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OPERATIONAL EXPERIENCE WITH WESTINGHOUSE CORES
(Through December 31, 1984)

J. Skaritka

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Approved: W. F. Staley, Manager
Engineering Services

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear Fuel Division
P.O. Box 3912
Pittsburgh, Pennsylvania 15230

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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 through December 31, 1984. This report, revised annually, is used for licensing purposes as a supporting document for safety analysis reports. 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. Fuel performance and generic concerns common to a number of plants, along with actions taken to resolve these concerns, are summarized in Section 4. Section 5 discusses and evaluates other core component experience.

The data on which the overview is based are given in Section 6.

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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 of 1968. Since then, over 3.5 million Zircaloy-clad fuel rods have seen in-reactor duty. The acceptance of Westinghouse nuclear fuel by the world-wide utility industry is demonstrated by the successful operation of fifty-two commercial PWRs which have used Westinghouse supplied Zircaloy-clad fuel. The burnup experience of Westinghouse PWR fuel includes 14x14, 15x15, 16x16, and 17x17 fuel assembly designs. Twenty-four plants were operating with the 17x17 fuel design during 1984.

Westinghouse experience during the past few years indicate a significant advance in the maturity of the Westinghouse fuel design. Westinghouse supplied PWR fuel has generated 159.1 million MWD (thermal) energy through 1984. More than 120 regions of fuel have been discharged after three cycles of operation, resulting in an increased number of fuel rods discharged and operating in higher burnup ranges. A total of 686 assemblies with assembly average burnups greater than 36,000 MWD/MTU, of which 75 had burnups greater than 40,000, provides direct evidence of the capability of Westinghouse fuel to operate at extended burnups.

During 1984, twenty-one plants were refueled with Westinghouse fuel, and three plants started initial commercial power operation. At the end of 1984, a total of 1,488,604 Zircaloy-clad fuel rods were in operation, representing 6266 fuel assemblies. The addition of discharged fuel brings the total number of irradiated Westinghouse Zircaloy-clad fuel rods to 3,581,675. This represents a total of 16,134 fuel assemblies containing 7361 metric tons of UO_2. The average burnup of the discharged fuel is 26080 MWD/MTU, and the average burnup of all Westinghouse fuel (in-core plus discharged) is 21910 MWD/MTU. Table 6-1 presents a burnup summary of Westinghouse fuel rods discharged and still in use through the end of 1984. As shown, there are a significant number of fuel rods which have burnups greater than 40,000 MWD/MTU.

Table 6-2 presents a Westinghouse fuel performance summary on a plant-by-plant basis. The peak burnup data in Table 6-2 shows the highest region average burnup for each plant after Cycle 1 startup. Many plants have fuel region burnups in the range of 32,000 to 37,000 MWD/MTU. Also tabulated are maximum coolant activities during 1984 for a number of plants.

The highest burnups of discharged fuel assemblies are in the range of 39,000-55,000 MWD/MTU. As part of an EPRI high burnup program, four high burnup demonstration assemblies were discharged from the Zion Unit 1 Cycle 6 core at an average burnup of 55,000 MWD/MTU. Zion Units 1 and 2, Point Beach Units 1 and 2, and Prairie Island Unit 2 have discharged regions of fuel with average burnups in the range of 36,000 to 37,500 MWD/MTU. Indian Point Unit 2 has discharged fuel with a batch (12 assemblies) average burnup of 37,200 MWD/MTU. Coolant activities at these plants have remained low, and no correlation between extended burnups and increased coolant activities has been observed. This experience, and additional details given in Section 3.0, demonstrates that Westinghouse fuel is capable of successful operation to assembly average burnups of 55,000 MWD/MTU.

Significant 17x17 fuel assembly burnup experience has been obtained with six plants completing three or more cycles of operation. Region average burnups up to about 35,000 MWD/MTU have been achieved.

Ten demonstration Optimized Fuel Assemblies (OFAs), six 17x17 and four 14x14 assemblies, have been examined after achieving assembly burnups of up to 39,170 MWD/MTU. All assemblies were in good condition except for one 14x14 OFA (see Section 3.2.2).

From late 1983 through 1984, a number of plants commenced irradiation of 14x14, 15x15 or 17x17 OFA reloads, and one plant started irradiation of a full core of 17x17 OFAs (Section 3.2.2). First cycle region average burnups up to 14,400 MWD/MTU were attained by the end of 1984. During 1985, a number of first cycle OFA visual examinations will be performed, and the second cycle irradiation of OFAs will commence.

With the exception of fuel rod fretting failures caused by debris in the reactor coolant (see Sections 3.2.1 and 4.5), the in-pile performance of Westinghouse fuel during this report period was very good. The average coolant activity of the plants operating with Westinghouse fuel dropped from an initial level of 0.030 micro curies per gram at the end of 1983 to 0.008 curies per gram at the end of 1984, a 70% reduction.

In summary, a significant amount of electricity has been generated by Westinghouse fuel, achieving fuel region burnup levels up to 37,500 MWD/MTU. Nuclear reactors now operating with Westinghouse Zircaloy-clad fuel have not experienced availability limitations due to fuel defects. This excellent record is supported by numerous onsite and hot cell examinations designed to confirm the adequacy of current fuel designs and to develop design improvements. This fuel performance record provides considerable evidence for the continued reliability of Westinghouse fuel assemblies.

SECTION 3

FUEL EXPERIENCE OVERVIEW

3.1 BURNUP EXPERIENCE HIGHLIGHTS

The total Westinghouse fuel burnup experience from 1968 through 1984 includes 16,134 fuel assemblies, accounting for approximately 3.6 million irradiated fuel rods. Table 6-2 contains a list, in chronological order of plant startup, of plants in which extensive operating experience has been obtained with Westinghouse Zircaloy-clad rods. Taken as an aggregate, the data indicates the extent of this experience. Figures 6-1 and 6-2 contain graphic representations of the fuel burnup data and Zircaloy-clad experience from 1974 through 1984.

Although much of the fuel is still operating at the lower burnup ranges (i.e., in the first and second cycles of irradiation) more than 650 Westinghouse fuel assemblies and approximately 141,200 fuel rods have achieved assembly average burnups in excess of 36,000 MWD/MTU. Higher burnups in excess of 40,000 MWD/MTU were achieved on 75 assemblies and approximately 14,500 fuel rods. Four assemblies irradiated in Zion Units 1 and 2 achieved an extended burnup of 55,000 MWD/MTU with a peak fuel rod burnup of 60,000 MWD/MTU. Sipping tests of these assemblies after 5 cycles detected no leaking fuel rods. Also, on-site post-irradiation visual examinations and later detailed hot cell examinations showed that the assemblies were in good condition with no fuel rod and assembly structure material anomalies. Zion Units 1 and 2, Point Beach Units 1 and 2 and Prairie Island Unit 2 have discharged Westinghouse fuel with region average burnups in the range of 36,000 to 37,500 MWD/MTU, and coolant activities have remained low at these plants.

