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Boiling-Water Reactor Internals Aging Degradation Study

Phase 1

Prepared by
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Prepared for
U.S. Nuclear Regulatory Commission

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Abstract

This report documents the results of a study on the effects of aging degradation on 25 selected Boiling-Water Reactor (BWR) internal components. The operating environment inside a BWR pressure vessel produces stressors that could lead to the development of aging-related degradation mechanisms. A data base containing aging-related failure information for the selected internal components is established using data from Licensee Event Reports. Results of the failure information survey identified two major aging-related degradation mechanisms for reactor internals: stress corrosion cracking (SCC) and fatigue. SCC includes intergranular SCC and irradiation-assisted SCC (IASCC).

Strategies for controlling and managing aging degradations are based on understanding the relationship between stressors and the associated aging-related degradation mechanisms. The implementation of a plant Hydrogen Water Chemistry (HWC) program is considered to be a promising method for controlling SCC, which is the more prevalent problem for BWRs. Flow-induced vibration (FIV) is the major cause of fatigue problems in BWR internals. FIV problems are resolved either by eliminating the excitation sources or by detuning the structure from input excitations. Questions remain concerning the effectiveness of HWC in mitigating SCC (including IASCC) in internals and in the assessment of high-cycle fatigue in a corrosive environment.

Vibration monitoring, based on neutron noise measurements and trending studies, is an inspection method that can provide early failure detection capability and can improve the effectiveness of current plant in-service inspection programs. However, the large water gap and the lack of existing ex-core neutron flux monitors may hinder the use of neutron noise vibration measurements in BWRs.

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Acknowledgments

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Summary

Reactor internals are located inside the pressure vessel, and the operating environment is favorable to the development of time-dependent or aging-related degradation mechanisms. The main objective of this study is to assess the effects of aging degradations on Boiling-Water Reactor (BWR) internal components.

Twenty-five BWR internals have been selected for the aging study; they serve five basic functions: (1) balance-of-plant steam quality control, (2) core support, (3) reactor coolant flow control and core heat transfer enhancement, (4) housings for in-core instruments and specimen samples, and (5) core power shaping and reactivity control. Fuel assemblies, control rods, and in-core monitors are not included in the scope of the present study.

Stressors are conditions that can initiate and sustain the growth of aging degradations. Reactor internals are subjected to stressors associated with applied loadings (thermal and mechanical), the reactor cooling water, the exposure to fast (E > 1 MeV) neutron fluxes, and manufacturing processes. Flow-generated oscillatory hydrodynamic forces and temperature fluctuations are major applied load stressors. The reactor cooling water can create an environment that is conducive to the development of stress corrosion cracking (SCC). Cast austenitic stainless steels can become embrittled by prolonged exposure to high temperatures. Long-term exposure to fast neutron fluxes can lead to changes in the mechanical properties of a material, which can become susceptible to brittle fracture and SCC. Chromium depletion in grain boundaries of austentic stainless steels and residual stresses in weld heat affected zones (HAZs) are primary stressors associated with manufacturing processes.

Potential aging-related degradation mechanisms for BWR internals include SCC, fatigue, erosion, embrittlement, creep, and stress relaxation. These aging mechanisms develop at different rates and may not be equally important in the design life of a reactor. Results of a component failure information survey identified SCC and fatigues as the two major aging-related degradation mechanisms for BWR internals.

Major reported failures of BWR internals include cracks in jet pump supports caused by SCC and fatigue, SCC in core spray spargers, fatigue cracks in feedwater spargers, and SCC (in some cases irradiation assisted) in in-core monitor dry tubes. Some of these failures have led to lengthy repair times, but there is no indication that these failures compromised the safety of reactor operations.

Aging degradations, if left unmitigated, will eventually cause a failure in an affected component. It is essential to control or eliminate stressors associated with the major aging-related degradation mechanisms. The development of SCC requires three conditions: (1) a susceptible material, (2) a corrosive environment, and (3) the presence of tensile stresses. The elimination of any one of the three conditions will reduce the likelihood of the development of SCC. The effects of the welding-related chromium depletion process, which can make stainless steel components susceptible to SCC, can be reduced by the use of low-carbon-content stainless steels. The tensile stress level in a structural component can be lowered by the use of larger components or by reducing the magnitude of the applied loads. Heat sink welding methods can control the level of residual stresses in a weld HAZ. The implementation of a plant Hydrogen Water Chemistry (HWC) program is intended to reduce the corrosiveness of the reactor cooling water by lowering its dissolved oxygen content. HWC is effective in mitigating SCC in the reactor recirculating water piping system, but there are insufficient data to assess its effectiveness in reactor internals. A major disadvantage of the HWC program is the increase in radiation level in the power-generating areas. Irradiation-assisted SCC (IASCC) is not well understood, and many active research projects are trying to gain more insight and understanding into this phenomenon.

Flow-induced vibration is the major cause of fatigue problems in reactor internals. Low-cycle fatigue is caused by large-amplitude vibrations and can be prevented by detuning the system's natural frequency from the dominant flow-generated excitation frequency. Large-amplitude vibrations are usually detected and corrected during reactor preoperational flow testings. Reactor internals are susceptible to high-cycle fatigue caused by small-amplitude vibrations, which are much more difficult to control and eliminate. The synergistic effects of high-cycle fatigue and a corrosive medium are not well quantified at the present time. High-cycle fatigue is an active aging-related degradation mechanism for reactor internals.

In the presence of active aging-related degradation mechanisms, it is essential to maintain a vigorous inspection program to ensure the structural integrity of reactor internals. The plant in-service inspection (ISI) program calls for the visual inspections of accessible areas of internals during refueling outages. The limitations of the visual inspection method are well known, and various alternate inspection methods, such as ultrasonic and eddy-current inspections, have been tried on an experimental basis. Reactors licensed after 1978 are equipped with loose part monitoring systems (LPMSs), which can indicate the presence of loose parts in the reactor primary system. Appropriate actions taken by plant operators can limit further damage to other reactor components and systems. LPMSs do not possess the capability of indicating the current status of flaws that may exist in a component. Neutron noise vibration measurements and trending studies can provide the information that can be used to evaluate the current status of flaws that may exist in selected core internal components. However, these practices have not been fully exploited by the domestic utility industry. Visual inspection, supplemented by ultrasonic and eddy-current inspection, remains the major method for inspecting reactor internals. The implementation of a monitoring system with early failure detection capability will further enhance the safety and efficiency of reactor operations.

1 Introduction

The effects of aging or time-dependent degradation on systems, structures, and components of a commercial nuclear power plant are of concern to the U.S. Nuclear Regulatory Commission (NRC). The Office of Nuclear Regulatory Research (RES) has initiated and sponsored a Nuclear Plant Aging Research (NPAR) Program.[1] The main objective of the program is to study aging-related degradation mechanisms and their effects on major reactor systems, structures, and components. The NPAR Program also includes evaluation of the effectiveness of inspection, monitoring, and maintenance methods for managing aging effects. The basic NPAR approach is to identify major reactor systems, structures, and components that are susceptible to the effects of aging degradation. Detailed aging studies are then conducted for these systems, structures, and components. One of the Oak Ridge National Laboratory's (ORNL's) assignments is to evaluate the effects of aging degradation on reactor internals.

Reactor internals are components located inside the reactor pressure vessel (thus, the term "internal"). The components perform several different functions. Many internals are parts of the core support structure, while other components direct the coolant flow through the vessel and control the heat transfer in the core. Housings for in-core monitors and specimen samples are also considered as reactor internals. Internals for Boiling-Water Reactors (BWRs) and Pressurized-Water Reactors (PWRs) are different in design and construction, and they will be treated in separate studies. This Phase 1 report will concentrate on the effects of aging degradation on BWR internals.

Thirty-seven BWRs are licensed for commercial operations in the United States.[2] Using the commercial operation starting date as the reference for counting reactor ages, 7 reactors (or ~19% of the total) are over 20 years old; 16 reactors (or 43%) are between 10 and 20 years old; and 14 reactors (or 38%) are <10 years old. There is a total of ~522 reactor-years of BWR operations; the operating histories of these reactors provide useful information for studying aging effects in selected reactor components.

Aging assessment is a multiple-step process. The first step is the identification and description of the selected internal components. The second step is to identify stressors that are present in the operating environment of the selected components. The third step is to establish the relationship between the stressors and potential aging-related degradation mechanisms. The final step is the identification of major aging-related degradation mechanisms based on a review of the operating history of the reactors and reported component failure information. The understanding of the relationship between major stressors and associated aging-related degradation mechanisms also provides the basis for formulating strategies for controlling and managing aging effects.

Twenty-five BWR vessel internals selected for the present study are identified in Chap. 2, which will also provide a brief description of the function, design, and construction of each component. Major stressors for these selected reactor internals are discussed in Chap. 3, and potential aging-related degradation mechanisms associated with the stressors are identified in Chap. 4. Chapter 5 summarizes the reported aging-related BWR internal failure information. The NPAR Program also addresses issues involving the inspection and maintenance methods used to control and manage aging effects. The effectiveness of current in-service inspection (ISI) programs and strategies for managing aging degradations are discussed in Chap. 6. Important results of this Phase 1 aging assessment of BWR internals are summarized in Chap. 7.

References

1. J. P. Vora, "Nuclear Plant Aging Research (NPAR) Program Plan," USNRC Report NUREG-1144, Rev. 1, September 1987.*

2. M. D. Muhlheim and E. G. Silver, "Operating U. S. Power Reactors," *Nucl. Saf.* 32(3) (July–September 1991).†

*Available for purchase from National Technical Information Service, Springfield, VA 22161.
†Available in public technical libraries.

2 BWR Internal Components

This study considers the effects of aging degradations on reactor internal components in BWRs designed by the General Electric Company (GE). It includes reactor design versions from BWR-2 to BWR-6. The only significant difference between these reactors is that jet pumps are used beginning with BWR-3 and in all subsequent BWR designs. Following are the 25 BWR internals selected for this study:

1. steam dryer
2. steam separator
3. steam separator support ring
4. shroud head
5. shroud head bolts
6. top guide
7. core shroud
8. core plate
9. orificed fuel supports (OFSs)
10. shroud support including access hole covers
11. core spray internal piping
12. core spray sparger
13. feedwater sparger
14. jet pump assembly
15. in-core neutron flux monitor housings
16. in-core neutron flux monitor guide tubes
17. in-core neutron flux monitor dry tubes
18. control rod drive (CRD) housing
19. jet pump sensing line
20. neutron source holder
21. control blades
22. vessel head cooling spray nozzle
23. control rod guide tubes
24. surveillance sample holders
25. differential pressure and liquid control line.

Most of the 25 selected components are commonly identified as "internals" in the Final Safety Analysis Report (FSAR) for a BWR plant; the three exceptions are the in-core neutron flux monitor housings, the CRD housing, and in-core neutron flux monitor dry tubes. The CRD and in-core neutron flux monitor housings are welded to the reactor vessel and are usually considered parts of the pressure vessel. Dry tubes can be considered as a part of the in-core neutron flux monitor. These components are included as internals because they perform a basic internal function (i.e., providing housings to in-core instruments).

The fuel assemblies, control rods, and in-core monitors are reactor components also located inside the reactor vessel. They perform unique functions that are different from functions of reactor internals. Fuel assemblies, control rods, and in-core monitors will not be included in the scope of the present study.

A simplified sketch of the arrangement of major BWR internal components with jet pumps is shown in Fig. 2.1(*a*), and the arrangement of internals in a BWR without jet pumps is shown in Fig. 2.1(*b*).

2.1 Functions of BWR Internals

The 25 selected BWR internals perform a variety of functions. They can be grouped into five general areas: (1) balance-of-plant (BOP) steam quality control, (2) core support structures, (3) reactor coolant flow control and core heat transfer enhancement, (4) housings for in-core instruments and specimen samples, and (5) core power shaping and reactivity control.

Two of the selected internals, the steam dryer and steam separators, are used for BOP steam quality control. Eight of the 25 components provide structural supports to the core and other internals: steam dryer support ring, shroud head, shroud head bolts, top guide, core shroud, core plate, OFSs, and access hole cover. The next largest group of internals are housings for in-core instruments and specimen samples: in-core neutron flux monitor housings, in-core neutron flux monitor guide tubes, in-core neutron flux monitor dry tubes, CRD housings, neutron source holder, control rod guide tubes, jet pump sensing line, and surveillance sample holders. Six components are used for coolant flow control and heat transfer enhancement: core spray internal piping, core spray sparger, feedwater sparger, vessel head cooling spray nozzle, differential pressure and liquid control line, and the jet pump assembly. Control blades and jet pumps are used for core power shaping and reactivity control.

The majority of the BWR internals are made of type 304 stainless steel. They are designed in accordance with the rules and regulations of Sect. III of the *American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME B&PV) Code*.[1] Specifically, internal components that are parts of the core support system and those that are used for reactor coolant flow control and core heat transfer enhancement are treated as safety-class items and are designed to meet the requirements as stipulated in Appendix 1 of Sect. III of the *ASME B&PV Code*. CRD housings, control rod guide tubes, and in-core monitor housings that are a part of the reactor primary pressure boundary are also designed to meet all Sect. III requirements. Allowable stress values from Sect. III are used as design guides for internal components that are not safety-class items. Safety-class internal components are fabricated in accordance with requirements of Sect. III, Subsection NG; other internals are fabricated to meet requirements of Sect. IX of the

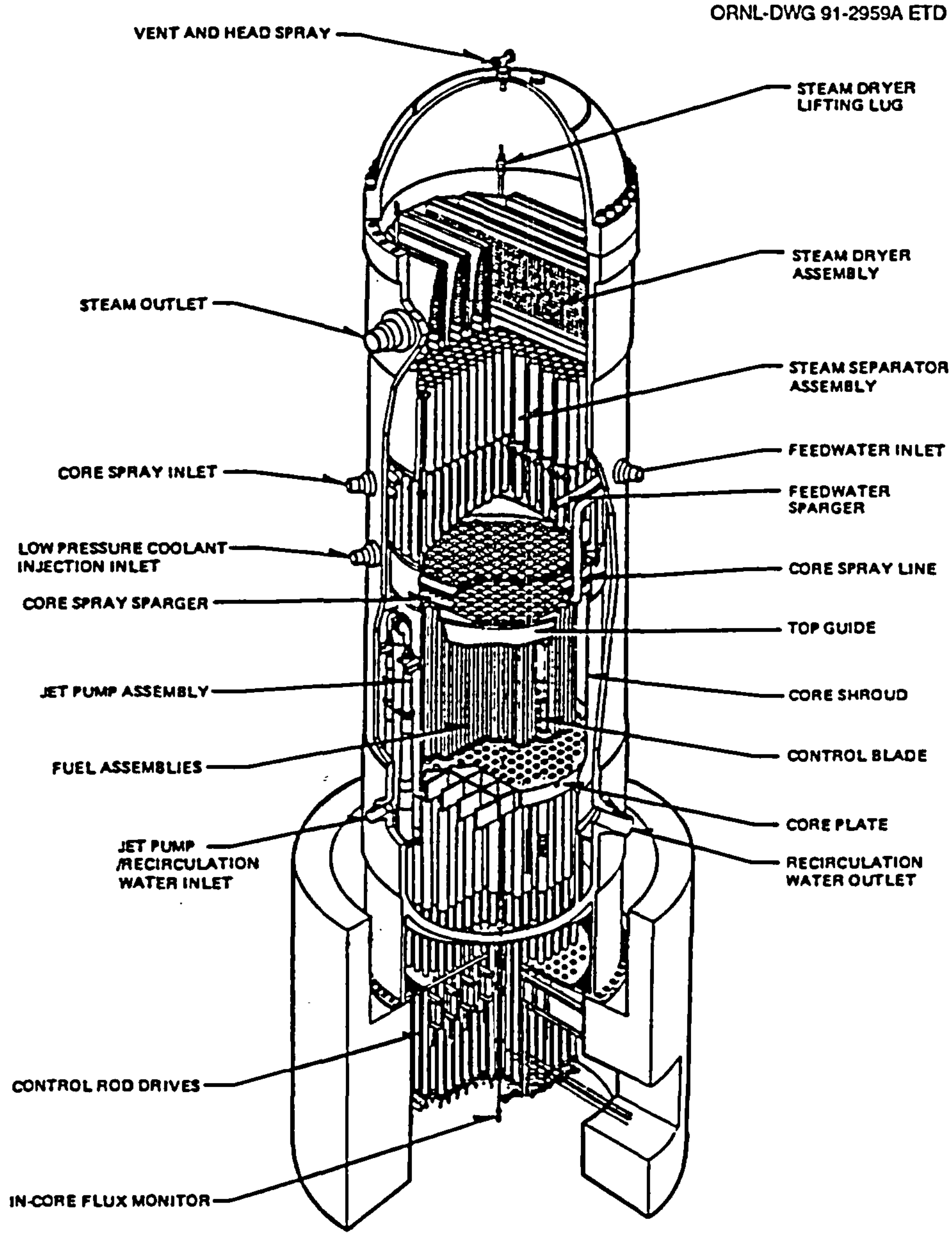

Figure 2.1(*a*) BWR internals with jet pumps

ASME B&PV Code.[2] For reactors that were designed and built before the establishment of Sect. III, analyses are performed to ensure that calculated stress values for such reactor components meet the intent of the Code under specified design conditions.

The following is a brief description of the function, design, and construction of the 25 selected BWR reactor internal components.

