3
COMPONENT CATEGORIZATION AND AGING MANAGEMENT STRATEGY DEVELOPMENT

This section of the I&E guidelines provides a summary of the design characteristics for B&W, CE, and Westinghouse PWR internals; a summary of the screening process used for the preliminary categorization of PWR internals; and a summary of the categorization and aging management strategy development results.

3.1 Design Characteristics Summary

The functions of PWR internals are to:

1. provide support, guidance, and protection for the reactor core;
2. provide a passageway for the distribution of the reactor coolant flow to the reactor core;
3. provide a passageway for support, guidance, and protection for control elements and in-vessel/core instrumentation; and
4. provide gamma and neutron shielding for the reactor vessel.

3.1.1 B&W Internals Design Characteristics

The seven B&W-designed operating units share common design characteristics with minor variations. The B&W-designed PWR internals consist of two major structural assemblies that are located within, but not welded to the reactor vessel. These two major assemblies are called the plenum assembly and the core support assembly (CSA). The latter includes three principal sub-assemblies – the core support shield (CSS) assembly, the core barrel assembly, and the lower internals assembly. The general arrangement of the B&W-designed PWR internals is shown in Figure 3-1. A brief summary of the design characteristics for these internals is provided in the following sub-section. For a more complete discussion, see Reference 8.

Plenum Assembly

The plenum assembly is a cylindrical structure with perforated grid plates on top and bottom, and is comprised of: (1) the plenum cover assembly; (2) the plenum cylinder assembly; (3) the upper grid assembly; and (4) the control rod guide tube assemblies. The plenum assembly fits inside the core support shield, positions the top of the fuel assemblies, supports the control rod guide tube assemblies, and provides the core hold-down required for hydraulic lift forces. The plenum assembly also provides continuous guidance and protection for the control rods, and directs flow out of the core to reactor vessel outlet nozzles. The plenum assembly is removed at the beginning of every refueling outage, in order to permit access to the fuel assemblies.

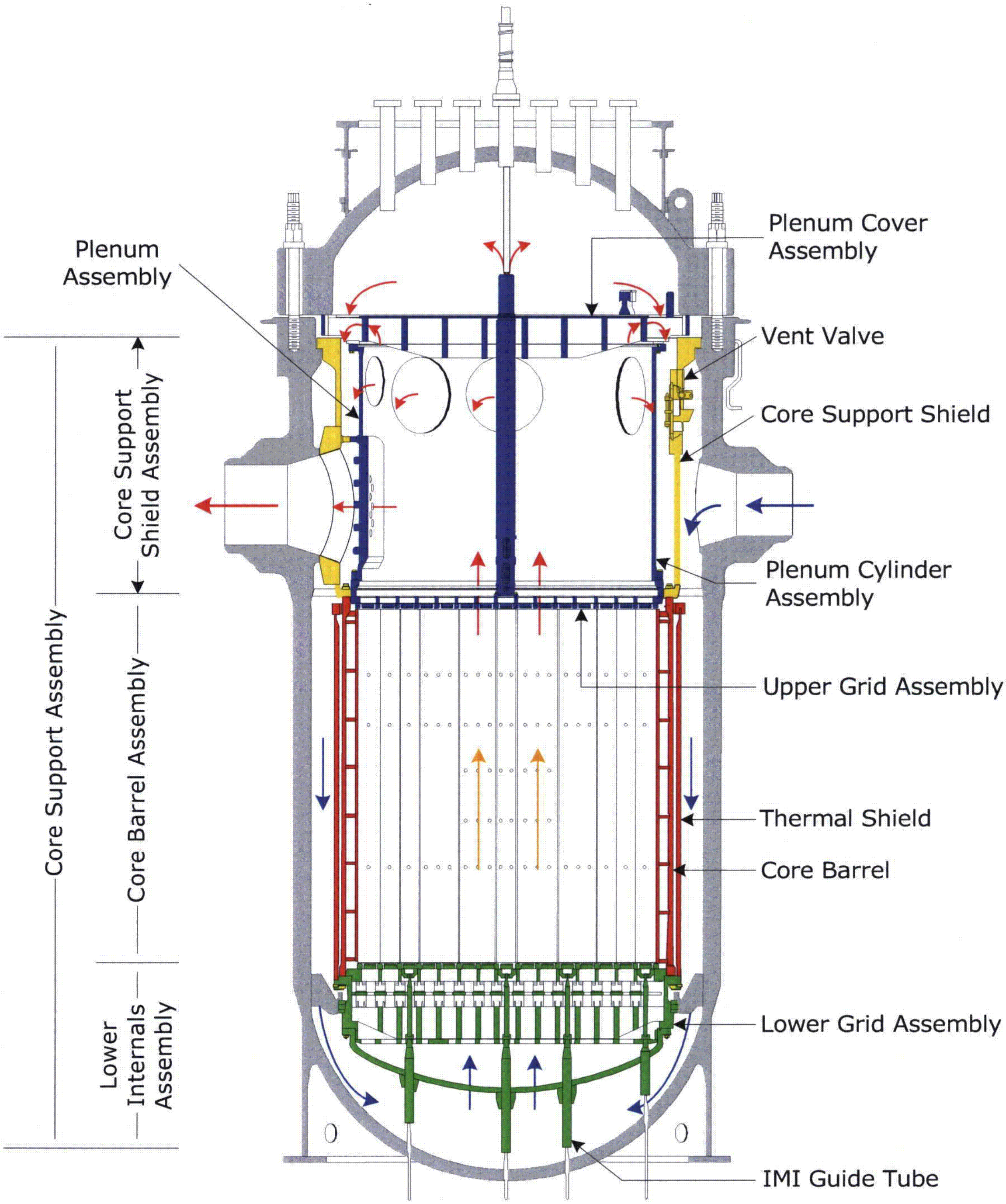

Figure 3-1
Overview of typical B&W internals

The plenum cover assembly is bolted to the top of the plenum cylinder, and consists of a weldment, a bottom flange, a support ring and flange, a cover plate, and lifting lugs. The plenum cover assembly provides support for the top of the control rod guide tube assemblies. The lifting lugs are used to lift the plenum assembly out of the reactor vessel.

