Integrity of Neutron-Absorbing Components of LWR Fuel Systems

W. J. Bailey
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March 1991

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ABSTRACT

A study of the integrity and behavior of neutron-absorbing components of light-water reactor (LWR) fuel systems was performed by Pacific Northwest Laboratory (PNL) and sponsored by the U.S. Department of Energy (DOE). The components studies include control blades (cruciforms) for boiling-water reactors (BWRs) and rod cluster control assemblies for pressurized-water reactors (PWRs). The results of this study can be useful for understanding the degradation of neutron-absorbing components and for waste management planning and repository design. The report includes examples of the types of degradation, damage, or failures that have been encountered. Conclusions and recommendations are listed.

SUMMARY

Considerable attention has been directed toward evaluating the integrity and behavior of spent light-water reactor (LWR) fuel assemblies and fuel rods during shipping, handling, and extended storage. The integrity of other fuel system components, such as boiling-water reactor (BWR) control blades, pressurized-water reactor (PWR) rod cluster control assemblies, and burnable poison rods and assemblies, is also worthy of consideration.

Control blade cracking has been observed at domestic and foreign BWRs in the past and involved cracks in the stainless steel tubing and loss of poison (boron) from some neutron absorber tubes. Perforated burnable poison (boron) rods in fuel assemblies were noted at a U.S. PWR in 1976; boron was lost from, and redistributed within, the perforated rods. The potential for swelling of hafnium control rods, especially at PWRs, has developed into a safety concern since the phenomenon was first noted in 1988. The concern is that a control rod could swell beyond the inner diameter of the guide tube in the associated fuel assembly. Cladding wear has also been noted on rod cluster control assemblies at some U.S. PWRs. Irradiation-assisted stress corrosion cracking (IASCC) has been encountered in BWRs and PWRs. In the early 1960s, IASCC was observed in fuel elements and a critical factor was the high stress associated with fuel swelling; however, IASCC has been observed more recently in lower stress components such as control blade handles and sheaths.

These observations suggest that spent fuel management should include consideration of all degraded, damaged (e.g., as a result of handling), or failed spent LWR fuel system components in the design of waste repository systems and the planning for spent fuel transfers. Fuel system components such as control blades (cruciforms) for BWRs and rod cluster control assemblies and burnable poison assemblies for PWRs must also be monitored for damage or degradation to permit appropriate preparations for shipping, handling, and extended storage. It is equally important that equipment operators involved in these activities understand the information concerning the integrity and behavior of both fuel assemblies and neutron-absorbing components.

The types of damage expected include cladding wear from fretting and friction, intergranular stress corrosion cracking, and irradiation enhanced stress corrosion cracking. These forms of degradation can lead to such difficulties as release of burnable poisons to the system, separation of components which then fall into positions where other components are damaged, and damage due to debris in regions of close clearance.

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1.0 INTRODUCTION

Considerable attention has been directed toward evaluating the integrity of spent light-water reactor (LWR) fuel assemblies and fuel rods during handling, extended storage, and shipping.⁽¹⁻⁵⁾ The integrity and behavior of other stored fuel system components such as neutron-absorbing (poison)^(a) components should also be considered in waste management planning and repository design. Such poison components include boiling-water reactor (BWR) control blades (cruciforms), pressurized-water reactor (PWR) control rod assemblies, and PWR burnable poison assemblies. These poison components have important functions in LWR systems. Control blades and control rods perform the dual functions of shaping power distributions and controlling reactivity in reactors. Burnable poisons provide temporary reactivity control.

Most nonfuel assembly hardware components (e.g., neutron absorbers) remain in use for more than one fuel assembly^(b) lifetime; therefore, it is anticipated that they will become more highly activated than corresponding nonfuel-bearing components of spent fuel assemblies.⁽⁶⁾ Notable exceptions may include channels (if not reused) for BWR fuel bundles, burnable poison assemblies that are used for only one cycle in PWRs, and poison curtains for BWRs.

The integrity and behavior of spent LWR fuel system components are affected by the service time in the reactor and the effects of subsequent handling, storage, and transportation to a spent fuel repository. As a response to these concerns, a study of the integrity and behavior of neutron-

(a) "Poison" material, as referred to in the nuclear industry, is defined as a material that absorbs neutrons unproductively and hence removes them from the fission chain reaction in a reactor, thereby decreasing its reactivity.

(b) The terms "fuel assembly" and "fuel bundle" are used interchangeably by the nuclear industry, although the former term is associated with fuel for PWRs and the latter term with fuel for BWRs. A BWR fuel assembly consists of a fuel bundle and the open-ended channel that encloses the bundle.

absorbing materials was performed by PNL^(a) under the Commercial Spent Fuel Management Program, which is sponsored by DOE. The results of the study which are described in this report can be useful for understanding degraded, damaged (e.g., during handling), or failed fuel components, including neutron-absorbing materials. Results can also be useful for waste management planning and repository design.

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2.0 CONCLUSIONS AND RECOMMENDATIONS

The results of the study of the integrity and behavior of neutron-absorbing (poison) materials have led to the conclusions and recommendations provided below in Sections 2.1 and 2.2, respectively.

2.1 CONCLUSIONS

The study of the integrity and behavior of neutron-absorbing (poison) components has provided the following key observations and conclusions:

- Most neutron-absorbing components contain stainless steel parts. The integrity of these stainless steel parts is a concern, if the conditions for intergranular stress corrosion cracking (i.e., material susceptibility, media conditions, and stresses within the material) exist when the neutron-absorbing components are handled, stored, or shipped following irradiation. One way to reduce the possibility for intergranular stress corrosion cracking is to control at least one of the three parameters necessary for that failure mechanism; for waste management and repository design, one parameter to consider controlling is the environment condition.
- In the 1970s, control blades at several BWRs were found to have cracks in some of the stainless steel tubing and a loss of boron from some of the absorber tubes.⁽⁷⁾ The cracking was attributed to intergranular stress corrosion cracking as a result of densification (sintering) of the B_4C particles in the tubes, followed by swelling (due to helium and lithium) of the sintered B_4C .⁽⁸⁾
- The influence of the environment (e.g., water chemistry) on intergranular stress corrosion cracking can be greatly increased if crevices are present on components; early BWRs experienced problems in components with crevices, such as irradiated stainless steel control blade sheaths with spot welds.⁽³⁾
- In the early 1960s, irradiation-assisted stress corrosion cracking occurred in fuel assemblies (a critical factor in such cases was the high stress associated with fuel swelling); more recently, irradiation-assisted stress corrosion cracking has been detected in lower stress components, including control blades and sheaths.⁽³⁾
- Some old BWR control blades in storage might have inverted B_4C absorber rods. A manufacturing deviation, which consisted of inverted B_4C absorber rods in control blades, was detected during 1973 and 1974 at three BWRs.⁽⁹⁻¹¹⁾
- At one BWR, handles have broken off control blades and the remainder of the control blades has fallen.^(12,13)

- Crud-induced localized corrosion (CILC) failures of burnable poison rods were first detected at a U.S. BWR in 1979.⁽¹⁴⁾ Later, four more U.S. BWRs encountered CILC failures. The CILC failures occurred at burnups of 15,000 to 20,000 MWd/MTU, were highly concentrated in the urania-gadolinia rods, were associated with a previously undetected variability in the susceptibility of the Zircaloy cladding to in-reactor nodular corrosion, and were confined to BWRs with copper alloy condenser tubes and filter demineralizer condensate cleanup systems. Of the urania-gadolinia rods in use, only a very small number (less than about 0.1%) have been susceptible to CILC.⁽¹⁵⁾ Some of the fuel assemblies with failed burnable poison rods were reconstituted; hence, the failed burnable poison rods that were replaced could be in storage in containers at spent fuel storage pools.
- During operation of rod cluster control assemblies (RCCAs) containing Ag-In-Cd in PWRs, the expected limiting factor is cladding wear. A disadvantage of such assemblies is the poor corrosion resistance of the Ag-In-Cd if the cladding is breached.⁽¹⁶⁾
- The potential for swelling of hafnium control rods has developed into a safety concern since the phenomenon was first detected at a U.S. PWR in 1988.^(1,2) The enlargement is of concern to reactor operations because a control rod could swell beyond the inner diameter of its guide tube and potentially interfere with its movement into and out of the core.⁽¹⁷⁾ Apparently the swelling occurs when hydrogen, under PWR operating pressures, diffuses through the stainless steel cladding, pierces the protective film on the hafnium, and results in localized hydriding of the hafnium that exacerbates the effects of uniform hydriding that were taken into consideration in the design. There are 14 PWRs (eleven are operating, three are under construction) in the U.S. that are equipped with hafnium control rods.
- The cladding on some control rods may have wear marks caused by fretting (surface damage); some of the rods may have cracked tips. In some cases, the wear marks extend through the cladding wall. In a number of cases, the affected control rod assemblies were replaced; as a result, the damaged assemblies may be in storage at spent fuel storage pools. Monitoring the reactor's coolant (or the pool water) for silver and hafnium is a reliable means for detecting fretting wear through the cladding for Ag-In-Cd and hafnium type RCCAs.
- In a few cases, vanes and rodlets have separated from the remainder of RCCAs and fallen back into the cores of PWRs. Problems with RCCAs have included failure of brazed or welded joints, bent or broken vanes, bent rodlets, control rod follower distortion, and stress corrosion cracking (a result of using a chloride-containing lubricant).
- Debris, especially metallic particles, can cause problems with close-fitting components and can make disassembly operations difficult. The infrequent malfunctions in control rod drive systems are usually related to the presence of metallic debris in the systems (i.e., if debris is

present, the control rod drive systems are susceptible to binding problems because of the small clearances that are required).⁽¹⁸⁾

- One fuel vendor has developed a two-part eddy current inspection technique to examine RCCA rodlets and control element assembly rodlets for evidence of wear and cracks.⁽¹⁹⁾ Early in the development of this technique, visual confirmation of cracks was obtained, which is considered now to be an unnecessary step for confirming the existence and cause of a crack.
- One of the advantages of an integral burnable absorber over the conventional absorber rod employing the borosilicate glass design is the elimination of all problems at the reactor site regarding the handling and disposal of burnable poison rods.⁽²⁰⁾
- There is no routine surveillance of burnable absorber assemblies in operating PWRs.⁽¹⁸⁾ A few problems (e.g., cladding cracks, stuck rods, broken rods, and a vane and rod that separated from rest of the assembly) have been noted with burnable absorber assemblies.
- There have been very few cases of damage to neutron-absorbing components due to mishandling.
- Cladding perforations in a number of burnable poison rods were detected at one U.S. PWR in 1976.⁽²¹⁾ Several other PWRs also experienced failures of similar burnable poison rods, but the frequency of failure was lower. The failure mechanism was primary hydriding. Boron loss from, and redistribution within, the perforated poison rods was detected; leaching of the B_4C from the Al_2O_3 - B_4C pellets occurred during reactor service. When leached boron from B_4C comes in contact with an excess of water at elevated temperatures a boric acid solution is produced.⁽²²⁾ Therefore, all 1300 poison rods in 108 fuel assemblies in U.S. PWRs were removed and replaced.
- At PWRs, only a few incidents have occurred with sources and plugging devices.^(a) These incidents include source assemblies or plugging devices stuck in fuel assemblies, binding of components, bent thimble plugs, and damage to a source caused by mishandling.

2.2 RECOMMENDATIONS

The recommendations from the study of the integrity and behavior of neutron-absorbing (poison) components are listed below.

(a) A plugging device is a device used to cap the space intended for a control rod or core instrumentation when neither of these is to be used during a given reactor cycle.