2.2 Component Description

Information on the design of the 25 selected BWR internals is obtained from various plant FSARs and supplemented by the Electric Power Research Institute (EPRI) report on the Monticello pilot plant life extension study.[3]

Reactor design is an evolving process, and many design features are plant specific. Internal components included in this report are those for a "typical" reactor designed by the vendor. The report will attempt to identify major changes

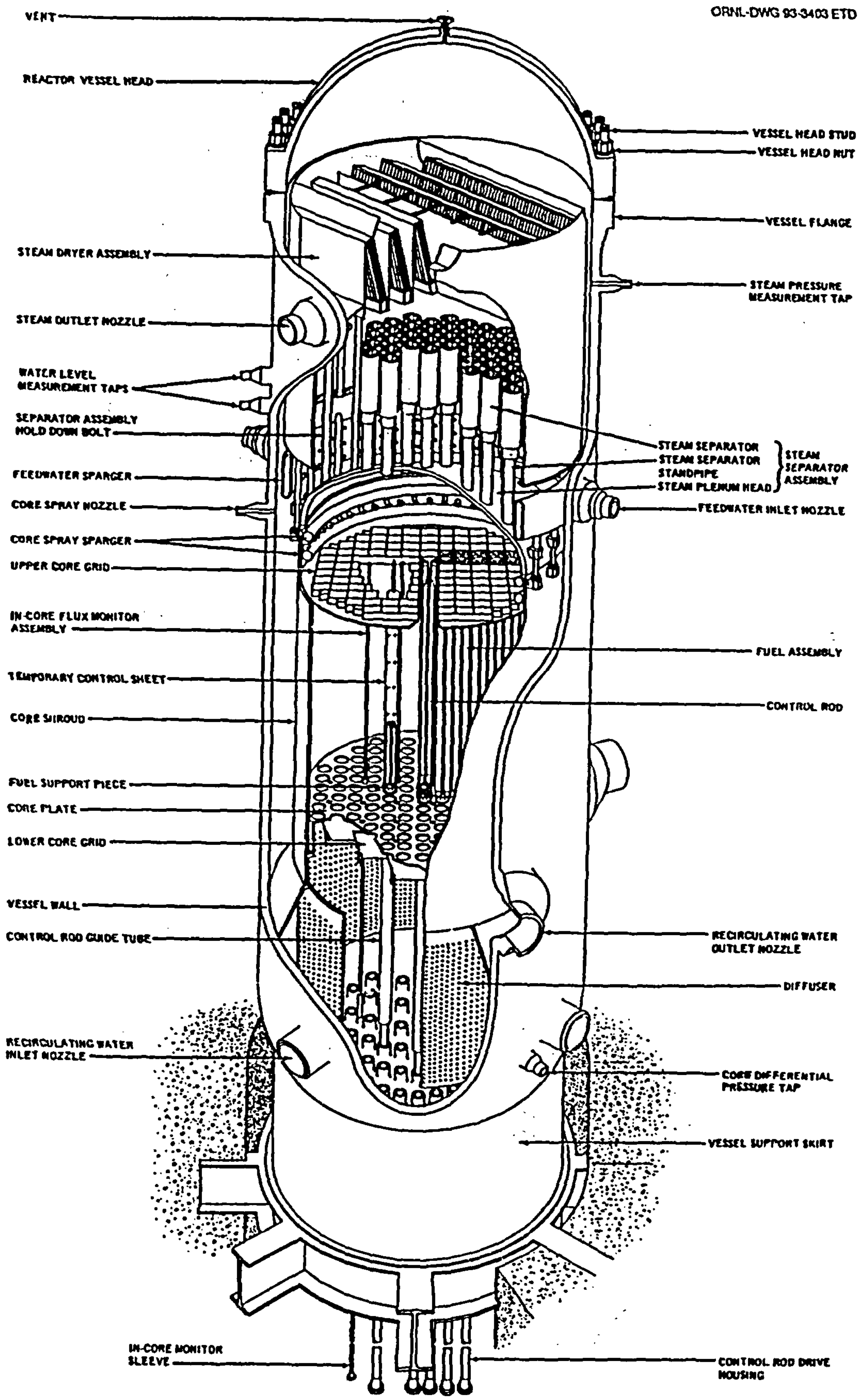

Figure 2.1(*b*) BWR internals without jet pumps

in internal components of the different reactor design versions, but no attempt will be made to identify all design changes.

Component No. 1: Steam Dryer

Function: The steam dryer removes the excess moisture in the steam exiting from steam separators. It is not needed for steady-state operations in the newer BWRs.

Description: The steam dryer assembly consists of the drying vanes and the top and bottom structural support members. They form a single structural unit that is mounted above the steam separators. The entire unit is supported by brackets extending inward from the pressure vessel wall. The moisture removed by the dryer is carried by a system of troughs and drains to the downcomer, which is the annular space between the core shroud and the pressure vessel wall. A sketch of a steam dryer unit is shown in Fig. 2.2.

Design Code: The steam dryer is designed to meet the intent of Sect. III of the *ASME B&PV Code.*

Material of Construction: The steam dryer is made of type 304 stainless steel.

Component No. 2: Steam Separators

Function: The steam separators separate water droplets from the steam in the steam-water mixture generated in the core.

Description: The steam separator assembly consists of 129 standpipes welded to openings in the shroud head, and an axial-flow steam separator is attached to the top of each standpipe. Fixed vanes are located inside the separator. As the steam-water mixture rises through the standpipes, a vortex is generated by the spinning action imparted to the mixture by the vanes. The vortex flow separates the water from the steam in each of the three stages of the separator. The steam passes through the top of the separators and into the dryer assembly. The water flows down the standpipes and into the annular space between the core shroud and the pressure vessel wall. A sketch of the steam separator assembly is shown in Fig. 2.3.

Design Code: The steam separator is designed to meet the intent of Sect. III of the *ASME B&PV Code.*

Material of Construction: Standpipes and steam separators are made of type 304 stainless steel.

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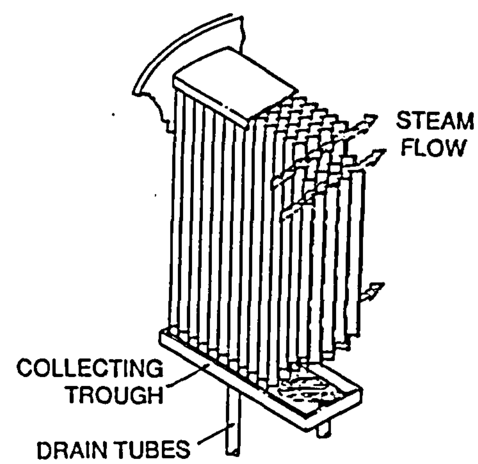

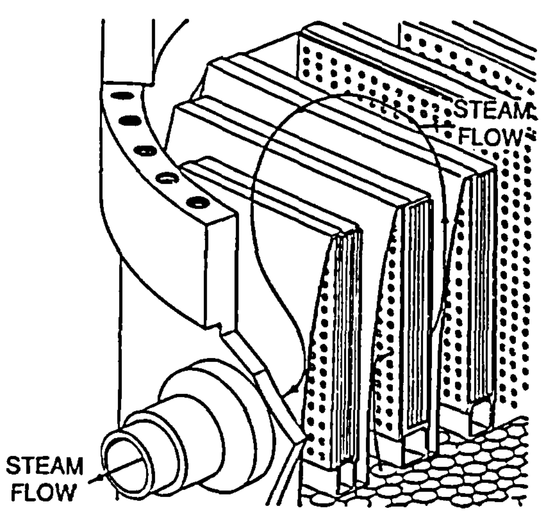

Figure 2.2 Steam dryer assembly

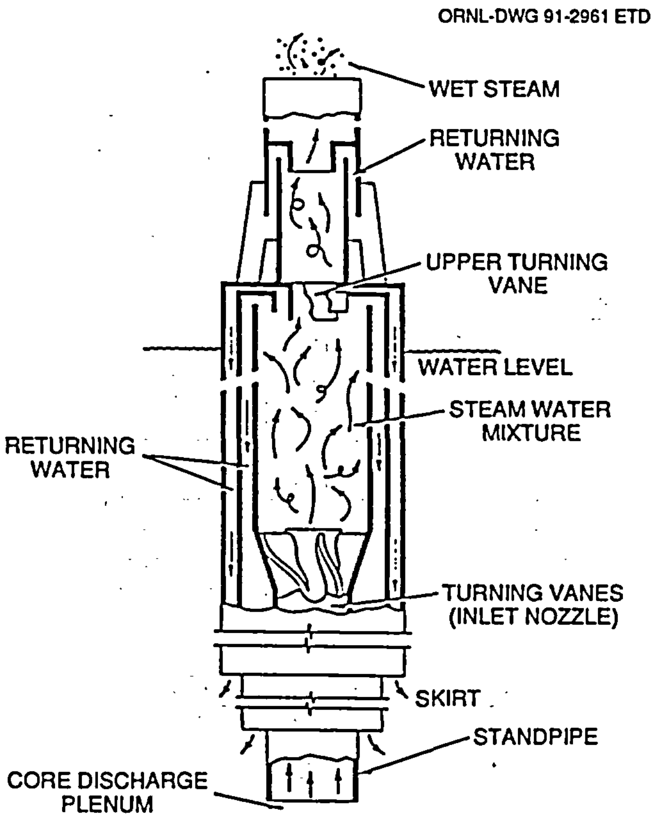

Figure 2.3 Steam separator assembly

Component No. 3: Steam Dryer Support Ring

Function: The steam dryer support ring supports the steam dryer assembly. The ring is attached to four support brackets that are welded to the pressure vessel wall.

Description: The circular support ring is made of cold-formed steel sections. The steel section is rectangular and the dimensions are 3 by 6 in. Some BWR plants have steam dryer support rings made of non-cold-worked steel sections.

Design Code: The support ring is designed to meet the intent of Sect. III of the *ASME B&PV Code*.

Material of Construction: The steam dryer support ring is made of type 304 stainless steel.

Component No. 4: Shroud Head

Function: The shroud head covers the discharge plenum region located at the top of the core. The plenum serves as a mixing chamber for the steam-water mixture before it enters the steam separators. The shroud head also provides structural support to the steam separators.

Description: The shroud head is a dome-shaped steel structure attached to the core shroud top flange by shroud head bolts. Steam separator standpipes are welded to openings in the shroud head. A sketch of the shroud head is shown in Fig. 2.4.

Design Code: The shroud head is designed to meet the intent of Sect. III of the *ASME B&PV Code*.

Material of Construction: The shroud head is made of type 304 stainless steel.

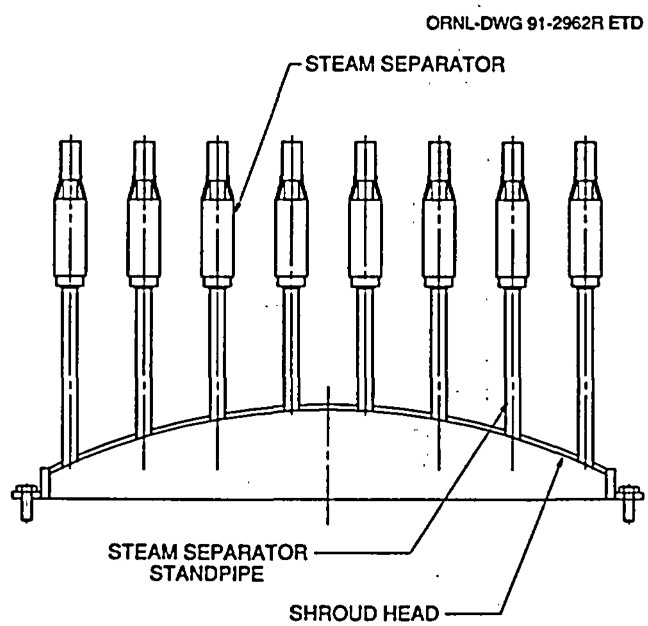

Figure 2.4 Shroud head

Component No. 5: Shroud Head Bolts

Function: Shroud-head bolts fasten the shroud head to the core shroud top flange. Typically, 36 bolts are used; however, the exact number of bolts and bolt sizes can vary from plant to plant.

Description: A typical shroud-head bolt is 1.75 in. in diameter and 14 ft long. A nut is screwed onto one end of the bolt, and a tee head is welded to the other. A sleeve covers the rest of the bolt, and the base of the sleeve is joined to a collar that is welded to the shaft near the tee head. A part of the collar is cut out to provide space for the alignment pin window. A sketch of a shroud-head bolt is shown in Fig. 2.5.

Design Code: The shroud head bolt is designed to meet the requirements of Sect. III of the *ASME B&PV Code*.

Material of Construction: The bolt, the nut, and the tee head are made of Inconel 600 alloy. The sleeve and the collar are made of type 304 stainless steel.

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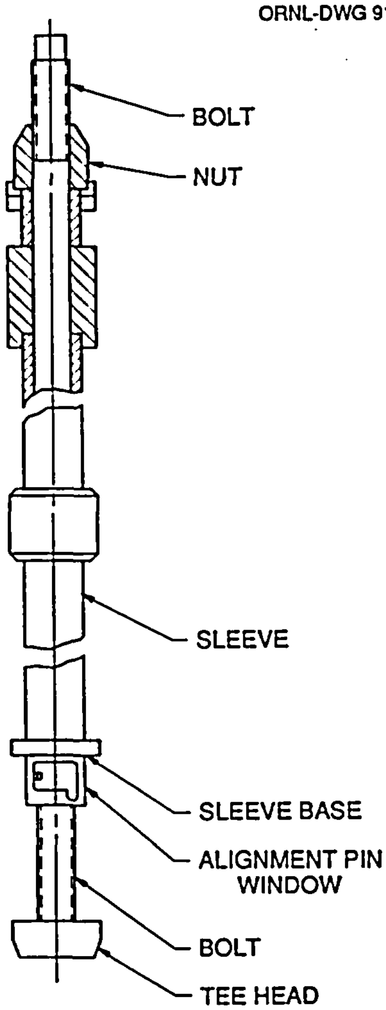

Figure 2.5 Shroud head bolt

Component No. 6: Top Guide

Function: The top guide provides lateral support and maintains proper spacing of the upper ends of the fuel assemblies.

Description: Top guides, with the exception of those in BWR-6s, are made by welding intersecting steel beams to a circular rim. At the intersecting joints, the upper beams have slots cut on the lower part, and the lower beams have slots cut on the upper part. The upper and lower beams are interlocked to form the grid. A crevice-free design, in which the rim and intersecting beams are machined from a single piece of steel, is used in BWR-6s. Each square opening formed by the intersecting beams provides lateral support and maintains proper spacing for four fuel assemblies. Holes are drilled into the bottoms of the beams, and they provide support to in-core neutron flux monitor dry tubes and the start-up neutron sources. The top guide is aligned by positioning pins that fit into slots in the top of the core shroud. A simplified sketch of a top guide is shown in Fig. 2.6.

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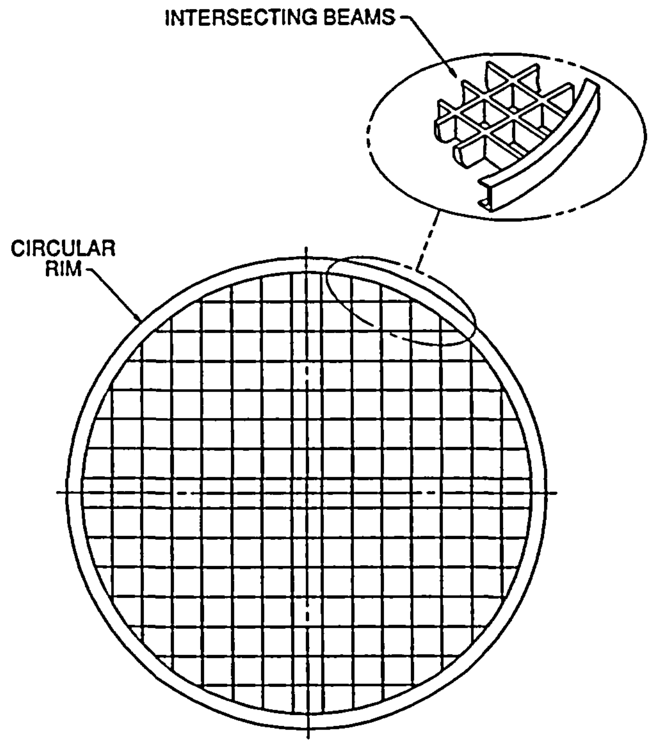

Figure 2.6 Top guide

Design Code: The top guide is designed to meet the requirements of Sect. III of the *ASME B&PV Code.*

Material of Construction: The top guide is made of type 304 stainless steel.

Component No. 7: Core Shroud

Function: The core shroud provides lateral restraint to the core. The top guide and the core plate are also supported by the core shroud. The core shroud serves as the partition separating the upward coolant flow from the downward recirculating flow in the pressure vessel.

Description: The core shroud is a segmented cylindrical steel structure. Each segment has a different shroud diameter. The steam separator/shroud head assembly is bolted to the flange of the top segment, which has the largest diameter. Core spray spargers are attached to the inside surface of the top shroud segment. The top guide is the lower boundary of the top segment. The middle shroud segment contains the core. It is the longest segment with an intermediate diameter. The top guide and the core plate define the top and bottom boundaries of the middle segment. The bottom segment of the core shroud contains the upper portion of the reactor lower plenum. The bottom shroud segment has the smallest diameter of the three. The bottom of the core shroud is welded to the shroud support.

For older reactors that are not equipped with jet pumps, the bottom shroud segment is in the form of a truncated cone. The top end of the cone segment is welded to the lower end of the middle shroud segment, and the bottom of the cone is welded to the reactor vessel wall.

Design Code: The core shroud is designed to meet the requirements of Sect. III of the *ASME B&PV Code*.

Material of Construction: The core shroud is made of type 304 or 304L stainless steel plates. The shell sections are solution heat-treated and then joined together by longitudinal and circumferential welds.

Component No. 8: Core Plate

Function: Perforations in the core plate provide lateral support and guidance to control rod guide tubes, peripheral fuel support pieces, in-core neutron flux monitor housings, and start-up neutron sources.