The plenum cylinder assembly is bolted to the bottom of the plenum cover assembly and consists of a cylinder, top and bottom flanges, reinforcing plates, and round bars. Its function is to direct the flow of reactor coolant from the core region to the reactor vessel outlet nozzles.

The upper grid assembly sits inside the lower flange of the core support shield and is bolted to the plenum cylinder bottom flange. It is comprised of an upper grid ring forging, an upper grid rib section, and fuel assembly support pads. Its function is to support and provide a seating surface for the tops of the fuel assemblies located within the core barrel below, and to restrain and align the bottoms of the control rod guide tubes.

The control rod guide tube assemblies each consist of a pipe (the guide housing), a flange, spacer castings, guide tubes, and rod guide sectors. The assemblies are welded to the plenum cover plate and bolted to the upper grid assembly. Their function is to provide control rod assembly guidance, protect the control rod assembly from the effects of potential coolant cross-flow, and structurally connect the upper grid assembly to the plenum cover.

Core Support Assembly

The core support assembly is fabricated by bolting together the core support shield assembly, the core barrel assembly, and the lower internals assembly to form a tall cylinder. The core support assembly remains in place in the reactor vessel during refueling, and is removed only to perform scheduled inspections of the reactor vessel interior surfaces or of the core support assembly itself.

The top portion of the core support assembly is the core support shield assembly, a cylinder with an upper flange that rests on a circumferential support ledge in the reactor vessel closure flange, thereby supporting the entire core support assembly. It sits directly on top of the core barrel, and consists of a cylinder, top and bottom flanges, outlet nozzles, vent valve nozzles, vent valves, round bars, flow deflectors, and lifting lugs. Its function is to provide a boundary between the incoming cold reactor coolant on the outside of the cylinder and the heated reactor coolant flowing on the inside of the cylinder.

The core barrel assembly is a second flanged cylinder, with its top flange bolted to the bottom flange of the core support shield assembly and its bottom flange bolted to the top flange of the lower internals assembly. The core barrel assembly consists of a cylinder, top and bottom flanges, baffle and former plates, and a thermal shield cylinder. Its functions are to direct the flow of coolant and to support the lower internals assembly. In addition, the thermal shield reduces the amount of radiation that reaches the reactor vessel. The incoming reactor coolant is directed downward along the outside of the core barrel cylinder and upward through the fuel assemblies contained inside the core barrel. A small amount of coolant flows upward through the space between the core barrel cylinder and the baffle plates. A small portion of the coolant also runs down the annulus between the thermal shield and the core barrel cylinder, through holes drilled in the core barrel cylinder bottom flange, and then upward through the core.

The lower internals assembly consists of a lower grid assembly, a flow distributor assembly, and in-core monitoring instrumentation guide tube assemblies. The lower internals assembly is bolted to the bottom flange of the core barrel cylinder, and its function is to direct coolant flow upward through the fuel assemblies. The lower grid assembly consists of three grid structures or flow plates: (1) the lower grid rib section, (2) the flow distributor plate, and (3) the lower grid forging. Each of these flow plates has holes or flow ports to direct coolant flow upward toward the fuel assemblies.

3.1.2 CE Internals Design Characteristics

In general, the 14 operating CE-designed PWRs in the U.S. are divided into three groups: (1) those with a bolted core shroud and top-mounted in-core instrumentation (ICI); (2) those with a welded core shroud and top-mounted ICI; and (3) those with a welded core shroud and bottom-mounted ICI.

The CE-designed PWR internals consist of three major structural assemblies, plus three other sets of major components. The three major assemblies are the: (1) upper internals assembly, (2) core support barrel assembly, and (3) lower internals assembly. In addition, the three other sets of major components are the control element assembly shroud assemblies, core shroud assembly, and in-core instrumentation support system. The general arrangement of the CE-designed PWR internals is shown in Figure 3-2. A brief summary of the design characteristics for these internals is provided in the following sub-section. For a more complete discussion, see Reference 10.

Upper Internals Assembly

The upper internals assembly is located above the reactor core, within the core support barrel assembly, and is removed during refueling as a single component in order to provide access to the fuel assemblies. The upper internals assembly consists of the upper guide structure support plate, the fuel assembly alignment plate, the control element assembly shroud assemblies, the upper guide structure grid assembly, the upper guide structure cylinder, the in-core instrumentation support system and the hold-down ring (or expansion compensating ring). The functions of the upper internals assembly are to provide alignment and support to the fuel assemblies, to maintain control element assembly shroud spacing, to prevent movement of the fuel assemblies in the case of a severe accident condition, and to protect the control rods from cross-flow effects in the upper plenum. The flange on the upper end of the upper internals assembly rests on the core support barrel.

Core Support Barrel

The core support barrel assembly consists of the core support barrel, the core support barrel upper flange, core support barrel alignment keys, and the core support barrel snubbers. In one CE plant, a thermal shield is part of the core support barrel assembly.