- When handling, storing, and shipping BWR control blades, operators should be made aware that 1) a few old control blades could have absorber tubes with cracks leading to possible loss of some boron carbide, small particles of which can damage the water-circulating pumps, and 2) the handles have broken off a few control blades.
- When handling, storing, and shipping PWR RCCAs, operators should be made aware that a few assemblies have had vanes and rodlets separate from the remainder of the assemblies.
- It will be important to control handling, storing, and shipping conditions to reduce the possibility that the stainless steel portions of neutron-absorbing components will undergo intergranular stress corrosion cracking.
- Efforts should be made to keep debris from being introduced during handling, storing, and shipping. Debris, especially metallic particles, can make control rod assemblies difficult to remove from fuel assemblies or can cause jamming or binding of close-fitting components.

3.0 INTEGRITY AND BEHAVIOR OF NEUTRON-ABSORBING COMPONENTS OF LWR FUEL SYSTEMS

Background information is presented on the neutron-absorbing components of LWR fuel systems and the experiences concerning the integrity and behavior of the neutron-absorbing components in Sections 3.1 and 3.2, respectively.

3.1 BACKGROUND INFORMATION ON NEUTRON-ABSORBING COMPONENTS

Background information on the neutron-absorbing components used by B&W Fuel Company (BWFC)^(a); Combustion Engineering, Inc. (C-E); General Electric Company (GE); and Westinghouse Electric Corporation is provided below.

3.1.1 B&W Fuel Company (BWFC)

Information on BWFC's^(a) control rod assemblies, burnable poison assemblies, and other nonfuel components is presented below.

3.1.1.1 Control Rod Assemblies

The control rod assemblies used by BWFC consist of 16 individual rods with their upper ends fastened to a spider assembly.⁽⁶⁾ The control rod drive mechanism engages the spider assembly to withdraw and position the control rod assembly. The control rods are about 13 ft long and 0.440 in. in diameter; the cladding is Type 304 stainless steel; the end plugs are Types 304 or 308 stainless steel. The neutron absorber in the rod is an Ag-In-Cd alloy.

The axial power shaping assemblies used by BWFC have a similar spider assembly.⁽⁶⁾ The axial power shaping rods are the same length and diameter as the control rods and employ Type 304 stainless steel for cladding; the end plugs, intermediate plugs, and nuts are usually Types 304 or 308 stainless steel. The rods contain Ag-In-Cd absorbers.

Gray axial power shaping rod assemblies also have a similar spider assembly.⁽⁶⁾ The gray axial power shaping rods are 155.56 in. long and 0.440 in. in diameter; the cladding is Type 304 stainless steel and the end

(a) Previously known as Babcock and Wilcox Company (B&W); the B&W Fuel Company (BWFC) is a partnership between B&W and the American subsidiary of a French consortium of Cogema, Framatome, and Uranium Pechiney.

plugs, intermediate plugs, and nuts may be made of Types 304 or 308 stainless steel. The absorber material in the rod is Inconel.

3.1.1.2 Burnable Poison Rod Assemblies

Burnable poison rod assemblies (BPRAs) are similar to control rod assemblies with individual burnable poison rods attached to a spider hub and with the rods occupying control rod guide tube positions in the fuel assemblies.⁽²³⁾ However, unlike the control rod assemblies that are moveable and attached to the end of a drive mechanism, the BPRAs are held in a fixed position by means of a latching mechanism on the upper end fitting of the fuel assembly.

The spider for the burnable poison assemblies is made of CF3M, Types 304 and 316 stainless steel.⁽⁶⁾ The 16 burnable poison rods employ Zircaloy-4 for cladding, end fittings, and nuts to fasten the rods to the spider; each rod is about 150 in. long and 0.430 in. in diameter. The hold-down spring is made of Types 302 or 304 stainless steel. The burnable poison is a BWFC proprietary mixture of Al_2O_3 and B_4C .

3.1.1.3 Other Nonfuel Components

For completeness, other nonfuel components that are part of the fuel system but do not contain neutron absorbers are described below.

- Orifice Rod Assemblies - An orifice rod assembly is installed in those fuel assemblies that do not have BPRAs or control rods and serves to reduce the bypass coolant flow through the empty guide tubes. The spider assembly is made of CF3M and Type 304 stainless steel; the orifice rods are about 12 in. long and 0.480 in. in diameter.⁽⁶⁾ The rods are made of Type 304 stainless steel and the nuts are made of Types 304 or 308 stainless steel.
- Primary Neutron Source Clusters - These clusters consist of 10 components; the shroud tube, intermediate plug, and lower end plugs are made of Type 304 stainless steel.⁽⁶⁾ The source is Am-Be-Cu; the composition is considered proprietary by BWFC.
- Regenerative Neutron Source Clusters - These clusters consist of a coupling spider assembly and eight rods; the spider assembly is made of CF3M or Types 304 or 316 stainless steel.⁽⁶⁾ The cladding and end plugs for the rods are Type 304 stainless steel. The length of the source rods is 140 in. and the diameter is 0.440 in. The source is antimony-beryllium; the composition is considered proprietary by BWFC.

3.1.2 Combustion Engineering, Inc. (C-E)

Information about C-E's control element assemblies, burnable poison assemblies, and other nonfuel components is presented below.

3.1.2.1 Control Element Assemblies

Control element assemblies (CEAs) provide a means of controlling core reactivity. CEAs designed by C-E consist of tubes or "fingers" containing neutron-absorbing materials.^(6,24) When the CEAs reach the end of their useful life (4000 full-power days), they are housed in spent fuel assemblies that are in the utility's spent fuel pools.

The CEAs for most C-E reactors consist of four or five fingers attached to a spider at their upper ends; the three Palo Verde reactors have CEAs with 12 fingers attached to a spider.⁽⁶⁾ Some CEAs have full-length rods; some have part-length rods. The cladding is Inconel; the neutron absorber is B_4C , an Ag-In-Cd alloy, or stainless steel.

For its System 80 PWRs, C-E has replaced the Ag-In-Cd slugs at the tip of the control rods with additional B_4C pellets of reduced diameter; also, C-E introduced a compressible metal liner (a Feltmetal material made of Type 347 stainless fibers consolidated to 22.5% of theoretical density) in this region to accommodate swelling of the B_4C .⁽²²⁾ When this hybrid control rod design is subjected to high exposures, C-E observes that the cladding strain induced by B_4C swelling is more likely to limit the lifetime of the CEAs than will Ag-In-Cd swelling.⁽²²⁾

The CEAs for Palisades are a special case.⁽⁶⁾ The CEAs are in the form of cruciform blades and are anticipated to last the lifetime of the unit. The neutron absorber is Ag-In-Cd alloy clad with Type 302 stainless steel. The control rod drive mechanism engages a hanger (made of Types 304 and 308 stainless steel) to withdraw and position the CEA.

3.1.2.2 Burnable Poison Assemblies

C-E uses integral burnable poisons and does not manufacture burnable poison nonfuel assembly hardware.⁽⁶⁾ C-E uses two basic burnable poison designs to optimize a fuel cycle.⁽²⁵⁾ The first involves rods containing homogeneous pellets of $Al_2O_3-B_4C$ in Zircaloy-4 tubing. As of September 1987,

C-E has successfully irradiated over 20,000 burnable poison rods of this design. The second design uses urania-gadolinia as the poison. Both designs of poison rods replace fuel rods in the lattice.

3.1.2.3 Other Nonfuel Components

For completeness, additional nonfuel components that are part of the fuel system but do not contain neutron absorbers are described below.

- Orifice Rod Assemblies - C-E does not describe any orifice rod assemblies in CEND-428.⁽²⁴⁾
- Neutron Source Assemblies - Two neutron source assemblies, which are inserted in guide tubes of perimeter fuel assemblies on opposite sides of the core, are used in each core.⁽⁶⁾ The life expectancies of the neutron source assemblies are 3500 full-power days for 16x16 fuel and 4000 full-power days for the other fuel. The neutron source assembly consists of two subassemblies. The lower subassembly contains the sources (a column of antimony-beryllium pellets with a plutonium-beryllium capsule above and below the pellet column). The plunger, upper subassembly, cladding, and spacer for the lower subassembly are made of Type 316 stainless steel.

3.1.3 General Electric Company (GE)

Information on GE's control element assemblies, burnable poison assemblies, and other nonfuel components is presented below.

3.1.3.1 Control Element Assemblies

Movable blades composed of many small rods containing B_4C are employed to control the power in BWRs. For control elements in their reactors, GE uses cruciform blades made of Type 304 stainless steel; the blades contain a boron carbide (B_4C) absorber in 84 Type 308 stainless steel tubes.⁽⁶⁾ The B_4C is packed into the tubes at about 70% of its theoretical density. The cruciform blades, which are 9.75 in. wide and have a control length of 143 in., have an estimated lifetime of 3 to 25 years. The end bearings of the cruciforms are made of an alloy high in cobalt (e.g., Stellite-3 or Haynes-25). During irradiation, the natural cobalt activates to cobalt-60; the bearings are sometimes removed and packaged for use as cobalt-60 sources. As a result, the heat output from these bearings can yield a distorted picture of the heat output from the cruciform as a whole.⁽⁶⁾

3.1.3.2 Burnable Poison Assemblies

Temporary poison curtains fabricated of borated (3800 to 5400 ppm boron) stainless steel sheets were employed to control reactivity in the initial cores of early GE reactors.⁽⁶⁾ The curtains were placed in water gaps between fuel assemblies without control rods. The curtains were 141.25 in. long, 9.20 in. wide, and 0.0625 in. thick.

Currently, GE employs integral gadolinia poisons (i.e., some fuel rods in a fuel assembly contain urania-gadolinia pellets) to control reactivity. There are typically four rods with urania-gadolinia pellets in an 8x8 BWR fuel assembly; there can be as many as eight in more recent designs. Except for a very small number of fuel rods (less than about 1%) that were susceptible to CILC, the in-reactor performance of the $(U,Gd)O_2$ fuel is similar to that of UO_2 . The Gd_2O_3 content in the burnable absorber rods has ranged from 0.5 to 6 wt%. The blended UO_2 and Gd_2O_3 , if processed correctly (requires a very complex sintering process), results in a near-homogeneous solid solution of $(U,Gd)O_2$. With few exceptions, this burnable absorber has been used in all GE-designed BWR reloads.⁽¹⁵⁾

3.1.3.3 Other Nonfuel Components

Other nonfuel components that are part of the fuel system but do not contain neutron absorbers are described below.

- Orifice Rod Assemblies - Orifice rod assemblies are not applicable to GE reactors.
- Neutron Sources - GE uses five to seven antimony-beryllium neutron source rods per reactor; each source consists of two irradiated antimony rods within a single beryllium cylinder.⁽⁶⁾ Both the antimony and the beryllium are encased in stainless steel tubes.
- BWR Fuel Channels - The fuel channels are made of Zircaloy-4. The channels for all BWR/2, 3, 4, and 5 reactors have a square cross section (inside width is 5.278 in.); the nominal length is 162 in. for BWR/2 and 3 reactors and 167 in. for BWR/4 and 5 reactors.^(6,26) Channel wall thicknesses can be 0.080, 0.100, or 0.120 in.

3.1.4 Westinghouse Electric Corporation

Information on Westinghouse's rod cluster control assemblies (RCCAs), burnable poison assemblies, wet annular burnable absorber assemblies, and some other nonfuel components is provided below.