Description: The core plate assembly consists of a perforated top plate stiffened by a circular rim and grid support beams. With the exception of BWR-6s, the top plate is also supported by tie rods. Control rod guide tubes, peripheral fuel support pieces, and the in-core neutron flux monitor housings are inserted into the perforations of the top plate. Grid support beams are attached to the core shroud, and they are fillet-welded at regular intervals to the top plate. Tie rods, when they are used, provide lateral support to the grid support beams. One end of the tie rods is fillet-welded to grid support beams through holes in the top plate, and the other end is attached to the rim at the periphery of the top plate. A simplified sketch of a core plate is shown in Fig. 2.7.

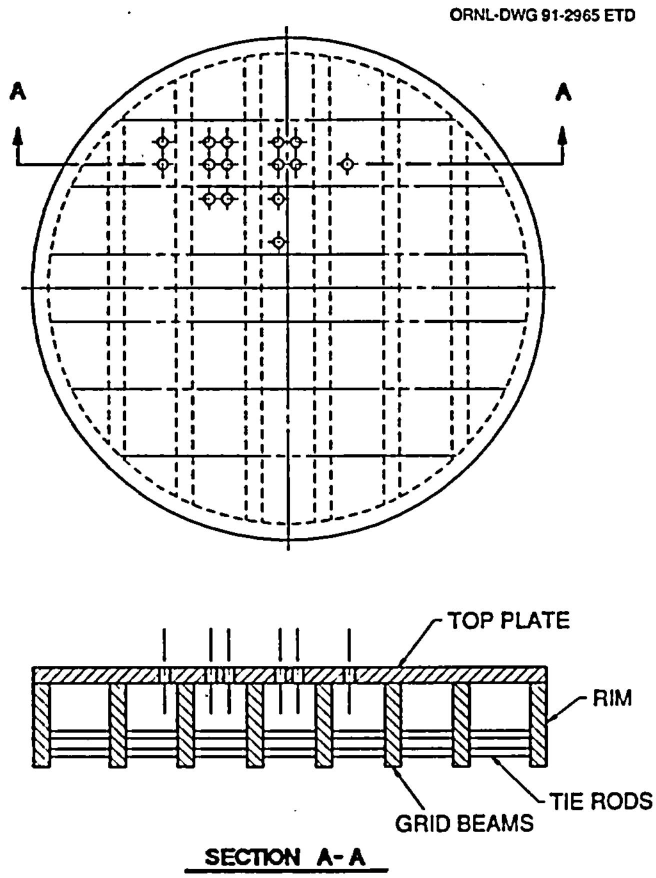

Figure 2.7 Core plate

Design Code: The core plate is designed to meet the requirements of Sect. III of the *ASME B&PV Code*.

Material of Construction: The core plate assembly is made of type 304 stainless steel.

Component No. 9: OFS Pieces

Function: OFS pieces support the weight of the fuel assemblies, as well as distributing cooling water to them.

Description: There are two types of OFS pieces. The standard or four-lobed OFS is a cylindrical structure with four internal compartments and a central opening for the positioning of a control blade. Each of the four internal compartments also has a lobe-shaped opening on the top surface. The standard OFS piece rests on top of a control rod guide tube. The bottom of a fuel assembly is inserted into one of the lobe-shaped openings, and the OFS piece provides lateral support and alignment to the bottom of the fuel assemblies. The weight of the fuel assembly is transferred to the control rod guide tubes through the OFS piece. The coolant flow into the fuel assemblies is regulated by an office located on the side of the lower portion of the standard OFS piece. The peripheral OFS piece is a one-opening cylindrical structure that provides lateral support to only one fuel assembly. The peripheral OFS is welded to perforations in the core plate. The orifice of the peripheral OFS, which regulates coolant flow into the fuel assemblies, is located directly below the top opening. A simplified sketch of a standard OFS piece is shown in Fig. 2.8.

Design Code: The OFS is designed to meet the requirements of Sect. III of the *ASME B&PV Code*.

Material of Construction: The center or standard OFS piece is cast from Grade CF-3 or CF-8 steel. The peripheral OFS is made of type 304 or 304L stainless steel.

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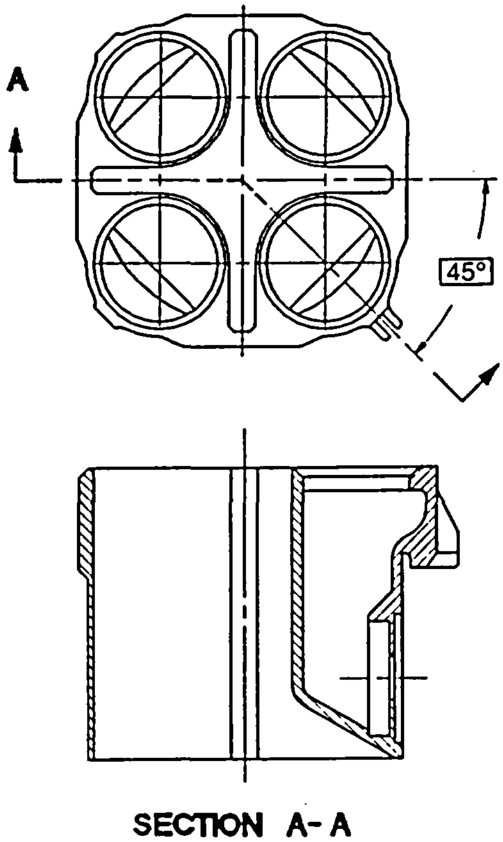

Figure 2.8 Standard OFS piece

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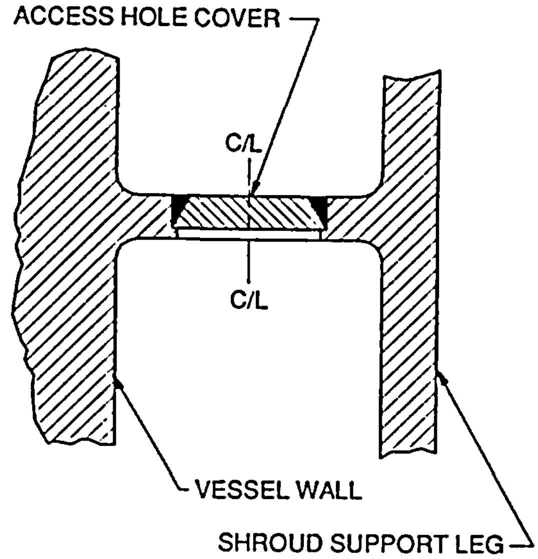

Figure 2.9 Access hole cover

Component No. 10: Shroud Support Including Access Hole Covers

Function: The shroud support, also known as the shroud baffle plate, carries the weight of the shroud, shroud head, steam separators, peripheral fuel assemblies, startup neutron sources, core plate, top guide, and jet pump diffusers. The shroud support also provides lateral support to the fuel assemblies. Jet pump diffusers penetrate the plate to inject cooling water to the inlet plenum below the core.

Description: The shroud support is an annular plate. The outer edge of the plate is welded to the inside surface of the reactor pressure vessel, and the inner edge is welded to the bottom segment of the core shroud. The inner edge of the annular plate is also supported by columns welded to the vessel bottom head. Jet pump diffusers are attached to openings in the shroud support plate. Two access holes, 180° apart, provide access to the jet pumps during construction. The access hole covers plugged the two access openings. The access hole cover is a circular steel plate. It is put into the access opening and rests on a ledge near the bottom of the hole. A full penetration weld attaches the cover to the shroud support. A sketch of the access hole cover is shown in Fig. 2.9.

Design Code: The access hole cover is designed to meet the requirements of Sect. III of the *ASME B&PV Code.*

Material of Construction: The access hole cover is made of alloy 600.

Component No. 11: Core Spray Line Internal Piping

Function: The core spray line internal piping, commonly referred to as the core spray line, is a component of the core spray system that supplies cooling water to the reactor fuel assemblies during a loss-of-coolant accident (LOCA). The objective is to supply and distribute sufficient coolant to the fuel assemblies so that the maximum fuel cladding temperature of 1204°C (2200°F) is not exceeded during a LOCA.

Description: The core spray line connects the external core spray piping to the core spray spargers. It is made of a 5-in. Schedule 40 steel pipe, and the content in the pipe is in a stagnant condition during normal plant operations. Two core spray lines enter the reactor vessel through two core spray nozzles. The two nozzles are located 180° apart in the vessel wall. Upon entry into the reactor vessel, a core spray line divides into two halves that are routed to opposite sides of the vessel. Along the way, the pipe lines are supported by clamps attached to the vessel wall. The pipes then go down the downcomer and are butt-welded to one end of the tee box pipe sections. The tee box pipe sections enter the top core shroud segment below the top flange. They then pass through the rim of the top guide and are connected to the center of a semicircular core spray sparger. The tee box pipe section is a component of the core spray sparger. The route of the other core spray line is identical, except it is connected to a core spray sparger at a different elevation. When the core spray system is activated in a LOCA, the core spray spargers inject cooling water into the core.

Design Code: The core spray line internal piping is designed to meet the requirements of Sect. III of the *ASME B&PV Code.*

Material of Construction: The core spray line internal piping is made of type 304 stainless steel.

Component No. 12: Core Spray Sparger

Function: The core spray sparger is a part of the Core Spray System. It injects cooling water into the core in the event of a LOCA. The objective is to supply and distribute sufficient coolant to the fuel assemblies, so the maximum fuel cladding temperature of 1204°C (2200°F) is not exceeded during a LOCA.

Description: The core spray spargers consist of two circular headers at different elevations and two tee box pipe sections for each header. The upper header has bottom-mounted nozzles, and the lower header has upper-mounted nozzles. Each circular header is composed of two 180° segments made of 3-1/2-in. Schedule 40 steel piping. Each of the 180° header segment is supplied by a tee box pipe section. The tee box pipe section, made of a 5-in. Schedule 40 steel pipe, connects the core spray spargers to the core spray line internal piping. One end of the tee box pipe section extends through the shroud wall and is butt-welded to the core spray line. The other openings of the tee box pipe sections are connected to the 180° header segments. The tee box pipe section is attached to the shroud by the seal rings with the attachment welded to the 5-in. pipe and the exterior surface of the shroud. The content in a core spray sparger is in a stagnant condition during normal operations. A simplified sketch of a core spray sparger is shown in Fig. 2.10.

Design Code: The core spray sparger is designed to meet the requirements of Sect. II of the *ASME B&PV Code.*

Material of Construction: The core spray sparger is made of type 304 stainless steel.

Component No. 13: Feedwater Sparger

Function: During normal operation, the feedwater sparger distributes cooler feedwater to the saturated reactor recirculating water before it comes into contact with the vessel

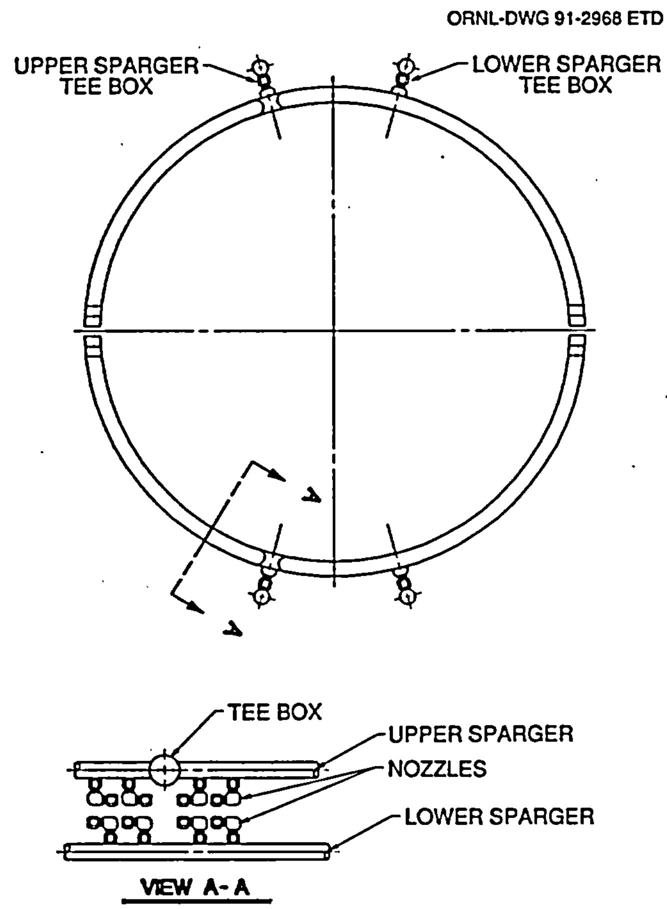

Figure 2.10 Core spray sparger

wall. The mixing of the two flows produces a flow of subcooled water to the jet pump and recirculation pump inlets and prevents the occurrence of pump cavitation problems. The homogeneous and uniform temperature mixture will also help to prevent the development of an asymmetrical power distribution in the reactor core. In the event of a LOCA, the feedwater sparger becomes a component of the High-Pressure Coolant Injection (HPCI) System. The HPCI System injects cooling water through the feedwater sparger into the core to maintain an adequate reactor water level in a LOCA.

Description: The feedwater sparger is a segmented circular header made of 5-in. Schedule 40 curved pipe sections. The number of segments is either four or six and is plant specific. Each segment receives water from a feedwater nozzle. The water inlet is located in the middle of each sparger segment, and the segment is shaped to fit the contour of the reactor vessel wall. A thermal sleeve is welded to the water inlet; the other end of the thermal sleeve is connected to the safe end of the feedwater nozzle by a slip-fit joint. Each sparger is supported by the thermal sleeve and a bracket mounted to each end of the segment. The end brackets are bolted to vessel wall brackets. The weight of the segment is supported by the vessel wall brackets, which also locate the header segments away from the vessel wall. Radial differential thermal expansions of the header segments are accommodated by the thermal sleeves; tangential slots are cut into the end brackets to account for tangential differential thermal expansions. The sparger segments are mounted at one elevation in the vessel. Feedwater is ejected through multiple small elbow and nozzle assemblies in an inward and downward direction. A simplified sketch of a feedwater sparger segment is shown in Fig. 2.11.

Design Code: The feedwater sparger is designed to meet the intent of Sect. III of the *ASME B&PV Code.*

Material of Construction: The feedwater sparger is made of types 316L and 316NG stainless steel.

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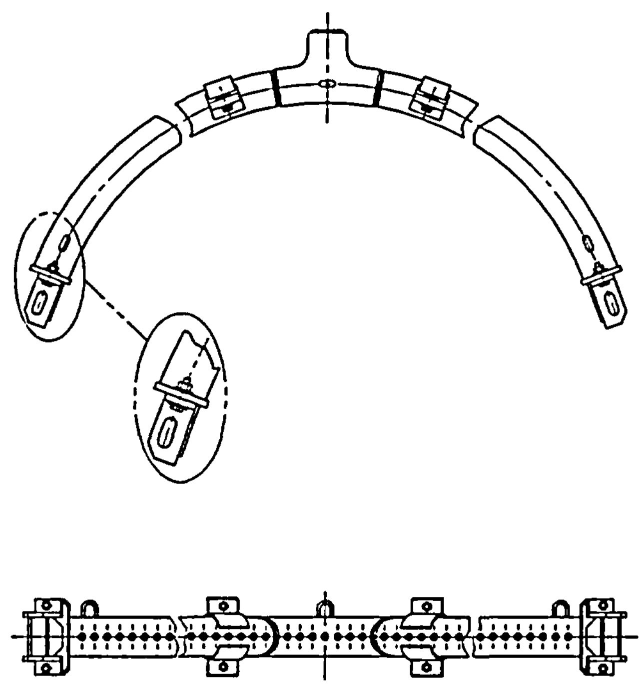

Figure 2.11 Feedwater sparger

Component No. 14: Jet Pump Assembly

Function: Jet pumps provide coolant flow for forced convection heat transfer in the reactor core, and they also provide power distribution shaping and reactivity control. They are used beginning with BWR-3 designs.

Description: Jet pump assemblies are located in two semicircular groups in the downcomer annular region between the core shroud and the reactor vessel wall. A jet pump assembly is composed of the riser pipe (shared with an adjacent jet pump), an inlet section, a suction inlet, the throat or mixing section, and a diffuser. The inlet section consists of a transition piece casting, a holddown beam, an inlet elbow, and a driving nozzle. The inlet section, the suction inlet, and the throat section form a removable unit. The diffuser is welded to an opening in the ring segment of the reactor vessel shroud support structure.

The riser pipe is welded to the thermal sleeve of the recirculating water inlet nozzle; the riser is also supported by two brace arms welded to beams extending from pads on the vessel wall. The connection between the throat and the diffuser is a slip-fit joint. The throat section is also restrained laterally by a bracket attached to the riser pipe. A metal-to-metal, spherical-to-conical seal joint is used between the riser pipe and the transition piece casting. A tight contact at the seal joint is maintained by clamps that fit under ears in the riser pipe, and a holddown beam is used to exert a downward force on a pad on top of the transition piece casting. A simplified sketch of a jet pump assembly is shown in Fig. 2.12.

