWER RATINGS

The performance of the fuel in three reactors with high average specific linear power ratings is shown in the following tabulation in terms of typical steady-state coolant activities observed (toward the end of each cycle). Since one of these reactors has operated for four cycles with no defects and the other two have operated with decreasing numbers of defects, as indicated by coolant activities, it is apparent the operation of pressurized water reactors at these high heat ratings has no measurable effect on the frequency of formation of cladding defects.

Plant	Average Specific Linear Power Rating kw/ft	Cycle	Typical Steady-State Activities, μCi/g Iodine 131	Iodine 133	Cesium 138
L	6.8	1	.00003	.0002	.001
L	6.8	2	.00004	.0004	.001
L	6.8	3	.00003	.0003	.001
L	6.8	4	.00002	.0003	-
M	6.8	1	.008	.01	.03
M	6.8	2	.0001	.0008	.002
M	6.8	3	.00005	.0005	.002
N	6.7	1	.06	.08	.1
N	6.7	2	.007	.008	.08
N	6.7	3	.008	.03	.05

3-11 OPERATION WITH DEFECTED FUEL

In recent Westinghouse-designed PWRs containing fuel manufactured to rigid specifications on moisture and pellet parameters affecting densification, coolant radioactivity levels have been maintained at low fractions of technical specification limits. Nevertheless, some defects do occur, and plants are designed to continue to operate until design burnup levels are achieved.

Significant experience has been accumulated with fuel operated for one or two reactor cycles after formation of defects. This experience, summarized in Table 5-3, has shown that continued operation after location of defects does not result in increased coolant activity and that, in many cases, the activity decreases with time. No Westinghouse reactor has ever been shut down because of deterioration of defected fuel nor have steadystate coolant activity levels exceeded a fraction of the limits for continued operation.

3-12 OTHER RANDOM DEFECTS AND HANDLING INCIDENTS

Instances of fuel assembly damage due to fuel handling operations are summarized in the following paragraphs.

During a May 1974 refueling of Point Beach Unit 1 (14x14 fuel) and a January 1977 refueling of D. C. Cook Unit 1 (15x15 fuel), it was discovered that one assembly in each core had a torn corner grid. Evaluations showed that the damage occurred while the assemblies were being moved during core refueling operation. Both assemblies were discharged. Adjacent assemblies were not damaged.

During the 1978 Cycle 3/4 refueling shutdown for Zion Unit 1, one Region 5 fuel assembly was observed to have an outside strap at one grid corner torn-off. This fuel assembly was placed into the Cycle 4 reload after determining that this assembly would not adversely affect operations.

At a May 1978 Cycle 1/2 refueling of the Trojan plant, a fuel assembly was damaged during fuel shuffling operations. Inspection of the assembly revealed torn outer grid straps at the corner of one grid. Apparently, during lifting of this assembly, a grid corner caught under an adjacent assembly's grids. The damaged assembly was not reinserted into the core pending evaluation of damage for potential reuse in a later cycle.

During the 1979 Cycle 1/2 refueling shutdown for Salem Unit 1, it was discovered that the outer grid strap on a number of assemblies were damaged. Inspection of all 193 17x17 fuel assemblies showed 31 assemblies with some indication of grid damage, of which 19 were considered reusable with special handling precautions. Fifteen of the damaged assemblies were reused in the cycle 2 core. Evaluation indicate that the grid damage occurred during the fuel handling operations, and, for the most part, was confined to the corners of the grid assemblies which interacted with grids on adjacent fuel assemblies. No fuel rods sustained any damage.

Grid damage also occurred for 2 or 3 fuel assemblies for each the following plants undergoing refueling in 1979: Indian Point Unit 2, D. C. Cook Unit 2, Ko-Ri Unit 1 and Farley Unit 1. A number of these assemblies had only minor damage and were reused for the next cycle.

Westinghouse has recommended revised fuel handling procedures which would minimized such grid damage for the current fuel designs. Recent refueling, using the revised procedures, have revealed no indication of grid damage. In addition, modifications to the grid corners of all newly fabricated fuel are expected to further reduce the probability of grid damage during fuel handling.

SECTION 4

CORE COMPONENT EXPERIENCE

4-1 ROD CLUSTER CONTROL (RCC) ASSEMBLIES

Full-length RCCs are being successfully used in the San Onofre Unit 1, in Connecticut Yankee, and in all of the reactors listed in Table 5-1.

Very little difficulty has been experienced with the large numbers of control rods in use, despite the fact that they are constantly exercised. However, because they are of vital importance to safety, a record of anomalous behavior of RCCs and their drive lines has been compiled. (During operation, the drive line is integral with the RCC; its inclusion in this record is necessary). The service report in Table 5-4 places these anomalies in perspective.

Several problems have occurred which involved rod cluster control assemblies and drive lines. The problems, causes, and solutions are summarized chronologically in Table 5-5.

Infrequent drive line malfunctions are usually related to the presence of metallic debris in the system. All RCC systems require the use of small controlled clearances to maintain control rod position and alignment. As a result, the systems are susceptible to possible binding problems in the presence of debris. Such problems have always been detected during testing or have occurred very early in the life of the plant. Once eliminated, they have not recurred.

In regard to binding forces, it should be noted that the buckling force of individual absorber rods of either the 15 x 15 or the 17 x 17 design is controlled through spacing of guide tube "cards" to locate them well above available operating forces. With the Westinghouse guide tube card arrangement, the buckling force of individual rods in the 17 x 17 array will be greater than either the combined drive line weight or the operating force on the control rod drive mechanism.

During the Cycle 1/2 refueling of the Salem Unit 1 plant, it was noted that a total of eight rodlets had separated from six Rod Cluster Control assemblies. Subsequent hot cell examination revealed incipient cracks in the fingers that supported the rodlets. The failures were attributed to stress corrosion cracking resulting from reworking the internal threads of a small quantity of fingers and coating than with a trithlorethane- based lubricant to facilitate assembling into control rods. To preclude further failures, all control rods containing fingers from the suspect group were replaced during the refueling.

4-2 BURNABLE POISON ASSEMBLIES

There are 30 plants which are using or have used Westinghouse burnable poison assemblies. Most of these assemblies are intended to serve for only one cycle and are not removed during service.

Two burnable poison rods of shorter length, but similar in design to those currently used, were exposed to in-pile test conditions in the Saxton test reactor. A visual examination of the rods was made in early June 1968, and a visual and profilometer examination was made July 30, 1968, after an exposure of 1900 effective full-power hours (25 percent B_{10} depletion). The rods were found to be in excellent condition; profilometry results verified that no dimensional variations from the initial condition occurred. An experimental verification of the reactivity worth calculations for borosilicate glass tubing has been performed.

Although there is no routine surveillance of burnable poison assemblies in operating reactors, with the exception of the instances detailed below no problems have been encountered.

During removal of a secondary source assembly from its fuel assembly during the RGE Cycle 3/Cycle 4 refueling, several cracks/openings were noticed in the cladding at the lower end of an associated burnable poison rod. Some bubbles were seen, but checks for tritium were negative. The source assembly was moved to the change fixture and the rod was video-scanned. As agreed to by Westinghouse and RGE personnel, the source assembly was loaded into the appropriate fuel assembly for use only in the (fourth) cycle of operation.

At Prairie Island Unit 2, difficulty was experienced in removing a burnable poison assembly from a fuel assembly. The burnable poison/fuel assembly was subsequently reinserted into the core and has functioned without incident.

