WESTINGHOUSE CLASS 3

WCAP-8183
Rev. 10

OPERATIONAL EXPERIENCE WITH WESTINGHOUSE CORES
(Up to December 31, 1980)

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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, 1980. 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 and generic concerns 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, 1980, there were 35 commercial PWRs that have used Westinghouse-supplied Zircaloy-clad fuel. During this reporting period nineteen plants have been refueled and three plants have started initial commercial power operation. A total of 901,700 fuel rods are operating in these reactors. The addition of previously discharged fuel brings the total number of Westinghouse Zircaloy-clad fuel rods to 1,996,400. This represents 4,500 MTU and a thermal energy production of 102,000 GWD. The average burnup of the discharged fuel is 24,500 MWD/MTU and the average burnup of all fuel is 20,600 MWD/MTU. (The burnup for all fuel is lower because of the new (unburned) fuel included in that number). Table 5-1 presents a burnup summary of Westinghouse fuel rods discharged and being irradiated through 1980. As shown, there are numerous fuel rods included in assembly burnups greater than 36,000 and 40,000 MWD/MTU.

Table 5-2 presents a Westinghouse fuel performance summary on a plant-by-plant basis. The peak burnup data in Table 5-2 reflect an average of the high burnup assemblies (batch) in a region. There are a number of plants which have fuel assembly burnups (batch) in the range of 32,000 to 37,100 MWD/MTU.

The highest burnups of individual discharged fuel assemblies have been in the range of 39,000-46,500 MWD/MTU. As part of an EPRI high burnup program, four high burnup demonstration assemblies have been irradiated in the Zion Unit 2 Cycle 4 core to 46,500 MWD/MTU. These four assemblies will have their burnups extended to a maximum of about 55,000 MWD/MTU in the Zion Unit 1-Cycle 6 core (startup early 1981). A significant amount of high burnup experience data has been evaluated for Westinghouse fuel assemblies.[2,3] Three plants have discharged fuel with region average burnups in the range of 36,000 to 37,100 MWD/MTU.

Significant 17x17 fuel assembly burnup experience has been obtained with two plants completeing two cycles of operation and three plants one cycle. Region average burnups up to 29,000 MWD/MTU were obtained, and, in general, the 17x17 fuel assemblies were in excellent condition. Four demonstration 17x17 Optimized Fuel Assemblies (OFA) have completed their first cycle of operation in the fall of 1980, with burnups in the range of 7000 to 9,600 MWD/MTU. End-of-Cycle 1 examinations showed the assemblies to be in excellent condition, and the four OFAs are in their second cycle of operation. Additional details are presented in Section 3-2.

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. With the exception of a few fretting failures in peripheral assemblies (see Section 3-6.3), no cause can be assigned to the apparent defects due to the low levels of occurrance.

Nuclear reactors now operating with Westinghouse Zircaloy-clad fuel have not experienced availability limitations due to fuel defects. As indicated in section 3-6, fuel concerns have been successfully and promptly resolved.

SECTION 3

FUEL EXPERIENCE OVERVIEW

3-1 BURNUP EXPERIENCE

Table 5-2 contains a list, in chronological order of plant startup, of plants in which extensive operating experience has been obtained with Westinghouse Zircaloy-clad rods in open lattice cores. Taken as an aggregate, the data 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 fuel burnup data through 1980 (also includes burnup performance figures from several previous years). The unshaded portion of the figure represents the fuel assemblies in service and the shaded 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 Figure 5-2 for plants containing Westinghouse fuel. 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 equivalent to 100% of coolant design basis can be used for purposes of comparison. In all cases, the activity levels are below the coolant activity levels allowed in the technical specifications.

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. The reported coolant activity levels, combined with the burnup experience of Figure 5-1, provide substantial proof of the reliability and good performance of Westinghouse Zircaloy-clad fuel.