The riser pipe receives high-pressure water from the recirculating water inlet nozzles and delivers the flow up to a transition piece casting. The flow from the riser pipe is turned downward into the driving nozzle by the inlet elbow. The high-speed nozzle exit flow entrains additional

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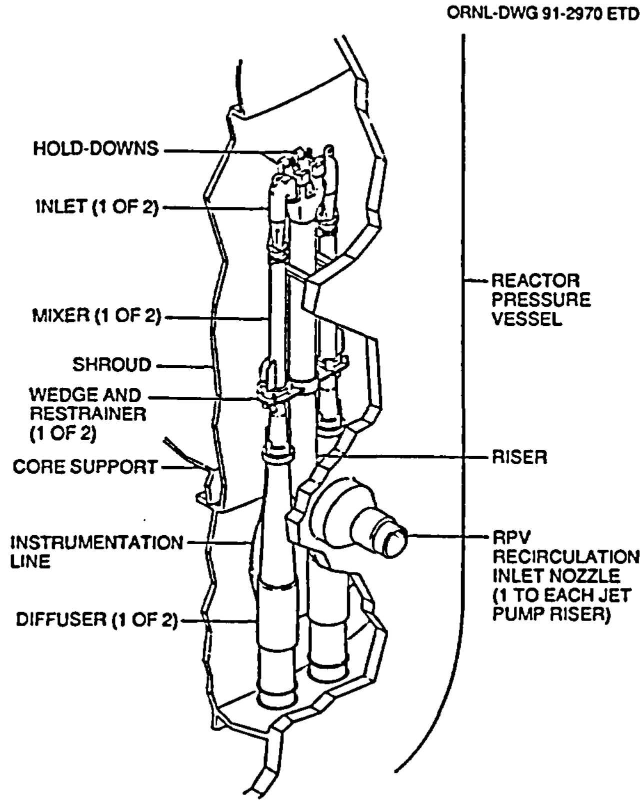

Figure 2.12 Jet pump assembly

cooling water through the suction inlet. The two flows mix in the throat or mixing section before discharge through the diffuser. The hydrodynamic forces acting on the jet pumps are reacted by holddown beams placed across the top of the transition piece casting.

Design Code: The jet pump is designed to meet the requirements of Sect. III of the *ASME B&PV Code.*

Material of Construction: Most components of the jet pump are made of type 304 stainless steel. Parts that were cast, such as the transition piece casting and inlet elbows, are made of either Grade CF-3 or CF-8 steel. A portion of the diffuser is made of alloy 600. The holddown beam and two parts at the inlet of the mixing section are made of alloy X-750.

Component No. 15: In-Core Neutron Flux Monitor Housings

Function: In-core neutron flux monitor housings provide a path for the insertion of neutron flux monitors into the reactor pressure vessel. Source Range Monitors (SRMs), Intermediate Range Monitors (IRMs), Local Power Range Monitors (LPRMs), and calibration monitors enter the reactor vessel through monitor housings.

Description: The in-core neutron flux monitor housing is a pipe segment with a seal-ring flange at the bottom. The pipe segment is inserted into the reactor vessel through penetrations in the vessel bottom head and welded to the inside surface of the bottom head. An in-core neutron flux monitor guide tube is welded to the top of each housing. Either an SRM/IRM drive unit, an LPRM, or a calibration monitor is bolted to the seal-ring flange at the bottom. The housing unit is considered part of the reactor primary pressure boundary. The configuration of a typical in-core neutron flux monitor is shown in Fig. 2.13. A simplified sketch of an in-core neutron flux monitor housing is shown in Fig. 2.14.

Design Code: The in-core neutron flux monitor housing is designed to meet the requirements of Sect. III of the *ASME B&PV Code.*

Material of Construction: The in-core neutron flux monitor housing is made of type 304 stainless steel.

Component No. 16: In-Core Neutron Flux Monitor Guide Tubes

Function: The in-core neutron flux monitor guide tube provides a path for the insertion and positioning of neutron flux monitors into the core. IRM, SRM, LPRM, and calibration monitors are inserted into the core through monitor guide tubes.

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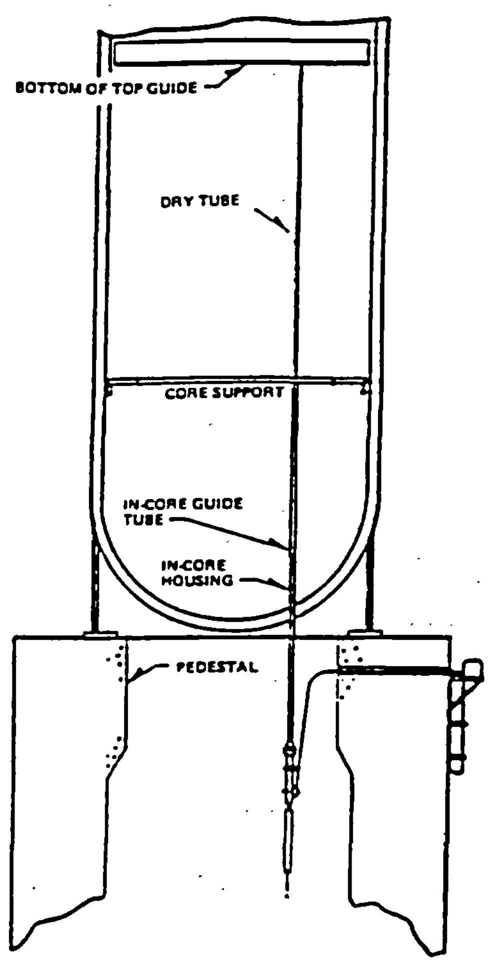

Figure 2.13 In-core neutron flux monitor

Description: The in-core neutron flux monitor guide tube is a pipe welded to the top of an in-core neutron flux monitor housing. The pipe segment extends from the top of the neutron flux monitor housing to the top of the core plate. A latticework of clamps, tie bars, and spacers provide structural support to the guide tube. To prevent loosening during reactor operations, bolts and clamps used in building the support latticework are welded after assembly. The in-core neutron flux monitor guide tube is considered part of the reactor primary pressure boundary. A simplified sketch of the in-core neutron flux monitor guide tube is shown in Fig. 2.14.

Design Code: The in-core neutron flux monitor guide tube is designed to meet the requirements of Sect. III of the *ASME B&PV Code.*

Material of Construction: In-core neutron flux monitor guide tubes are made of type 304 stainless steel.

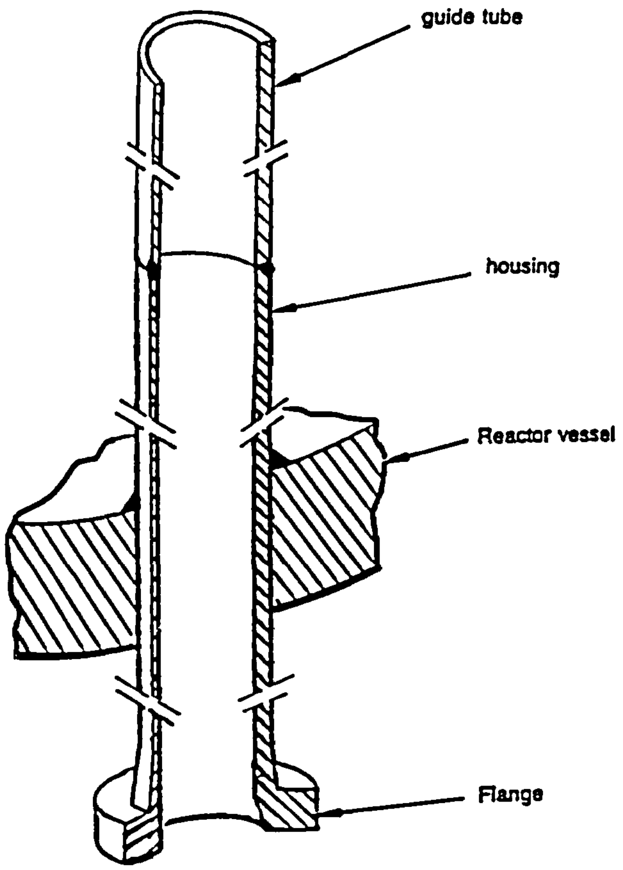

Figure 2.14 In-core neutron flux monitor housing and guide tube

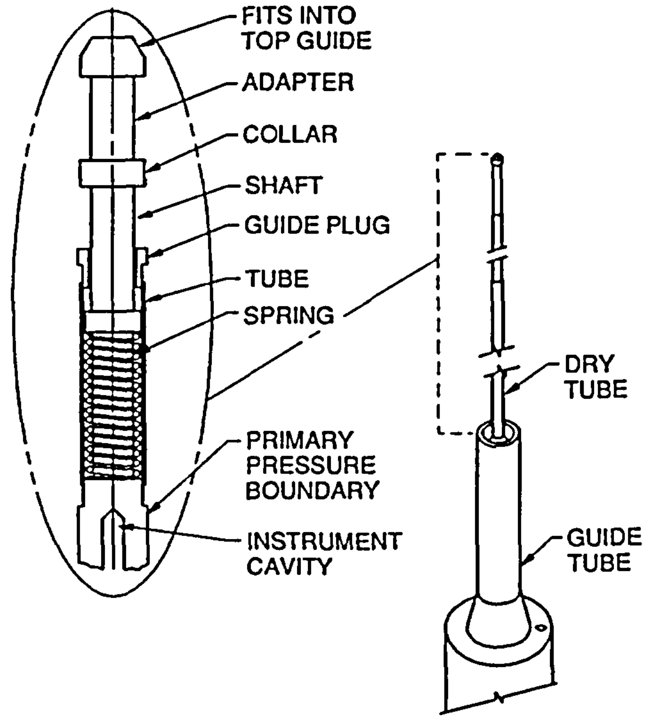

Figure 2.15 In-core neutron flux monitor dry tube

Component No. 17: In-Core Neutron Flux Monitor Dry Tubes

Function: In-core neutron flux monitor dry tubes house neutron monitors inside the core. Monitors include the IRM, SRM, LPRM and calibration monitors.

Description: The dry tube is a thin-walled tubing. The lower end of the tubing is welded to a thick-walled tube that contains the instrument cavity. The thick-walled tubing is a part of the reactor primary pressure boundary. A guide plug is welded to the top of the dry tube. An adapter unit is inserted into the dry tube through the guide plug and is pressed against a spring located inside the dry tube. The top of the adapter is inserted into slots in the top guide. A simplified sketch of a typical dry tube is shown in Fig. 2.15.

Design Code: The in-core neutron flux monitor dry tube is designed to meet the requirements of Sect. III of the *ASME B&PV Code.*

Material of Construction: The dry tube is made of annealed type 304 stainless steel.

Component No. 18: Control Rod Drive (CRD) Housing

Function: The CRD housing provides access into the reactor pressure vessel for the CRD. It supports the CRD and the control blade. The CRD housing also provides the necessary restraint needed to counteract the forces caused by the activation of the CRD system.

Description: The CRD housing is a tubular structure. The main portion consists of two tubes welded together. A flange is welded to the lower end, and a cap is welded to the upper end. The CRD is bolted to the lower flange. The CRD housing penetrates the pressure vessel bottom head and is J-welded to the top of the bottom head-stub tube. In the newer BWR plants, the CRD housing unit is welded directly onto the bottom head. There is a thermal sleeve in the inside surface along the entire length of the CRD housing. The CRD housing is a part of the reactor primary pressure boundary. A simplified sketch of a CRD housing is shown in Fig. 2.16.

Design Code: The CRD housing is designed to meet the requirements of Sect. III of the *ASME B&PV Code.*

Material of Construction: The CRD housing is made of type 304 stainless steel.

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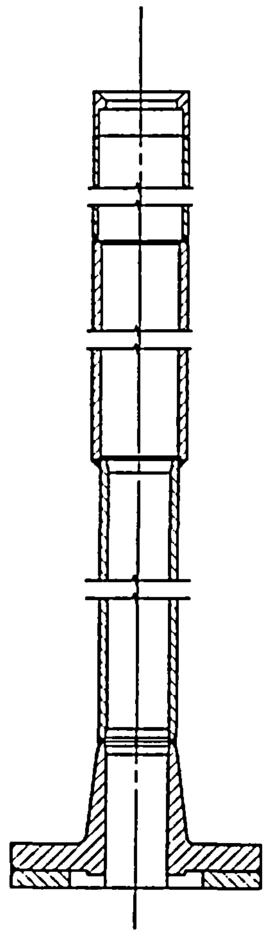

Figure 2.16 CRD housing

Component No. 19: Jet Pump Sensing Line

Function: The jet pump sensing line provides housing for instruments that are used to monitor the jet pump flow.

Description: The jet pump sensing line is made of a 1/4-in. Schedule 40 pipe (larger pipe for BWR-6s). The sensing line is welded to the jet pump diffuser at two square support brackets. The line is routed down the diffuser, around the jet pump, and exits the reactor pressure vessel through one of the two jet pump instrumentation nozzles. There is one sensing line for each jet pump in the reactor.

Design Code: The jet pump sensing line is designed to meet the intent of Sect. III of the *ASME B&PV Code.*

Material of Construction: The jet pump sensing line is made of type 304 stainless steel.

Component No. 20: Neutron Source Holder

Function: The neutron source holder contains the antimony–beryllium start-up sources that are needed to provide additional neutrons for attaining criticality in the first operating cycle. Start-up sources are not needed after the first fuel cycle.

Description: The neutron source holder is a cylindrical capsule containing the antimony gamma sources and the beryllium sleeve. The holders are fit into a slot in the top guide and inserted into a hole in the lower core plate. The reactor vendor has recommended that neutron source holders be removed from the reactor after the first fuel cycle.

Design Code: The neutron source holder is designed to meet the intent of Sect. III of the *ASME B&PV Code.*

Material of Construction: The neutron source holder is made of type 304 stainless steel.

Component No. 21: Control Blade

Function: Control blades are used for power distribution shaping and reactivity control. Power distribution shaping is achieved by the manipulation of the pattern of neutron-absorbing rods in the control blades. The neutron-absorbing rods can also be positioned to neutralize the effects of steam voids in the top of the core for reactivity control.

Description: The major component of a control blade is the neutron-absorbing rod assembly. Neutron-absorbing rods are made of stainless steel tubings filled with boron-carbide powder. The tubes are seal-welded with end plugs at the two ends. The rods are held in a cross-shaped pattern by a steel sheath extending the full length of the rod. The sheath is spot-welded to the top of the tubes. An upper handle aligns the tubes and provides the necessary structural rigidity at the top of the rod assembly. The lower handle and the velocity limiter serve the same purpose at the bottom of the assembly. Control blades are inserted into the core from the bottom of the pressure vessel. A simplified sketch of a control blade is shown in Fig. 2.17.

Design Code: The control blade is designed to meet the requirements of Sect. III of the *ASME B&PV Code.*

Material of Construction: Control blades are made of type 304 stainless steel.

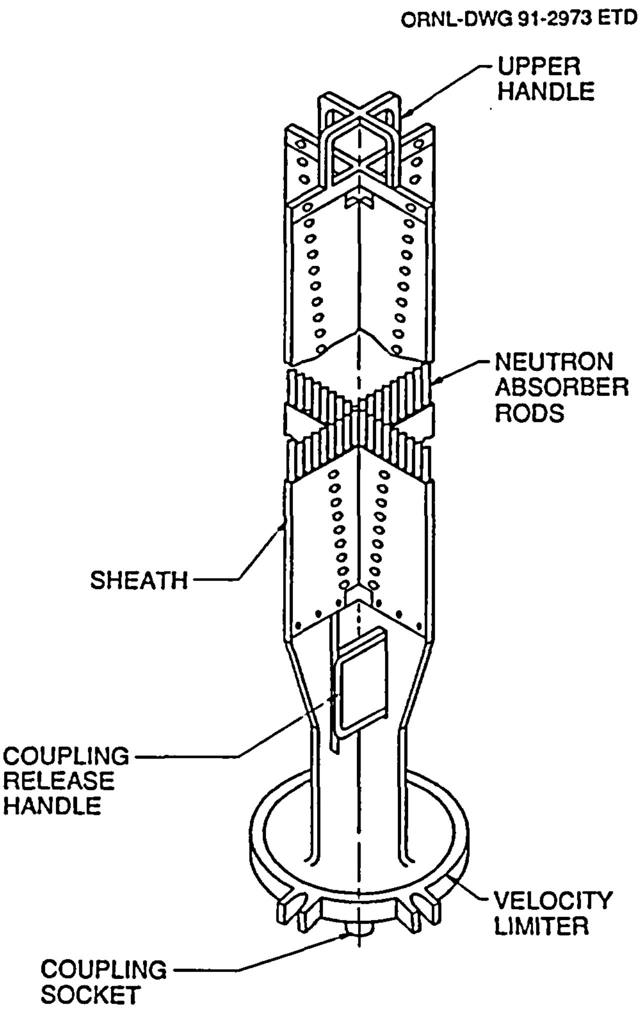

Figure 2.17 Control blade

Component No. 22: Vessel Head Cooling Spray Nozzle

Function: The vessel head cooling spray nozzle maintains saturated conditions in the reactor head region by condensing steam being generated by the hot reactor vessel walls and other internals. The spray also has the effect of reducing thermal stratification in the reactor vessel coolant so that the water level in the vessel can rise if needed. A higher water level can provide effective cooling to more of the reactor components inside the reactor vessel.

Description: A part of the reactor coolant flow returning to the reactor vessel can be diverted to a spray nozzle in the reactor head. The spray nozzle is mounted to a short length of pipe and a flange. The spray nozzle flange is bolted to a mating flange on the reactor vessel head nozzle.

Design Code: The vessel head cooling spray nozzle is designed to meet the requirements of Sect. III of the *ASME B&PV Code.*

Material of Construction: The vessel head cooling spray nozzle is made of type 304 stainless steel.

Component No. 23: Control Rod Guide Tubes.

Function: A control rod guide tube provides lateral guidance to a control rod and vertical support for a four-lobed OFS piece and the four fuel assemblies surrounding a control rod.

Description: The control rod guide tube is a cylindrical tube extending from the top of the CRD housing up through holes in the core plate. The bottom of the guide tube is supported by the CRD housing. A thermal sleeve is inserted into the CRD housing from below and then rotated to lock the control rod guide tube in place. The top of the guide tube goes through a hole in the core plate, and a four-lobed OFS piece is inserted into the top opening of the guide tube. A simplified sketch of a control rod guide tube is shown in Fig. 2.18.

Design Code: The control rod guide tube is designed to meet the requirements of Sect. III of the *ASME B&PV Code.*

Material of Construction: The control rod guide tube is made of type 304 stainless steel.