Two burnable poison rods and their vanes were separated from a source assembly during refueling operations at Beznau Unit 1. Separation occurred while the source assembly was being removed. The burnable poison rods remained fixed in the fuel assemblies for unknown reasons. Additionally, the bottom 18 inches of one burnable poison rod was found to have been removed. After separation of one burnable poison rod, the source assembly functioned through one operating cycle without incident. The source assembly was then replaced with another source assembly.

During February 1978, after two duty cycles of operation at Indian Point Unit 2, one burnable poison rod broken 24 inches from the top was found "jammed" in the thimble tube of the fuel assembly. The broken rod piece was forced into the fuel assembly by a thimble plug, and the fuel assembly has functioned without incident during the third duty cycle.

Four demonstration assemblies, each containing two pre-characterized improved burnable poison rodlets, have been inserted into the Indian Point Unit 3 Cycle 3 core for irradiation starting in early 1980. The improved BP rodlet design contains annular pellets of aluminum oxide - boron carbide burnable poison material in Zircaloy tubing. As of April 1980, indications are that the demonstration assemblies are operating satisfactorily.

4-3 SOURCES AND PLUGGING DEVICES

Primary sources, which are in service for only one cycle, are removed after secondary source activation. Secondary source rods have successfully operated for many years in cores. Recently (1978) there were two instances of stuck secondary source/depleted burnable poison assemblies during refuelings. During the Zion Unit 1 Cycle 3/4 Fall 1978 refueling, a secondary source assembly could not be removed from its fuel assembly. Also, during the Prairie Island Unit 1 Cycle 3/4 Spring 1978 refueling, two source assemblies could not be removed. For both plants, the fuel assemblies with the stuck sources (scheduled for discharge) were not reinserted into the core for the next cycle.

No problems have been encountered with plugging devices, except for one device being stuck in a fuel assembly (scheduled for discharge) during the Indian Point Unit 2 Cycle 2/3 refueling. This assembly was not reused for Cycle 3.

In 1979, at H. B. Robinson 2 and D. C. Cook 1, two secondary source assemblies were loaded into improper core locations. As a result, they sustained an interference fit between the fuel assembly and the flow mixers in the upper internals. In each of the incidences, one of the two source assemblies was found to be acceptable for future use.

Control Rods - (See table 5-5)

SECTION 5

OPERATING EXPERIENCE DATA

The data on which this report is based are presented in this section.

TABLE 5-1

BURNUP EXPERIENCE WITH ZIRCALOY-CLAD FUEL IN

Plant[a]	Plant Operation Date	Region 1 No. of Rods	Region 1 Region Average Burnup (GWD/MTU)	Region 2 No. of Rods	Region 2 Region Average Burnup (GWD/MTU)
Jose Cabrera (UEM)	June 1968	3580[b,c]	25.95[d]	4117[b,c]	24.35[d]
Beznau 1 (NOK)	June 1969	7339[b,c]	21.70[d]	7160[b,c]	20.28[d]
Ginna (RGE)	Nov. 1969	7339[b,c]	21.12[d]	7160[b,c]	18.69[d]
Mihama 1 (KEP)	July 1970	7339[b,c,f]	14.80[d]	7160[b,f]	18.54[d]
Robinson 2 (CPL)	Aug. 1970	10812[b,c]	16.16[d]	10608[b]	26.70[d]
Point Beach 1 (WEP)	Nov. 1970	8771[b,c]	20.61[d]	7876[b]	30.86[d]
Beznau 2 (NBK)	Oct. 1971	7339[b]	17.82[k]	7160[b]	29.67[d]
Mihama 2 (MEP)	Apr. 1972	7339[b,f]	23.95[d]	7160[b,f]	28.36[d]
Point Beach 2 (WIS)	May 1972	5907[b]	19.29[d]	6444[b]	29.90[d]
Surry 1 (VPA)	July 1972	10812[b]	19.11[d]	10608[b]	23.08[d]
Turkey Point 3 (FPL)	Oct. 1972	10812[b]	15.43[d]	10608[b]	26.04[d]
Surry 2 (VIR)	Mar. 1973	10812[b]	19.77[d]	10608	22.59[d]
Indian Point 2 (IPP)	May 1973	13260	15.43[d]	13056[b]	26.67[d]
Turkey Point 4 (FLA)	June 1973	10812[b]	15.85[d]	10608	29.64[d]
Zion 1 (CWE)	June 1973	13260[b]	19.81[s]	13056	31.45[d]
Prairie Island 1 (NSP)	Dec. 1973	7339	17.37[d]	7160	30.36[d]
Zion 2 (COM)	Dec. 1973	13260	18.90[t]	13056	30.20[d]
Takahama 1 (TAK)	Mar. 1974	10812[f]	15.89[d]	10608[f]	26.82[d]
Kewaunee (WPS)	Mar. 1974	7339	19.76[k]	7160	29.78[d]
Ringhals 2 (SSP)	June 1974	10812	19.87[y]	10608	26.12
Prairie Island 2 (NRP)	Dec. 1974	7339	18.46[d]	7160	30.74[d]
D.C. Cook 1 (AEP)	Jan. 1975	13260	18.28[d]	13056	29.59[d]
Trojan (POR)	Dec. 1975	17160	17.79[z]	16896	24.30[a,a]
Indian Point 3 (INT)	Apr. 1976	13260	18.93[k]	13056	30.24[d]
Beaver Valley 1 (DLW)	May 1976	13992	16.10[k]	13728	16.66
Salem (PSE)	Dec. 1976	17160	18.35[c,c]	16896	19.31[a,a]
Kori (KOR)	June 1977	7339	16.42[e,e]	7160	16.78
J. M. Farley (ALA)	Aug. 1977	13992	16.88[f,f]	13728	17.59[g,g]
OHI (OHI)	Dec. 1977	16632	9.670	16368	9.850
D.C. Cook 2 (AMP)	Mar. 1978	17160	16.93	16896	17.58[h,h]
North Anna (VRA)	Apr. 19[illegible]	13992	1[illegible].36(k)	13728	1[illegible].87

1. The notes for this table immediately follow the table.

OPEN LATTICE CORES[1]

Region 3 No. of Rods	Region 3 Region Average Burnup (GWD/MTU)	Region 4 No. of Rods	Region 4 Region Average Burnup (GWD/MTU)	
4654[b,c]	19.90[d]	3222[b]	28.15[d]	
14499[b,c]	12.12[d]	7876[b]	28.44[d]	(UEM)
7160[b,c]	9.77[d]	13067[c]	26.72[d]	(NOK)
7160[b,f]	12.87	7518[e,f,d]	2.78	(RGE)
10608[b]	22.94[d]	10812[e]	24.50[d]	
7518[b]	25.07[d]	7876	30.21[d]	(CPL)
7160[b]	29.18[d]	6444[b]	25.59[d]	(WEP)
7160[b,f]	25.31[d]			(NBK)
6802[b]	35.14[d]	8592	32.52[d]	
10608[b]	22.27[d]	18156	23.98[k]	(WIS)
10608[b]	29.14[d]	10608	28.92[d]	(VPA)
10608[b]	16.73[d]	16932[e]	27.14[k]	(FPL)
13056	33.51[d]	14688	24.10[r]	(VIR)
10608	29.77[d]	10608	25.80[d]	(IPP)
13056	36.12[d]	12240	31.63	(FLA)
7160	34.94[d]	7160	28.76[k]	(CWE)
13872	37.14[u]	12240	27.26	(NSP)
10608[f]	21.76[d]			(COM)
7160	31.89[d]	7160	21.75[v]	
10608	20.84[k]			(WPS)
7160	35.09[k]	7160	33.45	
13056	33.62[d]			(NRP)
16896	20.94	16896	7.17	
13056	26.08[b,b]	13056	11.06	
13728	11.18	13728	0.01[w]	(INT)
16896	14.65[d,d]	10560	1.62[w]	
7160	10.80[o]	7160	0.01[w]	
13728	12.07	13728	0.46[w]	
17952	6.510			
16896	12.09	[illegible]1120	0.01[w]	
13728	11.41	13728	0.01[w]	