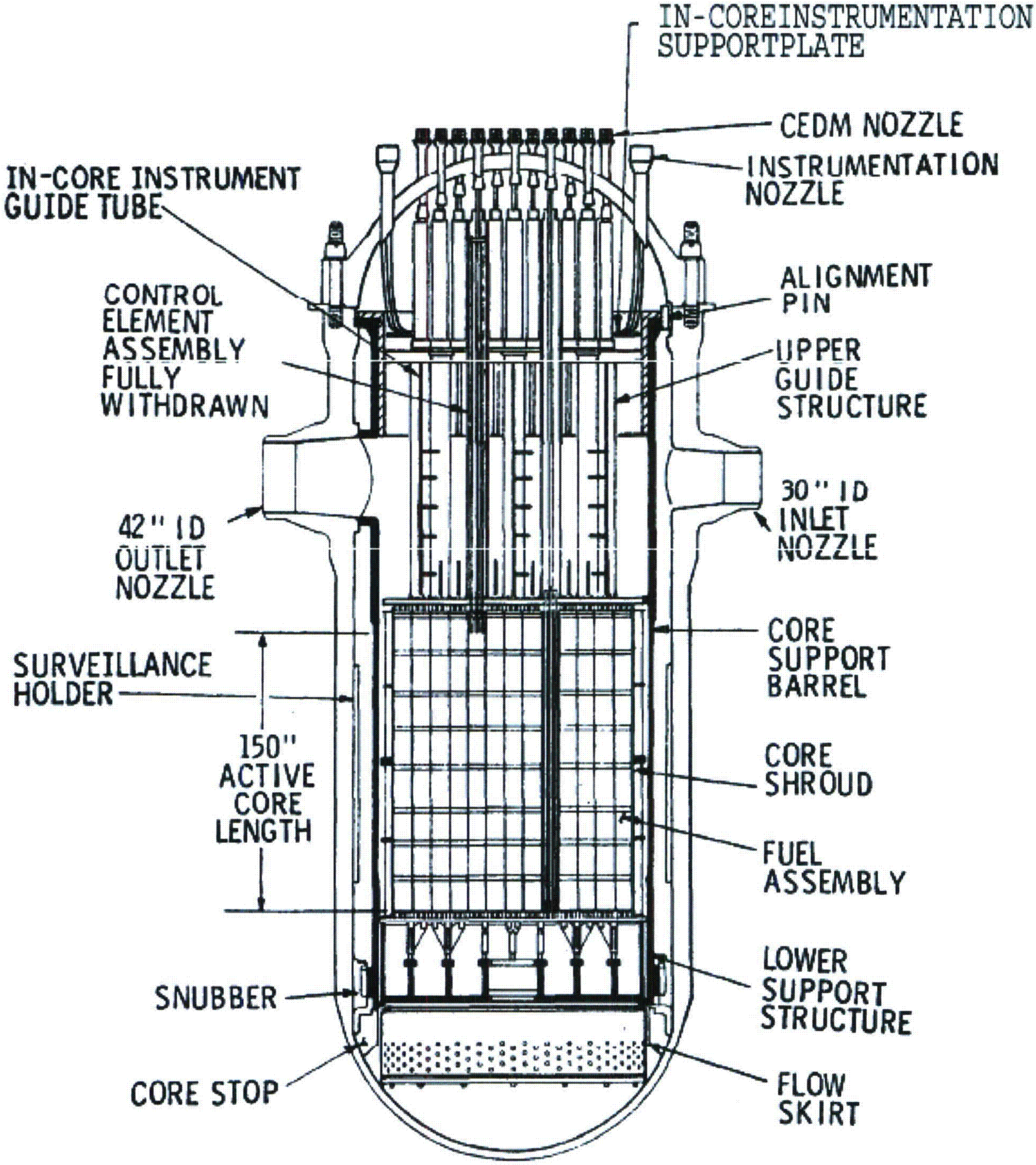

Figure 3-2
Overview of typical CE internals

The core support barrel is a cylinder which contains the core and other internals. Its function is to resist static loads from the fuel assemblies and other internals, and dynamic loads from normal operating hydraulic flow, seismic events, and loss-of-coolant-accident (LOCA) events. The core support barrel also supports the lower internals assembly and its core support plate, upon which the fuel assemblies rest.

The core support barrel upper flange is a thick ring that supports and suspends the core support barrel from a ledge on the reactor vessel.

Lower Internals Assembly

The lower internals assembly consists of the core support plate, the fuel alignment pins, the core support columns, the in-core instrumentation (ICI) support system, and the lower support structure beam assemblies. The core support plate functions are to position and support the reactor core, and to provide control of reactor coolant flow into each fuel assembly. The core support plate transmits the weight of the core to the core support barrel by means of the vertical core support columns, an annular skirt, and the lower support structure beams. The fuel alignment pins protrude from the core support plate and provide guidance and limit lateral movement of the individual fuel assemblies. CE plants with a welded core shroud and bottom-mounted ICI have no core support plate, in which case the fuel alignment pins are attached directly to the core support deep beams.

Core Shroud Assembly

The core shroud assembly is located within the core support barrel and directly below the upper internals assembly. The core shroud assembly is attached to the core support barrel by threaded structural fasteners for those internals with a bolted core shroud and top-mounted ICI. The core shroud assembly is attached to the core support plate – an element of the lower internals assembly – by tie rods or welds for the internals with a welded core shroud and top-mounted ICI (Figure 3-3). The core shroud assembly is attached to the lower internals assembly cylinder by welding for those internals with a welded core shroud and bottom-mounted ICI (Figure 3-4). The core shroud assembly functions are to provide a boundary between reactor coolant flow on the outside of the core support barrel and the reactor coolant flow through the fuel assemblies, to limit the amount of coolant bypass flow, and to reduce the lateral motion of the fuel assemblies.