3.1.4.1 Rod Cluster Control Assemblies

Westinghouse introduced the RCCA in 1967. Through June 1986, Westinghouse had manufactured 2215 RCCAs.⁽⁶⁾ RCCAs for Westinghouse reactors a) using 17x17 fuel assemblies consist of 24 individual rods attached at their upper ends to a spider assembly, b) using 15x15 fuel assemblies have 20 control rods, and c) using 14x14 fuel assemblies have 16 control rods.⁽⁶⁾ The control rod drive mechanism engages the spider assembly to withdraw and position the control rod assembly. The spider assembly a) for use with 17x17 fuel consists of Types 304 and 308 stainless steel and Inconel 718 springs, b) for use with 15x15 fuel consists of Types 304 and 308 stainless steel and Inconel X-750 and c) for use with 14x14 fuel consists of Types 304 and 308 stainless steel and Inconel X-750.⁽⁶⁾

The control rods for 17x17 fuel have Type 304 stainless steel cladding and Type 308 stainless steel end plugs, are 151.885 in. long and 0.385 in. in diameter, and have a cladding wall thickness of 0.0185 in. (except for hybrid rods, which have a cladding thickness of 0.038 in.).⁽⁶⁾ Full-length rods contain 142 in. of Ag-In-Cd alloy or hafnium; part-length rods contain 36 in. of Ag-In-Cd alloy with 106 in. of aluminum oxide above the absorber. Hybrid rods contain 40 in. of Ag-In-Cd alloy and 102 in. of B₄C above it. The hold-down springs in full-length and part-length rods are made of Type 302 stainless steel; in hybrid rods, the springs may be made with Inconel 718. The overall length of the control rod assemblies is 161 in.

The control rods for 15x15 fuel have Type 304 stainless steel cladding and Type 308 stainless steel end plugs, are 150.92 to 152.75 in. long and 0.443 in. in diameter, and have a cladding wall thickness of 0.020 in.⁽⁶⁾ The absorber is 142 in. of Ag-In-Cd alloy. The hold-down springs in the rods are made of carbon steel. The overall length of the control rod assemblies is 156.6 to 158.5 in.

The control rods for 14x14 fuel have Type 304 stainless steel cladding and Type 308 stainless steel end plugs, are 120 to 153 in. long and 0.435 in. in diameter, and have a cladding thickness of 0.0185 in.⁽⁶⁾ The absorber is 118 to 142 in. of Ag-In-Cd alloy. The full-length rods contain carbon steel hold-down springs; part-length rods contain Inconel X-750 hold-down springs. Overall length of the control rod assemblies is 134 to 158 in.

3.1.4.2 Burnable Poison Assemblies

Westinghouse has fabricated over 5000 burnable poison assemblies that involve 329 different configurations; the assemblies contain up to 24 burnable poison rods (thimble plugs were inserted into the remaining locations).⁽⁶⁾ The 16 to 24 rods and plugs are fastened to a spider pack or hold-down assembly, which is held down by the same hold-down plate that keeps the fuel assemblies in place. The spider packs and hold-down assemblies are made of Types 304 or 308 stainless steel and have springs made of Inconel 718 (alternative spring materials can be carbon steel, Type 302 stainless steel, or Inconel X-750).⁽⁶⁾ The overall length of the burnable poison assemblies ranges from 151.6 to 156.9 in.

Burnable poison rods for 17x17 fuel assemblies have Type 304 stainless steel cladding and Type 308 stainless steel end plugs, are 152.4 in. long, and 0.385 in. in diameter.⁽⁶⁾ The radial spacer material is Type 304 stainless steel. The neutron absorber is borosilicate glass tubing with 12.5% natural B₂O₃. The tubing is 142 in. long and 0.073 in. thick.

For other fuel assemblies, the burnable poison rods have Type 304 stainless steel cladding and Type 308 stainless steel end plugs, are 150.4 to 152.8 in. long, and have diameters from 0.437 to 0.445 in. The spacer material is Type 304 stainless steel. The neutron absorber is borosilicate glass tubing with 12.5% natural B₂O₃. The tubing is 141.6 to 142.7 in. long and is 0.072 to 0.076 in. thick.

The thimble plugging devices range from 5.15 to 9.17 in. in length.

3.1.4.3 Burnable Absorber Assemblies

Westinghouse's wet annular burnable absorber (WABA) rods consist of annular pellets of aluminum oxide-boron carbide burnable absorber contained

between two concentric Zircaloy tubes. Reactor coolant flows inside the inner tubing and outside the outer tubing of the annular rod.⁽¹⁸⁾ An advantage of the WABA rod is that it reduces fuel cycle cost because of reduced parasitic neutron absorption in the Zircaloy cladding and increased water fraction in the WABA cell. The WABA design replaced the previous borosilicate glass design in the early 1980s.⁽¹⁾

Described in documentation from Westinghouse are 143 burnable absorber assembly configurations using 3 to 24 WABA rods; all of the configurations except one use thimble plugs in the remaining locations.⁽⁶⁾ The neutron absorber rods are 143.1 to 150.1 in. long and contain 105.5 to 134 in. of annular $B_4C-Al_2O_3$ pellets; the pellet wall thickness is 0.070 in. The pellet column is clad inside and outside with Zircaloy-4. The rod outside diameter is 0.381 in.; the inner cladding wall thickness is 0.020 in. and the outer cladding wall thickness is 0.026 in. Rods with shorter absorber lengths have spacers of Zircaloy-4 up to 13.5 in. long. The top connector is made of Type 304 stainless steel. The hold-down assemblies are made of Type 304 stainless steel and have Inconel 718 springs. The thimble plugs are 5.15 to 8.08 in. long.

3.1.4.4 Other Nonfuel Components

Other nonfuel components that are part of the fuel system but do not contain neutron absorbers are described below.

- Thimble Plug Assemblies - Westinghouse has fabricated over 8200 thimble plug assemblies.⁽⁶⁾ The assemblies may remain in reactors for more than one cycle. The thimble plugs are 0.424 to 0.502 in. in diameter and 5.15 to 9.16 in. long.
- Primary Neutron Source Assemblies - The fuel assemblies that do not have control rod assemblies may be fitted with primary neutron source assemblies, although presumably only one or two per reactor and then probably only for the first reactor operating cycle or two [Westinghouse had described 106 configurations in WSTD-TME-148⁽²⁷⁾ but has only fabricated 92].⁽⁶⁾ Information on the configurations is provided:
a) there is only one primary neutron source rod in each of the configurations; b) six configurations have 20 thimble plugs but no other rods in addition to the source rod; c) twenty-three configurations have a primary source rod; 1, 2, 3, or 4 secondary source rods; and 0, 12, or 16 burnable poison rods with the balance being thimble plugs; and d) all other configurations have a primary source rod and 12 to 23 burnable poison rods, with the balance (if any) being thimble plugs.

Primary neutron source rods of 17x17 fuel assemblies have Type 304 stainless steel cladding and Type 308 stainless steel end plugs.⁽⁶⁾ The source is californium.

For other fuel assemblies, the primary neutron source rods have PDS 10708BN or Type 304 stainless steel cladding and PDS 10708BN or Type 308 stainless steel end plugs.⁽⁶⁾ The sources are plutonium-beryllium, polonium-beryllium, or californium.

- Secondary Neutron Source Assemblies - Westinghouse has manufactured 120 secondary neutron source assemblies (an assembly consists of 16 to 24 rods and plugs) that involved 95 different configurations.⁽⁶⁾ The configurations included the following: a) one configuration with 3 secondary neutron source rods, 12 burnable poison rods, and 5 guide tube positions that are left vacant; b) one configuration with 4 secondary neutron source rods, 12 burnable poison rods, 3 thimble plugs, and 5 guide tube positions that are left vacant; c) two configurations with 4 secondary neutron source rods and 16 guide tube positions that are left vacant; d) fifty-five configurations with 4 secondary neutron source rods and all remaining positions filled with thimble plugs; e) seven configurations with 4 secondary neutron source rods and all remaining positions filled with burnable poison rods; and f) twenty-eight configurations with 4 secondary neutron source rods, 7 to 16 burnable poison rods, and 4 to 12 positions filled with thimble plugs.

The secondary neutron source rods for 17x17 fuel assemblies have Type 304 stainless steel cladding and Type 308 stainless steel end plugs. The source material is an alloy of antimony (77%) and beryllium (22%).⁽⁶⁾

The secondary neutron source rods for other fuel assemblies have PDS 10708BN or Type 304 stainless steel cladding and PDS 10708BN or Type 308 stainless steel end plugs. The source material is an antimony-beryllium alloy.⁽⁶⁾

3.2 EXPERIENCE CONCERNING THE INTEGRITY AND BEHAVIOR OF NEUTRON-ABSORBING COMPONENTS

Information on the experience with integrity and behavior of neutron-absorbing components (and a few nonfuel components) of fuel systems at BWRs and PWRs is provided below in Sections 3.2.1 and 3.2.2, respectively.

3.2.1 Experience with Components at BWRs

The integrity and behavior of neutron-absorbing components, and a few other nonfuel components, of fuel systems at BWRs is described in the 10 sections below.

3.2.1.1 Control Blade Failures

Control blades at two domestic BWRs (Dresden-1 and Oyster Creek) and at one foreign BWR (KRB) were found to have cracks in some of the stainless steel tubing and a loss of boron from some of the absorber tubes.⁽⁷⁾ The cracking was believed to have been caused by intergranular stress corrosion as a result of densification (sintering) of the B_4C particles in the tubes followed by swelling (due to helium and lithium) of the sintered B_4C .⁽⁸⁾

A potential problem with BWR absorber tubes was observed initially in the late 1970s.⁽²⁸⁾ Four absorber rods were inserted at Dresden-1 in 1968 for irradiation. The rods were examined after Cycles 1 and 4 and were sound; two of the rods continued to be irradiated in Cycle 5. At the end of Cycle 5 (rod design life for internal pressure was exceeded), the two rods were found to have extensive cracking and loss of B_4C .

Five absorber tubes that had attained 80% of design life in a control blade at Dresden-1 were examined in 1974; one tube had through-wall cracks but no apparent loss of B_4C , and one tube had incipient cracks. In 1978, 45 absorber tubes from a foreign BWR, KRB, were examined; all the tubes had through-wall cracks and many of the tubes had B_4C missing. Also in 1978, 7 of 13 inspected absorber tubes from a control blade at the Oyster Creek BWR exhibited through-wall cracks and 5 of the absorber tubes had B_4C missing.

In November 1979, the NRC issued IE Bulletin 79-29,⁽²⁹⁾ which describes the problem of BWR control blade cracking and the associated loss of boron. Revision 1 to that bulletin was issued in August 1980. The bulletin also required that certain actions be taken by BWR operators, including identifying high-exposure blades and conducting destructive postirradiation examinations of high-exposure control blades.

An early estimate of control blade life was ~12 cycles, a later estimate was ~8 cycles, and, as a result of the observed cracking, the estimate in 1980 of blade life was 6.4 cycles.⁽³⁰⁾

3.2.1.2 Stress Corrosion Cracking of Components

At an August 1989 conference, intergranular stress corrosion cracking (IGSCC) was discussed. It was indicated that the influence of the environment

(e.g., water chemistry) on IGSCC can be greatly increased if crevices are present in components. Early BWRs experienced problems in components with crevices such as on irradiated stainless steel control blade sheaths with spot welds.⁽³⁾ Efforts were made in later BWR designs to eliminate the crevices. Also, improvements have been made in the water chemistry; for example, the conductivity of the water has tended to show a downward trend during the period from 1981 to the present.

Also discussed at the 1989 conference⁽³⁾ was the recent growing interest in the phenomenon of irradiation-assisted stress corrosion cracking (IASCC). In the early 1960s, IASCC occurred in fuel assemblies; a critical factor in such cases was the high stress associated with fuel swelling. However, it was indicated in a paper (by P. L. Anderson et al.) that more recently IASCC has been detected in lower stress components, including control blades and sheaths.