TABLE 5-1 (Continued)

BURNUP EXPERIENCE WITH ZIRCALOY-CLAD FUEL IN OPEN L

Region 5 No. of Rods	Region 5 Region Average Burnup (GWD/MTU)	Region 6 No. of Rods	Region 6 Region Average Burnup (GWD/MTU)	Region 7 No. of Rods
6265	28.66[d]	2864	30.46[d]	1611
6444	27.79[d]	5370	33.19[k]	5728
8592	25.25[d]	7160	26.06	3580
10608	21.56[d]	10608	22.93[d]	
5012	25.79[d]	6444	31.63[d]	5728
8950	25.39[d]	7160	26.95[d]	
8771	32.40[n]	6444	23.31	6444
8160[e]	22.00[o]	12240[e]	22.18	13056[e]
9792[e]	31.81[p]	8160	27.93	8160
4896	22.02	16320[e]	13.14[q]	13056
12240	12.71	13872	2.26	
8160	25.52	8160[e]	16.49	2448
13056	20.25	12240	5.65	
7160	28.25	7160	14.25	7160
13056	16.26	12240	5.65	
7160	18.33[x]	7160	11.89	
7160	23.32	7160	10.84	
15504	0.01[w]			

LATTICE CORES[1]

	Region 8		Region 9		
Region Average Burnup (GWD/MTU)	No. of Rods	Region Average Burnup (GWD/MTU)		No. of Rods	Region Average Burnup (GWD/MTU)
30.32[g]	3580	26.73[h]			
28.67[l]	5728	24.43[m]	(UEM)	3580	20.65[i]
24.64[d]	5728	17.01	(NOK)	4654	17.92
			(RGE)	5728	7.75
32.52	3759	23.40			
			(WEP)	5728	20.60
13.81	5728	4.40			
12.32					
19.16	8160	8.22			
0.02					
7.94	13260	3.67			
8.94					

TABLE 5-1 (Continued)

BURNUP EXPERIENCE WITH ZIRCALOY-CLAD FUEL IN OPEN LATTICE CORES(1)

Region 10		Region 11/99*		Region 12	
No. of Rods	Region Average Burnup (GWD/MTU)	No. of Rods	Region Average Burnup (GWD/MTU)	No. of Rods	Regi Aver Burn (GWD
3580	11.62[j]	3580	3.24	716	.01[w
5728	8.29	5728/716	.77/10.91		
6444	9.40	5728	1.09		

NOTES (TABLE 5-1)

a. All these plants contain Zircaloy-clad rods in rod cluster control (RCC) assemblies. Distinctions among them on the basis of plant power level, assembly size (14x14, 15x15, or 17x17), stainless steel or Zircaloy guide thimbles, or core length (8, 10, or 12 feet) are immaterial for this assignment. Coolant pressures and average rod powers are all similar to current designs.

b. Assemblies with rods seated on bottom nozzle plate

c. Nonpressurized rods

d. Discharged regions

e. Regions composed of assemblies from a variety of sources (for example 4 + 4a + 4b) are treated the same as homogeneous regions. The discharge (or present accumulated) burnup is the weighted average among all these assemblies.

f. No further data as of 12/29/75 (Cutoff date for Revision 4 data)

g. 2 of the [illegible] assemblies are still operating.

h. [illegible] of the 20 assemblies are still operating.

i. 19 of the 20 assemblies are still operating.

j. 18 of the 20 assemblies are still operating.

k. One assembly is still operating.

l. 2 of the 32 assemblies are still operating.

m. 20 of the 32 assemblies are still operating.

n. 17 of the 49 assemblies are still operating.

o. 32 of the 40 assemblies are still operating.

p. 37 of the 48 assemblies are still operating.

q. 68 of the 80 assemblies are still operating.

r. 65 of the 72 assemblies are still operating.

s. 9 of the 65 assemblies are still operating.

t. 5 of the 65 assemblies are still operating.

u. 4 of the 68 assemblies are still operating.

v. 16 of the 40 assemblies are still operating.

w. This region has not generated any significant burnup.

x. 24 of the 40 assemblies are still operating.

y. 36 of the 53 assemblies are still operating.

z. 5 of the 65 assemblies are still operating.

aa. 60 of the 64 assemblies are still operating.

bb. 52 of the 64 assemblies are still operating.

cc. 36 of the 65 assemblies are still operating.

dd. 57 of the 64 assemblies are still operating.

ee. 9 of the 41 assemblies are still operating.

ff. 2 of the 53 assemblies are still operating.

gg. 51 of the 52 assemblies are still operating.

hh. 49 of the 64 assemblies are still operating.

*. Region 99 consists of 4 assemblies from Beznau Unit 2.

TABLE 5-2

COOLANT ACTIVITY LEVEL[a]

Plant	Percentage of Design Basis Coolant Activity in Indicated Year, Quarter																
	1968	1969				1970				1971				1972			
	4	1	2	3	4	1	2	3	4	1	2	3	4	1	2	3	4
Jose Cabrera (UEM)	[b]	1.0	1.0	1.0	2.0	3.0	2.0	3.0	3.0	4.0	[c]	3.0	6.0	10.0	9.0[c]	1.0	1.1
Ginna (RGE)					0.0	2.0	25	42	31	42[c]	19[c]	18	22	26	22[c]	9.0	3[c]
Beznau 1 (NOK)				0.0	36	67	72	55	76	77	75[c]	[c]	5.0	[c]	7.0	7.0	6.0
Beznau 2 (NBK)													0.2	0.1	0.1	0.1	0.1
Robinson 2 (CPL)													0.0	0.0	0.6	0.2	0.9
Mihama 1 (KEP)								[b]	[b]	3.0	5.0	5.0	4.0[c]	5.0	6.0	[c]	4.0[c]
Point Beach 1 (WEP)									0.1	0.1		0.2	0.5	3.0	9.0	12	[c]
Mihama 2 (MEP)															0.4	1.1	2.0
Point Beach 2 (WIS)															[b]	1.1	[b]
Surry 1 (VPA)																0	[c]
Turkey Point 3 (FPL)																	[b]
Surry 2 (VIR)																	
Zion 1 (CWE)																	
Indian Point 2 (IPP)																	
Turkey Point 4 (FLA)																	
Prairie Island 1 (NSP)																	
Zion 2 (COM)																	
Takahama 1 (TAK)																	
Kewaunee (WPS)																	
Ringhals 2 (SSP)																	
Prairie Island 2 (NRP)																	
D. C. Cook 1 (AEP)																	
Trojan (POR)																	
Indian Point 3 (INT)																	
Beaver Valley 1 (DLW)																	
Salem 1 (PSE)																	
Ko-Ri 1 (KOR)																	
J. M. Farley (ALA)																	
Ohi 1 (OHI)																	

a. A decline in the coolant activity level, due to any cause other than discharge of affected fuel at normal refueling, is a reflection of decreased release of fission product iodine.