3.2 17 x 17 FUEL ASSEMBLY EXPERIENCE

Four demonstration assemblies of the 17 x 17 design with 7 grids 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 each cycle 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 now in its fourth cycle of irradiation with a December 1980 burnup of 32,800 MWD/MTU. 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 complete two cycles of operation with Westinghouse standard 17 x 17, eight grid fuel assemblies. Except for two rods damaged in Cycles 1 and 2 by coolant jetting at leaking baffle joints (see Section 3-6.3), the Trojan fuel was in excellent condition after two cycles of operation. No anamalous corrosion was observed, grids were intact, and rod bow was within the data base for the NRC approved rod bow correlation[4].

The Farley Unit 1, Salem Unit 1, D. C. Cook Unit 2 and Beaver Valley Unit 1 reactors have completed one cycle of operation with no anamolies noted for the 17x17 fuel assemblies. Some grids were damaged during refueling operations (See Section 3-6.6). D. C. Cook Unit 2 completed Cycle 2 in March 1981.

Eight plants of the 17x17 design are now in commercial operation in the United States. Ten 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. Seven 17x17 fueled domestic plants are included in Figure 5.2.

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. The four demonstration OFAs successfully completed their first cycle of operation in the fall of 1980, achieving burnups of 7000 and 9600 MWD/MTU respectively in the Salem and Farley plants. Post-irradiation examinations revealed that the assemblies were in excellent condition with no observable anomalies. The four OFAs are in their second cycle of operation in the Salem and Farley plants. Two additional demonstration OFAs began a first cycle of operation in the Beaver Valley Unit 1 Cycle 2 core, commencing in December 1980.

3-3 HIGH BURNUP AND MIXED OXIDE FUEL PERFORMANCE

Three plants have discharged fuel with region average burnups in the range of 36,000 to 37,100 MWD/MTU. A total of 67 assemblies have assembly average burnups greater than 38,000 MWD/MTU. Coolant activities have remained low at these plants, and no correlation between high burnups and high coolant activities have been observed. Four assemblies have been irradiated for four cycles in the Zion reactors to assembly average burnups of approximately 46,500 MWD/MTU as part of an EPRI/WESTINGHOUSE PROGRAM. Sipping these assemblies has confirmed that they are defect free. In early 1981 these four assemblies will start a fifth cycle of irradiation in Zion Unit 1 Cycle 6 to extend their burnup to about 55,000 MWD/MTU.

Four fuel assemblies with Westinghouse mixed oxide fuel rods are being irradiated for their first cycle of operation in the R. E. Ginna Cycle 10 core starting in the spring of 1980. The Cycle 10 core has not indicated any abnormal coolant activities during 1980.

3-4 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

3-5 OPERATION WITH DEFECTED FUEL

Rigid specifications on moisture and pellet parameters affecting densification have resulted in coolant radioactivity levels being 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.

3-6 GENERIC FUEL CONCERNS

The main concerns discussed are fretting in peripheral assemblies, ramp rate effects on fuel performance, fuel rod bowing and fuel assembly grid damage during fueling operations. No damage to Westinghouse fuel assemblies has been observed the past few years due to ramp rate effects or rod bow. For historical completeness, brief summaries are given on clad hydriding and fuel densification which have not been generic concerns since the mid-seventies.

3-6.1 Moisture and Hydriding

In the early 70's coolant chemistry indicated a number of cladding defects in the Beznau Unit 1 and R. E. Ginna reactors. Visual examinations during refueling indicated local hydriding leading to cladding defects. The hydriding was confirmed to be caused by moisture released from low density fuel pellets. Since then, fuel has been fabricated with an improved fuel powder and pelletization process, and tighter specifications on fuel moisture content. As a result, cladding defects due to hydriding have rarely been observed.

3-6.2 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.

By the end of 1972, the mechanism was fully characterized as a low-temperature densification caused by the fission-induced resolution of fine pores in the fuel. It was determined that densification could be practically eliminated by modifying powder and pelletization processes. Primary changes included porosity modification through increased density and sintering temperatures.