3.2.1.3 Blade Wear

In 1989, Monticello removed 9 core blades because of wear and moved 30 others to maximize their useful lives.⁽³¹⁾

3.2.1.4 Inverted B₄C Absorber Rods

A manufacturing deviation, which consisted of inverted B₄C absorber rods in control rod blades, was detected at three BWRs (Dresden-3, Quad Cities-1, and Millstone-1) in the U.S. The deviation was initially observed in July 1973 at Millstone-1. Those events are described in the following three subsections.

- Millstone-1 - During July 1973, a manufacturing deviation (inverted B₄C absorber rods) in U.S. BWR control rod blades was reported; this abnormal occurrence was initially noted at Millstone-1.^(10,11)
- Dresden-3 - All 177 control rod blades were examined with an eddy current tester; it was found that some of the B₄C absorber rods were not properly installed.⁽⁹⁾ The eddy current tester could locate the steel wool that was inserted in each control rod in the bottom end of the absorber tube before the end plug was installed. Of the 177 control rod blades, 25 blades had a total of 187 inverted absorber rods. During the April 1974 refueling shutdown, 7 control rod blades were replaced; the remaining 32 absorber rods would not affect safe operation even if the B₄C were to experience maximum slumping.

- Quad Cities-1 - Eddy current testing of all 177 control rod blades was performed at Quad Cities-1 in April 1974.^(10,11) The tests revealed that 91 B₄C absorber rods were inverted in 32 of the 177 control rod blades examined. During the April 1974 refueling outage, two control rod blades that were not acceptable were replaced; this left 57 inverted B₄C rods in the reactor.

3.2.1.5 Broken Handles on Control Blades and/or Dropping of Control Blades

There have been three events at a U.S. BWR, Duane Arnold, that involved broken handles on control blades and/or dropping of control blades. Those events are described below.

- Duane Arnold (first event) - An attempt was made in 1974 to lift two blade guides that were connected by a yoke at the top.⁽¹²⁾ When the guides had been lifted about three inches, the yoke handle broke and two blade guides fell back into the storage location and a piece of the handle fell to the floor of the dry fuel pool.

All blade guides were inspected by loading them to 1.5 times their static load and then subjecting them to liquid penetrant examination. That examination revealed a crack in the corner of one blade guide handle that was connected to the failed yoke. A yoke handle and a blade guide handle were also rejected because of porosity.

- Duane Arnold (second event) - A report issued in 1974 indicates that during operator training with no fuel in the reactor, the blade guide from position 33-18 was lifted until the nose piece cleared the upper grid.⁽¹³⁾ From that height, the blade guide fell back into position 33-18. The blade guide nose piece sustained a few minor surface imperfections, the upper grid was undamaged, and the fuel support piece was not damaged. Incomplete grappling caused the incident. The grapple hook was inspected and found to be in satisfactory condition. Apparently the operator failed to verify a "slack cable" indication before lifting the blade guide.
- Duane Arnold (third event) - In 1989, the handle (bail) of a control element assembly broke off the cruciform-shaped assembly and the remainder of the assembly fell to the bottom of the spent fuel storage pool. Results of the preliminary inspections indicate that the failure may have been caused by stress corrosion cracking.

3.2.1.6 Mechanical Interference of Channels on Fuel Assemblies with Control Blades

An event at a foreign BWR involved mechanical interference of channels with control blades. It is indicated in a 1973 report that mechanical

interference between fuel-bundle channels and control blades was observed at Japan's Fukushima-2.(32)

3.2.1.7 Shrinkage and Loosening of Control Blade Pads

There was one event at a foreign BWR (and the potential for events at two other foreign BWRs) involving shrinkage and loosening of control blade pads. Shrinkage and loosening of graphite pads on the control blades was observed at Sweden's Oskarshamn-1, according to a 1974 report.(33) The graphite pads were attached to the control blades with steel rings. The graphite pads were installed to prevent the control blades from rubbing against and scratching the channels on the fuel bundles. Neutron bombardment caused the shrinkage of the graphite pads, which were replaced with Inconel pads. The corrective action was also taken at two other Swedish BWRs, Oskarshamn-2 and Ringhals-1, that had not started up yet.

3.2.1.8 Burnable Poison Rods

Burnable poison rods are an integral part of BWR fuel assemblies; as a result the burnable poison rods, which typically contain urania-gadolinia pellets, are included in the handling and storage of spent fuel assemblies rather than being handled and stored as separate entities. There have been a few events of interest pertaining to burnable poison rods. Those events are described in the two subsections below.

3.2.1.8.1 Crud-Induced Localized Corrosion - A fuel rod failure mechanism, CILC, was first detected in a BWR (Vermont Yankee) in 1979.(14) The CILC failures occurred at burnups of 15,000 to 20,000 MWd/MTU, were highly concentrated in the urania-gadolinia rods, were associated with a previously undetected variability in the susceptibility of the Zircaloy cladding to in-reactor nodular corrosion, and were confined to BWRs with copper alloy condenser tubes and filter demineralizer condensate cleanup systems. Besides Vermont Yankee, four other BWR plants (Browns Ferry, Hatch, Limerick, and Peach Bottom) have subsequently encountered CILC failures. GE's experience with gadolinia began in 1965 when rods containing Al_2O_3 -4.7 wt% Gd_2O_3 pellets were inserted in Dresden-1.(15) GE has produced over 240,000 (U,Gd) O_2 fuel rods since 1971; except for a very small number of such rods (less than about 0.1%) that were susceptible to CILC, the in-reactor performance of the

(U,Gd) O_2 rods is similar to that of UO_2 fuel rods.⁽¹⁵⁾ Events involving failures of the burnable poison rods at two of the BWRs are described below.

- Hatch-1 - In the fall of 1981, a number of fuel assemblies were identified as leakers at Hatch-1.⁽³⁴⁻³⁸⁾ A detailed inspection was made of 33 fuel assemblies; in all of the assemblies inspected, the burnable poison rods containing urania-gadolinia pellets were the ones that had failed. The inspections involved a combination of eddy current and ultrasonic techniques, as well as visual examinations. Small through-wall pits were noted at the failure sites on the rods. It was concluded that the burnable poison rod failures occurred as a result of CILC. All of the burnable poison rods (failed and nonfailed ones) were replaced.⁽³⁴⁾
- Browns Ferry-2 - Leak testing (sipping) of the full core of Cycle-4 fuel indicated the presence of 31 leaking fuel assemblies.^(39,40) The fuel assemblies were inspected during October and November 1982. The fuel rod failures involving both gadolinia- and nongadolinia-containing rods were caused by CILC. Except for four damaged ones, all the fuel assemblies were reconstituted to conditions acceptable for continued core operation.

3.2.1.8.2 Underpressurized Gadolinia Rods - It was discovered that some underpressurized urania-gadolinia fuel rods (probably less than 100) were inadvertently installed in some BWR fuel assemblies that were shipped during 1980.^(41,43) There were 124 suspect fuel assemblies; of those, 122 had gone to Quad Cities-1 and 2 to Vermont Yankee. The underpressurization of the helium fill gas was the result of a malfunction of a weld station at the fabrication plant. The station was repaired; no new incidents have been reported.

3.2.1.9 Damaged Neutron Source Pin

At Limerick-2 in June 1989, the center stringer assembly holding seven startup neutron source pins unthreaded from the top of the assembly and fell about 35 ft through water to the cask pit floor.⁽⁴⁴⁾ The damage to one pin amounted to over \$2000; that pin was judged to be unacceptable for use.

3.2.1.10 Excessive Channel Bow

Fuel rod failures in a fuel assembly at a Swedish BWR in 1988 due to dryout were the result of the combined influence of excessive bow of channels on two high burnup fuel assemblies adjacent to the failed assembly.^(14,45) The large-bow channels were being recycled and had exposures of 54,000 to

68,000 MWD/MTU. The Swedish investigators indicate that bow measurements reveal a marked acceleration of channel bowing in the 40,000 to 50,000 MWD/MTU range.

In a U.S. paper,⁽⁴⁶⁾ it is indicated that differential growth of Zircaloy affects fuel rod and assembly bow. In a BWR, the result can also cause excessive channel bow.

3.2.2 Experience with Components at PWRs

Information on the experience with the integrity and behavior of neutron-absorbing (poison) components and a few other nonfuel components of fuel systems at PWRs is described in the five sections below.

3.2.2.1 Experience with Control Rod Assemblies

Experience with control rod assemblies (includes B&W control rod assemblies, C-E CEAs and Westinghouse RCCAs) is described in the following eight subsections.

3.2.2.1.1 Swelling of Hafnium Control Rods - The potential for swelling of hafnium control rods has developed into a safety concern since the phenomenon was first noticed at a PWR, Wolf Creek-1, in 1988.^(1,2) The enlargement is of concern because a control rod could swell beyond the inner diameter of its guide tube in the fuel assembly.⁽¹⁷⁾ In the U.S., there are 14 PWRs (11 units are operating, 3 units are under construction) that are equipped with hafnium control rods. Westinghouse Electric Corporation believes that the swelling occurs when hydrogen, under PWR operating pressures, diffuses through the stainless steel cladding and pierces the protective film on the hafnium.⁽⁴⁷⁾ [The NRC has commented^(14,48) that trapped moisture inside the rod could also result in a similar swelling effect.] Subsequent localized hydriding of the hafnium results in swelling (0.020 to 0.025 in. of expansion were measured at Wolf Creek-1). The localized hydriding was found by Westinghouse to exacerbate the effects of uniform hydriding that were considered in the design of these RCCAs.⁽¹⁾ Studies by Westinghouse indicate that the maximum swelling could be 0.030 in. in the worst condition, which would leave a clearance of about 0.010 in. for control rod insertion, assuming a symmetric geometry. Westinghouse indicated in January 1990 that it will no longer "actively market" hafnium control rods and would only fabricate them by

special request. Recent evaluations of the hafnium RCCAs indicate that safe operation of the affected PWRs is not compromised at least through the end of either a fourth annual cycle or a third 18-month cycle.(1)

In addition to the swelling, thinning of the control rod cladding, a result of wear caused by coolant-induced vibration of guide cards, was also observed at Wolf Creek-1.(14,48) Some through-wall wear marks were noted. A total of 53 RCCAs was found to be affected by swelling and wear.

A Westinghouse PWR in Taiwan had an identical problem with swelling and wear with a RCCA. A hafnium control rod could not be fully re-inserted into Maanshan-1 in September 1988--the rod had a broken tip and the cause of the failure was identified as hydriding.(49)

Concerning control rod insertability at BWRs, the swelling may be less of a problem for BWRs because the hafnium rod is encased in a stainless steel tube that is surrounded, in turn, by a stainless steel sheath to form a control blade.(14,48)

3.2.2.1.2 Fretting Wear and Cladding Cracking - While the RCCAs were fully withdrawn and stationary, fretting wear of RCCAs against upper internal guide cards was detected in 1983 at two operating Westinghouse-designed plants, Point Beach-1 and -2.(1,18) The guide cards are employed to position the rodlets in the guide housing when the RCCAs are parked in their normal full-out position; the wear is the result of fretting (vibratory interaction) between the guide cards and the rodlets during extended periods of steady state power operation. Based on the results of a visual examination, one RCCA at Point Beach-1 and 21 RCCAs at Point Beach-2 were replaced. Examinations in 1984 also revealed hairline cracks at the rodlet tips for two rodlets at Point Beach-1 and one rodlet at Point Beach-2; however, neither the cracks nor the wear caused operational problems. The cracks were not expected to interfere with RCCA travel and the absorber material was confirmed to remain stable (i.e., no detectable loss of absorber reactivity worth) in the presence of a crack. As a result of these observations, Westinghouse issued guidelines to all utilities suggesting axial repositioning of the RCCAs; Westinghouse also suggested inspection of RCCAs for replacement before through-wall wear of the cladding. Also, monitoring of the reactor's primary coolant for silver

and hafnium isotopes is a reliable means for detecting fretting wear through the cladding for Ag-In-Cd and hafnium RCCA types. The same type of guide card fretting wear and tip cracking was found at other plants during subsequent inspections of RCCAs; the plants included ones that use 14x14, 15x15, and 17x17 fuel assemblies. Replacement RCCAs were designed with modifications to increase the wear resistance of the cladding and reduce the susceptibility to rodlet tip cracking.