3.2 FUEL PERFORMANCE

Fuel performance is monitored by coolant activity levels during reactor operation and by on-site examinations at select locations during refueling. The iodine-131 activity in the coolant is reported in terms of average coolant activity level (µCi/gm). 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 is typical and can be used for purposes of comparison. Figures 6-3 show the yearly change (December basis) of coolant activity distributions for Westinghouse-fueled plants. In order for all plant data to be on a common comparison activity basis for Figure 6-3, all plants with an increased coolant letdown rate have their coolant activity normalized to a single letdown rate. In all cases the activity levels were below the maximum allowable coolant activity levels as set forth in the technical specifications. Figure 6-3 shows a small increase in coolant activity levels from 1980 to 1983. This increase is believed due to fuel rod fretting failures caused by baffle jetting and debris in the coolant (See Sections 4.4 and 4.5). Increased coolant activities were not related to fuel manufacturing or reactor operating history. In 1984, there was a 70% reduction in average coolant activity in the plants operating with Westinghouse fuel. The reported coolant activity levels, combined with the burnup experience of Figure 6-1 and Table 6-2, provide substantial proof of the reliability and good performance of Westinghouse Zircaloy clad fuel.

The performance of Westinghouse fuel in reactors with high average linear power ratings (e.g., Zion Units at 6.7 kw/ft) have shown that high power ratings have no measurable effect on increasing clad defects or coolant activity levels.

Over the years, onsite fuel examinations at selected sites have been performed to examine and evaluate the performance of Westinghouse fuel. The examinations included visual exams, dimensional and leak testing of both the fuel assemblies and individual fuel rods. For most plants, the fuel examinations revealed no evidance of damage to fuel rods, grids, thimble tubes, nozzles or holddown springs. For those plants where damaged fuel was observed, details are given in Sections 3.2.1, 3.2.2, and 4.0.

3.2.1 Fuel Assembly Abnormalities - 1984

During routine refueling outages between 1981 and 1984, fuel rod failures have been observed in several plants. These can be characterized in terms of hydriding, fretting, and top end plug weld failures. During the 1983

refuelings of Farley Unit 1, Surry Unit 1, Millstone Unit 2, Point Beach Unit 1, Point Beach Unit 2 (Section 3.2.2) and Turkey Point Unit 3 a total of about 90 Westinghouse fuel assemblies with leaking fuel rods were discovered. The evidence indicated that most failures in four plants (84 affected assemblies) were due to fretting wear of the clad (see Sections 3.2.2, 4.4, 4.5).

During 1984 refueling outages, Ko-Ri Unit 1, Farley Unit 1, D.C. Cook Unit 2, North Anna Unit 1, North Anna Unit 2, Salem Unit 1 and Surry Unit 1 were determined to have 46 fuel assemblies with leaking rods. Debris-induced fretting was determined to be a primary cause of fuel rod failures in the 30 fuel assemblies from Ko-Ri Unit 1 (one leaker), Surry Unit 1 (8 leakers), and North Anna Units 1 (17 leakers) and 2 (4 leakers). Additional details are given in Section 4.5. Farley Unit 1 (Cycle 5/6 refueling) had five leaking assemblies, and Cook Unit 2 (Cycle 3/4) had nine leaking assemblies. Both plants had fuel rod hydride failures from undetermined causes.

During late 1984 refuelings, it was observed that one irradiated assembly from Surry Unit 1 and one from Point Beach Unit 2 had failed top nozzle holddown spring clamp screws. The affected spring clamps became detached and were lying inside the assembly top nozzles. For Surry Unit 1, the spring clamp impeded the movement of an RCCA which became jammed in the associated fuel assembly during operation. A detailed investigation showed no structural damage of the RCCA and no safety hazard since the plant is designed to operate safely with a stuck RCCA. Also, there is a low probability of additional screw failures. Examinations showed failure of the clamp screws was due to stress corrosion cracking of the Inconel Alloy 600 material. The potential for this type of failure exists for 14x14 OFA, 14x14 LOPAR and 15x15 LOPAR fuel assemblies. To eliminate the possibility of this type of failure in future 14x14 and 15x15 fuel assemblies, clamp screws previously made with Inconel 600 will be fabricated from Inconel X-750 material beginning in early 1986.

During the 1984 refueling of Salem Unit 1 two assemblies were determined to have fuel rods with fretting marks, and each assembly had a leaking fuel rod. These assemblies were rejected for use in the next cycle. The leaking fuel rods and adjacent rods with fretting marks occurred at core locations adjacent

to another assembly which had torn and missing grid straps on two sides of the assembly. Evidence shows that the leaking fuel rods and marks in the fuel assemblies were caused by fretting wear with the adjacent torn grid and its debris. The assembly with the damaged grid, which was being normally discharged, also had evidence of two rods with clad failures.

3.2.2 Optimized Fuel Assemblies

Two demonstration 17x17 Optimized Fuel Assemblies (OFA) were placed into each of three reactors, Farley Unit 1, Salem Unit 1 and Beaver Valley Unit 1. Four demonstration 14x14 OFAs were placed into the Point Beach Unit 2 reactor. The 14x14 and 17x17 OFA employ a slightly reduced fuel rod diameter compared to the non-OFA fuel rod, while retaining the same fuel rod pitch. Also, for the intermediate grids, the OFAs use Zircaloy grids compared to Inconel grids used in non-OFAs. The top and bottom OFA grids retain the same non-OFA Inconel grid design. These changes result in a significant improvement in fuel efficiency by improving neutron moderation and reducing parasitic capture.

On-site post-irradiation examinations of the OFAs were performed following the completion of each cycle of irradiation. The demonstration OFAs employed the removable rod feature typical of Westinghouse demonstration assemblies, which permits the removal of 88 (eighty-eight) interior rods for individual fuel rod examination. These examinations included visual and dimensional inspections of the fuel assemblies and interior removable rods. The demonstration assemblies in Salem Unit 1 and Farley Unit 1 were discharged in 1984 after four cycles of irradiation and achieved burnups of 33,850 and 39,170 MWD/MTU respectively. The fuel assemblies were found to be in good condition. The two Farley OFAs have subsequently been re-inserted in Farley Unit 1 Cycle 7 for irradiation to a burnup of about 52,000 MWD/MTU.

The two demonstration assemblies in Beaver Valley Unit 1 completed their third cycle of irradiation and were discharged in 1984 with a burnup of 35,200 MWD/MTU. Post-irradiation examinations showed the assemblies to be in excellent condition.