Since 1972, appropriate controls on procedures and specifications for the product have reduced in-pile densification to acceptably low values, as shown by the reduction in size and number of gaps in the fuel columns. Fuel densification is considered in core designs by an NRC approved model[5]. Also fuel rods are prepressurized with helium to prevent clad flattening.

3-6.3 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 a Point Beach Unit 1 1976 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 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. The baffle joints on both plants were repaired to eliminate the leakage. Inspection and repair recommendations were made to other affected plants.

During more recent refueling of the KORI Unit 1 and Ringhals Unit 2 reactors in 1979 and the Trojan Reactor in 1980, fuel rod failures attributable to high velocity baffle leakage were again 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. During the 1980 cycle 2/3 refueling of the Trojan reactor, two failed rods due to baffle jetting were found. One of the failed rods occupied an internal core position during Cycle 2, and it is presumed that rod damage occurred when it was located adjacent to the baffle joint during the previous cycle 1. 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.

3-6.4 Startup 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 about 100 refuelings through 1980 without any indications of coolant activity increases attributed to pellet-clad interaction in Westinghouse fuel.

3-6.5 Fuel Rod Bowing

Rod bow has been observed in a large number of fuel assemblies over the past several years. Although no cladding defects of Westinghouse fuel has ever been observed due to rod bow, considerable attention has been applied toward understanding both the causes and possible effects of this phenomenon. 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 eva d for over 1700 fuel assemblies from approximately 70 regions of fuel at assembly burnups up to about 47,000 MWD/MTU. This base of experience represents a very large, 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(6) and additional NRC-Westinghouse correspondence(4,7) 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(7) was issued which essentially eliminated rod bow penalties for 15x15 (14x14) fuel and significantly reduced penalties for

17x17 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 for use in increasing operating capabilities.

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 were confirmed to be within the current Westinghouse licensing basis for rod bow. These data are included in the data base for the September 1979 revised rod bow report. The specifications for tubing characteristics suspected to contribute to rod bow have been or are being tightened.

3-6.6 Fuel Damage During Refueling Operations

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. Examination of this assembly at the conclusion of cycle 4 showed no degradation of fuel rod or assembly structure integrity.

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 of 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.

As a result of grid damage to a number of fuel assemblies in 1979, Westinghouse recommended revised fuel handling procedures in order to minimize the potential for such grid damage. In addition, modifications to the grid corners of all newly fabricated fuel were initiated during 1980 to further reduce the probability of grid damage during fuel handling.

During the October 1980 Cycle 2/3 refueling for Salem Unit 1, one fuel assembly was found to have a small portion of a grid strap torn. Part of the strap was removed and the assembly placed in the Cycle 3 core after evaluations showed satisfactory assembly performance would be expected. A second fuel assembly scheduled for discharge was observed to have a damaged grid. The type of damage on both assemblies indicates that the damage occurred during insertion of these assemblies prior to Cycle 2 startup. The satisfactory performance of these and other fuel assemblies with damaged grids reinserted in Salem is valuable evidence that some degree of damage can be tolerated without attendant fuel rod defects

SECTION 4

CORE COMPONENT EXPERIENCE

4-1 ROD CLUSTER CONTROL (RCC) ASSEMBLIES

Full-length RCCs are being successfully used in the reactors listed in Table 5-4.

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 record 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.

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 them with a trichlorethane-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. These failure mechanisms did not reoccur during the Salem 2/3 refueling and have not been observed at other plants.

4-2 BURNABLE POISON RODS

There are 33 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 and operated without problems.

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 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 pushed into the fuel assembly and captured by insertion of a thimble plug, and the fuel assembly 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 contained between two concentric Zircaloy tubes. The reactor coolant flows inside the inner tubing and outside the outer tubing of the annular rod. The new burnable poison rod reduces the fuel cycle cost due to reduced parasitic neutron absorption in the Zircaloy tubing and an increased water fraction in the burnable poison cell. 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.