b. Data not available

c. Shut down part of period

TABLE 5-2 (cont)

COOLANT ACTIVITY LEVEL[a]

Plant	Percentage of Design Basis Coolant Activity in Indicated Year, Quarter																			
	1973				1974				1975				1976				1977			
	1	2	3	4	1	2	3	4	1	2	3	4	1	2	3	4	1	2	3	4
Jose Cabrera (UEM)	1.3[c]	1.3[c]	[b]	0.1	0.3	0.1	[b]	0.6	0.8[c]	[b,c]	0.5	0.3	1.6	0.54	0.88	0.81	0.62	0.46[c]	0.53[c]	0.59
Ginna (RGE)	3.0	2.3	1.5	1.3	[c]	1.5[c]	1.0	2.6[c]	2.4[c]	9.0[c]	25	15	1.8	10.5	3.4	5.2	6.19	1.12[c]	1.48	1.27
Beznau 1 (NOK)	9.0	[c]	1.1	0.5	1.1	0.9[c]	0.5[c]	0.2	0.6	0.02[c]	[b,c]	[b]	2.0	0.9	1.0	1.59	1.59	1.21[c]	0.39[c]	1.20
Beznau 2 (NBK)	0.1	0.1	0.1	2.8	2.7	[b]	3.6[c]	2.9	1.8	1.4	[c]	[b]	2.4	2.2	[c]	0.36	0.45	0.66	0.50	0.41
Robinson 2 (CPL)	1.1	0.1[c]	0.1	0.1	0.3	0.8[c]	0.1	0.1	0.1	0.0	0.1	0.1[c]	0.03	0.06	0.11	0.09[c]	0.04	0.06	0.07	0.12
Mihama 1 (KEP)	6.0	[c]	[c]	0.1	0.1	[b,c]	[b,c]	[c]	[c]	[c]	[c]	[c]	[b,c]	3.1	[c]	[c]	[b]	[b]	[b]	[b]
Point Beach 1 (WEP)	1.0	2.8	1.5	1.9	2.8	[b,c]	[b]	21[c]	10.5[c]	9.0	8.0	3.8[c]	2.9	2.1	2.6	1.8	1.70	1.41	1.90	0.98
Mihama 2 (MEP)	3.0	2.6	2.4	[c]	[b]	3.3	2.1	1.9[c]	[c]	[c]	[c]	[c]	2.3	3.1	[b]	1.77	1.24	1.65	[b]	1.24
Point Beach 2 (WIS)	0.2	1.3	1.1	0.3	0.6	0.5	0.6	[c]	0.4[c]	0.6	0.5	0.6	0.03	0.07	0.14	0.23	0.19[c]	0.07	0.28	0.15
Surry 1 (VPA)	0.2	0.1	0.1	0.1[c]	[b,c]	0.6	1.1	0.7[c]	3.8[c]	4.1	2.9	[b,c]	0.8	2.6	4.1	0.94[c]	1.23	1.11	0.88	0.77
Turkey Point 3 (FPL)	0.1	0.1	2.1	1.2	1.1	1.5	1.0	[b,c]	2.4[c]	1.2	1.3	0.8[c]	1.04	0.88	0.72	1.2[c]	0.36	0.46	0.47	0.70
Surry 2 (VIR)	[b]	0.2	0.2	0.1	0.1	[c]	0.2[c]	[c]	0.3[c]	0.3[c]	0.5	0.5	[c]	0.14	0.16	[c]	0.13	0.13	0.11	0.02[c]
Zion 1 (CWE)			[b]	[c]	[c]	0.2	[c]	0.1	0.2[c]	0.1	0.1	0.5	[c]	1.1	1.5	4.1	2.67	1.27	1.74	1.23[c]
Indian Point 2 (IPP)			[b]	[c]	[c]	[b,c]	[b,c]	[b]	1.5[c]	1.4	0.8	0.8	[c]	[c]	[c]	2.2	2.05	0.82	0.28	0.78
Turkey Point 4 (FLA)			0.1	0.1	0.1	0.1	0.1	0.1	0.1	[c]	0.0	0.0	0.08	0.03	0.20	0.36	0.31	1.00[c]	0.40	0.37
Prairie Island 1 (NSP)					[b]	[c]	0.1[c]	0.1	0.1[c]	0.1	0.1	0.0	[c]	0.03	0.028	0.05	0.05	<0.01[c]	<0.01	0.30
Zion 2 (COM)					[b]	[c]	[b,c]	0.2	0.2	0.4	0.3	0.5	0.43	0.88	1.1	2.0	0.55[c]	0.07	0.31	0.39
Takahama 1 (TAK)					[b]	[b]	[b]	[b]	0.1	0.0	0.0	0.0	[c]	0.039	0.009	[b]	<0.01	<0.01	<0.01	<0.01
Kewaunee (WPS)					[b]	0.1	3.0	0.9	0.7	0.6	0.4	0.2	[b]	0.02	0.34	0.15	0.13	0.08[c]	0.06	0.08
Ringhals 2 (SSP)							[b]	[b]	[b]	[b]	[b]	[b]	0.13	[b]	[c]	[b]	0.02	0.28	0.36	0.36
Prairie Island 2 (NRP)									0.1	0.0	4.0	0.4	1.7	2.0	1.9	[c]	1.00	1.86	0.95	0.22[c]
D. C. Cook 1 (AEP)									[b]	0.0	0.0	0.0	0.12	0.26	0.42	0.66	0.26	0.25	0.22	0.32
Trojan (POR)												[b]	[b]	[b]	0.68	0.44	0.46	0.18	0.47	0.53
Indian Point 3 (INT)														[b]	[b]	0.18	0.03	0.04	0.14	0.14
Beaver Valley 1 (DLW)																[b]	[b]	<0.01	0.02	0.34
Salem 1 (PSE)																	[b]	<0.01	0.03	<0.01
Ko Ri 1 (KOR)																		[b]	[b]	[b]
J. M. Farley 1 (ALA)																			[b]	<0.01
Ohi 1 (OHI)																				[b]

a. A decline in the coolant activity level, due to any cause other than discharge of affected fuel at normal refueling, is a reflection of decreased release of fission product iodine.

b. Data not available

c. Shut down part of period

TABLE 5-2 (Cont)

COOLANT ACTIVITY LEVEL [a]

Percentage of Design Basis Coolant Activity in Indicated Year, Quarter

1978

PLANT		1	2	3	4
JOSE CABRERA	(UEM)	0.68 [c]	0.30	2.98	3.22
GINNA	(RGE)	1.62 [c]	0.90 [c]	1.29	1.14
BEZNAU 1	(NOK)	1.29	1.25 [c]	0.76	0.98
BEZNAU 2	(NBK)	0.35	0.98	0.19 [c]	0.20
MIHAMA 1	(KEP)	b	b	b	b
POINT BEACH 1	(WEP)	1.05	0.14	1.45 [c]	0.86 [c]
MIHAMA 2	(MEP)	b	b	b	b
POINT BEACH 2	(WIS)	0.14 [c]	0.27 [c]	0.22	0.26
SURRY 1	(VPA)	0.74	0.47 [c]	0.09	0.46
TURKEY POINT 3	(FPL)	0.50 [c]	0.37	0.24	0.33
SURRY 2	(VIR)	0.02	0.01	0.01	0.02
ZION 1	(CWE)	1.20	1.19	1.10 [c]	0.42 [c]
INDIAN POINT 2	(IPP)	0.73 [c]	0.45 [c]	0.61	0.25
TURKEY POINT 4	(FLA)	0.40	0.37	0.15 [c]	0.23 [c]
PRAIRIE ISLAND 1	(NSP)	0.03 [c]	0.14 [c]	0.68	1.98
ZION 2	(COM)	0.32 [c]	0.47 [c]	0.29	0.46
TAKAHAMA 1	(TAK)	b	b	b	b
KEWAUNEE	(WPS)	0.08	0.12 [c]	<0.01	<0.01
RINGHALS 2	(SSP)	b	b	b	0.29
PRAIRIE ISLAND 2	(NRP)	0.03	0.05	0.04	0.05 [c]