The four demonstration 14x14 OFAs in Point Beach Unit 2 completed their second cycle of irradiation in 1983 with an average burnup of 21,000 MWD/MTU. Post-irradiation examination of these assemblies showed that three of the four assemblies were in good condition and one assembly had nine failed fuel rods. The primary cause for failure was fretting wear at the bottom Inconel grid. Secondary hydriding was also observed at other axial locations in this assembly, with blisters and cracks in the cladding and a defected top end plug weld. The fretting wear was attributed to abnormally low grid spring-to-fuel-rod contact forces caused by bottom grid damage during fabrication. The failure mechanism was unique to the one failed assembly and has no generic implication of the OFA design. The manufacturing technique was corrected before the last three demonstration assemblies were fabricated. The satisfactory performance of the later assemblies confirms that the change in manufacturing procedure has precluded this type of failure from occurring again. The three non-failed assemblies were returned to the core and in 1984 completed a third cycle of irradiation with an average burnup of 36,300 MWD/MTU. The assemblies were in good condition. They are being irradiated for a fourth cycle to an expected burnup discharge of 40,000 MWD/MTU.

The initial reloads of 15x15 OFAs were introduced into the Cook Unit 1 Cycle 8 Core (criticality 10/83) and the Turkey Point Unit 3 Cycle 9 core (criticality 12/83). During 1984 the following additional plants began irradiating OFAs: Zion Units 1 and 2 (15x15 reloads), Turkey Point Unit 4 (15x15 reload), McGuire Unit 1 (17x17 reload), Maanshan Unit 1 (full core of 17x17), R. E. Ginna (14x14 reload) and Point Beach Unit 2 (14x14 reload). Substantial first cycle OFA performance data will be available during 1985.

3.2.3 VANTAGE 5 Assemblies

VANTAGE 5 is an improved Westinghouse PWR fuel assembly product. The VANTAGE 5 assembly has the same optimized fuel rod and Zircaloy grids as an OFA. VANTAGE 5 improves upon OFA by incorporating product features which reduce fuel cycle cost, increase core operating margins, and improve design and operating flexibility. VANTAGE 5 product features include (1) integral fuel burnable absorbers (IFBA), (2) intermediate flow mixers (IFM), (3) axial blankets, (4) increased discharge burnup, and (5) a reconstitutable top nozzle. The NRC has approved usage of the VANTAGE 5 fuel assembly.

In order to demonstrate the VANTAGE 5 design product features in a commercial reactor, four VANTAGE 5 demonstration assemblies (17x17 rod array) were loaded into the V.C. Summer Unit 1 Cycle 2 core and began power production in December 1984. In late 1985 the assemblies are expected to complete their first of three cycles of irradiation and will be subjected to post-irradiation examinations.

Each of the Summer demonstration assemblies contain 40 IFBA rods. In an IFBA rod, a thin boride burnable absorber coating is applied to the surface of the fuel pellets for about 80% of the fuel stack. IFBA demonstration fuel rods are currently being irradiated for their first cycle of operation in Turkey Point Units 3 and 4. Unit 3 has four IFBA rods which are being monitored during irradiation by in-core instrumentation. Unit 4 has 28 IFBA fuel rods in each of four demonstration assemblies which allow removal of some of the rods for post-irradiation examinations.

The IFM grids are small mixing vane grids which are located in the upper parts of the fuel assembly between the Zircaloy grids. The IFM grid typically increases the DNB margin by 25% over the OFA. During 1984, a characterized IFM demonstration assembly was irradiated in the McGuire Unit 1 Cycle 2 core.

3.2.4 Mixed Oxide Assemblies

Four fuel assemblies with Westinghouse mixed oxide fuel rods were irradiated for their third cycle of operation in the R. E. Ginna Cycle 13 core to a burnup of 29,650 MWD/MTU. These assemblies are being irradiated for a fourth cycle and are expected to be discharged with an average burnup of about 39,000 MWD/MTU. The Cycle 11 thru 14 cores, which contain the mixed oxide assemblies, have not shown any abnormal coolant activities through the end of 1984. This operation supplements earlier Westinghouse mixed oxide fuel experience in San Onofre, Beznau, and the Saxton Plutonium Program.

SECTION 4.0

GENERIC FUEL CONCERNS

The principle concerns discussed below are fuel rod fretting due to baffle jetting and debris in the reactor coolant, moisture and clad hydriding, ramp rate effects on fuel performance, fuel rod bowing and fuel assembly grid damage during refueling operations. No fuel rod clad failures due to corrosion have ever been observed in a Westinghouse fueled plant. Out-of-pile loop tests have shown that adverse crud deposition or corrosion can be avoided by appropriate control of reactor coolant chemistry. Fuel densification has not been a generic concern for operating fuel since the mid-seventies. 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[2].

4.1 MOISTURE AND HYDRIDING

In the early 70's, coolant activity indicated a number of cladding defects in the Beznau Unit 1 and R. E. Ginna reactors. Visual examinations during refueling indicated cladding defects caused by local hydriding. 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, clad hydriding failures due to as-fabricated fuel pellet moisture content have been practically eliminated. In the past few years hydride blisters in cladding has been a common observation in fuel failed due to fretting (Sections 3.2.1, 4.4, and 4.5). These blisters are considered to be mainly secondary hydriding failures caused by the primary clad fretting failures. A smaller number of hydriding failure have been observed from undetermined causes (Section 3.2.1). In order to ensure elimination of primary hydriding as a failure mechanism, additional manufacturing changes were made to further reduce hydrogen levels in fuel rods.

4.2 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 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. Prior to 1982 these restrictions applied only during the initial startup of a reload cycle; in 1982 these restrictions were also recommended for first core operation. 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 over 170 refuelings through 1984. Where these startup procedures have been followed, review of available data indicates there have been no coolant activity increases attributed to startup ramp rate effects on the fuel.

4.3 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 have 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 to predict the extent of rod bow that will be experienced during operation. Rod bow observations have been evaluated for over 1700 fuel assemblies from over 70 regions of fuel at assembly burnups up to about 47,000 MWD/MTU. This base of experience represents a very large, statistical sample around which future operating behavior may be assessed.

A revised fuel rod bow topical report was submitted for NRC review in September 1979. The report develops revised rod bow correlations for Westinghouse fuel. The bow correlations are used with the NRC approved partial rod bow function in a statistical manner to evaluate rod bow DNBR effects. The NRC has approved this report[3], and additional DNBR margins are available for Westinghouse fuel for use in increasing operating capabilities.

During 1984, 17x17 OFA demonstration (6 assemblies) showed rod bow data somewhat larger than the 17x17 assembly data used in the rod bow topical[3]. Due to improvements made in grid design and fabrication for production OFAs, it is expected that regions of OFAs being irradiated will have rod bows less than the of the OFA Demos. The NRC was informed of this information and concurred with the Westinghouse conclusion.

4.4 FUEL FRETTING DUE TO BAFFLE JETTING

Since 1971 baffle jetting has caused fuel rod failures in peripheral assemblies in a number of plants. The cause of the damage has been identified as high-velocity coolant crossflow through joints in the core baffle. The crossflow caused excessive rod vibration and eventual fretting through the cladding in the grid support areas.

Although not a fuel deficiency per se, this resulted in significant fuel damage in six plants: Jose Cabrera (1971), Point Beach Unit 1 (1976), Ko-Ri Unit 1 (1979, 1981, 1982), Ringhals Unit 2 (1979), Trojan (1980, 1982) and Farley Unit 1 (1983).