An underwater visual (TV camera) inspection of three RCCAs at a Westinghouse-designed PWR, Kewaunee, in April 1984 revealed apparent wear marks on the cladding of the RCCA rodlets; however, the amount of wear did not exceed the Westinghouse criteria for wear depth.⁽⁵⁰⁻⁵³⁾ The cladding wear occurred because of the design of the guide cards.

Eddy current examinations of RCCAs were conducted at a Westinghouse-designed PWR, Haddam Neck (also known as Connecticut Yankee), in March 1986.^(54,55) Evidence was found of cladding cracking, sliding wear, and guide card wear; two of the worst RCCAs were replaced. Also, five of the RCCAs were moved from the control rod bank to the shutdown bank; the fully withdrawn position of the shutdown bank was also raised.

As noted in Section 3.2.2.1.1, wear caused by coolant-induced vibration of guide cards was also noted at Wolf Creek-1 in 1988.^(1,2) Some through-wall wear marks were found in the control rod cladding.

It is indicated in an 1989 article⁽⁵⁶⁾ that the control rods at two Westinghouse-designed PWRs, Braidwood-1 and -2, are to be inspected for evidence of wear or fretting due to flow-induced vibrations.

3.2.2.1.3 Problems with RCCAs - Westinghouse-designed, full-length RCCAs have been successfully operated throughout the world in over 50 reactors.^(1,18) Currently, there are over 2300 RCCAs in service. Experience to date indicates that there has been little difficulty with large numbers of control rods. The infrequent drive line malfunctions are usually related to the presence of metallic debris in the system (i.e., if debris is present, the control rod drive systems are susceptible to binding problems because of the small clearances that are required). Several problems have

occurred with Westinghouse-designed RCCAs and control rod drive lines; these problems are described in the 19 subsections below.

- Connecticut Yankee (also known as Haddam Neck) - In April and August 1968, two RCCA drive lines were immobilized due to vane separation caused by faulty braze joints.⁽¹⁸⁾ The RCCAs were replaced; no failures were noted during subsequent cycles. It was also reported in 1970 that it was difficult to latch a fuel assembly; an examination revealed that a radial vane of the spider assembly, which holds the absorber rods, had broken.⁽⁵⁷⁾ A second fuel assembly was also found to have a severed vane on the spider assembly.

As will be noted below, similar problems with failures of brazed joints have occurred at Cook-1, Robinson-2, Surry-1, and at one foreign plant (Ko-Ri-1).

- Point Beach-1 - In October 1970, an RCCA drive line was immobilized because of a metallic chip of CF8 casting alloy that lodged between the rod cluster control spider and an intermediate guide tube guide plate.⁽¹⁸⁾ The chip was removed.
- Robinson-2 - In November 1970, an RCCA drive line was immobilized when a weld spatter nugget lodged between a rod cluster control rodlet and a lower guide tube guideway.⁽¹⁸⁾ The drive line was freed and the guide tube and rod cluster control were replaced.
- Indian Point-2 - In April 1972, three RCCA drive lines were immobilized and there was a short-term malfunction of another RCCA drive line.⁽¹⁸⁾ Those events are described below.

Two drive lines were immobilized by galling of one of the rod cluster control absorber rodlets with the associated thimble tube. The galling was considered to be a unique occurrence; the fuel assemblies and the RCCAs were repaired.

One RCCA drive line was immobilized by interference between an intermediate guide tube plate and a rod cluster control vane that had separated from the hub. The chip causing the interference was CF8 material. The RCCA was replaced.

The short-period malfunction of an RCCA was caused by jamming due to a metallic chip. The drive line was freed by dislodging the chip.

- Point Beach-1 - In October 1972, an RCCA would not insert into a fuel assembly.⁽¹⁸⁾ The problem was caused by three rodlets on the RCCA that were bent during manipulation. The RCCA was replaced.
- Yankee Rowe - During 1972, all the control rods were replaced, except for two hafnium control rods, because of control rod follower distortion (see Changes 97 and 104 to the Facility Operating License). All those rods,

including the hafnium rods, showed no evidence of distortion at that time.

During a controlled shutdown of the plant in 1981, one control rod became inoperable and would not scram.^(58,59) The event was caused by a bowed Zircaloy control rod follower on a cruciform-shaped hafnium control rod. The bowing was apparently caused by a Zircaloy growth phenomenon that may occur in rods manufactured in the 1960s as a result of the method for stress relieving that was employed. As a result of this event, both hafnium rods were replaced with spare Ag-In-Cd rods of the same design.

- Robinson-2 - In April 1973, a rod cluster control vane with two rodlets dropped into a fuel assembly.⁽¹⁸⁾ The vane and the rodlets had separated from the RCCA; the cause of the failure was not determined. The RCCA was replaced.
- Jose Cabrera - In January 1975, this Spanish PWR sustained a power tilt with power depression.⁽¹⁸⁾ The event occurred because an absorber rod, which had separated from an RCCA finger, was inserted into the core. The cause of the problem was failure of a weld associated with a finger-to-antilock pin joint. The RCCA was replaced.
- Cook-1 - In April 1976, sticking of an RCCA was experienced and a flux map indicated that one or two rodlets had been inserted into the core.⁽¹⁸⁾ Because of the failure of a brazed joint, the RCCA vane and two rodlets separated from the hub. The RCCA was replaced.
- Connecticut Yankee (also known as Haddam Neck) - In November 1977, interference with the manipulator crane gripper was noted during attempts to latch a rodged assembly.⁽¹⁸⁾ The event was caused by an RCCA vane with two rodlets attached that had become separated from the RCCA hub. The RCCA was replaced.
- Cook-2 - In January 1978, a control rod hung up in a guide tube during drag testing.⁽¹⁸⁾ A foreign object was found in the guide tube and was removed.
- Salem-1 - On May 31, 1979, some RCCAs were found with missing rodlets. The missing rodlets had broken loose from the spiders and fallen into the associated fuel assembly guide tubes. It was found that a total of eight rodlets had separated from six RCCAs.⁽¹⁸⁾ Incipient cracks in the fingers that supported the rodlets were noted during subsequent examinations. The failures were attributed to stress corrosion cracking, a result of contamination of the internal threads of a few fingers that had been reworked with a tri-chlorethane-based lubricant.⁽¹⁸⁾ All RCCAs with fingers from the suspect group were replaced; use of the chloride-containing lubricant was banned. Failures by this mechanism have not been observed since that incident.
- Ko-Ri-1 - In December 1979, an RCCA vane with two rodlets fell into a fuel assembly at this Korean PWR.⁽¹⁸⁾ The vane and rodlets had separated

from the RCCA; cause of the failure was not determined. The RCCA was replaced.

- San Onofre-1 - December 1981, one RCCA stuck in a fuel assembly but was freed after repeated manipulations.^(60,61) Surveillance techniques indicated at that time the presence of one or two broken rodlets. A 1987 report⁽¹⁸⁾ indicates that there was one broken rodlet. The event was attributed to failure (of undetermined cause) of the weld (brazed joint) attaching the vane supporting the rodlet to the hub of the RCCA. This failure mode results in full insertion of the affected vane and rodlet while the remainder of the RCCA remains operable. The RCCA operated through the cycle with the missing vane and rodlet; the RCCA was replaced in December 1985.⁽¹⁸⁾ Reactor coolant samples from San Onofre-1 were analyzed and the results showed no detectable ^{110m}Ag , which indicated that the cladding on the silver-indium-cadmium absorber material was intact. Westinghouse indicated that this kind of failure is not a generic concern for PWRs.
- Beaver Valley-1 - An RCCA was damaged during placement of the upper internals in the reactor vessel.⁽⁶²⁾ During installation of the upper internals in the reactor vessel, misalignment of a control rod tube support caused a downward deflection of the associated fuel assembly. Both the fuel assembly and RCCA were damaged in the incident; another fuel assembly with a new RCCA was inserted in that core position.
- Surry-1 - In June 1984, a loose top nozzle clamp on a fuel assembly interfered with RCCA travel.⁽¹⁸⁾ The clamp was removed and the RCCA replaced. This event is unrelated to the December 1984 event noted below.
- Surry-1 - In December 1984, an RCCA vane with a single rodlet separated from the spider and inserted into a fuel assembly.⁽¹⁸⁾ The event was caused by a failure of a brazed spider hub-to-vane joint; the cause of the failure of the brazed joint was not identified. The RCCA operated through Cycle 8 with a missing rodlet; the RCCA was then replaced.
- Point Beach-1 - In June 1985, an RCCA hung up in its guide tube because of debris in the guide tube.⁽¹⁸⁾ The hub and the vanes were found to be bent. The RCCA was repaired.
- Ko-Ri-6 - In September 1985, one RCCA was damaged.⁽¹⁸⁾ The RCCA bent in the guide tube; a bent vane was found and the RCCA was repaired. The event was caused by debris.

3.2.2.1.4 - Loose Guide Tube Nut - A loose part was detected in a steam generator by the loose parts monitoring system at a PWR, Crystal River-3, in March 1981.^(63,64) Babcock & Wilcox identified the part as a control rod upper guide tube nut. About 46,000 nuts of this design have been or are in service in Babcock & Wilcox plants; there have been no other reported

problems. It was judged that the plant was safe to operate with a fuel assembly missing the upper nut on one of the 16 control rod guide tubes. It was concluded that this was a unique, isolated occurrence, and the U.S. Nuclear Regulatory Commission (NRC) concluded that there was no evidence to suggest that there was a generic defect in the guide tube nut design.⁽⁶⁴⁾

3.2.2.1.5 Broken Bolt - A control rod drive could not fully insert one of the 37 control rod pairs at Fort St. Vrain, a high-temperature gas cooled reactor (HTGR), in August 1989.^(65,66) It was found that the head of one of the control rod's Inconel clevis pin bolts had cracked, broken off, and wedged itself between the control rod and its guide tube. Such bolts had been replaced at the plant in 1985.

3.2.2.1.6 Mishandling of Control Rods - Handling procedures were violated at Fort St. Vrain in March 1988 when personnel handled control rods with bare hands.⁽⁶⁷⁾

3.2.2.1.7 French Control Rod Problem - At Gravelines-4 in France in April 1989, a control rod could not be inserted into the core.⁽⁶⁸⁾ It was found that the control rod had broken off and fallen to the bottom of a fuel assembly, and its spring was stuck within the guide tube. It was also found that the local wear on the control rod casing was far more severe than studies had predicted.

3.2.2.1.8 Japanese Control Rod Problem - A Japanese research reactor failed to attain criticality in July 1989 when all the control rods were withdrawn.⁽⁶⁹⁾ The lower portion (about 20-cm long) of one B₄C-containing rod had become disconnected (the attachment screws were loose) and was still in the core.

3.2.2.1.9 RCCA Examinations - C-E has developed an inspection technique to examine approximately 1000 irradiated RCCA rodlets and a larger number of irradiated CEA rodlets.⁽¹⁹⁾ The technique is based on a two-part eddy current examination; several anomalies such as fretting wear, sliding wear, and cracks in the cladding have been identified by the eddy current data. During early development of the technique, visual confirmation of cracks in the cladding was obtained; this is considered now to be an unnecessary step for confirming the existence and cause of a crack.