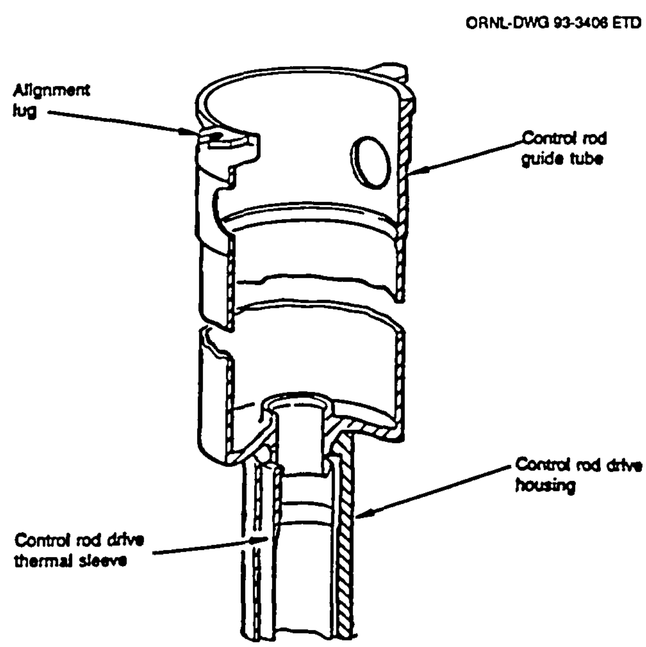

Figure 2.18 Control rod guide tube

Component No. 24: Surveillance Sample Holders

Function: The surveillance sample holders contain impact and tensile specimen capsules. The test specimens are used to measure irradiation effects on material properties of the reactor vessel wall materials.

Description: The surveillance sample holders are welded baskets hanging from brackets welded to the inside of the reactor vessel at the middle level of the core. The circumferential positions of the holders are chosen to expose the test specimens to the same maximum neutron fluxes as those experienced by the vessel wall.

Design Code: The surveillance sample holders are designed to meet the intent of Sect. III of the *ASME B&PV Code.*

Material of Construction: The surveillance sample holders are made of type 304 stainless steel.

Component No. 25: Differential Pressure and Liquid Control Line

Function: The differential pressure and liquid control line serves two functions: to sense the differential pressure across the core plate and to provide a means for injecting liquid control solution into the coolant stream.

Description: The differential pressure and liquid control line enters the reactor vessel as two concentric pipes at an elevation below the core shroud. The two pipes separate in the lower plenum. The inner pipe ends near the lower shroud with a perforated section below the core plate. The perforated pipe section senses the pressure below the core plate during normal operations and also serves as the outlet for the injection of liquid control solution, as needed. Injection of the cooler liquid control solution through the inner pipe will reduce the effects of thermal shock on the vessel nozzle. The outer pipe ends above the core plate and is used to sense the pressure in the region outside the fuel assemblies.

In some BWRs, the core spray line is used as the conduit for injecting liquid control solution. In these reactors, the differential pressure sensing lines enter the reactor vessel through two bottom head penetrations. One line ends near the lower shroud with a perforated pipe section for sensing pressure below the core plate. The other line ends just above the core plate and senses the pressure in the region outside the fuel assemblies.

Design Code: The differential pressure and liquid control line is designed to meet requirements of Sect. III of the *ASME B&PV Code.*

Material of Construction: The differential pressure and liquid control line is made of type 304 stainless steel.

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2. American Society of Mechanical Engineers, "ASME Boiler and Pressure Vessel Code, Section IX, Welding and Brazing," 1992.*

3. Northern States Power Company and Multiple Dynamics Corporation, "BWR Pilot Plant Life Extension Study at the Monticello Plant: Interim Phase 2," Interim Report, EPRI NP-5836M, October 1988.

*Available from American National Standards Institute, 1430 Broadway, New York, NY 10018, Copyrighted.

3 Primary Stressors

Reactor internals are submerged in a hot and moving liquid coolant during normal plant operations. The nominal system pressure inside the reactor vessel is 7.3 MPa (1055 psia). The coolant saturation temperature corresponding to the system pressure is 288°C (550°F). The average core inlet flow velocity is about 2.1 m/s (7 ft/s). Internal components located in the vicinity of the core are also exposed to fast neutron (E > 1 MeV) fluxes and gamma radiation. The operating environment inside a reactor pressure vessel generates many stressors that are considered as necessary to the development of aging-related or time-dependent degradation mechanisms.

The term "stressor" is used to represent conditions that may contribute to the initiation and the sustaining of the subsequent growth of an aging-related degradation mechanism. Applied loading is probably the most common stressor for structural components. In addition to applied loadings, environmental effects and manufacturing processes can also impose stressors on reactor internals. Environmental effects and manufacturing processes may cause changes in the mechanical and physical properties of the materials of construction. Changes in material properties can have a significant impact on the aging process.

Abnormal operating conditions can impose much more severe conditions on reactor components. However, they will not be investigated in the present study where the emphasis is on the assessment of aging effects under normal plant operations. Normal plant operations constitute the great majority of a reactor's operating history.

3.1 Applied Loadings

Applied loadings are mechanical and thermal loads associated with normal and transient (startup or shutdown) plant operations. External mechanical loads include static differential pressure loadings, preloads in bolts, and hydrodynamic forces produced by coolant flows in the pressure vessel. An important internally applied load is one caused by welding-induced residual stresses.

Flow-generated hydrodynamic forces can be static (steady state) or oscillatory in nature. Steady-state hydrodynamic forces are usually referred to as lift and drag forces; they are counterbalanced by the weight of the component and structural supports. Oscillatory hydrodynamic forces are generated by flow separations. Pump-generated pressure pulsations can also act as periodic external excitations to reactor internals. In the BWR environment, oscillatory hydrodynamic forces are the major concern because they can cause a component to vibrate. FIV is a major cause of fatigue problems in reactor internals.

Preloads on bolts are applied loads acting on reactor internals. Tensile stresses produced by the preloads may become a contributing factor to the development of cracks in bolts. Most of the BWR internals are not part of the primary pressure boundary and are not subjected to large static differential pressure loads. Exceptions are in-core instrumentation guide tubes and housings; differential pressure loads are incorporated into design loadings for these components.

Thermal loads are caused by the existence of temperature gradients in a component, by thermal expansions of different materials, and by restricted thermal expansions. Reactor internals are designed to accommodate differential thermal expansions; as a result, constraint-induced, steady-state thermal loads are kept at a low level. Some internal components are exposed to the mixing of fluid flows at different temperatures. The mixing actions produce rapid temperature changes in the components. These thermal cyclings can lead to the development of alternating stresses and fatigue cracks.[1]

The applied loadings of major concern to BWR internals are flow-generated cyclic loads. They produce alternating stresses that can lead to fatigue failures.

3.2 Environmental Stressors

Major environmental stressors imposed on reactor internals are the result of their contact with a hot and potentially corrosive coolant. The corrosiveness of the coolant is controlled primarily by the presence of dissolved oxygen in the reactor cooling water. Dissolved oxygen is a product of radiolytic reactions in the core. The level of dissolved oxygen in the BWR cooling water, 100 to 300 ppb, is sufficient to produce the electrochemical driving force needed to promote SCC in sensitized austenitic stainless steel components. The electrochemical potential of an austenitic stainless steel is increased by the buildup of the dissolved oxygen content in the cooling water, and the stainless steel becomes more susceptible to corrosion. Type 304 stainless steel, the most common material of construction for internals, is susceptible to SCC in the sensitized condition. The presence of other impurities such as chlorides may accelerate the corrosion process.

The relatively high operating temperature can produce physical changes in some reactor internal components.

Primary

Parts made from cast austenitic stainless steel (CASS) can become embrittled after prolonged operations in the BWR operating temperature.[2] Embrittled materials are more susceptible to fracture failures.

The exposure to fast neutron (E > 1 MeV) fluxes is another important environmental stressor for internal components.[3] The effects are most noticeable for components located in close proximity to the core. Long-term exposure to fast neutron fluxes can change the mechanical and physical properties of the materials of construction for internals. Irradiation effects can increase the yield and ultimate strengths and decrease the ductility and fracture toughness, as well as the critical flaw size, of a material. Neutron irradiation effects can also lower the temperature at which creep can become a significant deformation mechanism. These changes may adversely affect the structural integrity of reactor internals.

3.3 Manufacturing Stressors

Methods for fabricating reactor internals may impose stressors on the components. Manufacturing stressors can also promote the development of aging-related degradation mechanisms. Welding and cold-working are two common manufacturing methods used in making internals, and they can introduce stressors to the finished parts.

The conventional welding process sensitizes austenitic stainless steels (such as type 304) and makes them susceptible to corrosion. Residual stresses in weld heat-affected zones (HAZs) may contribute to the development of SCC.

Chromium and nickel are the two major alloying elements used in the making of austenitic stainless steels. Carbon is also used as a minor alloying element. Austenitic stainless steels, such as type 304, have generally good corrosion resistance. However, sensitization can make a welded austenitic stainless steel component vulnerable to SCC. When a structural component made of an austenitic stainless steel is cooled down slowly through the temperature range from ~820 to 480°C (1500 to 900°F), as in a welding process, chromium carbides precipitate at grain boundaries. The precipitation of chromium carbides will lead to a decrease in the chromium content in regions adjacent to grain boundaries. When the chromium content in type 304 stainless steel drops below 12%, the steel becomes susceptible to corrosion attacks at the grain boundaries.[4] The chromium depletion process is known as the sensitization process, and weld HAZs are common locations for sensitized austenitic stainless steel.

In addition to the material sensitization process, residual stresses are also generated in weld HAZs during cooldown. Residual stresses may contribute to the development of SCC.

Cold-working introduces new manufacturing stressors. Plastic strains are accumulated in the workpiece during cold-working, and excessive plastic strain accumulation can lead to the development of cracks. Contacts with a hot and corrosive coolant may accelerate the crack initiation process. Evidence also suggests that cold-working can induce martensite formation in the workpiece. Martensite formation may help the material sensitization process and make a cold-worked component more susceptible to corrosion attacks.[5]

Applied loadings, environmental effects, and manufacturing processes create stressors that, by themselves or in conjunction with each other, create conditions favorable to the development of aging-related degradation mechanisms in BWR internals.

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*Available in public technical libraries.
†Available for purchase from National Technical Information Service, Springfield, VA 22161.

4 Aging-Related Degradation Mechanisms

The presence of stressors is essential to the development of aging-related degradation mechanisms. Major potential aging-related degradation mechanisms for BWR internals are identified as SCC, fatigue, embrittlement, erosion, creep, and stress relaxation. SCC includes intergranular SCC (IGSCC) and IASCC. Fatigue is caused by vibrations and thermal cycling. Embrittlement can develop as a result of prolonged exposure to fast neutron fluxes and thermal aging. Contacts with a high-speed liquid flow or flows with droplet impingements are common causes of erosion problems. Creep and stress relaxation are caused by prolonged exposure to high temperatures and fast neutron fluxes. They can lead to changes in material properties and deformation mechanisms. Stressors associated with these aging-related degradation mechanisms are present inside a BWR pressure vessel.

4.1 Stress Corrosion Cracking

Corrosion is the weakening of a structural component as a result of material deterioration caused by electrochemical reactions with the surrounding medium. The effects can be global or highly localized. Global effects are referred to as general corrosion. The localized effects usually involve some form of crack development.

The surface of a metallic structure can become oxidized when it is submerged in a corrosive fluid medium. The removal of corrosion products by the fluid medium will result in a thinning of the wall. This general corrosion process is global and occurs over the entire contact surface in a somewhat uniform manner. Austenitic stainless steels, such as type 304, have good resistance to general corrosion and wastage.

SCC is a common form of highly localized corrosion phenomena. SCC can occur in ductile materials with little or no plastic strain accumulation associated with the process. The development of SCC in a structural component requires the simultaneous presence of three conditions: a conducive environment, a susceptible material, and tensile stresses above a threshold level. SCC is not likely to develop when any one of these three conditions is absent from the operating environment. Thus, the elimination of one condition is the basis for formulating strategies to control SCC. Depending on the alloy compositions and the nature of stressors presented in the system, cracks can develop along grain boundaries; such SCC is known as IGSCC. When cracks propagate along certain crystallographic slip planes within the grains, the failures are known as transgranular stress corrosion cracking (TGSCC). More detailed discussions on the fundamentals of SCC can be found in texts by Logan[1] and Romanov.[2]

The hot and oxygenated reactor cooling water creates a corrosive environment in the BWR pressure vessel. The dissolved oxygen in the reactor cooling water increases the electrochemical potential of type 304 stainless steels and makes them vulnerable to corrosion attacks. The presence of impurities, such as chlorides and sulfates, in the coolant may accelerate the crack development process.

Most of the SCC failures in BWR internals are found in weld HAZ. In addition to the oxygenated reactor cooling water, the welding process can provide the other two conditions that are needed for the development of SCC.

When a weld is cooled down through the temperature range from 820 to 480°C (1500 to 900°F), type 304 stainless steel undergoes a sensitization process characterized by a chromium depletion at grain boundaries.[3] The sensitization process makes austenitic stainless steels susceptible to corrosion attacks. The presence of residual stresses in weld HAZs supplies the third requirement for SCC. The level of residual stresses may be reduced by proper heat treatment. Proper heat treatment for internals is not always possible because of their large size. As a result, residual stresses remain in weld HAZs. Residual stresses are self-equilibrating, and tensile stresses are always present in them. The threshold tensile stress level that is needed for the development of SCC varies, but it is generally accepted that the level is near the yield stress of the material. Residual stresses in weld HAZs can approach the yield stress value, and they can contribute to the development of SCC. Tensile stresses generated by preloads in bolts can also attain the threshold value needed for the development of SCC.

The increase in electrochemical potential of the material, the chromium depletion sensitization process, and the presence of tensile stresses provide the three necessary conditions needed for the development of SCCs. SCC is probably the most common aging degradation mechanism in BWR internals. The majority of SCC failures observed in BWR internals are intergranular. Other conditions may also have some influence on the development of SCC. The more significant ones are crevice condition and prolonged exposure to fast neutron fluxes.

A crevice is a general term describing a small and narrow region containing a stagnant fluid. High concentrations of impurities (such as chloride and sulfate) may be trapped in the stagnant fluid, creating a local environment that is favorable to the development of SCC. The impurities produce a highly acidic solution that is anionic in nature and can destroy the protective thin oxide film and increase the likelihood of corrosion attacks. The presence of crevices in an internal component can be a design feature, or they can actually be small cracks initiated by other degradation mechanisms. Narrow gaps between contact surfaces, such as those found in bolted joints, are design features that may create crevices in an internal component.

SCC has been observed in nonsensitized stainless steel internal components. In some cases, the stress levels in these components are relatively low and below the material yield stress. A common factor in these nonsensitized, low-stress internal components with SCC failures is a prolonged exposure to fast neutron ($E > 1$ MeV) fluxes. Reactor internals that are exposed to fast neutron fluxes in a corrosive environment are susceptible to IASCC.[4] Most IASCCs are intergranular.

Basic understanding of IASCC is not complete, and no unified theory can account for all radiation effects on SCC. Evidence suggests that a threshold neutron fluence level exists below which IASCC is not likely to occur. The best estimate for the threshold value is about 5×10^{20} (neutrons/cm^2) (Ref. 5). BWR internals, such as control blades, OFS pieces, the core shroud, the core plate, the top guide, and start-up neutron sources can experience lifetime neutron fluence levels exceeding the threshold values. These components are susceptible to IASCC.

It has been suggested that irradiated materials are weakened by gas bubbles formed by transmutation processes.[6] Trace quantities of boron, which can react with thermal neutrons to form lithium and helium,[6] are found in austenitic stainless steels. Hydrogen is also produced by a transmutation process involving fast neutrons and elements such as N, Ni, and Cr (Ref. 4). Bubbles are formed when these gases precipitate from the solid. The bubbles have a tendency to move to dislocations and grain boundaries,[6] and they create discontinuities that can weaken the material. The solid may disintegrate in the absence of high tensile stresses when the bubbles are fused together. While the gas bubble formation mechanism can account for some of the observed characteristics of IASCC, questions remain on bubble sizes, their weakening effects on the solid, and the driving forces behind the bubble movement.

The presence of impurities such as phosphorous and silicone increases the susceptibility of unirradiated stainless steels to IGSCC. It has been suggested that high-purity stainless steels are more resistant to IASCC than commercial-grade stainless steels.[4] However, conclusive evidence to support this assertion is lacking.

The basic understanding of IASCC is not complete, and there is still much active research work in the field. General agreement is that the exposure to fast neutron fluence above a threshold value, together with a corrosive environment, can provide conditions that are favorable to the development of IASCC. The threshold tensile stress level needed for the development of IASCC may be significantly lower than the material's yield stress.
In summary, most welded BWR internals are susceptible to SCC. Crevice conditions may aid the SCC process. Prolonged exposure to fast neutron fluxes can lead to IASCC in nonsensitized steel components with low stress levels. Of the 25 selected BWR internals, 18 operate in conditions that would make them susceptible to SSC; they are summarized in Table 4.1. The last six components listed in Table 4.1 are also vulnerable to IASCC.

4.2 Fatigue

Reactor internals are subjected to time-dependent or dynamic loads. A structure will vibrate as a response to dynamic loads. Vibrations can lead to crack developments, and structural failures caused by vibrations are known as fatigue failures. Cyclic loadings and flow-induced vibrations are the root causes of fatigue problems in BWR internals. The loadings can be either mechanical or thermal in nature. Vortex shedding and other unsteady flow effects can generate oscillatory hydrodynamic forces, while cyclic thermal loads are produced by contact with the turbulent mixing of flow streams at different temperatures. Fatigue failures can be divided into two types: high- and low-cycle fatigues. When stress is in the linear-elastic range and the amplitude of the vibration is small, the resulting failure is classified as high-cycle fatigue. Little or no plastic strain accumulation is associated with high-cycle fatigue failures. High stress values, large amplitude vibrations, and significant plastic strain accumulation are important characteristics associated with low-cycle fatigues. In addition to the stress and strain criteria, the number of cycles of vibrations before a failure occurs is used to identify low- and high-cycle fatigue. When a crack is initiated between 10^3 and 10^4 cycles, the failure is classified as a low-cycle fatigue failure.