A variety of corrective actions were undertaken which included conversion of plants from down flow to up flow in the core barrel/baffle annulus to reduce the pressure drop across the baffle and modification of fuel assemblies in select baffle locations. The assembly modifications included the use of dummy stainless steel rods in place of fuel rods in the areas of greatest flow impingement and, for some plants, the use of small sections of grids (partial grids) located at select mid-span locations to increase resistance for rod vibration and fretting.

In 1983, the modified fuel assemblies were examined at the Trojan and Ko-Ri Unit 1 refuelings. Modified assemblies at Trojan were found to be in excellent condition with no evidence of damage. The modified assemblies in Ko-Ri were in good condition, except for one assembly which had two failed rods located in the second row behind the stainless steel rods.

Evidence of fuel damage due to baffle jetting was observed in Farley Unit 1 in 1983, where eleven assemblies were observed to be damaged. These assemblies were discharged. In order to preclude further fuel damage due to baffle jetting, the plant was converted to upflow at the core baffle.

During a 1984 refueling shutdown, the Trojan plant was converted to upflow at the core baffle to eliminate fuel failures due to baffle jetting.

4.5 FUEL FRETTING DUE TO COOLANT DEBRIS

Evidence of fuel failures due to debris-induced fretting has been found in six plants (Surry Unit 1, Millstone Unit 2, Point Beach Unit 1, Ko-Ri Unit 1 and North Anna Units 1 and 2). The coolant in these plants contained metallic debris and all have had relatively high coolant activity levels. Fuel rod fretting failures can occur as a result of metallic debris from reactor repair operations being carried by coolant into the fuel rod coolant channels. Debris carried into the fuel channels can become trapped and can vibrate and wear through the fuel rod cladding.

Of about 115 failed assemblies, one of the plants mentioned above had fifty-one assemblies with failed fuel rods during a 1983 refueling and an additional 8 leakers during a 1984 refueling. A large quantity of metallic debris was observed in the fuel assemblies during on-site examinations at this plant. Many of the failed rods had hydride defects, and examinations showed that most of the rods failed due to debris - induced fretting. Utilities have been informed of this potential damage mechanism, and Westinghouse has recommended appropriate precautions to be taken to keep the coolant system as clean as possible during repair/maintenance operations.

4.6 ASSEMBLY DAMAGE DURING FUEL HANDLING OPERATIONS

Instances of fuel assembly grid strap damage due to fuel handling operations were observed during refueling operations at a number of plants. A detailed summary of these events through 1982 is given in a previous core experience report.(4)

Since the implementation of a two-phase grid corner design modification (the first in 1980 and the second in 1982) and revised Westinghouse fuel handling procedures (revised in 1982 to better define fuel assembly load deflection limits, restrictions in fuel assembly contact with adjacent surfaces, limits on load changes that occur during fuel movement and recommended fuel handling practices), grid damage occurring during in core fuel handling has been greatly reduced. In all isolated incidents occurring subsequent to these design and procedural modifications, grid damage was sustained only on those assemblies fabricated prior to the implementation of the modified grid corners. As the number of damaged fuel assemblies from all affected plants are slowly discharged, the chances of recurring damage in the future will be minimized. During a 1984 refueling of Salem 1, a torn grid with missing straps was observed in one assembly being normally discharged; the damaged grid resulted in fuel rod fretting wear and leaking rods in this assembly and two other adjacent assemblies (see Section 3.2.1).

During a December 1981 spent-fuel shuffle operation at the Prairie Island Plant Site, the top nozzle of a Unit 1 Region 4 fuel assembly separated from the remainder of the assembly while being transferred between storage cells.(5) The separation occurred at the uppermost bulge joint in each of the 16 stainless steel sleeves which join the top nozzle to the guide thimble tubes. The Zircaloy control rod guide thimble tubes were intact and unaffected by the failure. Hot cell examinations indicate that the bulge joint failures were the result of intergranular stress corrosion cracking in the vicinity of the bulge. This type of failure in Westinghouse fuel assemblies has not been observed at other nuclear plants where more than 12,000 Westinghouse fuel assemblies have been handled without similar

occurrences. There were no fuel rods damaged during the incident. The failed assembly had been irradiated for three cycles to a burnup of 29,000 MWD/MTU and had been discharged from the reactor in April 1979. This type of failure is considered an isolated occurrence.

4.7 OTHER FUEL DAMAGE

During the 1982 Cycle 5/6 refueling shutdown for Indian Point Unit 2, a visual examination showed a total of six failed fuel rods in five fuel assemblies. Three rods exhibited both hydriding and the separation of the top end plug. Two (2) rods exhibited hydriding only and one rod exhibited a detached end plug. One additional assembly is suspected as having one failed rod as evidenced by white areas possibly due to hydriding. It has not been determined whether the hydriding was the primary cause of rod failures or was a secondary event occurring after rods failed due to other causes (e.g. end cap weld failure). With the exception of one assembly, all assemblies exhibiting fuel rod failures were scheduled for discharge.

SECTION 5

CORE COMPONENT EXPERIENCE

5.1 ROD CLUSTER CONTROL (RCC) ASSEMBLIES

Full-length RCCs are being successfully used in over 45 reactors. Very little difficulty has been experienced with the large numbers of control rods in use, despite the fact that they are often 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.)

Several problems have occurred which involved rod cluster control assemblies and drive lines. The problems, causes, and solutions are summarized chronologically in Table 6-3.

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 tri-chlorethane-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.

During the 1983 Cycle 9/10 and 11/12 refueling outages at Point Beach Units 2 and 1 respectively, RCCAs were visually examined for absorber rod cladding wear. The visual examinations were conducted with an underwater periscope. Based on the visual wear results, one Unit 2 RCCA and 21 Unit 1 RCCAs were replaced. Subsequent 1984 examinations of the discharged RCCAs have shown cracks at the rodlet tips for two rodlets from Unit 1 and one rodlet from Unit 2. Investigations are continuing to identify the cause of cracks in the rodlet tips.

Axial repositioning of the normal "parked" position of the remaining Point Beach RCCAs by 2-3 steps have been implemented. Repositioning of the "parked" position will distribute the wear more evenly along the length of the absorber rod cladding. Wear observations indicated two types of wear patterns which are characterized as follows: (1) Axially continuous wear of varying depth caused by RCCA axial movement (i.e., stepping or scramming), (2) Intermittent "fretting" wear located at elevations corresponding to the guidance plates in the reactor upper internals RCCA guide tube. These wear indications were predominately oriented on that portion of the cladding which faces the center of the RCCA. By design this type of RCCA cladding wear is expected, due to contact between the RCCA absorber rods and the guidance surfaces. This type of wear (and cause) were observed on 3 RCCAs during the 1984 Cycle 9/10 refueling shutdown of the Kewaunee Reactor. These RCCAs were accepted for the next cycle since their clad wear did not exceed the Westinghouse wear criteria. The RCCAs were positioned at an alternate parked position for Cycle 10 operations to prevent excessive wear depth.