4-4 HOLDDOWN ASSEMBLIES

During the April, 1980, refueling of a non-domestic reactor, 32 of the individual helical coil springs used to provide holddown forces on each of the 132 burnable poison, source, and plug assemblies were found to have cracked or broken coils. The springs are mounted on top of the cited core component assemblies within associated fuel assembly nozzles and are exposed to the coolant flow exiting from the fuel assemblies. Holddown force is developed by compression of the springs when the upper coreplate is lowered into place.

The fracture surfaces indicate probable fatigue failure which is concluded to have resulted from flow excitation. The sensitivity to flow appears to have been a function of the barrel shape of the spring which caused the center coils to project into the flow stream. The springs were of a unique design utilized only in the ten plants having an upper head injection (UHI) system.

Failure of 9 of 132 springs of the same design were found subsequently during refueling of a twin unit of the non-domestic reactor. No failures have ever been encountered in holddown springs of a more standard design used in the 32 non UHI plants.

A review of the situation confirmed that these spring failures caused no safety concerns. However, W is cooperating with the utilities involved to minimize further use of springs of the failed design. These springs may be used for no more than one cycle before being replaced by a modified spring design developed to minimize potential fatigue failure.

SECTION 5

OPERATING EXPERIENCE DATA

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

TABLE 5-1

ZIRC FUEL BURNUP STATUS AS OF 12/31/80

ASSEMBLYWISE BURNUP DISTRIBUTION OF
WESTINGHOUSE ZIRCALOY-CLAD FUEL RODS

ACTIVE RODS ****

ASSEMBLYWISE BURNUP *	NUMBER OF RODS
0- 3.9	243552
4- 7.9	69328
8-11.9	79208
12-15.9	104712
16-19.9	98900
20-23.9	89331
24-27.9	116118
28-31.9	65194
32-35.9	30436
36-39.9	995
40---	895

TOTAL RODS

ASSEMBLYWISE BURNUP	NUMBER OF RODS
0- 3.9	243552
4- 7.9	72550
8-11.9	102919
12-15.9	186439
16-19.9	342785
20-23.9	225368
24-27.9	332379
28-31.9	298257
32-35.9	144099
36-39.9	41472
40---	3501

NEWLY DISCHARGED RODS **

ASSEMBLYWISE BURNUP	NUMBER OF RODS
0- 3.9	0
4- 7.9	0
8-11.9	2776
12-15.9	10207
16-19.9	46903
20-23.9	5304
24-27.9	41964
28-31.9	69472
32-35.9	38979
36-39.9	6207
40---	2427

TOTAL DISCHARGED RODS ***

ASSEMBLYWISE BURNUP	NUMBER OF RODS
0- 3.9	0
4- 7.9	3222
8-11.9	23711
12-15.9	81727
16-19.9	243885
20-23.9	136037
24-27.9	216261
28-31.9	233063
32-35.9	113663
36-39.9	40477
40---	2606

*Burnup Units in GWD/MTU = (MWD/MTU)/1000
**Fuel Rods Discharged during Year 1980
***Fuel Rods Discharged from 1969 through December 1980
****Fuel Rods not discharged from operating plants

TABLE 5-2

WESTINGHOUSE FUEL PERFORMANCE STATUS REPORT

(Fourth Quarter 1980)