Control Element Assembly Shroud Assemblies

The control element assembly shroud assemblies consist of control element assembly shrouds, the control element assembly shroud bolts, and the control element assembly shroud extension shaft guides. The shroud tubes protect the control rods from cross-flow effects in the upper plenum. The bottom part of the shrouds is bolted at their lower end to the fuel assembly alignment plate. The extension shaft guides also protect the control rods from cross-flow effects in the upper plenum, and provide lateral support and alignment of the control element assembly extension shafts during refueling operations. The control element drive mechanisms are positioned on the reactor vessel closure head and are coupled to the control element assemblies by the control element assembly extension shafts. Control element assembly shroud assemblies are attached to the upper guide structure support plate by tie rods.

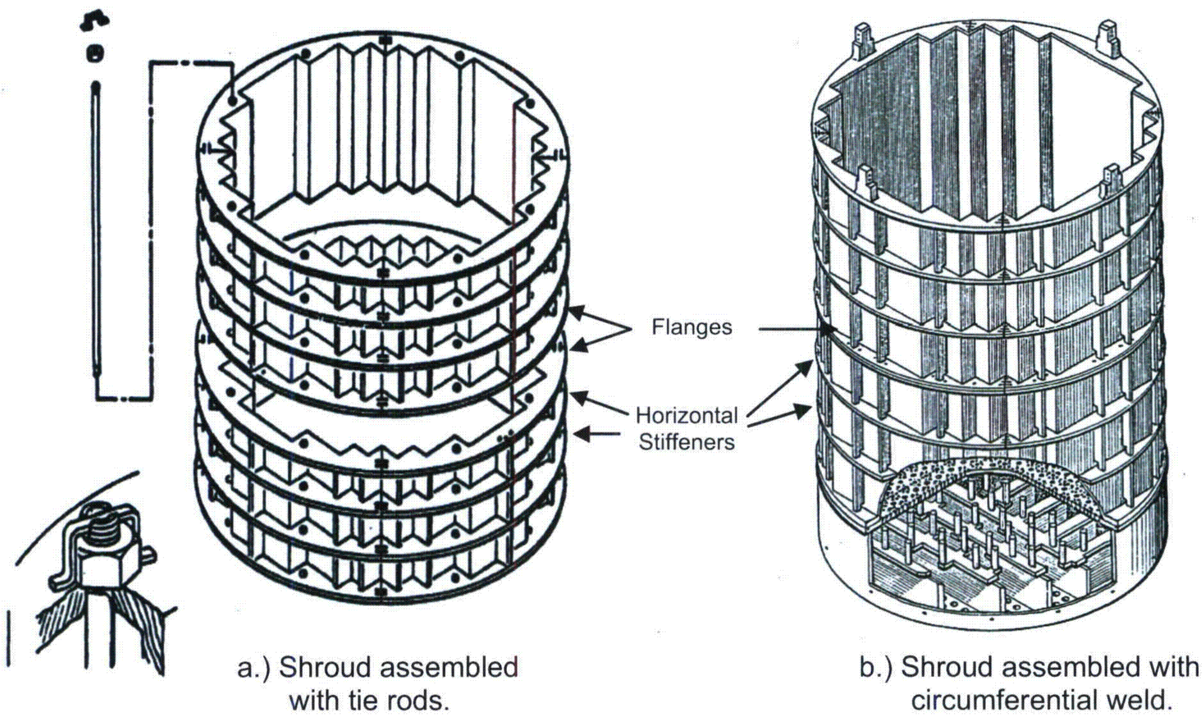

Figure 3-3
CE welded core shroud designs assembled in two vertical sections (with top-mounted ICI)

In-Core Instrumentation Support System

The in-core instrumentation support system consists of in-core instrumentation guide tubes and components which provide support to the in-core instrumentation.

For plants with top-entry in-core instrumentation assemblies, the in-core instrumentation is inserted through the reactor vessel head through a nozzle into a guide tube. The guide tubes interface with the thimble support plate, which is perforated to fit over the control element assembly extension shaft guides, with a connection to the upper guide structure support plate. ICI thimble tube assemblies extend downward from a flanged connection at the thimble support plate (in the original design) through the fuel alignment plate and into the reactor core. The upper portion of the ICI thimble tube exists between the thimble support plate and fuel alignment plate, while the lower ICI thimble tube is the zirconium alloy portion that extends into the fuel assemblies.

For plants with bottom-entry in-core instrumentation, the guide tubes are connected to and supported by the lower internals assembly, from which the in-core instrumentation enters the core.