During Surry Unit 1, Cycle 7 operations, an RCCA became stuck at 56 steps during June 1984. A safety analysis and Technical Specification changes permitted Cycle 7 operation to continue until the planned refueling shutdown. Site examinations showed that the stuck RCCA was caused by failed fuel assembly spring clamp screws (see Section 3.2.1). The RCCA had no structural damage.

5.2 BURNABLE ABSORBER ASSEMBLIES

There are about 50 plants which are using or have used Westinghouse burnable absorber (BA) assemblies which contain borosilicate glass as the BA. Most of these assemblies are intended to serve for only one cycle and are not removed during service.

Although there is no routine surveillance of BA 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 R. E. Ginna Cycle 3/Cycle 4 refueling, several cracks/openings were noticed in the cladding at the lower end of an associated burnable absorber 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. The source assembly was reloaded for an additional cycle of operation and operated without problems.

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

During the Cycle 4/5 refueling at Beznau Unit 1, a BA rod and vane were separated from a secondary source assembly and were stuck in the fuel assembly guide thimble tube. The vane was cut off the stuck BA rod at the BA rod flexure and the fuel assembly was used for one additional cycle with a modified plugging device and operated without incident. The secondary source with the missing vane and BA rod was reinserted into another fuel assembly for Cycle 5, which was scheduled for discharge following the cycle. During the Cycle 5/6 refueling, two additional BA rods from this same secondary source were stuck in the fuel assembly guide thimble tubes. One was completely broken off and the other was broken off 18 inches from the bottom of the rod. The fuel assembly containing the two broken BA rods was discharged along with the secondary source assembly. The cause of these stuck BA rod incidents has not been determined.

During February 1978, after two duty cycles of operation at Indian Point Unit 2, one BA rod which was broken 24 inches from the top was found stuck in the thimble tube of the fuel assembly. A thimble plug was inserted in the assembly to capture the unrecoverable piece of the BA rod. The fuel assembly functioned without incident during the third duty cycle.

Four demonstration assemblies, each containing two precharacterized Wet Annular Burnable Absorber (WABA) rods, were inserted into the Indian Point Unit 3 Cycle 3 core for irradiation starting in early 1980. The WABA rod design contains annular pellets of aluminum oxide-boron carbide burnable absorber 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 WABA rod reduces the fuel cycle cost due to reduced parasitic neutron absorption in the Zircaloy tubing and an increased water fraction in the WABA cell. The four WABA demonstration assemblies were examined during the Cycle 3/4 refueling outage, following their first cycle of irradiation. Results from these examinations indicated that the precharacterized WABA rods have performed as originally expected. No anomalous conditions were observed due to operation; however, one of the demonstration assemblies was mishandled during the fuel assembly changeout, which resulted in a broken WABA rod. Due to this handling error only two of the four demonstration assemblies were reinserted for a second cycle of irradiation. It is planned to re-examine the four WABA rods in the two assemblies following completion of their second cycle during 1985.

5.3 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 1978 there were two instances of stuck secondary source/depleted burnable absorber assemblies during refueling. During the Zion Unit 1 Cycle 3/4 Fall 1978 refueling, a secondary source assembly could not be removed from its fuel assembly. In 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.

During the Ko-Ri Unit 1 Cycle 3/4 refueling, a secondary source assembly could not be removed from its fuel assembly. After visual examination of the source assembly, spider, and fuel assembly, no apparent anomalies were noticed, and the fuel assembly with the stuck source was reinserted into the core for Cycle 4.

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 in 1978. This assembly was not reused for Cycle 3.

5.4 HOLDDOWN SPRINGS

During the April 1980 refueling of a nondomestic reactor, 32 of the individual helical coil springs used to provide holddown forces on each of the 132 burnable absorber, 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 indicated probable fatigue failure which was 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.

Failures of 9 of 132 springs of the same design were found subsequently during refueling of a twin unit of the nondomestic reactor. No holddown spring failures have ever been encountered in the 32 non-UHI plants which do not use springs with the barrel shape design.

A review of the situation confirmed that these spring failures caused no safety concerns. Core components utilizing these springs are being replaced during refuelings by a modified spring design developed to minimize potential

fatigue failure. These modified springs have been retrofitted on unirradiated core components, previously shipped to sites with the old barrel shape spring design.

During the mid-Cycle 1 maintenance outage at McGuire Unit 1 (17x17 UHI), fractured holddown springs were discovered on 21 of 94 burnable absorber assemblies scheduled for discharge at the end of Cycle 1. These failures were similar to those observed at three other 17x17 UHI reactors; however, three of the failed springs were fractured in two locations. In all three instances, all spring segments were retained and captured by their respective holddown assemblies, and no additional damage was observed. All holddown assemblies with the suspect spring design were discharged at this outage prior to start-up and were replaced with assemblies having the modified spring design.

SECTION 6

OPERATING EXPERIENCE DATA

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

TABLE 6-1

ZIRCALOY-CLAD FUEL BURNUP STATUS THROUGH 1984
ASSEMBLYWISE BURNUP DISTRIBUTION
OF WESTINGHOUSE ZIRCALOY-CLAD FUEL RODS

ZIRC ACTIVE ROD BURNUP STATUS AS OF 12/31/84 **

ASSEMBLYWISE * BURNUP	14X14 RODS	15X15 RODS	16X16 RODS	17X17 RODS	TOTALS RODS
0.- 3.9	15752	15504	11280	167376	209912
4.- 7.9	12100	28560	36425	125400	202485
8.-11.9	24152	36720	4700	128832	194404
12.-15.9	8771	51000	8225	169488	237484
16.-19.9	19642	22644	15745	143088	201119
20.-23.9	12601	32028	3290	82368	130287
24.-27.9	18974	38352	3760	75504	135590
28.-31.9	9579	39984	1880	44352	95795
32.-35.9	13165	23052	0	24816	61033
36.-39.9	2664	12444	0	2376	17684
40.-43.9	179	816	0	0	995
44.-47.9	0	816	0	0	816
48.-51.9	0	0	0	0	0
52.-55.9	0	0	0	0	0
56.-59.9	0	0	0	0	0
TOTALS	137779	301920	85305	983600	1488604

ZIRC DISCHARGED ROD BURNUP STATUS AS OF 12/31/84***

ASSEMBLYWISE * BURNUP	14X14 RODS	15X15 RODS	16X16 RODS	17X17 RODS	TOTALS RODS
0.- 3.9	0	0	0	4224	4224
4.- 7.9	5000	0	0	5016	10016
8.-11.9	24857	8160	6815	31944	71776
12.-15.9	18592	51408	11750	45144	126894
16.-19.9	61755	91800	470	194040	348065
20.-23.9	55454	112200	4935	49104	221693
24.-27.9	121708	113832	4700	121704	361944
28.-31.9	114202	209916	0	124608	446726
32.-35.9	147138	138516	0	92400	378054
36.-39.9	30788	50796	0	27456	109040
40.-43.9	5728	5916	0	0	11644
44.-47.9	179	0	0	0	179
48.-51.9	0	0	0	0	0
52.-55.9	0	816	0	0	816
56.-59.9	0	0	0	0	0
TOTALS	585401	783380	28670	695840	2093071