TABLE 5-2 (Cont)

COOLANT ACTIVITY LEVEL [a]

Percentage of Design Basis Coolant Activity in Indicated Year, Quarter

1978

PLANT		1	2	3	4
D. C. COOK 1	(AEP)	0.36	0.47 [c]	0.04	0.04
TROJAN	(POR)	0.33	b	b	b
INDIAN POINT 3	(INT)	0.41	0.26	<0.01 [c]	0.11
BEAVER VALLEY 1	(DLW)	0.48	0.32	0.04	b
SALEM 1	(PSE)	<0.01	b	<0.01	<0.01
KORI 1	(KOR)	b	b	b	b
J. M. FARLEY 1	(ALA)	0.03	0.02	0.01	0.05
OHI 1	(OHI)			b	b
D. C. COOK 2	(AMP)			<0.01	<0.01
NORTH ANNA 1	(VRA)			2.31	1.86
OHI 2	(OKB)				b

a. A decline in the coolant activity level, due to any cause other than discharge of affected fuel at normal refueling, is a reflection of decreased release of fission product iodine.

b. Data not available

c. Shut down part of period

TABLE 5-2 (Cont)

COOLANT ACTIVITY LEVEL [a]

Percentage of Design Basis Coolant Activity in Indicated Year, Quarter

1979

PLANT		1	2	3	4
JOSE CABRERA	(UEM)	.68[c]	3.45	2.81	(b)
GINNA	(RGE)	1.62[c]	2.17	1.75	2.23
BEZNAU 1	(NOK)	1.29	0.80	0.61	0.74
BEZNAU 2	(NBK)	-	-	-	-
MIHAMA 1	(KEP)	(b)	(b)	(b)	(b)
POINT BEACH 1	(WEP)	1.05	1.49	1.38	(c)
MIHAMA 2	(MEP)	(b)	(b)	(b)	(b)
POINT BEACH 2	(WIS)	.14[c]	0.04	0.04	0.04
SURRY 1	(VPA)	.77	0.30	(b)	0.20
TURKEY POINT 3	(FPL)	.52[c]	0.19	0.13	0.13
SURRY 2	(VIR)	.02[c]	(b)	(b)	(b)
ZION 1	(CWE)	1.19	0.81	0.91	(c)
INDIAN POINT 2	(IPP)	0.73	0.78	(b)	0.94
TURKEY POINT 4	(FLA)	0.40	0.18	0.08	0.15
PRAIRIE ISLAND 1	(NSP)	0.01	0.32	0.42	0.37
ZION 2	(COM)	0.32[c]	0.09	0.07	0.07
TAKAHAMA 1	(TAK)	(b)	(b)	(b)	(b)
KEWAUNEE	(WPS)	0.23	0.02	0.09	<0.01
RINGHALS 2	(SSP)	(b)	(b)	(b)	(b)
PRAIRIE ISLAND 2	(NRP)	0.03	0.03	0.03	0.03

TABLE 5-2 (Cont)

COOLANT ACTIVITY LEVEL [a]

Percentage of Design Basis Coolant Activity in Indicated Year, Quarter

1979

PLANT		1	2	3	4
D. C. COOK 1	(AEP)	0.37	0.40	-	-
TROJAN	(POR)	0.33	0.57	0.87	(b)
INDIAN POINT 3	(INT)	0.41	0.11	0.11	(c)
BEAVER VALLEY 1	(DLW)	0.43	(b)	0.21	0.02[c]
SALEM 1	(PSE)	0.01	0.15	(b)	(b)
KORI 1	(KOR)	(b)	0.81	1.21	0.17
J. M. FARLEY 1	(ALA)	.03[c]	<0.01	0.70	<0.01
OHI 1	(OHI)	(b)	(b)	(b)	(b)
D. C. COOK 2	(AMP)	(b)	0.02	0.01	(c)
NORTH ANNA 1	(VRA)	(b)	0.61	0.57	(c)
OHI 2	(OKB)	(b)	(b)	(b)	(b)

a. A decline in the coolant activity level, due to any cause other than discharge of affected fuel at normal refueling, is a reflection of decreased release of fission product iodine.

b. Data not available

c. Shut down part of period

TABLE 5-3

PREVIOUS EXPERIENCE IN RELOADING DEFECTED WESTINGHOUSE FUEL

Plant	Shutdown	Number of Assemblies Affected	Method of Defect Detection and Notes	Coolant Activity Trend
A	Cycle 1A/1B	24	Positive leak test	Activity stable during Cycle 1B
A	Cycle 1B/2	5	Positive leak test One assembly with collapse and bulge reinserted	Activity stable during Cycle 2
A	Cycle 5/6	3	Positive leak test	Activity slowly decreased during Cycle 6
B	Cycle 2/3	1	Positive leak test	Activity increased at the beginning and then slowly decreased during Cycle 3
C	Cycle 1/2	2	Positive leak test	Activity increased slightly during Cycle 2; additional defects forming
C	Cycle 3/4	1	Postive leak test Coolant activity indicated failures at start of Cycle 3, Region 4 indicated based on uranium samples	Activity decreased during Cycle 3; activity lower and stable in Cycle 4
C	Cycle 4/5	9	Part of Region 4 reinserted	Activity stable in Cycle 5
D	Cycle 1/2	1	Visible hole in one rod at EOC-1; no significant degradation observed EOC-2	Activity stable in Cycle 2

TABLE 5-4
SERVICE RECORD OF RCCs THROUGH DECEMBER 31, 1979

Plant	Number of Full-Length RCCs/Core	Service Months
San Onofre	45	152
Connecticut Yankee	53	151
Jose Cabrera	17	139
Beznau 1	25	127
Ginna	29	122
Mihama 1	29	114
Robinson 2	45	123
Point Beach 1	33	110
Beznau 2	25	99
Mihama 2	29	93
Point Beach 2	33	93
Surry 1	48	91
Turkey Point 3	45	87
Surry 2	48	82
Indian Point 2	53	80
Turkey Point 4	45	79
Zion 1	53	79
Prairie Island 1	29	74
Zion 2	53	73
Takahama	48	70
Kewaunee	29	70
Ringhals 2	48	66
Prairie Island 2	29	61
D. C. Cook 1	53	60
Trojan	53	49
Indian Point 3	53	44
Beaver Valley 1	48	43
Salem 1	53	37
Ko-Ri 1	29	30
J. M Farley	48	28
Ohi 1	53	25
D. C. Cook 2	53	20
North Anna 1	48	21
TOTAL 33 plants	1382	2592