Reactor	Location	Owner	Date Initial Criticality	Nominal MWe Net	Current Cycle Number	Peak Batch Avg Burnup as of 12/31/80 (MWD/MTU)	Generation, MWh(e) Fourth Quarter 1980	Generation, MWh(e) Cumulative	Percent of Design Basis Activity Release Rate(b)
Jose' Cabrera	Spain	Union Electric S.A.	June 1968	153	10	31,600	306,600	12,155,600	2.8
Beznau 1*	Switzerland	Nordostschweizerische Kraftwerke AG	June 1969	350	10	33,500	776,190	26,256,050	0.75
Mihama 1	Japan	Kansai Electric Power	July 1970	320	2	18,500	374,922(e)	6,629,910	(a)
Point Beach 1	U.S.	Wisconsin Electric Power	November 1970	497	9	36,500	537,170(d)	33,259,720	1.9
Mihama 2	Japan	Kansai Electric Power	April 1972	470	6	29,600	420,735	19,571,588	(a)
Point Beach 2	U.S.	Wisconsin Electric Power	May 1972	497	7	36,100	1,101,040	30,123,850	0.04
Surry 1	U.S.	Virginia Electric Power	July 1972	822	5	24,000	0(e)	31,545,342	(a)
Turkey Point 3	U.S.	Florida Power & Light	October 1972	693	7	33,000	867,510(e)	33,739,272	(a)
Surry 2	U.S.	Virginia Electric Power	March 1973	822	6	32,100	1,759,090	28,584,097	0.08
Zion 1	U.S.	Commonwealth Edison	June 1973	1040	5	36,100	1,861,055(e)	40,261,965	0.63
Indian Point 2	U.S.	Consolidated Edison	May 1973	873	5	34,100	318,570(d)	31,148,356	2.9
Turkey Point 4	U.S.	Florida Power & Light	June 1973	693	7	33,200	589,755(d)	30,248,308	(a)
Prairie Island 1	U.S.	Northern States Power	December 1973	530	6	34,900	1,206,566	23,500,970	≤.01
Zion 2	U.S.	Commonwealth Edison	December 1973	1040	5	37,100	1,866,683	35,656,606	0.14
Kewaunee	U.S.	Wisconsin Public Service	March 1974	535	6	31,900	1,171,600	24,368,771	≤.01
Prairie Island 2	U.S.	Northern States Power	December 1974	530	5	35,100	1,169,080	22,758,940	0.02
Trojan	U.S.	Portland General Electric	December 1975	1130	3	28,300	2,371,146	22,981,360	0.26
Indian Point 3	U.S.	Power Authority of the State of New York	April 1976	873	3	30,500	309,090(e)	22,894,790	(d)
Beaver Valley	U.S.	Duquesne Light	May 1976	852	2	19,300	349,400(e)	8,634,400	(c)
Salem 1	U.S.	Public Service Electric & Gas	December 1976	1090	3	26,900	63,130(d)	16,921,510	(d)
Ko-Ri 1	Korea	Korea Electric Company	June 1977	564	2	25,900	1,084,141	9,024,900	4.2
Farley 1	U.S.	Alabama Power Company	August 1977	829	3	27,300	761,730(d)	14,147,552	0.02
Ohi 1	Japan	Kansai Electric Power	December 1977	1122	2	16,900	1,792,210	10,384,072	(a)
D. C. Cook 2	U.S.	Indiana & Michigan Electric	March 1978	1060	3	28,700	936,090(e)	17,091,830	0.33
North Anna 1	U.S.	Virginia Electric Power	April 1978	907	3	26,600	1,578,895	14,336,921	0.24
Ohi 2	Japan	Kansai Electric Power	September 1978	1122	2	≤ 2,000	71,680(d)	10,167,365	(a)
North Anna 2	U.S.	Virginia Electric Power	June 1980	907	1	< 1,000	967,659(c)	1,189,679	(c)
Sequoyah 1	U.S.	Tennessee Valley Authority	July 1980	1148	1	< 1,000	557,091(c)	557,991	(c)
Ringhals 3	Sweden	Swedish States Power Board	July 1980	912	1	< 1,000	301,172(c)	381,523	(c)
Salem 2	U.S.	Public Service Electric & Gas	August 1980	1115	1	< 1,000	(a)	(a)	(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 inference from short periods of operation during period of report.

d. Plant refueling during period of report.

e. Maintenance and Inspection or Repair during period of report.