ZIRC TOTAL ROD BURNUP STATUS AS OF 12/31/84

ASSEMBLYWISE * BURNUP	14X14 RODS	15X15 RODS	16X16 RODS	17X17 RODS	TOTALS RODS
0.- 3.9	15752	15504	11280	171600	214136
4.- 7.9	17100	28560	36425	130416	212501
8.-11.9	49009	44880	11515	160776	266180
12.-15.9	27363	102408	19975	214632	364378
16.-19.9	81397	114444	16215	337128	549184
20.-23.9	68055	144228	8225	131472	351980
24.-27.9	140682	152184	8460	197208	498534
28.-31.9	123781	249900	1880	168960	544521
32.-35.9	160303	161568	0	117216	439087
36.-39.9	33652	63240	0	29832	126724
40.-43.9	5907	6732	0	0	12639
44.-47.9	179	816	0	0	995
48.-51.9	0	0	0	0	0
52.-55.9	0	816	0	0	816
56.-59.9	0	0	0	0	0
TOTALS	723180	1085280	113975	1659240	3581675

*   Burnup units in GWD/MTU; 1 GWD = 1000 MWD
** Fuel rods currently operating
*** Fuel rods discharged from 1969 through December 1984

TABLE 6-2

WESTINGHOUSE FUEL PERFORMANCE STATUS REPORT - 1984
(Through 12/31/84)

Reactor	Location	Owner	Date Initial Criticality	Nominal MWe Net	Current Cycle Number	Peak Region Avg. Burnup (MWD/MTU)	Generation, MWh(e) Cumulative	1984 Maximum I-131 Activity in Primary Coolant 10**-6 Ci/gm
Jose'de Cabrera	Spain	Union Electrica S.A.	6/68	153	12	30,800	15,407,600	0.043
Beznau 1	Switzerland	Nordostschweizerische Kraftwerke AG	6/69	350	14	35,000	37,148,560	(a)
R. E. Ginna	U.S.A	Rochester Gas & Electric	11/69	490	14	26,700	44,584,850	0.0054
Point Beach 1	U.S.A.	Wisconsin Electric Power	11/70	497	12	36,600	44,645,240	0.011
Point Beach 2	U.S.A.	Wisconsin Electric Power	5/72	524	11	37,500	44,640,140	0.0088
Surry 1	U.S.A.	Virginia Electric Power	7/72	822	7	34,120	47,095,560	0.062(c)
Turkey Point 3	U.S.A.	Florida Power & Light	10/72	693	9	31,900	48,269,270	0.0037
Surry 2	U.S.A.	Virginia Electric Power	3/73	822	7	33,600	49,635,580	0.00026(c)
Indian Point 2	U.S.A.	Consolidated Edison	5/73	873	7	33,400	48,845,930	0.022
Turkey Point 4	U.S.A.	Florida Power & Light	6/73	693	10	35,100	45,443,610	0.013
Zion 1	U.S.A.	Commonwealth Edison	7/73	1040	8	37,100	61,866,300	0.010
Prairie Island 1	U.S.A.	Northern States Power	12/73	530	9	35,700	40,311,560	(a)
Zion 2	U.S.A.	Commonwealth Edison	12/73	1040	8	36,700	59,133,380	0.00066
Kewaunee	U.S.A.	Wisconsin Public Service	3/74	535	9	34,100	40,231,870	(a)
Prairie Island 2	U.S.A.	Northern States Power	12/74	530	8	36,100	38,236,900	(a)
D. C. Cook 1	U.S.A	Indiana & Michigan Elec.	1/75	1050	8	33,600	61,771,890	0.016(c)
Trojan	U.S.A.	Portland General Electric	12/75	1130	7	34,100	40,002,008	0.043
Millstone 2(b)	U.S.A.	Northeast Utilities	12/75	870	6	30,300	44,839,196	0.121

2775L:6-850812

TABLE 6-2 (Con't)

WESTINGHOUSE FUEL PERFORMANCE STATUS REPORT - 1984
(Through 12/31/84)

Reactor	Location	Owner	Date Initial Criticality	Nominal MWe Net	Current Cycle Number	Peak Region Avg. Burnup (MWD/MTU)	Generation, MWh(e) Cumulative	1984 Maximum I-131 Activity in Primary Coolant 10**-6 Ci/gm
Indian Point 3	U.S.A.	Power Authority of the State of New York	4/76	873	4	35,700	33,925,530	0.011
Beaver Valley	U.S.A.	Duquesne Light	5/76	852	4	32,800	26,959,905	0.00077
Salem 1	U.S.A.	Public Service Electric & Gas	12/76	1090	6	32,700	33,916,050	0.0042
Ko-Ri 1	Korea	Korea Electric Company	6/77	564	6	30,000	22,385,581	0.071
Farley 1	U.S.A.	Alabama Power Company	8/77	829	6	32,000	33,461,330	0.015(c)
D. C. Cook 2	U.S.A.	Indiana & Michigan Electric	3/78	1100	5	35,300	43,785,430	(a)
North Anna 1	U.S.A.	Virginia Electric Power	4/78	907	5	30,700	31,365,446	0.051(c)
North Anna 2	U.S.A.	Virginia Electric Power	6/80	907	4	32,800	22,540,104	0.012(c)
Sequoyah 1	U.S.A.	Tennessee Valley Authority	7/80	1148	3	30,500	24,661,450	0.0021
Ringhals 3	Sweden	Swedish States Power Board	7/80	915	2	-	13,485,796	0.00014
Salem 2	U.S.A.	Public Service Electric & Gas	8/80	1115	2	23,700	15,598,340	0.0012
Almaraz 1	Spain	Union Electrica S.A.	4/81	902	2	-	13,370,770	0.020
Farley 2	U.S.A	Alabama Power Co.	5/81	829	3	30,400	22,041,256	0.00080(c)
McGuire 1	U.S.A.	Duke Power Co.	6/81	1180	2	24,300	16,687,171	0.0094
Krsko 1	Yugoslavia	Savske Electric and Elektroprivreda	9/81	615	3	25,300	11,151,198	0.014
Sequoyah 2	U.S.A.	Tennessee Valley Authority	10/81	1148	2	27,300	19,056,718	0.0193
Angra 1	Brazil	Furnas	3/82	626	1	5,700	2,571,987	0.00054

2775L:6-850312

TABLE 6-2 (Con't)

WESTINGHOUSE FUEL PERFORMANCE STATUS REPORT - 1984
(Through 12/31/84)