The fluid flows inside the reactor pressure vessel can impose mechanical and thermal loads to internals submerged in the flow streams. A common mechanical load is the drag force, which is a steady-state hydrodynamic force acting on a submerged structure. In transient operations, the propagation of pressure waves through the system can generate

Table 4.1 BWR Internals susceptible to SCC

Component	Stressors
Steam dryer support ring	Corrosive coolant, cold working, weld HAZ
Shroud head bolts	Corrosive coolant, bolt preloads, crevice conditions
Access hole cover	Corrosive coolant, crevice conditions, weld HAZ
Steam separator	Corrosive coolant, weld HAZ
Core spray sparger	Corrosive coolant, weld HAZ, cold-working
Shroud head	Corrosive coolant, weld HAZ
CRD housing	Corrosive coolant, weld HAZ
Feedwater sparger	Corrosive coolant, weld HAZ
Core spary line internal piping	Corrosive coolant, weld HAZ, crevice conditions
Jet pump	Corrosive coolant, preloads, weld HAZ
In-core neutron flux monitor housing	Corrosive coolant, weld HAZ
In-core neutron flux monitor guide tubes	Corrosive coolant, weld HAZ
In-core neutron flux monitor dry tubes	Corrosive coolant, weld HAZ, fast neutron fluxes
Neutron source holder	Corrosive coolant, fast neutron fluxes, crevice conditions
Core shroud	Corrosive coolant, fast neutron fluxes, weld HAZ
Core plate	Corrosive coolant, weld HAZ, fast neutron fluxes, crevice conditions
Control blades	Corrosive coolant, fast neutron fluxes, weld HAZ, crevice conditions
Top guide	Corrosive coolant, fast neutron fluxes, crevice conditions
OFS pieces	Corrosive coolant, fast neutron fluxes

oscillatory hydrodynamic forces. Core internals are designed to withstand such steady-state and transient loads. The primary cause of high-cycle fatigue failures in BWR internals is attributed to a phenomenon known as FIV. When a fluid medium flows over a blunt body, flow separations lead to the shedding of vortices from the body. Vortex shedding generates a cyclic load acting on the structure that would cause it to vibrate. In most cases, the amplitudes of the vibrations are small. A structure that is subjected to small-amplitude vibrations is susceptible to high-cycle fatigue failures. In rare instances, destructive large-amplitude resonant vibrations could develop in the structure when its fundamental natural frequency coincides with an input excitation frequency. However, reactor internals are designed to avoid operating under a resonant vibration condition. Key reactor internal components are monitored during reactor preoperation testings to ensure the absence of flow-induced resonant vibrations.

The operation of pumps in the reactor primary coolant system introduces flow-related excitations into the reactor coolant flow stream. They are pressure pulsations with dominant frequencies closely related to the pump rotating speed, blade passing frequency, and their harmonics. The effects of these pressure pulsations are strongest at the entrance regions around vessel inlet nozzles. Coincidence of a dominant pressure pulsation frequency with a structural fundamental frequency can also lead to resonant vibration problems. In any event, these pump-generated pressure pulsations are excitation sources for small-amplitude vibrations for reactor internals.

There are regions inside the reactor pressure vessel where reactor coolant flow streams at different temperatures are mixed in turbulent conditions. Reactor internal components located in these regions are exposed to a fluid medium with rapidly changing temperatures. The changing temperatures can impose thermal loads on components submerged in the mixing flow streams. The effects of the varying stresses caused by thermal cycling loads are similar to those caused by vibrations. The components are susceptible to high-cycle fatigue failures.[7] Usually little or no plastic strain accumulation occurs in a structure undergoing rapid thermal cyclings.

Aging

Low-cycle fatigue failures are mostly associated with excessive structural deformations under cyclic applied loads. The components are usually undersized for the loads that they are required to carry, and plastic strains are accumulated as a result of large structural deformations. Failure, generally in the form of crack development, will occur if the plastic strain accumulations exceed a critical value. Strengthening or "beefing up" the component is a common and effective solution to low-cycle fatigue problems.

BWR internals that are subjected to cyclic loadings are susceptible to fatigue failures. Major cyclic loadings include flow-generated oscillatory hydrodynamic forces and thermal cyclings caused by turbulent mixing of coolant flow streams at different temperatures. Of the 21 reactor internals, 6 are susceptible to fatigue failures and are summarized in Table 4.2.

Table 4.2 BWR internals susceptible to fatigue

Component	Stressors
Jet pump sensing line	Flow-generated oscillatory, hydrodynamic forces
Steam dryer	Large applied cyclic pressure loads
Feedwater sparger	Flow-generated oscillatory hydrodynamic forces, leakage flow-induced rapid thermal cyclings
Core spray sparger	Flow-generated oscillatory hydrodynamic forces
Jet pump	Flow-generated oscillatory hydrodynamic forces
In-core neutron flux monitor housings	Flow-generated oscillatory hydrodynamic forces
In-core neutron flux monitor guide tubes	Flow-generated oscillatory hydrodynamic forces
In-core neutron flux monitor dry tubes	Flow-generated oscillatory hydrodynamic forces

4.3 Embrittlement

Embrittlement is the loss of ductility and fracture toughness of a material, and it is usually accompanied by increases in the material yield and ultimate strengths. Prolonged exposures to fast neutron fluxes and high temperatures (thermal aging) can cause embrittlement in components made of stainless steels. The process can transform a ductile material to one that is susceptible to brittle fracture failures.

4.3.1 Radiation Embrittlement

In addition to fast neutron fluence level, effects of radiation embrittlement are also determined by the irradiation temperature and material compositions. Austenitic stainless steels, such as type 304, are considered susceptible to radiation embrittlement. The assessment of radiation embrittlement effects is based primarily on the interpretations of experimental results. The experimental data base is obtained from testings using surveillance specimens from commercial power reactors and other test reactor experiments. The reduction in uniform elongation is used as an indication of the decrease of ductility in a material after a prolonged exposure to fast neutron fluxes. Changes in fracture toughness are deduced from results of Charpy V-notch impact testings and other types of fracture-toughness testings.

Experiments[8] were conducted at a temperature range from room temperature to ~760°C (1400°F) and fast neutron ($E > 1$ MeV) fluence levels up to about 6×10^{21} neutrons/cm^2. At ~300°C (570°F), the unirradiated uniform elongation for type 304 stainless steel is ~38%. At the same test temperature, the uniform elongation was reduced to ~22% at a neutron fluence of ~1.5×10^{20} neutrons/cm^2. At ~1.5×10^{21} neutrons/cm^2 the uniform elongation at 300°C was further reduced to ~0.5%. The decrease in uniform elongation seemed to level off at higher neutron fluence levels. At a temperature of ~300°C (570°F), testing results[8] using type 304 stainless steel test specimens indicated that effects of radiation embrittlement begin to manifest at a neutron fluence level of ~5×10^{20} neutrons/cm^2. The expected fast neutron fluence for reactor internals is determined by the distance of the components from the core. In 40 years of power operations, the estimated maximum neutron fluence is ~1×10^{22} neutrons/cm^2 and is found at the central region of the top guide.[9] The estimated 40-year neutron fluence levels for other components located close to the core,[9] such as the core plate, core shroud, and fuel support pieces, range from ~3×10^{20} to 5×10^{21} neutrons/cm^2. These internal components are susceptible to effects of radiation embrittlement. The expected 40-year fast neutron fluence levels for other internals are below 1.5×10^{20} neutrons/cm^2, and their materials of construction should retain sufficient ductility and fracture toughness to ensure that brittle fracture is not likely to occur.

4.3.2 Thermal Embrittlement

Thermal embrittlement is caused by a prolonged exposure to high temperatures. It is a time-dependent degradation mechanism that affects mostly reactor internals made of CASS.[10] CASS is a two-phase alloy composed of austenite and ferrite. The determining factor is the ferrite content in the alloy; when it is <20%, CASS is not sensitive to thermal embrittlement. When the ferrite content is >20%, and when the operating temperature is in the range from 427 to 482°C (800 to 900°F), a chromium-rich precipitation in the ferrite phase would lead to a decrease in the fracture toughness of the material. Also indications are that the ferrite phase could become brittle when a CASS component is aged at a temperature of ~316°C (600°F). Thermal aging is a potential aging-related degradation mechanism because the ferrite content in CASS BWR internal components can go up to 25%, and the required temperature ranges are within the service conditions inside the pressure vessel. BWR internal components that are susceptible to radiation and thermal embrittlements are summarized in Table 4.3.

Table 4.3 BWR internals susceptible to embrittlement

Component	Stressors
Top guide	Fast neutron fluxes
Core plate	Fast neutron fluxes
Core shroud	Fast neutron fluxes
Control blades	Fast neutron fluxes
In-core neutron flux monitor dry tubes	Fast neutron fluxes
Cast components of the jet pump assembly	Thermal aging and ferrite precipitation in CASS
OFS pieces	Fast neutron fluxes, thermal aging, and ferrite precipitation in CASS components
Cast components of the steam separator assembly	Thermal aging and ferrite precipitation in CASS

4.4 Erosion

Erosion is a potential aging-related degradation mechanism when an internal component comes into contact with liquid flows. The abrasive effects are well-known when solid particles are present in a flow stream. Reactor cooling water is of high purity; solid particles, if they are present in the flow stream, will be very small in quantity. Solid particle abrasion is not a serious problem for BWR internals. The two potential problem areas associated with erosion in BWR internals are bubble formation in a liquid flow stream and liquid droplet impingements in two-phase flows.

Bubbles in a liquid flow stream can become attached to solid surfaces and can cause erosion by cavitation. In addition, the formation and collapse of bubbles can create disturbances in the flow that can disrupt the protective boundary layer film adjacent to solid surfaces and expose fresh metal surfaces to the erosion process. High-speed flow regions, such as those in the jet pump throat, nozzle, and diffuser, are vulnerable to bubble-induced erosion.

Erosion also can occur in a two-phase flow regime, such as the steam-water flows in steam separators. The impact of liquid droplets on a metal surface is analogous to the impact of solid particles and can cause abrasion problems. Of the 25 selected BWR internals, 2 are susceptible to erosion and are summarized in Table 4.4.

Table 4.4 BWR internals susceptible to erosion

Component	Stressors
Throat, nozzle, and diffuser sections in jet pump	Cavitation-induced bubbles
Steam separator	Liquid droplet impingements

4.5 Creep and Stress Relaxation

Creep is the progressive deformation of a structure under a constant stress. Stress relaxation is the reduction of internal stresses in a component under constraints. Creep can lead to fracture failures in a structural component. Stress relaxation can loosen bolts in bolted joints, which, in turn, could result in a loss of structural integrity and leakage problems.

Creep and stress relaxation are generally associated with structures that operate in a high-temperature environment. Exposures to fast neutron fluxes may also affect the development of creep and stress relaxation. As a general rule, when the operating temperature is less than half of the melting point temperature of the material, effects of creep and stress relaxation are not significant. The melting point temperature for a typical austenitic stainless steel is ~1430°C (2600°F). The normal operating temperature for BWR internals is about 288°C (550°F), less than half of the stainless steel melting point temperature. Thermally induced creep and stress relaxation are not considered as major aging-related degradation mechanisms for reactor internals.

Prolonged exposure to fast neutron fluxes lower the temperature at which creep and stress relaxation can become significant deformation mechanisms. Results of in-pile experiments conducted at 288°C (550°F) showed that effects of stress relaxation can be observed in bolts made of type 304 stainless steels[11] at a neutron fluence of 6×10^{19} neutrons/cm^2. At about the same temperature, significant stress relaxation has been observed in type 304 stainless steel test specimens at neutron fluence above 5×10^{20} neutrons/cm^2. BWR internals located in the vicinity of the core have 40-year neutron fluence levels in the range from 6×10^{19} to 5×10^{20} neutrons/cm^2. Of the 25 selected components, 4 are susceptible to creep and stress relaxation and are summarized in Table 4.5.

Table 4.5 BWR internals susceptible to radiation-induced creep and stress relaxation

Component	Stressors
Top guide	Fast neutron fluxes and high temperature (288°C)
Core shroud	Fast neutron fluxes and high temperature (288°C)
OFS pieces	Fast neutron fluxes and high temperature (288°C)
Control blades	Fast neutron fluxes and high temperature (288°C)
In-core neutron flux monitor dry tubes	Fast neutron fluxes and high temperature (288°C)

In summary, 19 of the 25 selected BWR internals are susceptible to SCC, including IASCC. Specifically, 6 out of the 19 are susceptible to IASCC. Of the 25 components, 7 are vulnerable to fatigue failures. Embrittlement is a potential aging-related degradation mechanism for eight components, and erosion is a potential aging-related degradation mechanism for two internal components. Five components are susceptible to the effects of creep and stress relaxation. The 25 selected components and their potential aging-related degradation mechanisms are shown in Table 4.6.

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*Available in public technical libraries.
†Available for purchase from National Technical Information Service, Springfield, VA 23161.

Table 4.6 BWR internals and potential aging-related degradation mechanisms

Component	SCC	Creep	Fatigue	Embrittlement	Erosion
Steam dryer			*		
Steam separator	*			*	*
Shroud head	*				
Shroud head bolts	*				
Steam separator support ring	*				
Top guide	*	*			
Access hole cover	*				
Core shroud	*	*			
OFS piece	*	*		*	
Core plate	*				
Core spray line internal piping	*				
Core spray sparger	*		*		
Feedwater sparger	*		*		
Jet pump	*		*	*	*
In-core neutron flux monitor housings	*		*		
In-core neutron flux monitor guide tubes	*		*		
In-core neutron flux monitor dry tubes	*	*	*		
CRD housing	*				
Neutron source holder	*				
Jet pump sensing line			*		
Control blade	*	*		*	

5 Survey of Aging-Related Failures

Stressors and potential aging-related degradation mechanisms have been identified for the 25 selected BWR internals. Aging-related degradation mechanisms do not develop at the same rate in the environment inside the reactor vessel. Therefore, component failure information is needed to properly assess the relative importance of the potential aging-related degradation mechanisms. The identification of major aging-related degradation mechanisms can also provide information that can be used to formulate strategies for managing aging effects.

As components of an operating nuclear power reactor, reactor internals are regularly inspected as part of the plant ISI program. The ISI programs for domestic nuclear power plants are established under the rules and regulations of Sect. XI of the *ASME B&PV Code.*[1] The Code is also the basis for ISI programs in most countries that utilize U.S.-developed reactor technologies.[2]

The program outlines specific procedures and requirements for inspecting reactor components, structures, and systems to ensure safe plant operation. The programs may include orders, rules, criteria, and guidelines established by regulatory agencies as well as industry groups.

5.1 ISI Program for Reactor Internals

The plant ISI program calls for the visual inspection of reactor internals. A complete inspection cycle is 10 years, and selected components are inspected at refueling outages.

Section XI of the *ASME B&PV Code*[1] specifies three classes of visual inspections: VT-1, VT-2, and VT-3. A VT-1 examination is conducted to determine the condition of the part, component, or surface examined, including such conditions as cracks, wear, corrosion, erosion, or physical damage on the surface of the part or component. The examination can be performed either directly or remotely.

A VT-2 examination is used to detect leakage from pressure retaining components. Most reactor internals are not pressure boundary components, and, as a result, VT-2 inspections are seldom used on internals.

A VT-3 examination is conducted to determine the general mechanical and structural conditions of components and their supports, such as verifications of clearances, settings, physical displacements, loose or missing parts, debris, corrosion, wear, erosion, or the loss of integrity at bolted or welded connections. VT-3 inspections can be performed either directly or remotely. More detailed information of the inspection methods and procedures can be found in the appropriate subsections of Section XI of the *ASME B&PV Code.*[1]

During a refueling outage, some internal components are removed from the pressure vessel and stored in a pool. VT-1 and VT-3 visual examinations are performed on accessible areas of reactor internals when they are in the storage pool. The underwater inspection is performed with remote television cameras under proper lighting. Internals that remain in the vessel are also visually examined by remote television cameras.

When a crack or flaw is detected, detailed information concerning the crack, such as its location and suspected causes, are recorded. The information is also sent to NRC as a Licensee Event Report (LER). Information on corrective actions taken are also included in the LER. For these reasons, LERs are considered reliable sources for reactor component failure information.

5.2 Reported Failure Information Survey

The survey of the aging-related component failure information for BWR internals is compiled from LERs and nonproprietary EPRI reports on nuclear power reactor operating experiences. Most of these failures were reported in a 10-year period from 1980 to 1989. It is difficult to extract reactor internal failure information from LERs, and it is possible that a few failure cases may have been missed. Therefore, the survey results are not exhaustive in nature, but they should provide a representative picture of the aging-related failure situation for BWR internals.

Because of access limitation problems, four BWR internals are not inspected: the OFS piece, steam separators, shroud head, and core plate. The other 21 components were inspected in accordance with the plant ISI program. Aging-related failures were detected in 19 of the 21 components. The survey results indicated that the core shroud and the top guide are the two regularly inspected components that had no reported failures during the survey period. However, the core shroud and the top guide are subjected to many significant stressors and are considered vulnerable to aging-related degradations.

5.3 Internal Component Failure Information Survey Results

A review of the survey results indicates that most reported aging-related failures in the 10-year survey period can be attributed to two aging-related degradation mechanisms: SCC (including IASCC) and fatigue. Radiation-induced embrittlement may have contributed to the failures of components that were exposed to high-energy neutron fluxes.

5.3.1 Reported SCC Failures

Of the 25 selected BWR internals, 19 are susceptible to SCC. SCC was detected in 11 of the 19 components. Most of the cracks were located at weld HAZs. The following sections provide a brief description of the more significant reported failures.