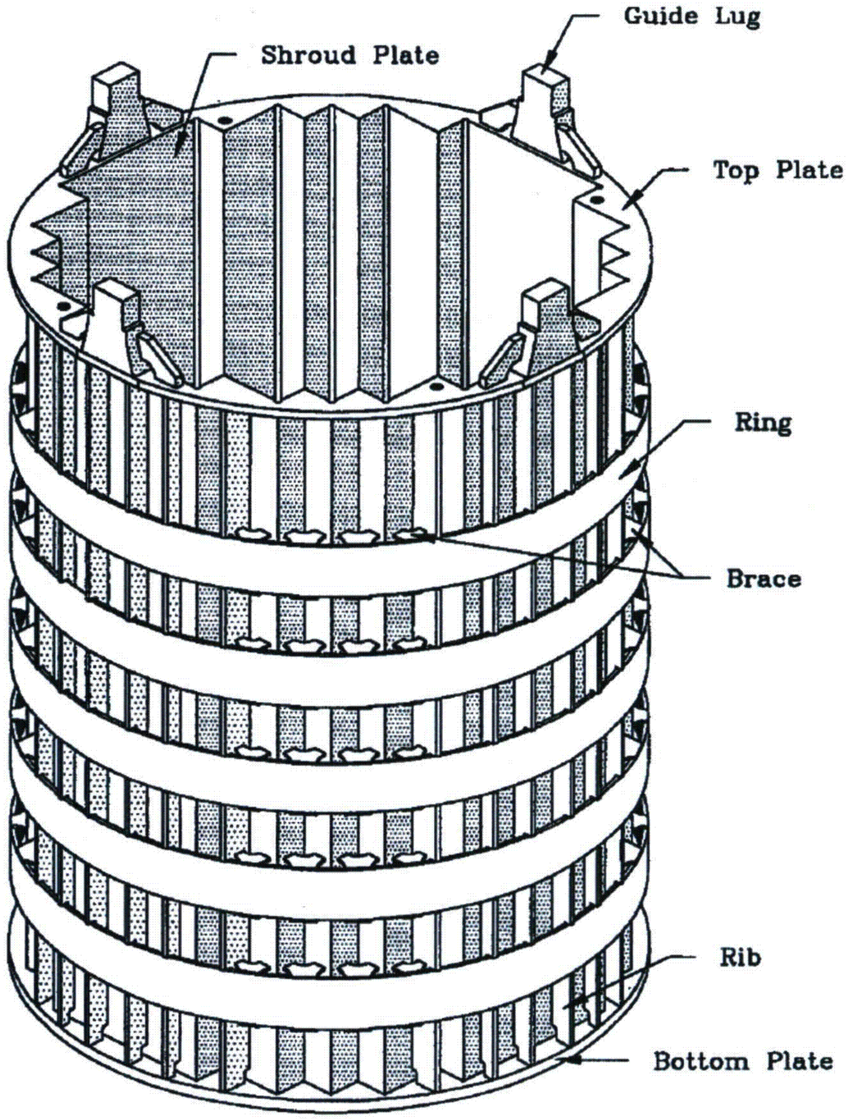

Figure 3-4
CE welded core shroud with full height panels (with bottom-mounted ICI)

3.1.3 Westinghouse Internals Design Characteristics

A schematic view of a typical set of Westinghouse-designed PWR internals is shown in Figure 3-5. However, because of the significant variation in design characteristics, the 48 operating Westinghouse PWRs in the U.S. are sub-divided into various groups, starting with the number of reactor coolant system (RCS) loops – two-loop, three-loop, and four-loop configurations. Other significant variations include the original thermal output, the baffle-barrel region flow design (downflow, upflow, and converted upflow), and upper support plate configuration. A complete set of these groups is provided in Section 4 of Reference 10.