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, 1980

Plant	Number of Full-Length RCCs/Core	Time From Cycle 1 Start (months)
San Onofre	45	164
Connecticut Yankee	53	163
Jose Cabrera	17	151
Beznau 1	25	139
Ginna	29	134
Mihama 1	29	126
Robinson 2	45	135
Point Beach 1	33	122
Beznau 2	25	111
Mihama 2	29	105
Point Beach 2	33	105
Surry 1	48	103
Turkey Point 3	45	99
Surry 2	48	84
Indian Point 2	53	92
Turkey Point 4	45	91
Zion 1	53	91
Prairie Island 1	29	86
Zion 2	53	85
Takahama	48	82
Kewaunee	29	82
Ringhals 2	48	78
Prairie Island 2	29	73
D. C. Cook 1	53	72
Trojan	53	61
Indian Point 3	53	56
Beaver Valley 1	48	55
Salem 1	53	49
Ko-Ri 1	29	42
J. M Farley	48	40
Ohi 1	53	37
D. C. Cook 2	53	32
North Anna 1	48	33
North Anna 2	48	8
Sequoyah 1	53	1
Ringhals 3	53	5
TOTAL 36 plants	1536	3002

TABLE 5-5

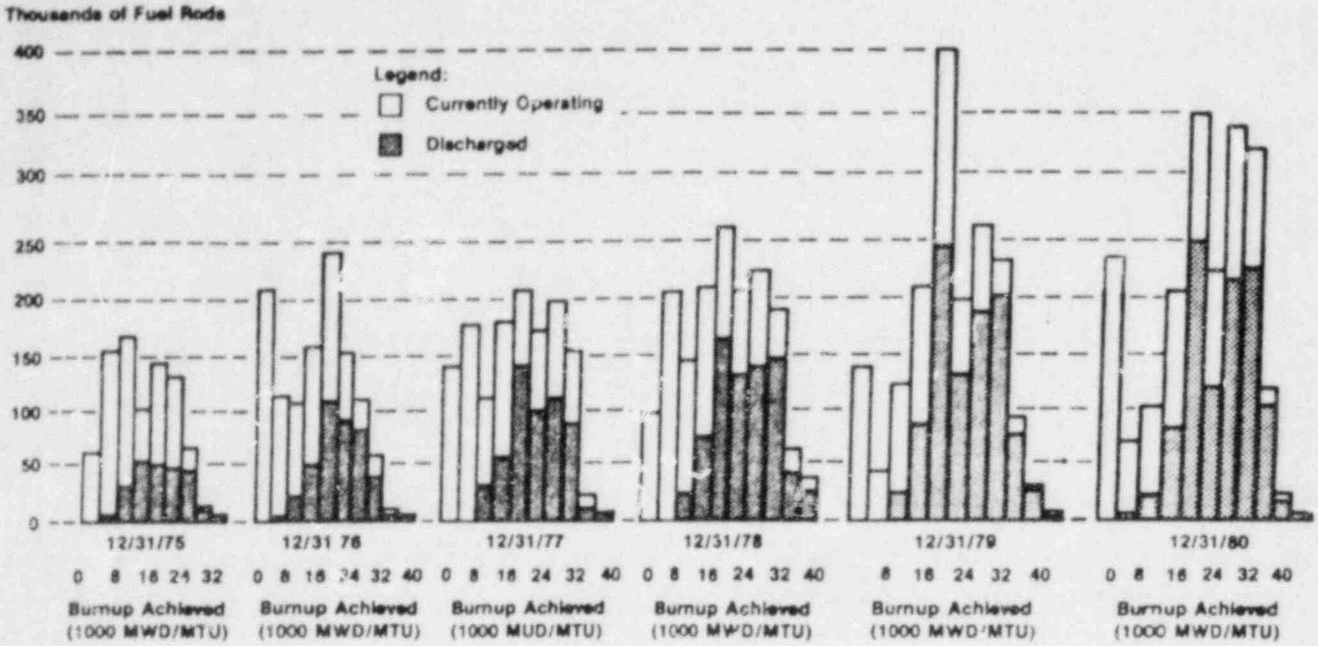
SUMMARY OF RCC AND DRIVE LINE PROBLEMS

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



Average Burnups						
All Fuel	14,300	15,100	17,300	18,800	20,000	20,600
Discharged Fuel	19,800	20,700	22,200	23,700	24,000	24,500