TABLE 5-5
SUMMARY OF RCC AND DRIVE LINE PROBLEMS

Plant	Date	Problem	Cause	Solution	Remarks
Connecticut Yankee	April, August 1968	Two assemblies' drive lines immobile	Vane separation due to faulty braze joint	RCCs replaced	No failures noted in subsequent cycles
Point Beach Unit 1	October 1970	Drive line immobile	Chip lodged between RCC spider and an intermediate guide tube guide plate	Freed by removal of vessel head and upper internals	Chip was CF8 casting alloy.
H. B. Robinson Unit 2	November 1970	Drive line immobile	Weld spatter nugget lodged between RCC rodlet and a lower guide tube guideway	Guide tube and RCC replaced; freed by removal of vessel head and upper internals	Nugget was a SS alloy which originated elsewhere in the system.
Indian Point Unit 2	April 1972	Three drive lines immobile and one with a short-period malfunction	Two drive lines immobilized by galling of one of the RCC absorber rodlets with the corresponding thimble tube	Fuel assemblies and RCCAs repaired	The galling was a unique occurrence.
			One drive line immobilized by interference between an intermediate guide tube plate and an RCC spider vane separated from the hub	RCC replaced	Chip was CF8 material.
			Short-period malfunction due to jamming caused by a metallic chip	Chip dislodged and drive line freed upon withdrawal	None

TABLE 5-5 (cont)

SUMMARY OF RCC AND DRIVE LINE PROBLEMS

Plant	Date	Problem	Cause	Solution	Remarks
Point Beach Unit 1	October 1972	RCC would not insert into assembly.	Three rodlets on RCC bent during manipulation	RCC replaced	None
H. B. Robinson Unit 2	April 1973	RCC vane with two rodlets inserted in fuel assembly	Vane and rodlets separated from RCC	RCC replaced	Cause of failure not determined
Jose Cabrera	January 1975	Power tilt with power depression	Absorber rod separated from an RCC finger was inserted into core.	RCC replaced	Failure of weld making finger-to-antilock pin joint
D. C. Cook Unit 1	April 1976	RCC stickling and flux map indication of one or two rod-lets inserted into core	RCC vane and two rodlets rodlets separated from RCC hub	RCC replaced	Failure of braze to base metal joint was cause.
Connecticut Yankee	November 1977	Manipulator crane gripper interference noticed while at-tempting to latch a rodded assembly	RCC vane with two rodlets attached separated from RCC hub	RCC replaced	None
D. C. Cook Unit 2	January 1978	C/R hangup in guide tube during drag testing	Foreign object found in guide tube	Object removed	None
KORI Unit 1	December 1979	RCC vane with two rodlets inserted into fuel assembly	Vane & Rodlets separated from RCC	RCC replaced	Cause of failure not determined

TABLE 5-6

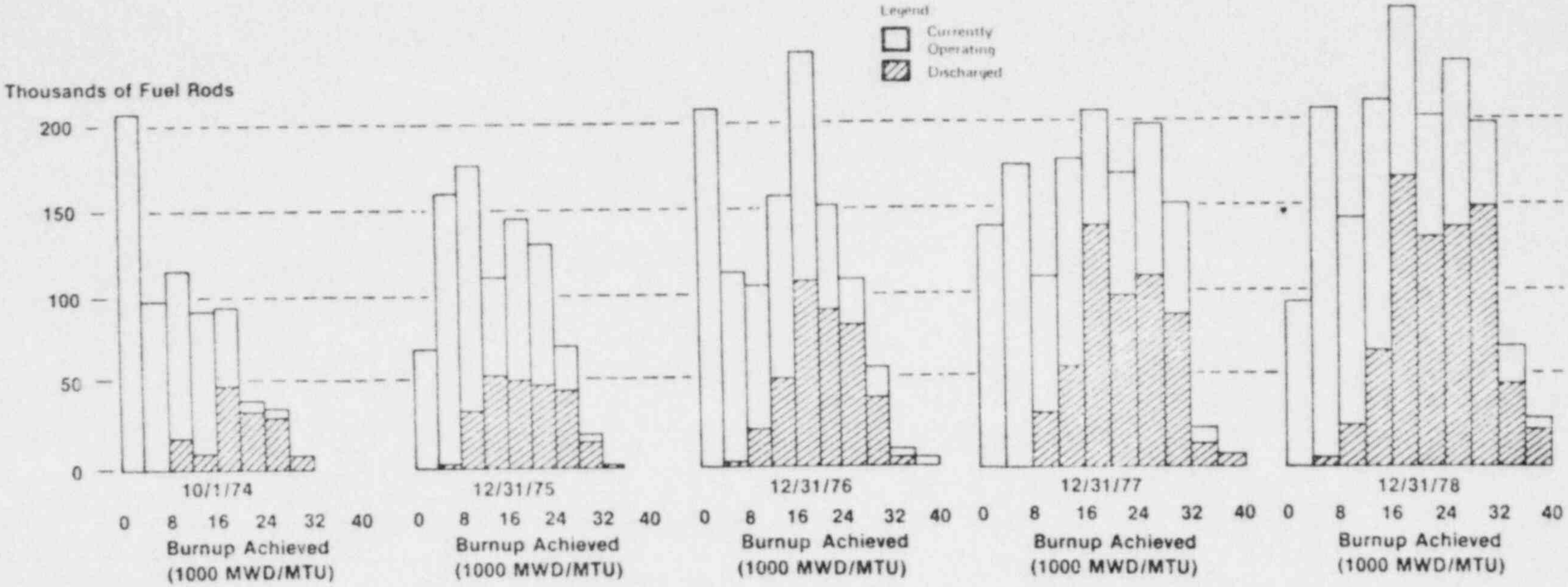
WESTINGHOUSE FUEL PERFORMANCE STATUS REPORT
(Fourth Quarter 1979)