Reactor	Location	Owner	Date Initial Criticality	Nominal MWe Net	Current Cycle Number	Peak Region Avg. Burnup (MWD/MTU)	Generation, MWh(e) Cumulative	1984 Maximum I-131 Activity in Primary Coolant 10**-6 Ci/gm
Ringhals 4	Sweden	Swedish States Power Board	8/82	915	1	14,000	8,918,491	0.00008
V. C. Summer	U.S.A.	South Carolina Electric & Gas	10/82	900	1	17,000	9,201,570	0.00024
Kori 2	Korea	Korea Electric Company	4/83	605	2	17,600	6,886,420	0.0046
McGuire 2	USA	Duke Power Company	5/83	1180	1	15,400	10,189,312	0.00053
ASCO 1	Spain	FECSA	6/83	887	1	-	4,695,962	(a)
Almaraz 2	Spain	Union Electrica S.A.	9/83	902	1	12,300	7,002,508	0.0016
Diablo Canyon	USA	Pacific Gas & Electric	5/84	1084	1	100	-	(a)
Maanshan 1	Taiwan	Taiwan Power Company	7/84	890	1	6,800	2,986,480	0.00021(c)
Callaway 1	USSA	Union Electric Company	10/84	1150	1	800	-	0.00028(c)

a. No data reported.
b. Currently fueled by Westinghouse.
c. Increased letdown rate. Coolant activity is not normalized to a single letdown rate.

TABLE 6-3
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 6-3

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 sticking and flux map indication of one or two rodlets 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 attempting 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
Salem Unit 1	June 1979	RCC Rodlets separated RCCAs and inserted into fuel assemblies	Stress corrosion cracking in the internal threads of the spider finger due to thread rework	RCCS replaced	All RCCAs having reworked fingers were replaced. No recurrence of failures in subsequence cycles
KORI Unit 1	December 1979	RCC vane with two rodlets inserted into fuel assembly	Vane and Rodlets separated from RCC	RCC replaced	Cause of failure not determined
San Onofre 1	January 1982	RCC vane and rodlet separated from spider body and inserted into fuel assembly.	Failure of the braze to base metal joint.	RCCA replaced	Cause of failure not determined.
Point Beach Unit 2 Point Beach Unit 1	June 1983 December 1983	Significant RCCA Absorber rod clad wear	Fretting at interface of RCCA rodlets and upper internals guide cards above upper core plate.	One RCCA (Unit 2) and 21 RCCAs (Unit 1) replaced. Reposition parked position of other RCCAs to distrib-ute wear.	Distribution of wear extends RCCA lifetime

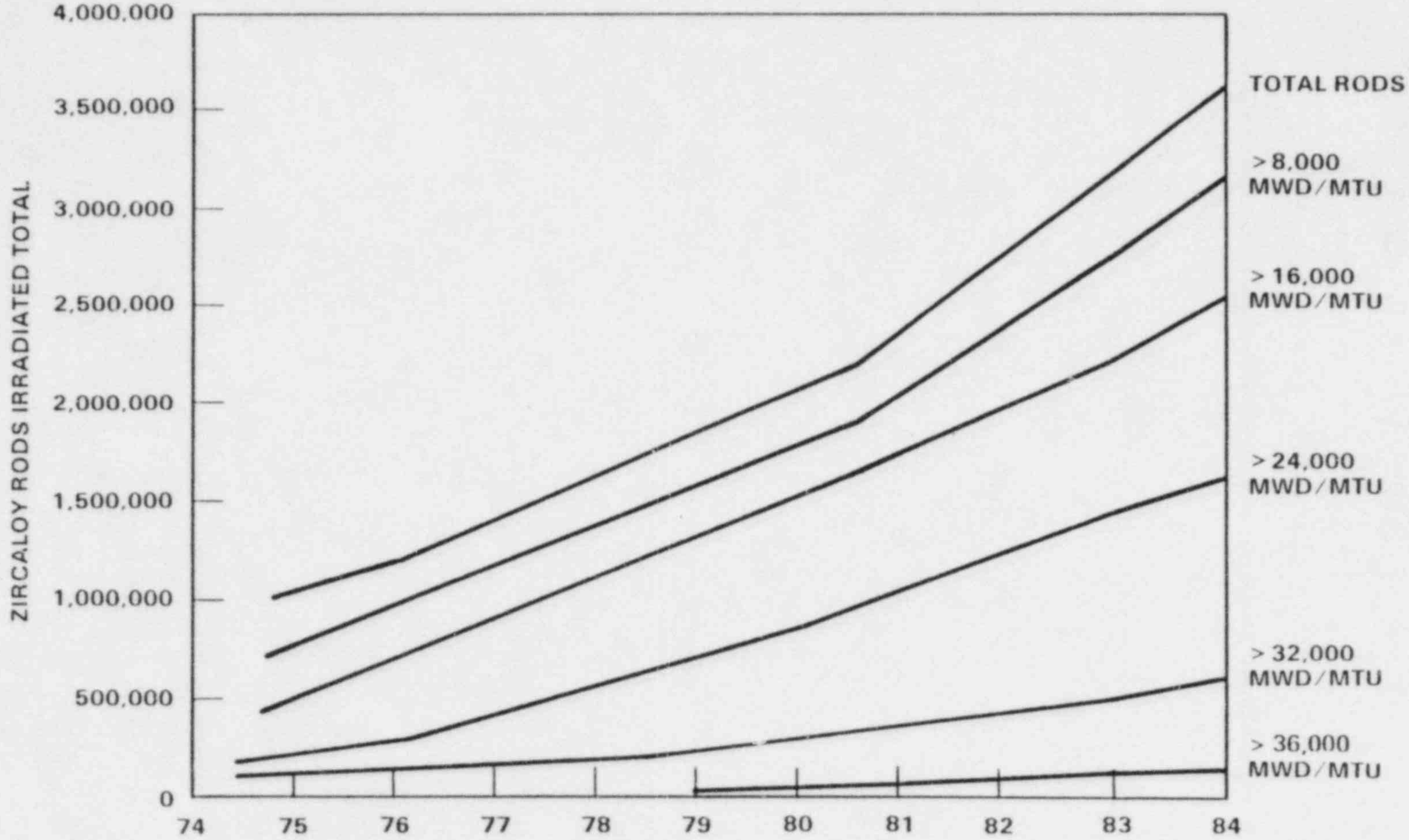

Figure 6-1. Assemblywise Performance of Westinghouse Zircaloy-Clad Fuel (Representative of All Zircaloy Fuel Operating and Discharged)