Figure 5-1 Burnup Performance of Westinghouse Zircaloy-Clad Fuel

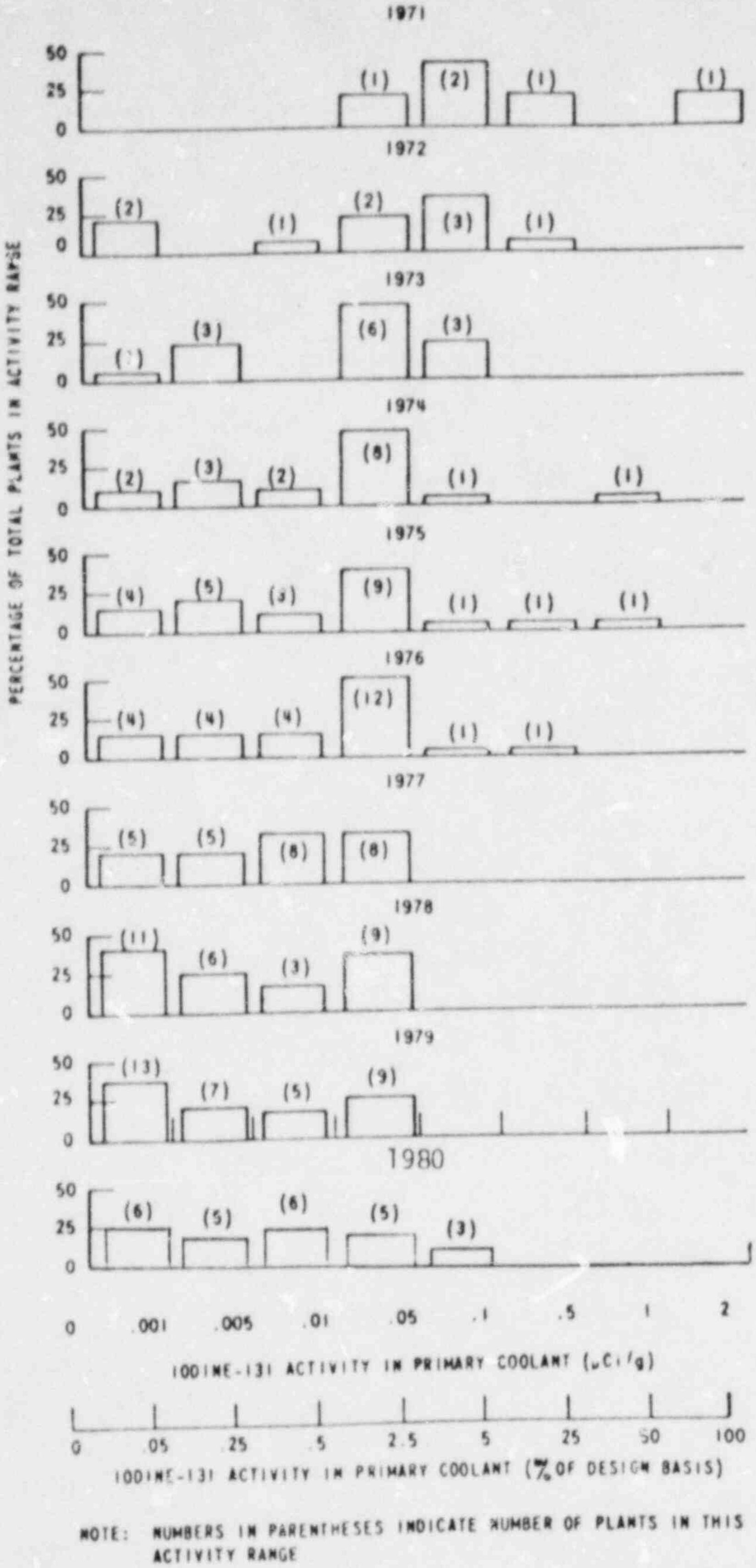


Figure 5-2. Westinghouse Fuel Performance — Iodine 131 Activity in Primary Coolant as of December 31, 1980