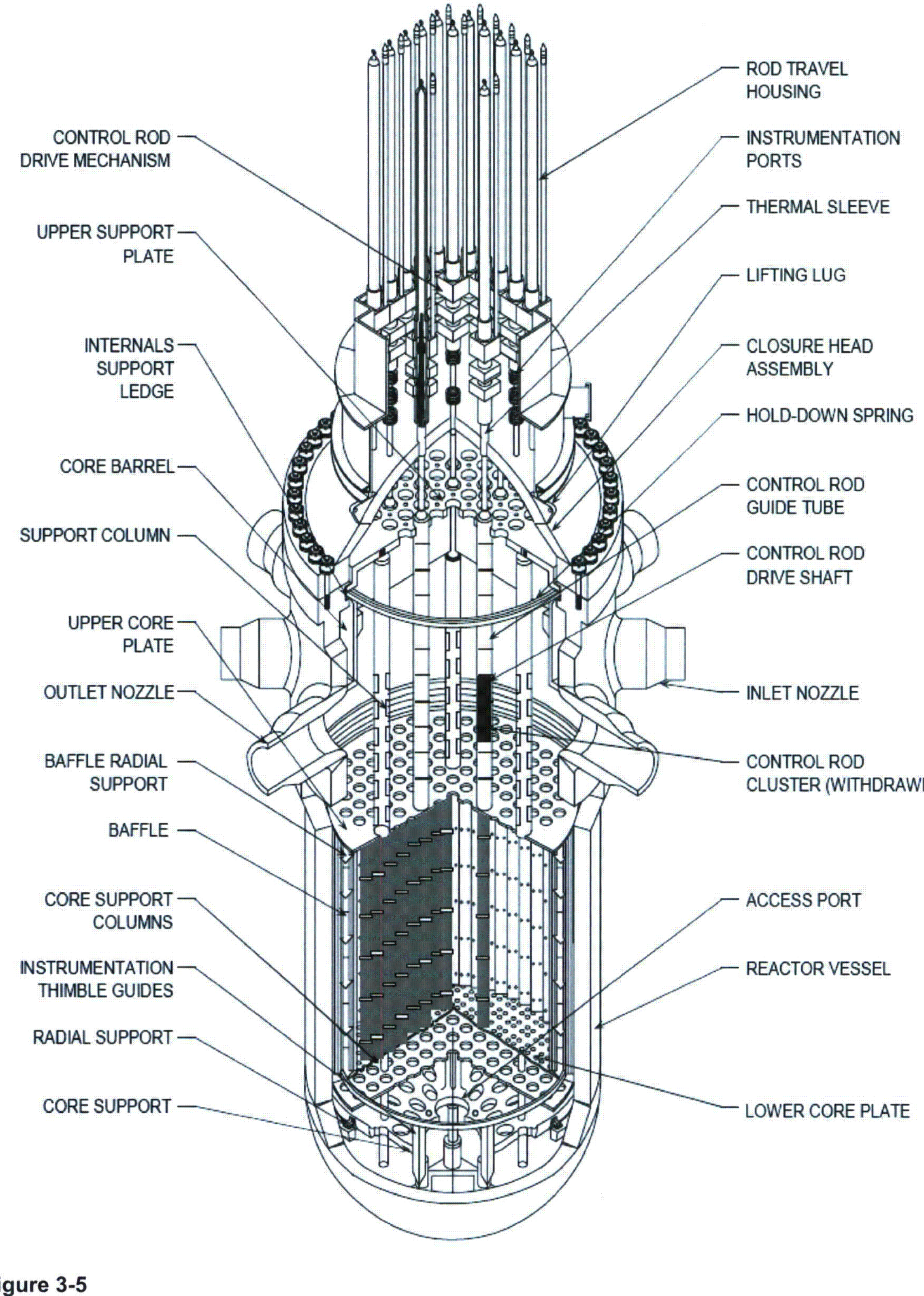

Figure 3-5
Overview of typical Westinghouse internals

All Westinghouse internals consist of two basic assemblies: an upper internals assembly that is removed during each refueling operation to obtain access to the reactor core and a lower internals assembly that can be removed following a complete core off-load.

The lower internals assembly is supported in the vessel by clamping to a ledge below the vessel-head mating surface and is closely guided at the bottom by radial support/clevis assemblies. The upper internals assembly is clamped at this same ledge by the reactor vessel head. The bottom of the upper internals assembly is closely guided by the core barrel alignment pins of the lower internals assembly.

Upper Internals Assembly

The major sub-assemblies that comprise the upper internals assembly are the: (1) upper core plate (UCP) and fuel alignment pins; (2) upper support column assemblies; (3) control rod guide tube assemblies and flow downcomers; (4) upper plenum; and (5) upper support plate assembly.

During reactor operation, the upper internals assembly is preloaded against the fuel assembly springs and the internals holddown springs by the reactor vessel head pressing down on the outside edge of the upper support plate (USP). The USP acts as the divider between the upper plenum and the reactor vessel head and as a relatively stiff base for the rest of the upper internals. The upper support columns and the guide tubes are attached to the USP. The UCP, in turn, is attached to the upper support columns. The USP assemblies are designated as one of three different designs: (1) a deep beam design, (2) a top hat design, or (3) an inverted top hat design.

The UCP is perforated to permit coolant to pass from the core below into the upper plenum defined by the USP and the UCP. The coolant then exits through the outlet nozzles in the core barrel. The UCP positions and laterally supports the core by fuel alignment pins extending below the plate. The UCP contacts and preloads the fuel assembly springs and thus maintains contact of the fuel assemblies with the lower core plate (LCP) during reactor operation.

The upper support columns vertically position the UCP and are designed to take the uplifting hydraulic flow loads and fuel spring loads on the UCP. The guide tubes are bolted to the USP and pinned at the UCP so they can be easily removed if replacement is desired. The guide tubes are designed to guide the control rods in and out of the fuel assemblies to control power generation. The guide tubes are also slotted in their lower sections to allow coolant exiting from the core to flow into the upper plenum.

The upper instrumentation columns are bolted to the USP. These columns support the thermocouple guide tubes that lead the thermocouples from the reactor head into the upper plenum to just above the UCP.

The UCP alignment pins locate the UCP laterally with respect to the lower internals assembly. The pins must laterally support the UCP so that the plate is free to expand radially and move axially during differential thermal expansions between the upper internals and the core barrel. The UCP alignment pins are the interfacing components between the UCP and the core barrel. The UCP alignment pins are shrunk-fit and welded into the core barrel and the core barrel bearing pad. The gap sizes between the alignment pins and the matching inserts are customized.

The USP, the upper support columns, and the UCP are typically considered core support structures.

Lower Internals Assembly

The reactor core is positioned and supported by the lower internals and upper internals assemblies. The individual fuel assemblies are positioned by fuel alignment pins in the LCP

and in the UCP. These pins control the orientation of the core with respect to the lower internals and upper internals assemblies. The lower internals are aligned with the upper internals by the UCP alignment pins and secondarily by the head/vessel alignment pins. The lower internals are orientated to the vessel by the lower radial keys and by the head/vessel alignment pins. Thus, the core is aligned with the vessel by a number of interfacing components.

The lower internals assembly is supported in the vessel by clamping to a ledge below the vessel-head mating surface and closely guided at the bottom by radial support/clevis assemblies. The upper internals assembly is clamped at this same ledge by the reactor vessel head. The bottom of the upper internals assembly is closely guided by the core barrel alignment pins of the lower internals assembly.

The fuel assemblies are supported inside the lower internals assembly on top of the LCP. The LCP is elevated above the lower support forging by support columns and bolted to a ring support attached to the inside diameter of the core barrel. The support columns transmit vertical fuel assembly loads from the LCP to the much thicker lower support forging. The lower support forging is welded to and supported by the core barrel, which transmits vertical loads to the vessel through the core barrel flange.

The functions of the LCP are to position and support the core and provide a metered control of reactor coolant flow into each fuel assembly. The LCP is located near the bottom of the lower support assembly, inside the core barrel, and above the lower support forging.