64-51-14086-1

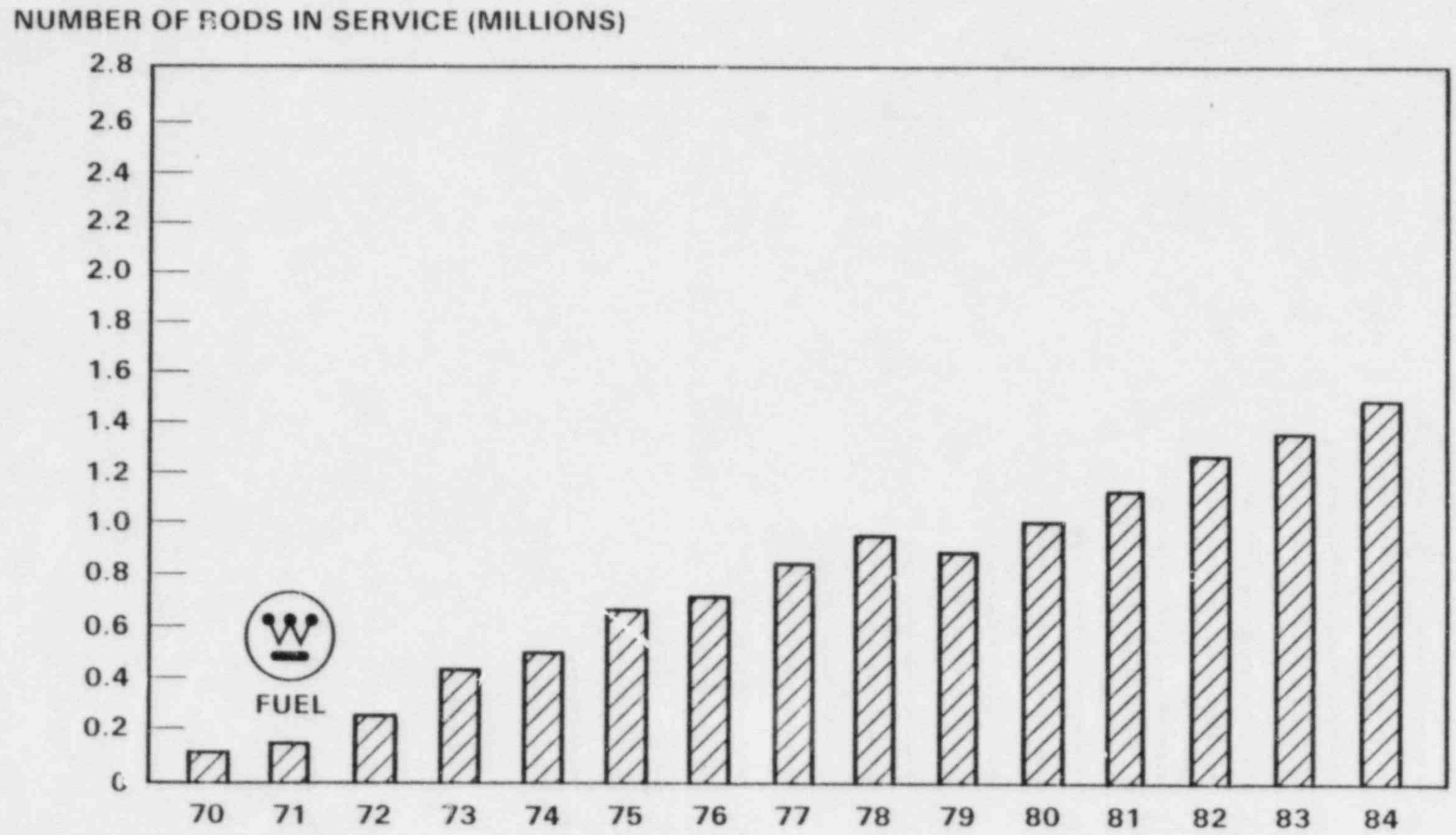

Figure 6-2. Westinghouse Zircaloy Clad Fuel Experience

64-51-14086-2

DATA POINTS SHOW THE NUMBER OF PLANTS IN A GIVEN ACTIVITY RANGE.

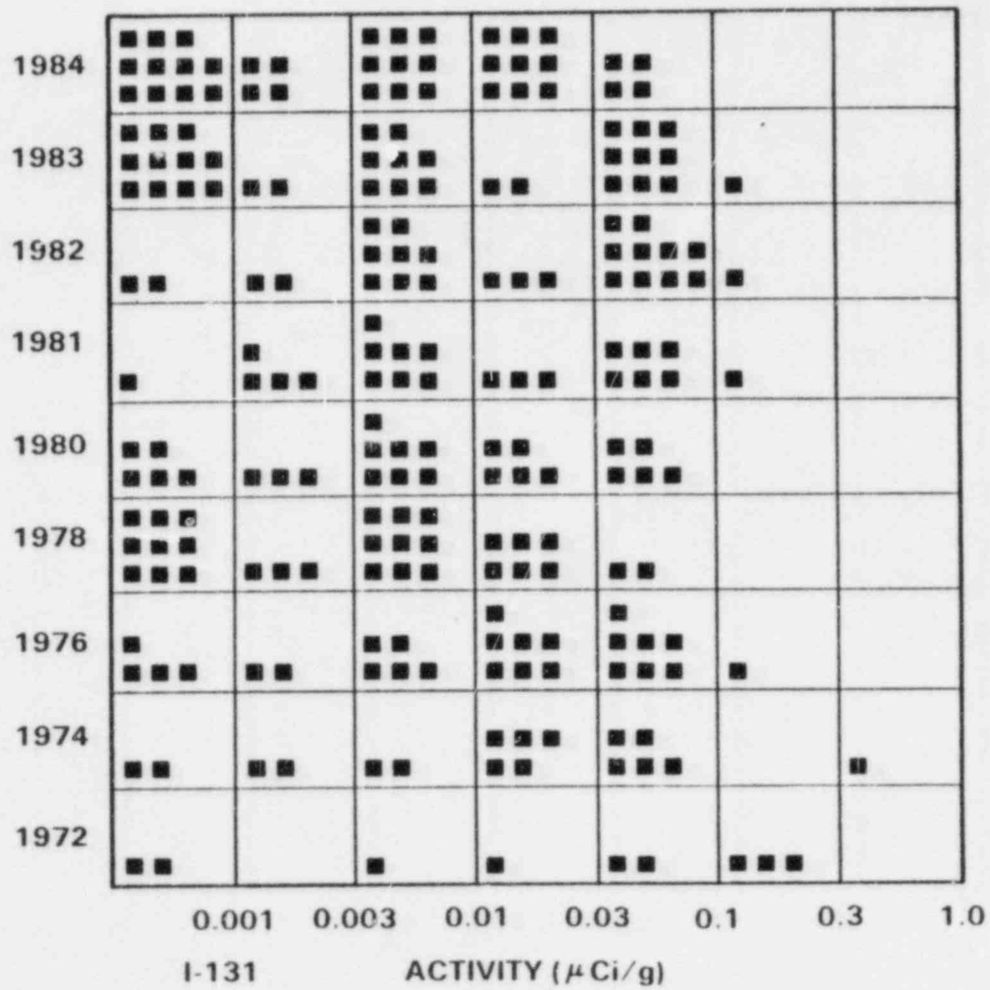

Figure 6-3. Reactor Coolant Activity
(Normalized to a Single Coolant Letdown Rate)

64-51-14086-3

SECTION 7
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