5.3.1.1 Jet Pump Holddown Beams

SCC was detected in jet pump holddown beams in six BWRs in the early 1980s. The beam is made of a nickel alloy (Inconel X-750). The cracks were located across the ligament of the beam section at the mean diameter of the bolt thread area as illustrated in Fig. 5.1. The cracks were intergranular. Low heat-treatment temperature (885°C), which led to the sensitization of the beam material, and high preloads on the beam bolt were major contributing factors to the development of SCC in jet pump holddown beams. Replacement holddown beams and beams in later models of BWRs are solution heat-treated at a higher temperature (1090°C), and preloads in the beam bolts were also reduced. These measures were effective in preventing IGSCC problems in new holddown beams.

A failure of the holddown beam could result in the disassembly of the jet pump and may also lead to a reduction of the operation safety margin during postulated accident situations. NRC has issued IE Bulletin No. 80-07 (Ref. 3) on IGSCC problems in jet pump holddown beams; also, the bulletin established requirements for the inspection of jet pump holddown beams during refueling outages.

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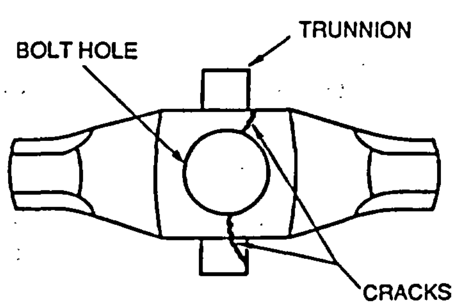

Figure 5.1 Crack locations of JET pump holddown beams

5.3.1.2 Core Spray Spargers

Core spray sparger crackings were detected in six BWRs. The cracks were initiated and propagated by IGSCC. Cold working and sensitization during the fabrication of the spargers and stresses incurred during installation were considered as major factors in the development of the observed cracks.

Cracks in the core spray spargers could alter the flow distribution to the core and might lead to the generation of loose parts in the pressure vessel. NRC IE Bulletin No. 80-13 (Ref. 4) addresses issues concerning the core spray sparger cracking problems and outlines requirements for the inspection of core spray spargers during refueling outages.

5.3.1.3 Access Hole Covers in Shroud Support

Ultrasonic (UT) inspections of one BWR showed indications of partial through-the-wall cracks in the welds attaching the access hold covers to the shroud support plate. These cracks were not detected by visual inspections. Welding-induced residual stresses and crevice conditions on the weld are major contributing factors to the initiation and propagation of cracks in the access hole cover to shroud support welds.

A failure of the access hole cover to shroud support weld could lead to a separation of the cover plate from the shroud support. The potential exists that a severed access hole cover may be swept into the recirculation pump suction line, causing damage to the pump. The unplugging of the access hole covers will also have the undesirable effects of creating an alternate flowpath that would allow some of the recirculating system flow to bypass the core during normal and accident operating conditions. NRC has issued Information Notice No. 88-03 (Ref. 5) on the importance of inspecting the access hole cover to shroud support welds, but the Information Notice did not specify any specific inspection requirement.

5.3.1.4 In-Core Neutron Flux Monitor Dry Tubes

Indications of cracks were detected by visual inspections of IRM and SRM dry tubes in several BWRs. The cracks were located in the thin tube segment surrounding the compression spring. Due to the high neutron fluence level at the locations of these components, it is generally accepted that IASCC and radiation-induced embrittlement may have contributed to the development of the observed cracks. The accumulation of cruds and oxide formation in crevices between the guide plug and the thin tube segment may have produced the tensile stresses needed for the development of SCC.

The thin tube is a nonpressure-retaining boundary of the dry tube. There was no indication of cracks in the adapter, shaft, guide plug, and thick instrument cavity (a primary pressure retaining boundary). Cracked dry tubes are replaced during refueling outages.

5.3.1.5 Stub Tube to CRD Housing Welds

During routine inspections of the pressure vessel of two older BWRs, leakage was detected in the gap between the vessel wall and the CRD housing. Reactor cooling water is leaked to the outside of the pressure vessel through a slip-fit clearance between the inside surface of the stub tube and the outside surface of the CRD housing. The leak is considered as unisolable.

Leakage was attributed to the development of through-the-wall cracks in the stub tube in the vicinity of the J-weld that joins the CRD housing to the top of the stub tube. The cracks were intergranular. A typical crack is illustrated in Fig. 5.2.

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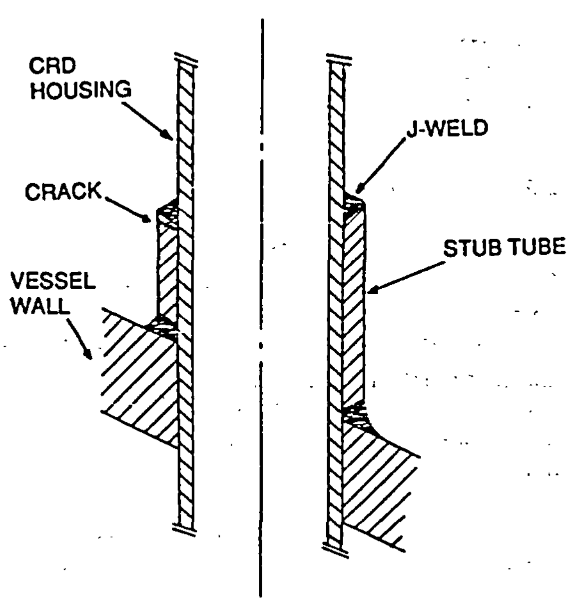

Figure 5.2 Crack in CRD stub tube

The leakage was stopped by using mechanical seals sized to slip over the outer surface of the stub tube and by roll-swaging the CRD housing to close the gap with the vessel wall.

Stub tubes are parts of the reactor pressure vessel, and their failures are included in the report because they were caused by the attachment of the CRD housing. This is considered as a generic problem for older BWRs with stub tubes. Stub tubes are not used in newer BWRs where the CRD housing is welded directly to the bottom head of the reactor vessel. A search of the reported failure data base did not locate any failure of the CRD housing to vessel bottom head welds.

Other SCC failures were also reported and are summarized in Table 5.1. The type of reactors involved and the number of failure cases were not available when the information was taken from EPRI reports.

Specifically, IASCC failures were reported in three BWR internals: control blades, IRM and SRM dry tubes, and neutron source holders. Radiation-induced embrittlement may also have contributed to the failures of the dry tubes.

5.3.2 Reported Fatigue Failures

Of the 25 selected BWR internals 7 are susceptible to fatigue, and failures were detected in 6 of the 7 components. Most of the reported cases are classified as high-cycle fatigue caused by flow-induced vibrations. Low-cycle fatigue was the cause of failure of cracks detected in thin steam dryer hoods, and excessive structural deformations under cyclic pressure loads were the responsible failure mechanism. The following is a brief description of the more important fatigue failure cases.

5.3.2.1 Feedwater Sparger

In 1972, large circumferential cracks were detected in a feedwater sparger segment in a BWR-3 unit. Subsequent inspections revealed similar cracks in other BWR-3 and -4 units. The cracks were located near the sparger water inlet in the vicinity of the feedwater nozzle. Failures were attributed to flow-induced vibrations and rapid thermal cycling.

The thermal sleeve connects the feedwater sparger to the feedwater nozzle. One end of the thermal sleeve is welded to the sparger water inlet, and the other end is joined to the safe end of the feedwater nozzle by a slip fit joint. A simplified sketch of the feedwater sparger to feedwater nozzle

Table 5.1 BWR internals with reported SCC

Component	Cases reported	Failure location
Core spray sparger	6 (BWR-3, BWR-4) (distribution unknown)	Circumferential cracks in sparger ring and at header arm to "T" box weld HAZs
CRD housing	2 (1 BWR-2, 1 BWR-3)	Stub tube at HAZ of CRD housing to stub tube weld
Shroud head bolts	2 (2 BWR-4)	Near root of collar to bolt weld
Access hole cover	1 (1 BWR-4)	Access hole cover plate to shroud support welds
Core spray line internal piping	2 (2 BWR-4)	Weld HAZs
Steam dryer support ring	Several	Cold-work locations
Feedwater sparger	Several	Weld HAZs in ejection nozzles
Jet pump assembly holddown beam	6 (5 BWR-3, 1 BWR-4)	Beam ligament sections at beam bolt diameter
Jet pump assembly riser pipe	2 (1 BWR-3, 1 BWR-4)	Riser pipe to safe end welds
Control blades	Several	Sheath to tie rod and handle welds, bail portion of handle
IRM dry tube	4 (2 BWR-2, 1 BWR-3, 1 BWR-4)	Top of thin tube segment surrounding compression spring
SRM dry tube	3 (2 BWR-2, 1 BWR-4)	Top of thin tube segment surrounding compression spring
Neutron source holder	Several	Holder near top of beryllium chamber

connection is shown in Fig. 5.3. In the original design, the slip fit joint is loose, and feedwater leaked into the narrow region between the outside surface of the thermal sleeve and the inside surface of the feedwater nozzle. Instability of the leakage flow caused the feedwater sparger to undergo vibrations. In addition, the cooler leakage feedwater is mixed with the hot recirculating downcomer reactor water in the region surrounding the sparger water inlet. Flow-induced vibrations and rapid thermal cyclings led to the development of fatigue cracks in the feedwater sparger and around the feedwater nozzle corner.

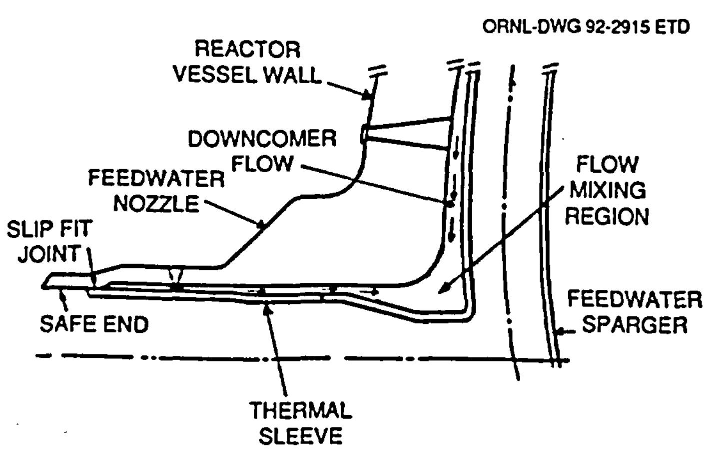

Figure 5.3 Feedwater sparger and feedwater nozzle connection

The feedwater sparger fatigue cracking problem was resolved by the implementation of a new sparger with a tight slip-fit joint, effectively reducing the leakage flow down to a very low and manageable level. A new feedwater nozzle was also used to prevent crackings in the nozzle with the new feedwater sparger design.

The BWR vendor has conducted extensive studies on the feedwater sparger cracking problem, and the results are summarized in an article by Kiss and Gerber.[6]

5.3.2.2 Jet Pump

Jet pumps in most early BWR-3 units were affected by flow-induced vibration problems, which led to the development of fatigue cracks in the pump support system. The design deficiency was recognized early, and the fatigue cracking problem was resolved by using a new and stronger holddown beam. The improved jet pump support was implemented to many units before they become operational.

Due to its operation characteristics, fatigue caused by flow-induced vibrations is always a concern with jet pumps in BWRs.

5.3.2.3 LPRM Dry Tube

Indications of leakage from three Local Power Range Monitor (LPRM) dry tubes were detected during a cold shutdown of an overseas BWR-6 unit. The reactor was operating in the low-pressure coolant injection (LPCI) mode. Damages to the LPRM dry tubes were confirmed in subsequent inspections of reactor internals, and 46 pieces of a broken dry tube were retrieved. The damaged dry tube also represented a degradation of a portion of the reactor coolant pressure boundary. Failures were attributed to fatigue caused by flow-induced vibrations generated by the LPCI flow.

The problem was unique to BWR-6 units and was resolved by the installation of a flow deflector at the LPCI discharge openings inside the shroud. The flow deflector shields the LPRM dry tubes from excitations generated by the LPCI flow.

The flow deflector was installed in domestic BWR-6 units. The reported failure data base showed no LPRM dry-tube failure in domestic BWR-6 units.

Fatigue failures in other BWR internals were reported and are summarized in Table 5.2. Although some reactor internals are susceptible to erosion, creep, and stress relaxation failures, an extensive search of the LER data base did not locate any reported failure in BWR internals that can be attributed to these particular aging-related degradation mechanisms.

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*Available from American National Standards Institute, 1430 Broadway, New York, NY 10018, Copyrighted.
†Availabe in public technical libraries.

Table 5.2 BWR internals with reported fatigue failures

Component	Cases reported	Failure location
Steam dryer	8 (BWR-2, BWR-4, BWR-5) (distribution unknown)	Cracks in thin dryer hoods and lifting rod straps
Steam dryer support ring	1 (BWR-4)	Cracks in support bracket
Jet pump sensing line	5 (1 BWR-3, 4 BWR-4)	Attachment welds to diffusers and in sensing lines
Jet pump restrainer gate	2 (2 BWR-3)	Tack welds in outboard clamp bolt keeper
In-core neutron flux monitor (LPRM) dry tube	1 (BWR-6)	Cracks in dry tube at upper cooling holes
Feedwater sparger	Several	Circumferential cracks in sparger near reactor pressure vessel feedwater nozzle

6 ISI and Aging Management Program

Visual inspection is a relatively simple inspection method used mainly to detect surface flaws and cracks. However, the method is limited to easily accessible regions of the component being examined and is not capable of detecting subsurface or partial through-the-wall cracks. Due to access limitations, interior regions of steam separators, OFS pieces, the shroud head, and the core plate are not inspected.

Alternate methods have been tried, on an experimental basis, to detect partial through-the-wall cracks, as well as flaws in inaccessible regions in BWR internals. A method that has been used with some success is ultrasonic (UT) inspection. A UT inspection has located partial through-the-wall cracks in the access hole cover that were not detected by visual examinations; however, UT inspection has its own access limitation problems, and the interpretations of UT inspection results are more complicated and difficult than are those for visual inspections. Many active research works are investigating alternate inspection methods; the objective is to improve the overall effectiveness of the plant ISI programs. Establishment of a plant ISI program is an evolving process. When new information or new inspection methods become available, the responsible organization will issue bulletins and letters to inform plant operators of the latest developments and their implications in inspection procedures and maintenance requirements.

One of the major concerns in reactor operations is the presence of loose parts in the primary system. Because of the long interval between inspections and the inability of visual inspections to detect subsurface or partial through-the-wall cracks, the scenario that a reactor may operate with flawed or cracked internal components cannot be ruled out. The failure of reactor internals during operation may generate loose parts that can cause damage to components in the reactor primary system. Such failures have occurred in PWRs,[1] but there is no reported incident of this nature in BWRs.

Loose parts can endanger the core support and core cooling functions of reactor internals. In addition to safety implications, loose parts in the reactor primary system can lead to extensive outages for repair work. For safety reasons, NRC is interested in the development of monitoring methods that can detect the presence of loose parts in the reactor primary system. Research and development in detecting loose parts has led to the establishment of *NRC Regulatory Guide* 1.133,[2] which outlines the operating requirements for the establishment of Loose Part Monitoring Systems (LPMSs) in domestic commercial nuclear power plants. Reactors licensed since 1978 are equipped with LPMSs. LPMSs have also been installed in many plants licensed before 1978.

6.1 LPMS

The LPMS is an acoustic-based monitoring system designed to detect the presence of loose metallic parts in the reactor primary system. It provides diagnostic information to the plant operator so the operator can take appropriate actions to ensure a safe plant operation. These corrective actions can also minimize the risks to other reactor components and systems.

Sound waves will be generated when a loose part, carried along by the reactor coolant flow, collides with a stationary reactor component. The sound waves, basically bending waves, will propagate to other parts of the reactor, including the pressure vessel. The effectiveness of the LPMS will depend on the system's capability to detect these impact waves and the operator's ability to interpret these structure-borne signals.

The LPMS uses a series of sensors (piezoelectric accelerometers) mounted on the outside surface of the reactor pressure vessel to detect collision-generated, structure-borne sound waves. Four rings of three sensors each are recommended for BWRs. One ring is located in the top head region and another at the bottom head region of reactor vessels. The other two rings are mounted in the cylindrical portion of the vessel, usually in the vicinity of outlet nozzles. Inputs from sensors are filtered and then fed to a monitor that will record and analyze them. More detailed information on LPMSs can be found in the reports by Kryter[3] and Mayo.[4]

The impact sound waves contain information that can be used to estimate the mass and energy of the moving object, as well as the location of the point of impact. A dominant frequency and the amplitude of the structure-borne sound waves at a sensor location can be extracted from the input signals. Differences in arrival times at various sensor locations are also measured. The mass and energy of the moving part are estimated based on a comparision of the measured dominant frequency and amplitude of the input sound waves with a known calibration curve. The impact location is determined by using differences in arrival times at known sensor locations and a triangulation process. Many uncertainties are associated with the signal processing procedures. As a general rule, the uncertainty level is low when the mass of the moving object is small and the propagation path of the structure-borne sound waves is simple. The uncertainty level increases with the mass of the loose part and the complexity of the sound wave propagation path.

The performance of LPMS is mixed. Because of the complexity and difficulty in processing and interpreting the structure-borne sound waves generated in a collision process, mixed performance is not unexpected. EPRI has conducted a study with the goal of improving the performance of LPMSs; recommendations from the study can be found in the paper by Weiss and Mayo.[5]

While an LPMS can provide information that can be used to detect the presence of loose parts in the reactor primary system, it lacks the capability of indicating the current status of system degradations that may exist in reactor components, structures, and systems. The safety of reactor operations and plant availability can be enriched by a monitoring system that has the capability of providing information regarding the status of system degradations in a component. Based on the information, appropriate corrective actions can be initiated before failures and malfunctions occur. Vibration monitoring and trending studies, which are common practices in preventive maintenance programs for rotating machineries, have potential applications in reactor inspection and maintenance programs.