The function of the lower support forging or casting is to provide support for the core. The lower support forging is attached with a full-penetration weld to the lower end of the core barrel. In this position it can provide uninterrupted support to the core. The core sits directly on the LCP, which is supported by the lower support columns that are attached to and extend above the lower support forging. Some four-loop plants employ a cast lower support instead of a forging. The functions, loads, and supporting hardware are the same except for dimensions.

The primary function of the core barrel is to support the core. A large number of components are attached to either the core barrel or the core barrel flange, including the baffle/former assembly, the outlet nozzles, the neutron panel assemblies or thermal shield, the alignment pins that engage the UCP and the LCP, the lower support forging, and the LCP. The radial keys restrain large transverse motions of the core barrel but at the same time allow unrestricted radial and axial thermal expansions.

The baffle and former assembly is made up of vertical plates called baffles and horizontal support plates called formers. The baffle plates are bolted to the formers by the baffle/former bolts, and the formers are attached to the core barrel inside diameter by the barrel/former bolts. The baffle/former assembly forms the interface between the core and the core barrel. The baffles provide a barrier between the core and the former region so that a high concentration of flow in the core region can be maintained. A secondary benefit, although not a requirement of the baffles, is to reduce the neutron flux on the vessel.

Baffle plates are secured to each other at selected corners by edge bolts. In addition, in some installations, corner brackets are installed behind and bolted to the baffle plates.

The function of the core barrel outlet nozzles is to direct the reactor coolant, after it leaves the core, radially outward through the reactor vessel outlet nozzles. The core barrel outlet nozzles are located in the upper portion of the core barrel directly below the flange and are attached to the core barrel, each in line with a vessel outlet nozzle.

Additional neutron shielding of the reactor vessel is provided in the active core region by neutron panels or thermal shields that are attached to the outside of the core barrel. Specimen guides that contain specimens for determining the irradiation effects of the vessel during the life of the plant are attached to the neutron panels/thermal shields.

The flux thimble is a long, slender stainless steel tube that passes from an external seal table, through the bottom mounted nozzle penetration, through the lower internals assembly, and finally extends to the top of the fuel assembly. The flux thimble provides a path for the neutron flux detector into the core and is subjected to reactor coolant pressure and temperature on the outside surface and to atmospheric conditions on the inside. The flux thimble path from the seal table to the bottom mounted nozzles is defined by flux thimble guide tubes, which are part of the primary pressure boundary and not considered to be part of the internals. The bottom-mounted instrumentation (BMI) columns provide a path for the flux thimbles from the bottom of the vessel into the core. The BMI columns align the flux thimble path with instrumentation thimbles in the fuel assembly.

The LCP and the fuel alignment pins, the lower support forging or casting, the lower support columns, the core barrel, the core barrel flange, the radial support keys, the baffle plates, and the former plates are typically classified as core support structures.

3.2 Initial Screening Summary

This sub-section contains a summary of the initial screening of PWR internals – screening those internals on the basis of susceptibility to eight different age-related degradation mechanisms – stress corrosion cracking (SCC), irradiation-assisted stress corrosion cracking (IASCC), wear, fatigue, thermal aging embrittlement, irradiation embrittlement, void swelling, and the combination of thermal and irradiation-enhanced stress relaxation or irradiation-enhanced creep. Development and justification of the screening criteria required knowledge of the specific aging mechanisms and their effects, some engineering judgment, and the use of empirical relations where data were lacking. The full explanation of the screening criteria for the eight age-related degradation mechanisms identified for PWR internals is provided in Reference 7.

For this initial screening, the group of PWR internals that were deemed not to be susceptible to any of the eight age-related degradation mechanisms (i.e., below the screening criteria) were placed into the A Category. The Category A components are listed in previous reports for the B&W PWR designs [8] and the CE and Westinghouse PWR designs [10]. The further categorization of the components is discussed in Section 3.3.

The age-related degradation mechanisms used for the initial screening are defined in the following sub-sections. More detailed discussions of these aging mechanisms are provided in Reference 7.

3.2.1 Stress Corrosion Cracking

Stress Corrosion Cracking (SCC) refers to local, non-ductile cracking of a material due to a combination of tensile stress, environment, and metallurgical properties. The actual mechanism that causes SCC involves a complex interaction of environmental and metallurgical factors. The aging effect is cracking.

3.2.2 Irradiation-Assisted Stress Corrosion Cracking

Irradiation-assisted stress corrosion cracking (IASCC) is a unique form of SCC that occurs only in highly-irradiated components. The aging effect is cracking.

3.2.3 Wear

Wear is caused by the relative motion between adjacent surfaces, with the extent determined by the relative properties of the adjacent materials and their surface condition. The aging effect is loss of material.

3.2.4 Fatigue

Fatigue is defined as the structural deterioration that can occur as the result of repeated stress/strain cycles caused by fluctuating loads and temperatures. After repeated cyclic loading of sufficient magnitude, microstructural damage can accumulate, leading to macroscopic crack initiation at the most highly affected locations. Subsequent mechanical or thermal cyclic loading can lead to growth of the initiated crack. Corrosion fatigue is included in the degradation description.

Low-cycle fatigue is defined as cyclic loads that cause significant plastic strain in the highly stressed regions, where the number of applied cycles is increased to the point where the crack eventually initiates. When the cyclic loads are such that significant plastic deformation does not occur in the highly stressed regions, but the loads are of such increased frequency that a fatigue crack eventually initiates, the damage accumulated is said to have been caused by high-cycle fatigue. The aging effects of low-cycle fatigue and high-cycle fatigue are additive.