3.2.2.2 Experience with Burnable Absorber Assemblies

Experience with burnable absorber assemblies is described in the two sections below.

3.2.2.2.1 Experience with Westinghouse Burnable Absorber Assemblies - Westinghouse-designed burnable absorber assemblies have been employed in PWRs throughout the world for more than 15 years.⁽¹⁾ In the late 1980s, the WABA was the most frequently used burnable absorber. The more cost efficient WABA replaced the previous borosilicate glass design beginning in the early 1980s. One of the advantages of an integral burnable absorber over the conventional boron glass rods is the elimination of all problems at the reactor site regarding the handling and disposal of burnable poison rods.⁽²⁰⁾

WABA rods have been routinely used in reload cycles since 1983; an August 1989 report⁽¹⁾ indicates that no abnormalities have been observed. According to a June 1987 report,⁽¹⁸⁾ over 3900 WABA rods have been successfully irradiated and discharged without incident, and over 7300 WABA rods were in service at that time. In 1987, it was indicated that there were about 50 PWRs that were using or had used Westinghouse burnable absorber assemblies that contained borosilicate glass as the burnable absorber.⁽¹⁸⁾ Most of those assemblies were intended for only one cycle of use and were not removed after service.

There is no routine surveillance of burnable absorber assemblies in operating PWRs; no problems have been encountered except those described in the six subsections below.

- Ginna - While a secondary source assembly was being removed from its fuel assembly during the Cycle-3/4 refueling, several cracks/openings were observed in the cladding at the lower end of an associated burnable absorber rod.⁽¹⁸⁾ Results from tests for tritium were negative, even though bubbles were observed.
- Prairie Island-2 - Difficulty was experienced in removing a burnable absorber assembly from a fuel assembly.⁽¹⁸⁾ The burnable absorber assembly/fuel assembly was subsequently reinserted into the core and functioned without incident.
- Beznau-1 - A burnable absorber assembly rod and vane became separated from a secondary source assembly and stuck in the fuel assembly guide thimble tube during the Cycle-4/5 refueling of this Swiss PWR. The vane was cut off the stuck burnable absorber rod; the fuel assembly was

irradiated for one more cycle with a modified plugging device and operated without incident. The secondary source with the missing vane and burnable absorber rod was inserted into another fuel assembly for irradiation in Cycle 5. During the Cycle-5/6 refueling, two more burnable absorber rods from this same secondary source were stuck in the fuel assembly guide thimble tubes. It was found that one rod was broken off 18 in. from the bottom end of the rod and the other rod was completely broken off. The fuel assembly containing the two broken burnable absorber rods was discharged along with the secondary source assembly. According to the June 1977 report, (18) the cause of these stuck burnable absorber events has not been determined.

- Indian Point-2 - In February 1978, after two duty cycles, one burnable absorber rod (broken 24 in. from the top) was found jammed in the thimble tube of the associated fuel assembly. (18,70) To capture the unrecoverable piece of the burnable absorber rod, a thimble plug was inserted in the fuel assembly; during the third duty cycle, the fuel assembly functioned without incident. (18)
- Indian Point-3 - Four demonstration assemblies, each containing two pre-characterized WABA rods, were inserted for irradiation in Cycle 3 in early 1980. (18) Examination of the four assemblies during the Cycle-3/4 refueling outage revealed no anomalous conditions; however, one of the demonstration assemblies was mishandled during fuel assembly changeout and resulted in the breaking of a WABA rod.
- Failed Borosilicate Rodlet - One failure involving a borosilicate glass rodlet was reported in 1988. (1) In that event, one rodlet became detached from its assembly and remained in the guide tube of a fuel assembly. An attempt to remove the rodlet was unsuccessful; no failure mechanism was identified. (1)

3.2.2.2.2 Ejected Burnable Poison Rod Assemblies - Two BPRAs were ejected from the core of Crystal River-3 (B&W design) while operating at power in 1978. (23,86) The ejected BPRAs were carried out of the reactor vessel through the hot leg and into the steam generator; the vibratory motion of the spider hub, which weighs about 10 pounds, resulted in significant damage to the steam generator tube ends. It was determined by B&W (87) that the ejection of the BPRAs from the core was caused by fretting wear in the hold-down latching mechanism. The fretting wear occurred when the BPRAs were levitated during four-pump operation.

3.2.2.3 Experience with Burnable Poison Rods

Because burnable poison rods are typically an integral part of PWR fuel assemblies, they are included in the handling and storage of spent fuel assemblies rather than being handled and stored as separate entities. Urania-

gadolinia is being developed and used as a fuel-burnable absorber admixture for use in some PWRs to control reactivity and power peaking in the more highly enriched extended-burnup fuel.⁽⁷¹⁾ Nearly 37,000 of Advanced Nuclear Fuels Corporation's (ANF's) gadolinia-bearing fuel rods have been irradiated in 25 reactors (16 BWRs and 9 PWRs); none of the gadolinia-bearing rods have failed.⁽⁷²⁾ The gadolinia concentrations ranged from 1 to 10 wt%.

Westinghouse is developing an integral fuel burnable absorber (IFBA).⁽¹⁾ An IFBA is a fuel rod that contains some fuel pellets coated with a thin layer (<25 microns) of zirconium diboride (ZrB_2), which serves as a burnable absorber. About 80% of the pellets in an IFBA rod are the coated type. Over 40,000 IFBA rods are scheduled for delivery to PWRs by the end of 1988.⁽⁷³⁾ Results from postirradiation examinations indicate that the ZrB_2 coating is a reliable burnable absorber, maintains its structural integrity during irradiation, and has no deleterious effects on fuel behavior. ZrB_2 is favored over gadolinium as an IFBA for PWRs for several reasons: improved power peaking control, reduced residual activity, and its unchanged uranium enrichment requirement.⁽⁷⁴⁾

A few events involving burnable poison rods are described in the following three subsections.

3.2.2.3.1 Perforated Burnable Poison Rods - Events in which perforated burnable poison rods were detected occurred at St. Lucie-1 and a number of other C-E plants in the U.S. The burnable poison rods were fabricated with Al_2O_3 - B_4C pellets and Zircaloy cladding. Information about those occurrences is presented below.

- St. Lucie-1 - During power ascension testing on July 6, 1976, a power tilt was detected at 80% power and it was determined that the power distribution in the core was not as predicted for 80% power.⁽⁷⁶⁻⁷⁸⁾ The power tilt and the attendant growth were confirmed at power. Physics tests were performed at hot zero power conditions. The plant was shut down until an evaluation was completed.⁽⁷⁶⁾

Visual inspection of fuel assemblies indicated that the burnable poison rods had cladding penetrations.⁽²¹⁾ The event was caused by the perforated burnable poison rods within the fuel assemblies. Boron loss from, and redistribution within, the perforated poison rods were the sources of the observed anomalies. Because the poison rods are inside the fuel assemblies, only about 12% of the surface of all poison rods in an assembly are observable using visual techniques. Only Batch B

(80 fuel assemblies) and part of Batch C (28 fuel assemblies) contained burnable poison rods. Inspection of these showed that most contained at least one poison rod with apparent hydride blisters.⁽²¹⁾

The poison rods and fuel assemblies at St. Lucie-1 were virtually identical to those of five other C-E reactors that had been operating for over three years and had accumulated burnups up to 25,000 MWd/MTU without unusual power anomalies. One clear difference was that the St. Lucie-1 poison rods were reworked.⁽²¹⁾ After the fuel assemblies had been completed, it was decided to remove and disassemble the poison rods to permit the B_4C pellets to be chamfered. The chamfered pellets were then reloaded into the cladding and the completed poison rods reloaded into the fuel assemblies. Rework records were scrutinized to determine a potential explanation for the anomalous behavior. It was found that the failure mechanism was primary hydriding of the cladding caused by excessive moisture content.^(22,23,77) The manufacturing procedures were modified to reduce the moisture content; as a result, failures of poison rods by this mechanism have been effectively eliminated.^(23,77)

All 1300 poison rods in the 108 Cycle-1 fuel assemblies were removed and replaced by new poison rods.^(75,76) The Cycle-1 fuel assemblies were not designed to be nondestructively disassembled and rebuilt. Hence one of the problems was the difficulty in removing the poison rods. (On C-E's new fuel assemblies, the end caps unscrew and the fuel rods and poison rods can be easily withdrawn.) The on-site fuel assembly rework was successfully accomplished.⁽⁷⁵⁾

The examination of the perforated rods provided an opportunity to characterize B_4C leaching from the Al_2O_3 - B_4C pellets; the leaching presumably occurred by reaction with the high-temperature water that entered the rod during reactor operation.⁽²²⁾ It is well known that a boric acid solution is produced when B_4C comes in contact with an excess of water at elevated temperatures.⁽²²⁾ It was found that at low ^{10}B depletion levels (about 10%) the B_4C particles were readily leachable from the porous Al_2O_3 matrix if there was sufficient access to the reactor coolant. In some regions, nearly 100% removal of the B_4C occurred in eight weeks or less of reactor operation.

- Other C-E Plants - Burnable poison rods with the higher moisture content were already in a number of operating C-E plants when the poison rod failure occurred at St. Lucie-1 (see preceding entry under "St. Lucie-1").⁽²³⁾ Visual inspections of the fuel at these plants was performed by C-E during 1977 at their normal refueling outages. Failures of burnable poison rods were noted at all of the plants, but the frequency of failure was less than that for St. Lucie-1. The circumstances surrounding these failures and the modifications of the fabrication procedures to eliminate the failure mechanism were reviewed by the NRC.⁽²³⁾ It was stated at that time that the fabrication process modifications (principally involved tightening of the moisture specifications) should eliminate this failure mechanism from current and future core loadings.

3.2.2.3.2 Debris-Induced Fretting - Fuel inspections at a B&W-designed PWR, Arkansas-2, at the end of Cycle 1 revealed seven fuel assemblies having perforated fuel and/or burnable poison rods.^(78,79) In two of the seven assemblies, the rod perforations were attributed to debris-induced fretting (debris had been trapped below the bottom Inconel spacer grid); the cause of the rod failures in the other five assemblies was unknown at that time.

Inspection of the fuel at the end of Cycle 2 revealed 10 damaged fuel assemblies, of which one had a perforated poison rod. Debris-induced fretting was again observed; however, it was not believed to have caused the Cycle-2 failures.^(78,79)

3.2.2.3.3 Bowed Poison Rods - During the Core 1A shutdown at Maine Yankee, seven fuel assemblies were found to contain bowed poison rods.⁽⁸⁰⁾ The results of an examination indicated that the poison rod bowing was independent of axial location top to bottom, except that in most cases the maximum degree of bowing was near the midpoint between grid spacer locations.

3.2.2.4 Experience with Sources and Plugging Devices

Primary sources are used for only one cycle and are then removed; on the other hand, secondary source rods which are activated during the first cycle have successfully operated for many years.⁽¹⁸⁾ It is indicated in a 1989 report⁽¹⁾ that no operational problems were reported in 1988 or 1987 with sources and thimble plugging devices. Thimble plugs have been bent during occasional handling incidents; a simple repair procedure is employed to straighten the bent thimble plugs and the repaired plugs are reused. According to the report,⁽¹⁸⁾ a number of PWRs have justified operation, and are operating without thimble plugging devices.

There have been a few events involving sources and plugging devices. Those events are described below.