Reactor	Location	Owner	Date Initial Criticality	Nominal MWe Net	Current Cycle Number	Peak Region Avg Burnup as of 12/31/79 (MWD/MTU)	Generation, MWh(e) Fourth Quarter 1979	Generation, MWh(e) Cumulative	Percent of Design Basis Activity Release Rate(b)
Jose'de Cabrera	Spain	Union Electric S.A.	June 1968	153	9	30,500	281,200	11,193,700	(a)
Beznau 1	Switzerland	Nordostschweizerische Kraftwerke AG	June 1969	353	9	33,200	801,720	23,489,430	0.74
Robert E. Ginna	U.S.	Rochester Gas & Electric	November 1969	490	9	26,700	290,101	28,685,791	2.23
Mihama 1	Japan	Kansai Electric Power	July 1970	320	2	18,500	33,907	5,540,274	(a)
Point Beach 1	U.S.	Wisconsin Electric Power	November 1970	497	8	32,500	214,790(d)	30,633,750	(d)
Mihama 2	Japan	Kansai Electric Power	April 1972	470	4	28,400	4,398	16,311,041	(a)
Point Beach 2	U.S.	Wisconsin Electric Power	May 1972	497	6	36,100	1,103,520	23,361,340	0.04
Surry 1	U.S.	Virginia Electric Power	July 1972	822	5	23,900	1,037,845	28,924,097	0.20
Turkey Point 3	U.S.	Florida Power & Light	October 1972	693	6	31,800	890,685(d)	29,107,062	0.13
Surry 2	U.S.	Virginia Electric Power	March 1973	822	4	27,100	0	26,213,947	(a)
Zion 1	U.S.	Commonwealth Edison	June 1973	1040	4	36,100	424,595(d)	33,456,560	(d)
Indian Point 2	U.S.	Consolidated Edison	May 1973	873	4	33,500	1,588,580	26,229,876	0.94
Turkey Point 4	U.S.	Florida Power & Light	June 1973	693	6	29,800	1,313,615	26,178,838	0.15
Prairie Island 1	U.S.	Northern States Power	December 1973	530	5	34,900	484,373	20,173,130	0.37
Zion 2	U.S.	Commonwealth Edison	December 1973	1040	4	37,100	59,900(d)	29,964,321	0.07
Kewaunee	U.S.	Wisconsin Public Service	March 1974	535	5	31,900	791,200	20,558,371	<0.01
Takahama 1	Japan	Kansai Electric Power	March 1974	781	3	26,800	1,720,029	16,533,844	(a)
Ringhals 2	Sweden	Swedish State Power Board	June 1974	822	4	26,100	1,144,798	20,483,423	(a)
Prairie Island 2	U.S.	Northern States Power	December 1974	530	4	35,100	1,148,670	18,050,130	0.03
Trojan	U.S.	Portland General Electric	December 1975	1130	2	24,300	1,420(d)	16,591,389	(a)
Indian Point 3	U.S.	Power Authority of the State of New York	April 1976	873	2	30,200	0	19,668,390	(d)
Beaver Valley	U.S.	Duquesne Light	May 1976	852	1	16,600	455,800(d)	8,285,000	0.02(d)
Salem 1	U.S.	Public Service Electric & Gas	December 1976	1090	2	19,300	12,920(d)	10,943,290	(a)
Ko-Ri 1	Korea	Korea Electric Company	June 1977	564	1	16,700	282,259(d)	5,547,746	0.17
Farley 1	U.S.	Alabama Power Company	August 1977	829	2	17,500	712,046(d)	9,252,194	< 0.01
Ohi 1	Japan	Kansai Electric Power	December 1977	1122	1	9,850	288,620(d)	6,341,877	(a)
D. C. Cook 2	U.S.	Indiana & Michigan Electric	March 1978	1060	1	17,600	488,370(d)	10,154,410	(d)
North Anna 1	U.S.	Virginia Electric Power	April 1978	907	1	16,900	0	8,346,349	(d)
Ohi 2	Japan	Kansai Electric Power	September 1978	1122	1	<1,000	1,431,195	4,497,155	(a)

a. No data reported.
b. Activity release rate calculated from coolant activity averaged over the quarter and presented as percent of that I-131 release rate which establishes the basis for design of plant shielding and coolant cleanup system equipment.
c. Reported information reflects last period of operation or inferrence from short periods of operation during period of report.
d. Plant refueled during period of report.

FIGURE 5-1

BURNUP PERFORMANCE OF WESTINGHOUSE ZIRCALOY-CLAD FUEL

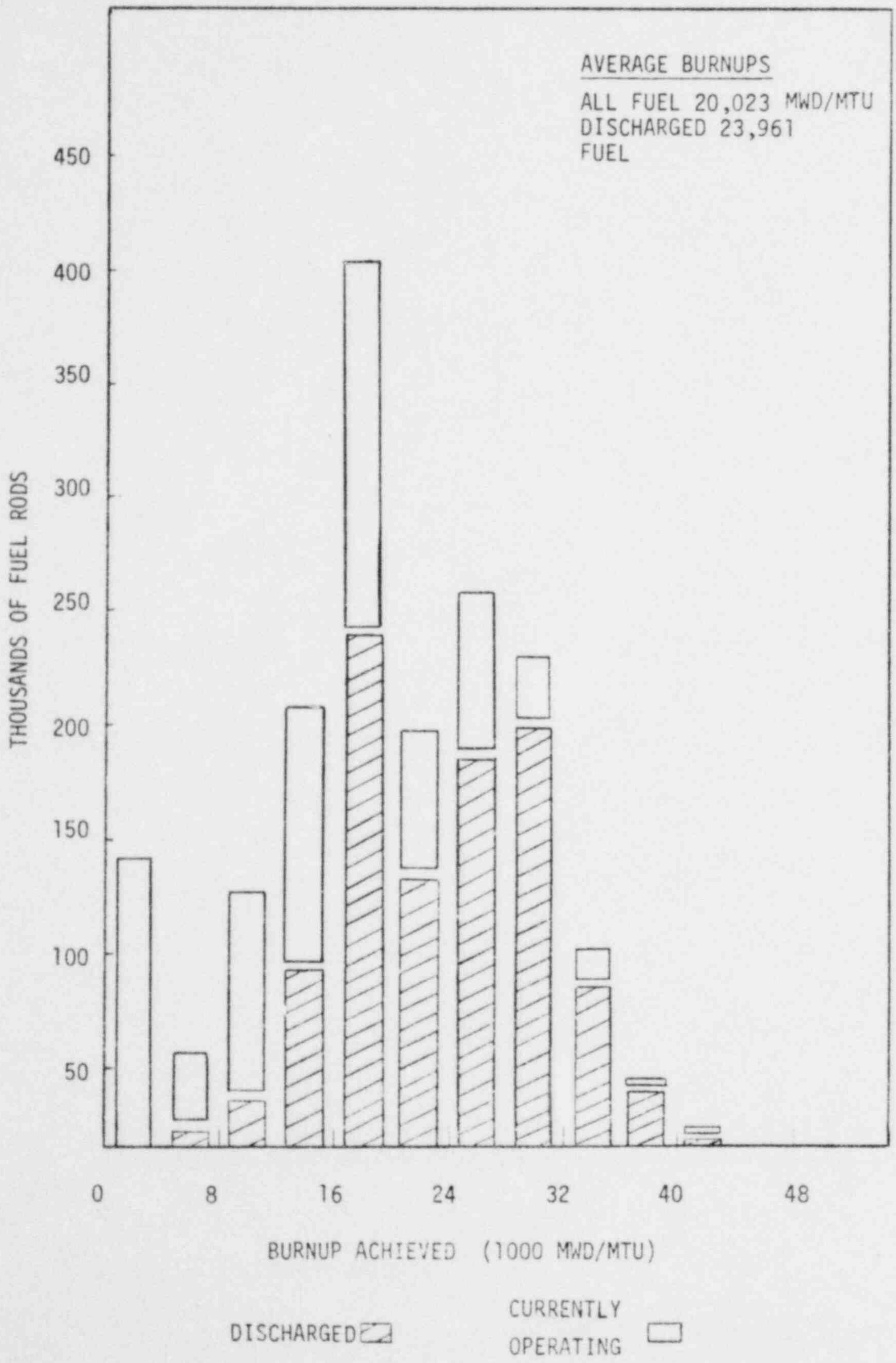


Average Burnups	10/1/74	12/31/75	12/31/76	12/31/77	12/31/78
All Fuel	10,200	14,300	15,100	17,300	18,800
Discharged Fuel	19,900	19,600	20,700	22,200	23,700

FIGURE 5-1 (Continued)

BURNUP PERFORMANCE OF WESTINGHOUSE
ZIRCALOY-CLAD FUEL

(As of 12/31/79)