6.2 Vibration Monitoring and Trending Studies

The hostile environment inside a reactor pressure vessel would rule out the use of accelerometers attached to internals for long-term vibration measurements. Most reactor internal vibrations are measured through indirect means. A common method is to infer internal component vibrations from neutron noises measured by ex-core detectors. Neutron noises are fluctuations in the neutron flux around a mean value. The neutron flux is moderated by the water layer between the core and the pressure vessel. Vibrations of internal components located around the core can change the thickness distribution of the water layer surrounding the core. Changes in the water layer thickness can lead to variations in the moderation effects of the water and can be correlated to neutron noises as measured by ex-core detectors. Ex-core neutron noise measurement is considered an effective method for detecting vibrations in internal components such as the core shroud, in-core neutron monitor dry tubes, and fuel channel boxes.

The key to the neutron noise vibration measurement method is the establishment of a reference noise spectrum at a given reactor power level and at specified sensor locations. The noise spectrum is usually given in the form of a plot of the normalized power spectral density (NPSD) curve over a specified frequency range. The ability to identify characteristic features or spikes in the noise spectrum with structural natural frequencies of internals is essential to the success of the method. Neutron noise vibration analysis is of limited value to internals whose natural frequencies cannot be clearly identified on the noise spectrum. Fundamentals of the theory and practice of reactor neutron noise vibration measurement can be found in the text by Thie.[6]

A reference noise spectrum can be obtained from results of reactor preoperational testings. Temporary in-core sensors can provide information that can be used to identify characteristic natural frequencies of internal components in the reference noise spectrum. The structural natural frequencies of reactor internals are also computed by analytical models. Trending studies are performed by comparing noise spectrums obtained during plant operations with the reference spectrum. Three to five neutron noise measurements are made during a fuel cycle. Deviations of an identifiable point in an actual noise spectrum from that in the reference spectrum can be interpreted as indications of degradations in an internal component. The ability to correlate deviations in the noise spectrums with the severity of system degradation is essential to the success of trending studies.

It may be more difficult to establish an effective neutron noise measurement system in BWRs than in PWRs. By comparision, the water layer thickness between the core and the pressure vessel wall is larger in BWRs than in PWRs. The increase in water gap may reduce the sensitivity of the water layer thickness vs changes in moderation effects and make it more difficult to detect neutron noise signals. Also, there are no existing ex-core neutron detectors in BWRs. As a result, neutron noise vibration monitoring and trending studies have not gained acceptance with BWR plant operators and reactor vendors.

Vibration monitoring and trending studies are used extensively in inspection and maintenance programs for German and French reactors,[7] but their potential applications to plant ISI programs have not been fully explored in the United States.

The ISI program is an inspection and maintenance program that is designed to detect failures before they can become a threat to the safety of plant operations. It offers little help in controlling or reducing the effects of aging degradations. Aging-related degradation mechanisms can be managed by controlling or eliminating stressors associated with the aging degradation mechanisms. Strategies used to manage the two major aging-related degradation mechanisms will be addressed separately.

6.3 Strategy for Managing SCC

Reported aging-related failure information survey results show that SCC is probably the most common aging degradation mechanism for BWR internals. Austenitic stainless steels, such as type 304, are used extensively in the construction of BWR internals. These steels provide good general corrosion resistance capability; however, the use of welding in the fabrication process and the reactor cooling water may combine to create conditions that are conducive to the development of SCC. As noted previously, three conditions are needed for the development of SCC: a corrosive environment, a susceptible material, and the presence of tensile stresses. All three conditions are present in BWR internals when weld HAZs come into contact with the reactor cooling water.

The basic strategy for controlling SCC is the elimination of any one of the three conditions. Material sensitization, which is responsible for the susceptibility of austenitic stainless steels, can be minimized by the use of stainless steels with a low carbon content and heat sink welding methods. However, for many existing operating reactors these are not viable options. The two remaining options for these reactors are the reduction of the corrosive agent in the coolant and the lowering of the tensile stress level in a component.

Tensile stresses in an internal component are caused by externally applied loads and by residual stresses in weld HAZs. Proper heat treatments can reduce residual stresses in weld HAZs, but they are not considered practical for most reactor internals because of the size of the components involved. When the dominant tensile stresses are generated by applied loads, the stress level in a component can be lowered by the use of larger structural components. Reducing the magnitude of the applied load is also effective when it is possible to do so. These remedies have been used to reduce the tensile stress level in jet pump holddown beams. The reduction of the tensile stress level is an option for controlling SCC, but it has limited applications.

The remaining option for controlling SCC is to lower the dissolved oxygen content in the reactor cooling water. The dissolved oxygen content can be reduced by the injection of hydrogen gas into the coolant flow stream. The hydrogen will react with oxygen to form water. The process becomes known as the Hydrogen Water Chemistry (HWC)[8] program.

The dissolved oxygen content in the reactor cooling water is a product of radiolytic reactions in the core and is generic to the reactor operation. A direct relationship exists between the dissolved oxygen content and the electrochemical potential (ECP) of austenitic stainless steels. When the ECP is increased, the steel becomes more susceptible to SCC. The removal of the dissolved oxygen and lowering the ECP to a value below –0.23 V (standard hydrogen electrode) will reduce the risk of the development of SCC in internals made of austenitic stainless steels.

Hydrogen injection is an effective method to remove the dissolved oxygen from the coolant flow stream. Measurements in high-purity water showed a significant drop in the ECP for austenitic stainless steels when the dissolved oxygen content is reduced to below 40 ppb.[9] The susceptibility to SCC for type 304 stainless steels is greatly reduced when the dissolved oxygen content is kept below 40 ppb. Laboratory testing results[10] indicated that HWC can inhibit the initiation of SCC, as well as retard the growth rate of existing flaws.

The implementation of a plant HWC program is an effective method for controlling SCC in primary reactor coolant piping systems. In principle, it should also be an effective method for mitigating SCC in internal components. The radiolytic reaction rate is higher around the core region, where most internals are located, and a higher hydrogen gas injection rate may be required to reduce the local dissolved oxygen content. However, field operating data with BWR internals in an HWC environment are very limited. As more plant HWC programs are implemented, a larger operating data base will become available, and it can be used to assess the effectiveness of HWC for reactor internals. The BWR vendor, GE, has issued a Service Information Letter (SIL)[11] to BWR plant operators recommending the implementation of a HWC program in all BWR plants.

Note that the implementation of a plant HWC program will increase the radiation fields in the power-generating areas such as the turbine building. The HWC program may come into conflict with other programs aiming at reducing radiation exposure levels to workers in nuclear power plants.

The presence of impurities such as chlorides and sulfates may also contribute to the development of SCC. Limits are imposed on the BWR water chemistry parameters to control the impurity contents in the reactor cooling water. The conductivity of water, which is a reliable indicator of the overall level of impurities in the water, is monitored during reactor operations. During normal BWR power operations, the operating specifications stipulate that the chloride and sulfate contents shall be maintained at 20 ppb or less, and the water conductivity at <0.3 μS/cm.[8] When the water conductivity exceeds 0.3 μS/cm, the reactor water cleanup system is activated to lower the water conductivity to below its allowable limit. Prolonged operation in an off-standard condition (in excess of 96 h) will require the implementation of an approved program of corrective measures that will bring the level down below the allowable value. When the water conductivity exceeds 1.0 μS/cm for a period of 24 h, the reactor will be brought to a cold shutdown within 16 h. When the water conductivity exceeds 5 μS/cm, the reactor will be shut down in an orderly manner as rapidly as other plant constraints will allow. The reactor water cleanup system and operating procedures will ensure that the cooling water is maintained at an acceptable purity level so that the degradating effects of impurities can be kept to a minimal level.

Other corrective actions can also be implemented to reduce the risk of SCC. The presence of crevices in an internal component is a feature that could aid the development of SCC, as well as reduce the effectiveness of HWC. A crevice-free design and manufacturing process, together with the use of stainless steels with a low carbon content and heat sink welding method, could produce internal components that are less susceptible to SCC. This combined approach should be used in making internals for newer reactors and for replacement parts.

IASCC is a major aging-related degradation mechanism for reactor internals located around the core. The effects of a plant HWC program on the development of IASCC are not well understood. A reduction of the corrosive agent in the cooling water may reduce the risk of IASCC, but there are not sufficient operating data to quantify the effects.

6.4 Strategy for Managing Fatigue

BWR internals are susceptible to high- and low-cycle fatigue failures. High-cycle fatigue is the more prevalent degradation mechanism.

The development of cracks in thin steam dry hoods is one of the few cases of reported aging-related failures in which low-cycle fatigue was the suspected cause. The thin steam dry hoods were subjected to large structural deformations under cyclic pressure loads. As a result of these early problems, thicker steam dryer hoods are used in the newer reactors and as replacement parts. Failures have not been detected in these new and thicker steam dryer hoods.

A second potential source for low-cycle fatigue failures in reactor internals is the development of flow-induced resonant vibrations. Reactor internal components are designed to prevent the development of large-amplitude flow-induced resonant vibrations; this is accomplished by separating the component structural fundamental natural frequency from the dominant flow-induced excitation frequency. As an added measure, preoperation flow testings are used to detect potential resonant and other types of vibration problems. The development of flow-induced resonant vibrations is a problem that has plagued the operations of heat exchangers, but it is not a common problem for reactor internals. In one reported case of fatigue failure, the development of flow-induced resonant vibrations may be the suspected cause. This case involved a LPRM dry tube in a BWR-6 reactor. The reported failure was detected during a shutdown operation with the reactor in the low-pressure coolant injection (LPCI) operating mode. Damage was attributed to fatigue failure at the dry tube upper cooling holes caused by FIV created by the LPCI flow. The problem was resolved by the installation of a flow deflector to shield the LPRM dry tubes from flow-induced excitations.

No evidence suggests that the development of resonant vibrations is a common problem for BWR internals. However, vessel internals are not immune to the effects of flow-induced small-amplitude vibrations, which could lead to high-cycle fatigue failures. The effects of small-amplitude vibrations are cumulative; depending on the stress level in the component, they could increase the probability of fatigue crack initiation in a component after a certain number of cycles of operation. For most stainless steels and at specified stress levels, the allowable number of cycles of operation can be evaluated by using the fatigue crack initiation curves specified in the *ASME B&PV Code*.[12] The fatigue crack initiation curves, commonly referred to as the S-N curves, tend to approach asymptotic stress values as the number of cycles increases. The S-N curves for some material level off beyond a certain number of cycles of operation, and the corresponding asymptotic stress values are known as endurance limits. Theoretically, there is no limit to the number of cycles of operation when the maximum stress level in a component is kept below the endurance limit. Maintaining a low stress level in a reactor internal component is the basic strategy for managing high-cycle fatigue problems. When the stress level cannot be kept below the endurance limit or when the S-N curves do not level off, a counting procedure is used to keep track of the number of cycles of operation; the information can be used to estimate the time to fatigue crack initiation. Note that environmental factors have not been taken into account in the development of the fatigue crack initiation curves. Flaws in a component could also affect the time to fatigue crack initiation and alter the rate of crack propagation.

A fracture-mechanics analysis can be used to evaluate the time interval between crack initiation and unstable crack propagation. The rate of fatigue crack propagation is sensitive to environmental conditions, and attempts are being made to incorporate these factors into the analyses. At the present time, the confidence level in these predictive methods is uncertain at best because environmental effects have not been fully incorporated into the analysis.

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7 Discussions and Conclusions

BWR internals operate in an environment that is favorable to the development of aging-related degradation mechanisms. The primary stressors are a high-temperature, corrosive, and moving liquid coolant; manufacturing processes; fast neutron fluxes; and cyclic loadings. Reported aging-related failures can be attributed to two major degradation mechanisms: SCC (including IASCC) and fatigue. The presence of crevices and cold-work locations in an internal component may contribute to and accelerate the development of SCC problems.

7.1 SCC

The three conditions needed for the development of SCC are a corrosive environment, a susceptible material, and tensile stresses. Dissolved oxygen is the primary corrosive agent in the reactor cooling water, and material sensitization can make austenitic stainless steels susceptible to corrosion attacks. Preloads in bolts and residual in weld HAZs are major sources of tensile stresses in an internal component.

The basis for formulating aging management strategies is the control or elimination of any one of the conditions associated with the development of aging-related degradation mechanisms. The use of stainless steels with a low carbon content and heat sink welding methods can make internal components less susceptible to the corrosive environment; however, these are not viable options for existing reactors that have not been built with these technologies. A popular approach taken by the nuclear power industry to mitigate SCC in BWRs is to reduce the dissolved oxygen content in the reactor cooling water. This is accomplished by the implementation of a plant HWC program in which hydrogen gas is injected into the coolant flow stream to remove the dissolved oxygen. Testing results indicated that the plant HWC is effective in suppressing SCC in reactor primary coolant piping systems. More operating data will be needed to assess the effects of HWC on reactor internals. The use of crevice-free design and manufacturing methods may aid in the control of SCC in reactor internals in new reactors and in replacement parts.

The exposure to fast neutron fluxes is a major contributing factor to the development of IASCC. IASCC has been detected in nonsensitized steel components with low stress levels. By reducing the corrosiveness of the reactor cooling water, the plant HWC program should be beneficial to the control of IASCC. However, more operating data will be needed to assess the effects of HWC in mitigating IASCC in reactor internals.

The implementation of a plant HWC program will increase the radiation fields around the power-generating areas. This may limit the use of HWC because the increase in radiation level is in conflict with the goal of reducing radiation exposures to workers in nuclear power plants.

7.2 Fatigue

Reactor internals undergo vibrations as a response to dynamic loads. Dynamic loads can be mechanical or thermal in nature. Vibrations will eventually lead to crack initiation and crack growth. The time to crack initiation is determined by the levels of stress and structural deformations in a component.

Most fatigue failures in BWR internals are caused by small-amplitude FIVs. Unlike large-amplitude FIVs, it is not possible to eliminate all small-amplitude vibrations, and high-cycle fatigue is an active aging degradation mechanism for reactor internals. Also uncertainties are associated with the synergistic effects of fatigue and a corrosive environment. A vigorous inspection program is needed to control fatigue cracking problems in reactor internals.

7.3 Inspection Programs

When an aging-related degradation mechanism is not controlled or mitigated, it would eventually lead to failures in the affected components. Effective inspection and aging management programs are essential to the prevention of aging-related failures.

BWR internals are visually inspected in accordance with the rules and regulations of Sect. XI of the *ASME B&PV Code*. Visual inspections can detect surface flaws, but the method has shortcomings. The major one is access limitation; namely, the inspection is only performed on parts that can be visually observed. As a result, internal components with a complex geometry or located in inaccessible regions are not inspected on a regular basis. Another shortcoming of visual inspections is that they cannot detect subsurface or partial through-the wall cracks. Other inspection methods, such as UT and eddy-current inspections, have been used to examine internals. Both the UT and eddy-current inspection methods have their own limitations. For these reasons, the scenario in which a reactor may operate with flawed or cracked internal components cannot be ruled out. Reactors licensed since 1978 are equipped with LPMS. LPMS is an acoustic-based monitoring system and can detect the presence of loose parts in the reactor primary system. When a loose part is detected, corrective actions can be taken to limit potential damages to other reactor components and systems. LPMS does not possess the capability of detecting incipient failures. Vibration monitoring and trending studies can provide such capability. Neutron noise measurement is a common method for detecting vibrations in selected reactor internals. Trending study is an assessment of the relative vibration amplitude of an internal component as a function of time. Increases over a predetermined limit are interpreted as signs for the incipient failure of a component. The large water gap between the core and the reactor vessel wall may reduce the effectiveness of neutron noise vibration measurements for BWR internals. The lack of ex-core neutron flux monitors may also handicap the use of neutron noise vibration measurement in BWRs.

An effective vibration monitoring and trending studies program can enhance reactor safety, as well as plant efficiency. These preventive maintenance practices have been used sparingly but should warrant more consideration and exploitation by the U. S. utility industry.

Reactor internals, while they perform safety-related functions such as core cooling and core support, are not components of the reactor primary containment system. They can be replaced if necessary. As a result, reactor internals are not considered to be high-ranking safety-class components. Failures of internals could create conditions that may challenge the integrity of the reactor primary containment system, but they do not affect the effectiveness of the primary containment systems. However, aging-related failures may require extensive shutdown time for repair works. The implementation of effective aging degradation management and improved monitoring programs can improve the safety, as well as the efficiency, of long-term reactor operations.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This report documents the results of an aging assessment study for boiling water reactor (BWR) internals. Major stressors for BWR internals are related to unsteady hydrodynamic forces generated by the primary coolant flow in the reactor vessel. Welding and cold-working, dissolved oxygen and impurities in the coolant, applied loads and exposures to fast neutron fluxes are other important stressors. Based on results of a component failure information survey, stress corrosion cracking (SCC) and fatigue are identified as the two major aging-related degradation mechanisms for BWR internals. Significant reported failures include SCC in jet-pump holddown beams, in-core neutron flux monitor dry tubes and core spray spargers. Fatigue failures were detected in feedwater spargers. The implementation of a plant Hydrogen Water Chemistry (HWC) program is considered as a promising method for controlling SCC problems in BWR. More operating data are needed to evaluate its effectiveness for internal components. Long-term fast neutron irradiation effects and high-cycle fatigue in a corrosive environment are uncertainty factors in the aging assessment process. BWR internals are examined by visual inspections and the method is access limited. The presence of a large water gap and an absence of ex-core neutron flux monitors may handicap the use of advanced inspection methods, such as neutron noise vibration measurements, for BWR.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

NPAR, reactor internals, stressors, aging-related degradation mechanisms, component failure information, inservice inspections boiling-water reactor (BWR), stress corrosion cracking (SCC), fatigue

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