- Zion-1 - In the fall of 1978, a secondary source assembly could not be removed from a fuel assembly during the Cycle-3/4 refueling outage.⁽¹⁸⁾ The fuel assembly with the stuck source (scheduled for discharge) was not reinserted into the core for irradiation in the next cycle.
- Prairie Island-1 - In the spring of 1978, two source assemblies could not be removed during the Cycle-3/4 refueling activities.⁽¹⁸⁾ The fuel

assemblies with the stuck sources (scheduled for discharge) were not reinserted for additional irradiation in the next cycle.

- Ko-Ri-1 - During the Cycle-3/4 refueling, a secondary source assembly could not be removed from its fuel assembly.⁽¹⁸⁾ The source assembly, spider, and fuel assembly were visually inspected; no apparent anomalies were observed. The fuel assembly with the stuck source was reinserted into the core for irradiation in Cycle 4.
- Turkey Point-3 - While regenerative secondary sources were being relocated to new fuel assemblies in the spent fuel pit in April 1987, the operating personnel erred by not following the procedure and damaged a source.⁽⁸¹⁾ That secondary source was moved horizontally before it was fully withdrawn from a fuel assembly; as a result, the source contacted the fuel assembly and was damaged.
- Indian Point-2 - During the Cycle-2/3 refueling in 1978, a plugging device stuck in a fuel assembly (scheduled for discharge).⁽¹⁸⁾ This fuel assembly was not reinserted for irradiation in Cycle 3.

3.2.2.5 Binding of Components

Binding of orifice rod assemblies occurred at Arkansas-1 and binding of control rod blades occurred at Palisades. Those events are described below.

- Arkansas-1 - Inspection of 40 PWR fuel assemblies containing orifice rod assemblies (ORAs) at Arkansas-1^(a) revealed that the ORAs were binding in the guide tubes of seven fuel assemblies.^(82,83) This event occurred on December 28, 1973, before initial fuel loading. Fabrication misalignment combined with the small clearance between the orifice rod and guide tube nut inner diameter caused binding.⁽⁸⁷⁾ It is indicated in a subsequent report⁽⁸⁴⁾ that cocked upper guide tube nuts forced the orifice rods to bend and bind even though an examination showed no out-of-specification conditions.
- Palisades - After inspection of control blade upper end fittings, one blade bound slightly as it was being reinstalled into the core.^(82,85) The binding was caused by bent guide rod nut capture devices on two adjacent Type A fuel assemblies. Further inspection revealed 20 bent nut capture devices in the core, all on Type A fuel.

The bent nut capture devices were the result of design and/or procedural deficiencies.⁽⁸⁾ The 0.060-in. thick nut capture devices bend very easily. Also, it is difficult to insert the blade by crane without the blade catching on the edges of fuel assemblies.

(a) Arkansas Nuclear One, Unit 1 (also known as ANO-1).

4.0 REFERENCES(a)

1. Skaritka, J. 1989. Operational Experience with Westinghouse Cores (through December 31, 1988). WCAP-8183, Rev. 17, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.
2. "NRC Issues Notice to PWR Operators on Westinghouse Hafnium Control Rods." NuclearFuel 14(7):13-15 (April 3, 1989). (Wolf Creek-1)
3. "Examining In-Vessel Problems." Nuclear Engineering International 35(426):43-45 (January 1990). (ISSN 0029-5507)
4. Bailey, W. J., and A. B. Johnson, Jr. April 1980. "The Expected Condition of Spent LWR Fuel on Delivery to a Repository." High Level Radioactive Waste Management 2:1437-1443. (PNL-SA-17334)
5. Bailey, W. J. May 1990. "Experience with Failed or Damaged Spent Fuel and Its Impacts on Handling." Journal of the Institute of Nuclear Materials Management XVIII(3):30-33.
6. Office of Civilian Radioactive Waste Management. December 1987. Characteristics of Spent Fuel, High-Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation. DOE/RW-0184, Vol. 1, U.S. Department of Energy, Richland, Washington.
7. Eisenhut, D. G. (U.S. Nuclear Regulatory Commission), Memorandum to Division Directors, Office of Nuclear Reactor Regulation, "Information Memorandum No. 18--New Failure Mode for BWR Control Blades," dated October 22, 1979.
8. Brayman, K. W., and K. W. Cook. December 1979. Evaluation of Control Blade Lifetime with Potential Loss of B₄C. NEDO-24226, General Electric Company, San Jose, California.
9. Abel, J. A. (Commonwealth Edison Company). 1974. Letter to Directorate of Licensing, U.S. Atomic Energy Commission, "Dresden Unit 3 Inverted Control Rod Tubes," Docket No. 50249-468, dated July 2, 1974.
10. Reactor Control Blade Evaluation, MNPS-1 Special Report, July 23, 1973. (see Docket No. 50254-445, p. 13/14).
11. Commonwealth Edison. 1974. Letter to Directorate of Licensing, U.S. Atomic Energy Commission, "Quad-Cities Unit 1 Inverted Control Rod Tubes," Docket No. 50254-445, dated July 24, 1974.

-
- (a) The following references are available from the U.S. Nuclear Regulatory Commission's Public Document Room in Washington, D.C.: 7, 9-13, 28, 29, 32, 36, 38-43, 48, 51, 53, 55, 57, 59, 61-64, 71, 76, 78-80, and 83-87.

12. Directorate of Regulatory Operation, U.S. Atomic Energy Commission. 1974. Letter to Iowa Electric Light and Power Company, "Duane Arnold Energy Center 1 - Failure of Handle of Blade Guide Yoke," Docket No. 50331-165," dated March 27, 1974.
13. Directorate of Regulatory Operations, U.S. Atomic Energy Commission. 1974. Letter to Iowa Electric Light and Power Company, "Duane Arnold Energy Center 1 - Dropping of Blade Guide from Grapple," Docket No. 50331-165, dated April 11, 1974.
14. Bailey, W. J., and S. L. Wu. March 1990. Fuel Performance Annual Report for 1988. NUREG/CR-3950 (PNL-5210), Vol. 6, Pacific Northwest Laboratory, Richland, Washington.
15. Baily, W. E., and R. L. Crowther. November 1985. "Gadolinia Fuel Performance in BWRs." Trans. Am. Nucl. Soc. 50:552-553. Copyright 1985 by the American Nuclear Society, La Grange Park, Illinois.
16. Hoth, A. C., H. W. Keller, and J. M. Shallenberger. 1981. "Development of Hafnium and Comparison with Other PWR Control Materials." Trans. Am. Nucl. Soc. 39:397-399. Copyright 1981 by the American Nuclear Society, La Grange Park, Illinois.
17. "Westinghouse to Study Swelling of Hafnium Control Rod Cladding." NuclearFuel 13(25):10-11 (December 12, 1988).
18. Skaritka, J. June 1987. Operational Experience with Westinghouse Cores (through December 31, 1986). WCAP-8183, Rev. 15, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.
19. Ruzauskas, E. J., and V. P. Roy. April 1988. "Combustion Engineering's Technique for RCCA Examinations." In Proceedings of the International Topical Meeting on LWR Fuel Performance, p. 282. Williamsburg, Virginia. Copyright 1988 by the American Nuclear Society, La Grange Park, Illinois.
20. Lindberg, M., et al. April 1988. "Advanced Axial Gradation of the ASEA-ATOM PAAD Fuel." In Proceedings of the International Topical Meeting on LWR Fuel Performance, pp. 370-376. Williamsburg, Virginia. Copyright 1988 by the American Nuclear Society, LaGrange Park, Illinois.
21. Masters, R. 1976. "Investigating the Power Anomalies at St. Lucie 1." Nuclear Engineering International, pp. 33-34, November 1976.
22. Fuhrman, N., E. J. Ruzauskas, and G. P. Smith, Jr. April 1985. "Irradiation Behavior of Neutron Absorber Materials in Combustion Engineering's Burnable Poison and Control Elements." Light Water Reactor Fuel Performance DOE/NE/34130-1, Vol. 2, pp. 5-13 through 5-28.
23. Houston, M. D. December 1979. Fuel Performance Annual Report (Period through December 1978). NUREG-0633, U.S. Nuclear Regulatory Commission, Washington, D.C.

24. Hayduk, D. M. March 1987. Reference Manual of Core Components Fabricated By Combustion Engineering. CEND-428, Combustion Engineering, Windsor, Connecticut.
25. Anderson, C. K., et al. September 1987. "Advanced PWR Fuel Design Concepts." Trans. Am. Nucl. Soc. 56(Supplement No. 1):439-445. Copyright 1987 by the American Nuclear Society, La Grange Park, Illinois.
26. General Electric Company. September 1976. BWR Fuel Channel Mechanical Design and Deflection. NEDO-21354, General Electric Company, San Jose, California.
27. Westinghouse Electric Corporation. September 1986. Nuclear Fuel Data. WSTD-TME-148, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.
28. Gridley, R. L. (General Electric), Letter to P. S. Check (U.S. Nuclear Regulatory Commission), "Response to NRC Additional Question on Control Blade Life (B₄C Loss)," dated June 29, 1979.
29. Moseley, N. C. (U.S. Nuclear Regulatory Commission), Memorandum to B. H. Brier, et al., IE Bulletin No. 79-26, "Boron Loss from BWR Control Blades," dated November 19, 1979.
30. Franklin, D. G. November 1980. "Economic Incentives for Improved Core Materials Performance." Trans. Am. Nucl. Soc. 35:201-202. Copyright 1980 by the American Nuclear Society, La Grange Park, Illinois.
31. "Monticello Needs Longer Outage After 20-Month Run," Nucleonics Week 30(50)13-14 (December 14, 1989). Reprinted by permission of copyright holder, McGraw-Hill, Inc. Reproduction prohibited without express permission of McGraw-Hill, Inc.
32. The Millstone Point Company. 1973. Letter to Directorate of Licensing, U.S. Atomic Energy Commission, "Control-Blade Inspection at Millstone Point-1," Docket No. 50245, dated January 30, 1973.
33. Nucleonics Week 15(36):6 (September 5, 1974). Reprinted by permission of copyright holder, McGraw-Hill, Inc. Reproduction prohibited without express permission of McGraw-Hill, Inc.
34. Massaro, S. A. (U.S. Nuclear Regulatory Commission), ed., Power Reactor Events, November-December 1981, 4(1)5-6, U.S. Nuclear Regulatory Commission, May 1982.
35. "Hatch-1, LER No. 81-114/01T-0," LER Monthly Report - December 1981, February 19, 1982 (p. 57).
36. Manry, M. (Georgia Power Company), Letter to J. P. O'Reilly (U.S. Nuclear Regulatory Commission), "Hatch Unit 1, Reportable Occurrence Report 50-321/1981-114," dated November 10, 1981. (LER No. 81-114/01T-0)