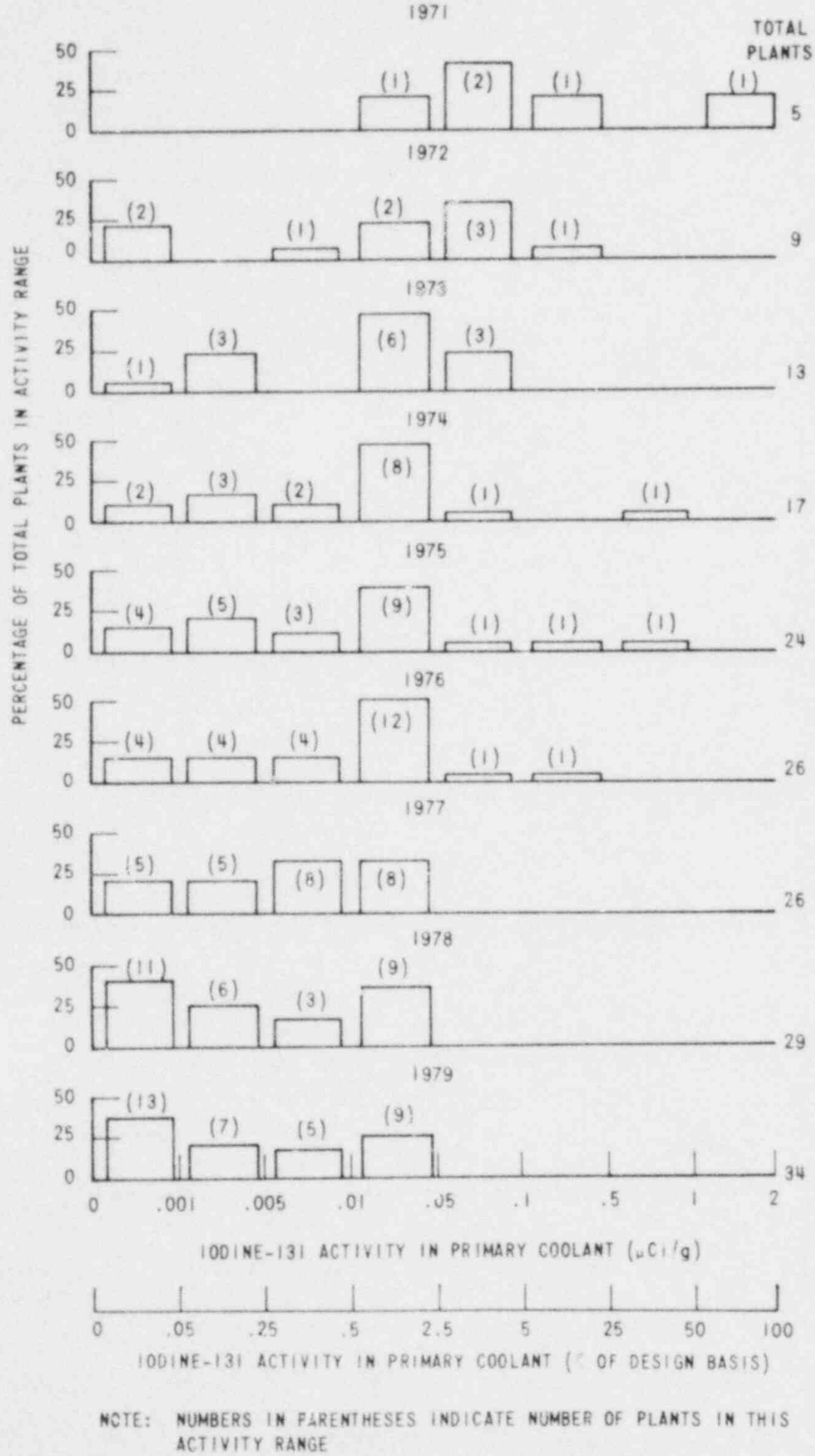


Figure 5-2. Westinghouse Fuel Performance – Iodine 131 Activity in Primary Coolant as of December 31, 1979

Zircaloy – Clad Fuel Performance and Experience

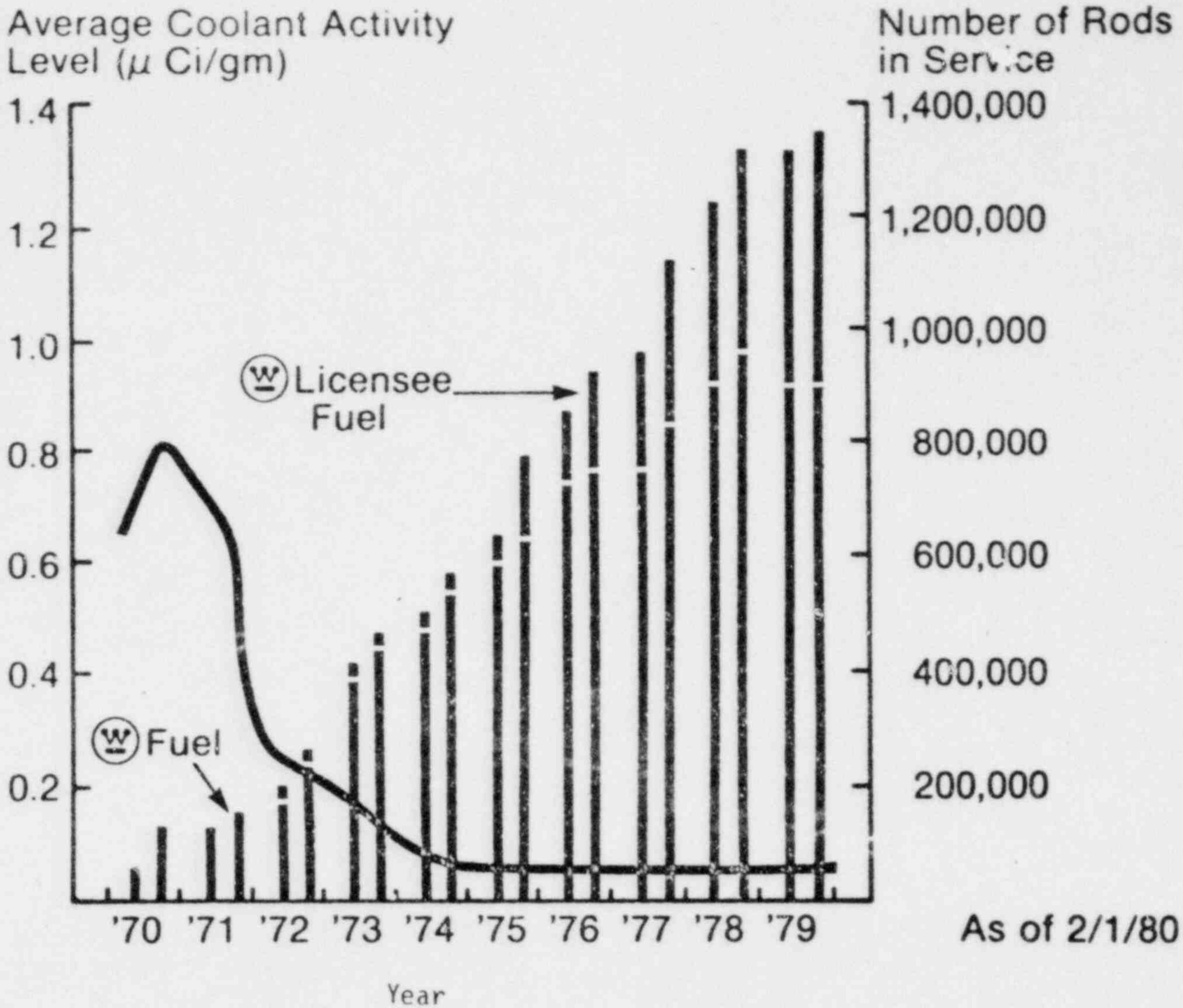


Figure 5-3. Performance Record - Average Coolant Activity Level/Number of Rods in Service

SECTION 6

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