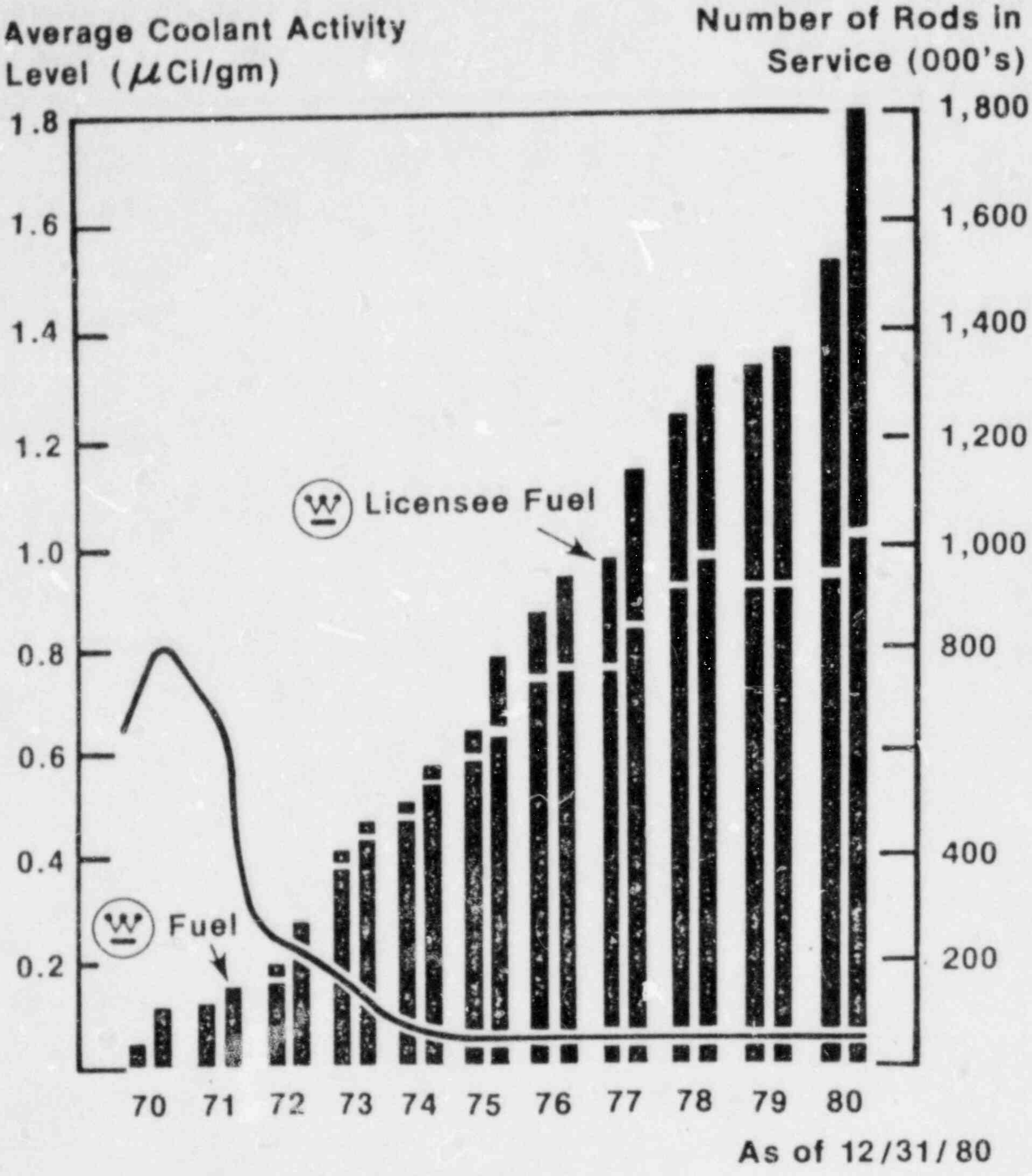


Figure 5-3 Performance Record - Average Coolant Activity Level/Number of Rods in Service

SECTION 6

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