37. "Reactor Notes," Nucleonics Week 23(4):12 (January 28, 1982). Reprinted by permission of copyright holder, McGraw-Hill, Inc. Reproduction prohibited without express permission of McGraw-Hill, Inc.
38. Georgia Power Company, Letter to J. P. O'Reilly (U.S. Nuclear Regulatory Commission), "Summary Report of Plant Hatch Unit 1 Fall 1981 Fuel Failures," October 5, 1983. (Docket No. 50-321)
39. Paulk, G. L. (U.S. Nuclear Regulatory Commission), Inspection at Browns Ferry Site Near Decatur, Alabama, Report Nos. 50-259/83-09, 50-260/83-09, and 50-296/83-09, Docket Nos. 50-259, 50-260, and 60-269, April 13, 1983.
40. Williamson, II, B. T., Results of Browns Ferry Unit 2 Reload 2 Fuel Inspection, Special Test No. 82-15, Tennessee Valley Authority, Browns Ferry Nuclear Plant, undated. (Docket No. 50-260)
41. Gridley, R. L. (General Electric), Letter to H. E. Blis and J. A. Silady (Commonwealth Edison Company), "Quad Cities 1 Reload 5 Gadolinia Rods," November 11, 1980.
42. Johnston, W. V. (U.S. Nuclear Regulatory Commission), Memorandum to T. A. Ippolito (U.S. Nuclear Regulatory Commission), "Underpressurized Gadolinia Fuel Rods," dated December 4, 1980.
43. Janecek, R. F. (Commonwealth Edison), Letter to Director of Nuclear Reactor Regulation (U.S. Nuclear Regulatory Commission), "Quad Cities Station Unit 1 Cycle 6, Operation per Provisions of 10 CFR 50.59, NRC Docket No. 50-254," dated November 25, 1980.
44. "Limerick-2, Docket 50-353, LER 89-001," Licensee Event Report (LER) Compilation for Month of August 1989, NUREG/CR-2000 (ORNL/NSIC-200) 8(8):64 (September 1989).
45. "World News...Fuel Failures Traced to Dryout," Nuclear Engineering International 34(418):10-11 (May 1989). (ISSN 0029-5507)
46. Strasser, A. A., and K. D. Sheppard. 1987. "Achieving Improved Fuel Reliability." Trans. Am. Nucl. Soc. 54(Supplement No. 1):165-167. Copyright 1987 by the American Nuclear Society, La Grange Park, Illinois.
47. "Westinghouse Says It Will Cease Making All-Hafnium Control Rods." Nucleonics Week 30(6):1, 13, and 14 (February 9, 1989). Reprinted by permission of copyright holder, McGraw-Hill, Inc. Reproduction prohibited without express permission of McGraw-Hill, Inc.
48. Wu, S. L. (NRC), Memorandum to M. W. Hodges (NRC), "Status of RCCA Wear and Swelling," dated December 13, 1988.
49. "Maanshan-1 to Restart After Replacing Hafnium Control Rods," Nucleonics Week 30(7):8 (February 16, 1989). Reprinted by permission of copyright holder, McGraw-Hill, Inc. Reproduction prohibited without express permission of McGraw-Hill, Inc.

50. "Kewaunee, LER 84-003," Licensee Event Report (LER) Compilation for Month of June 1984, NUREG/CR-2000 (ORNL/NSIC-200), 3(6):24 (July 1984).
51. Giesler, G. W. (Wisconsin Public Service Corp.), Letter to the U.S. Nuclear Regulatory Commission, "Docket No. 50-305, Operating License DPR-43, Kewaunee Nuclear Power Plant, Reportable Occurrence 84-003-00," dated May 1, 1984. (Kewaunee-1, LER 84-003-00)
52. "Kewaunee, Docket 50-305, LER 84-003, Rev. 1," Licensee Event Report (LER) Compilation for Month of October 1985, NUREG/CR-2000 (ORNL/NSIC-200), 4(10):41-42 (November 1985).
53. Hintz, D. C. (Wisconsin Public Service Corp.), Letter to U.S. Nuclear Regulatory Commission, "Docket No. 50-305, Operating License DPR-43, Kewaunee Nuclear Power Plant, Reportable Occurrence 84-003-01," dated October 31, 1984. (LER No. 84-003-01)
54. "Connecticut Yankee, Docket 50-213, LER 86-015," Licensee Event Report (LER) Compilation for Month of August 1986, NUREG/CR-2000 (ORNL/NSIC-200), 5(8):24-25 (September 1986).
55. Graves, R. H. (Connecticut Yankee Atomic Company), Letter to U.S. Nuclear Regulatory Commission, "Facility Operating License No. DPR-61, Docket No. 50-213, Reportable Occurrence LER 50-213/86-015-00," April 23, 1986. (Haddam Neck)
56. "Commonwealth Awards Multiple Outage Contract to Westinghouse." Nucleonics Week 30(33):1 (August 17, 1989). Reprinted by permission of copyright holder, McGraw-Hill, Inc. Reproduction prohibited without express permission of McGraw-Hill, Inc.
57. Connecticut Yankee Atomic Power Company. 1970. Letter to U.S. Atomic Energy Commission, "Preliminary Report of Two Failed Control Rod Cluster Assemblies." Docket No. 50213, dated May 15, 1970.
58. "LER No. 81-005/03L-0," LER Monthly Report - August 1981, September 25, 1981, p. 71.
59. Autio, H. A. (Yankee Atomic Electric Company), Letter to B. H. Grier (U.S. Nuclear Regulatory Commission), "Licensee Report 50-29/81-05/03L, Failure of #17 Control Rod to Scram," dated June 1, 1981. (LER No. 81-005/03L-0)
60. "San Onofre-1, LER 82-003," Licensee Event Report (LER) Compilation for Month of May 1982, NUREG/CR-2000 (ORNL/NSIC-200), 1(5):74, June 1982.
61. Ray, H. B. (Southern California Edison Company), Letter to R. H. Engelken (U.S. Nuclear Regulatory Commission), "Docket No. 50-206, San Onofre Unit 1, Licensee Event Report No. 82-003/99X-0," dated March 5, 1982.

62. Carey, J. J. (Duquesne Light), Letter to S. A. Varga (U.S. Nuclear Regulatory Commission), "Beaver Valley Power Station, Unit No. 1, Docket No. 50-334, License No. DPR-66, Cycle 4 Reload Safety Evaluation Report Revision," dated September 1, 1983.
63. Barnard, P. Y. (Florida Power Corporation), Letter to J. F. Stolz (U.S. Nuclear Regulatory Commission), "Missing Control Rod Upper Guide Tube Nut," dated April 2, 1981. (Crystal River-3)
64. Schwenk, G. A. (U.S. Nuclear Regulatory Commission), Memorandum to E. L. Jordan (U.S. Nuclear Regulatory Commission), "Loose Guide Tube Nut Discovered in OTSG at Crystal River Unit No. 3," dated April 10, 1981.
65. "Ft. St. Vrain, Docket 50-267, LER 89-009," Licensee Event Report (LER) Compilation for Month of December 1989 8(12):19 (January 1990).
66. "Ft. St. Vrain Problems Won't Affect MHTGR, Says GA's Blue," Nucleonics Week 30(37):5-6 (September 14, 1989). Reprinted by permission of copyright holder, McGraw-Hill, Inc. Reproduction prohibited without express permission of McGraw-Hill, Inc.
67. "Inspections...Public Service Company of Colorado," Atomic Energy Clearing House 35(33):A7-A9 (August 18, 1989). A publication of Congressional Information Bureau, Inc., Washington, D.C. 20005.
68. "Control Rod Incident Leads to Multiple Headaches for EDF," Nucleonics Week 30(21):1-2 (May 25, 1989). (Gravelines-4) Reprinted by permission of copyright holder, McGraw-Hill, Inc. Reproduction prohibited without express permission of McGraw-Hill, Inc.
69. "Research Reactor Rod Malfunction Rates Bottom of Severity Scale," Nucleonics Week 30(31):7 (August 3, 1989). Reprinted by permission of copyright holder, McGraw-Hill, Inc. Reproduction prohibited without express permission of McGraw-Hill, Inc.
70. Skaritka, J., and J. A. Iorri. 1981. Operational Experience with Westinghouse Cores (up to December 31, 1980). WCAP-8183, Rev. 10, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.
71. Mitchell, D. B., et al. October 1988. B&W Fuel Company 1987 Fuel Performance Report. B&W Fuel Company, Lynchburg, Virginia.
72. Skogen, F. B., A. H. O'Leary, and K. E. Karcher. April 1988. "Performance of Gadolinia Burnable Absorber Assemblies at Burnups to 50,000 MWd/MTU." In Proceedings of the International Topical Meeting on LWR Fuel Performance, pp. 348-355. Williamsburg, Virginia. Copyright 1988 by the American Nuclear Society, LaGrange Park, Illinois.
73. Cunningham, S. E., et al. April 1988. "ZrB₂ Integral Fuel Burnable Absorber Rod Performance." In Proceedings of the International Topical Meeting on LWR Fuel Performance, pp. 343-347. Williamsburg, Virginia. Copyright 1988 by the American Nuclear Society, LaGrange Park, Illinois.

74. Srinilta, S., R. F. Schmidt, and J. E. Pritchett. November 1987. "A Comparison of Gadolinium and ZrB_2 as Integral Fuel Burnable Absorbers in PWRs." Trans. Am. Nucl. Soc. 55:124-126. Copyright 1987 by the American Nuclear Society, La Grange Park, Illinois.
75. Harris, K. N., J. J. Hutchinson, and E. H. Smith, Jr. 1977. "On-Site Reconstitution of 108 St. Lucie 1 Fuel Assemblies." Trans. Am. Nucl. Soc. 26(Supplement 1):58-59 (August 1977). Copyright 1977 by the American Nuclear Society, La Grange Park, Illinois.
76. Florida Power and Light Company. 1976. Letter to Office of Inspection and Enforcement (Region II), U.S. Nuclear Regulatory Commission, "Update Report of Power Distribution Anomalies at St. Lucie 1 (RO 335-76-35)." Docket No. 50335, dated December 20, 1976.
77. Combustion Engineering, Inc. February 1977. St. Lucie Unit 1 - Repair Report. CEN-38(F), Rev. 1, Combustion Engineering, Inc., Windsor, Connecticut.
78. Clark, R. A. (U.S. Nuclear Regulatory Commission), Letter to J. M. Griffin (Arkansas Power and Light Company), "ANO-2 Fuel Inspection Results," dated July 24, 1983. (Docket No. 50-368)
79. Rubenstein, L. S. (U.S. Nuclear Regulatory Commission), Memorandum to G. C. Lainas (U.S. Nuclear Regulatory Commission), "ANO-2 Fuel Inspection Results," dated June 17, 1983.
80. Maine Yankee Engineering Office. 1976. Letter to U.S. Nuclear Regulatory Commission, "Poison Rod Bowing." Docket No. 50309-488, dated April 13, 1976.
81. "Inspections: Florida Power and Light Company," Atomic Energy Clearing House 33(49):A8-A9 (December 9, 1987), a publication of Congressional Information Bureau, Inc., Washington, D.C. 20005. (Turkey Point-3, Inspection Report 50-250/87-14).
82. Bobe, P. E. January 1976. Fuel Performance of Licensed Nuclear Power Plants Through 1974. NUREG-0032, U.S. Nuclear Regulatory Commission, Washington, D.C.
83. Phillips, J. D. (Arkansas Power & Light Company, Arkansas Unit 1) 1974. Letter to Directorate of Regulatory Operations, U.S. Atomic Energy Commission, "Fuel Assembly - Orifice Rod Assembly Binding." Docket No. 50313, dated February 13, 1974.
84. U.S. Atomic Energy Commission. 1974. Letter to Arkansas Power & Light Company, "Arkansas-1 Fuel Element-Orifice Rod Assembly Binding," Docket No. 50313-142, dated March 7, 1974.

85. Haueter, R. L. (Consumers Power Company, Palisades Plant) 1971. Letter to Directorate of Licensing, U.S. Atomic Energy Commission, Docket No. 50255, dated December 16, 1971.
86. Stewart, W. P. (Florida Power Corporation). 1978. Letter to C. Nelson (U.S. Nuclear Regulatory Commission), "Crystal River Unit 3 Status Report - May 1, 1978," Docket No. 50302, May 4, 1978.
87. Houston, M. D. (U.S. Nuclear Regulatory Commission). 1978. Memorandum to D. F. Ross (U.S. Nuclear Regulatory Commission), "Summary of Meeting with Babcock & Wilcox on Orifice Rod Assembly Operating Experience," June 13, 1978.

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