Fatigue crack initiation and growth resistance is governed by a number of material, structural and environmental factors, such as stress range, loading frequency, surface condition and presence of deleterious chemical species. Cracks typically initiate at local geometric stress concentrations, such as notches, surface defects, and structural discontinuities. The aging effect is cracking.

3.2.5 Thermal Aging Embrittlement

Thermal aging embrittlement is the exposure of delta ferrite within cast austenitic stainless steel (CASS) and precipitation-hardenable (PH) stainless steel to high inservice temperatures, which can result in an increase in tensile strength, a decrease in ductility, and a loss of fracture toughness. Some degree of thermal aging embrittlement can also occur at normal operating temperatures for CASS and PH stainless steel internals. CASS components have a duplex microstructure and are particularly susceptible to this mechanism. While the initial aging effect is loss of ductility and toughness, unstable crack extension is the eventual aging effect if a crack is present and the local applied stress intensity exceeds the reduced fracture toughness.

3.2.6 Irradiation Embrittlement

Irradiation embrittlement is also referred to as neutron embrittlement. When exposed to high-energy neutrons, the mechanical properties of stainless steel and nickel-base alloys can be changed. Such changes in mechanical properties include increasing yield strength, increasing ultimate strength, decreasing ductility, and a loss of fracture toughness. The irradiation embrittlement aging mechanism is a function of both temperature and neutron fluence. While the initial aging effect is loss of ductility and toughness, unstable crack extension is the eventual

aging effect if a crack is present and the local applied stress intensity exceeds the reduced fracture toughness.

3.2.7 Void Swelling and Irradiation Growth

Void swelling is defined as a gradual increase in the volume of a component caused by formation of microscopic cavities in the material. These cavities result from the nucleation and growth of clusters of irradiation produced vacancies. Helium produced by nuclear transmutations can have a significant impact on the nucleation and growth of cavities in the material. Void swelling may produce dimensional changes that exceed the tolerances on a component. Strain gradients produced by differential swelling in the system may produce significant stresses. Severe swelling (>5% by volume) has been correlated with extremely low fracture toughness values. Also included in this description is irradiation growth of anisotropic materials, which is known to cause significant dimensional changes in in-core instrumentation tubes fabricated from zirconium alloys. While the initial aging effect is dimensional change and distortion, severe void swelling may result in cracking under stress.

3.2.8 Thermal and Irradiation-Enhanced Stress Relaxation or Irradiation-Enhanced Creep

The loss of preload aging effect can be caused by the aging mechanisms of stress relaxation or creep. Thermal stress relaxation (or, primary creep) is defined as the unloading of preloaded components due to long-term exposure to elevated temperatures, such as seen in PWR internals. Stress relaxation occurs under conditions of constant strain where part of the elastic strain is replaced with plastic strain. Available data show that thermal stress relaxation appears to reach saturation in a short time (< 1000 hours) at PWR internals temperatures.

Creep (or more precisely, secondary creep) is a slow, time and temperature dependent, plastic deformation of materials that can occur when subjected to stress levels below the yield strength (elastic limit). Creep occurs at elevated temperatures where continuous deformation takes place under constant strain. Secondary creep in austenitic stainless steels is associated with temperatures higher than those relevant to PWR internals even after taking into account gamma heating. However, irradiation-enhanced creep (or more simply, irradiation creep) or irradiation-enhanced stress relaxation (ISR) is an athermal process that depends on the neutron fluence and stress; and, it can also be affected by void swelling should it occur. The aging effect is a loss of mechanical closure integrity (or, preload) that can lead to unanticipated loading which, in turn, may eventually cause subsequent degradation by fatigue or wear and result in cracking.

3.3 Component Categorization and Aging Management Strategy Development Results Summary

3.3.1 Method and Definitions

This sub-section provides a summary of the results of the categorization of PWR internals after the initial screening. In this exercise, Failure Modes, Effects, and Criticality Analyses (FMECA) were applied to the PWR internals. Based upon the FMECA results, the most affected PWR internals were placed into Category C, while the components that are only moderately affected were placed into Category B. In addition, the FMECA process determined that some components

not initially Category A were sufficiently unaffected by consequences to be subsequently placed into Category A.

In addition to this categorization using FMECA, a more refined assessment involved functionality assessment of some of the components other than Category A components with the intent to determine the tolerance of components and systems of components to aging degradation effects. When the functionality assessments were completed, all PWR internals were placed into four functional groups, as summarized below:

- **Primary:** those PWR internals that are highly susceptible to the effects of at least one of the eight aging mechanisms were placed in the Primary group. The aging management requirements that are needed to ensure functionality of Primary components are described in these I&E guidelines. The Primary group also includes components which have shown a degree of tolerance to a specific aging degradation effect, but for which no highly susceptible component exists or for which no highly susceptible component is accessible.
- **Expansion:** those PWR internals that are highly or moderately susceptible to the effects of at least one of the eight aging mechanisms, but for which functionality assessment has shown a degree of tolerance to those effects, were placed in the Expansion group. The schedule for implementation of aging management requirements for Expansion components will depend on the findings from the examinations of the Primary components at individual plants.
- **Existing Programs:** those PWR internals that are susceptible to the effects of at least one of the eight aging mechanisms and for which generic and plant-specific existing AMP elements are capable of managing those effects, were placed in the Existing Programs group.
- **No Additional Measures:** those PWR internals for which the effects of all eight aging mechanisms are below the screening criteria were placed in the No Additional Measures group. Additional components were placed in the No Additional Measures group as a result of FMECA and the functionality assessment. No further action is required by these guidelines for managing the aging of the No Additional Measures components.

The categorization and analysis processes described herein are not intended to supersede any ASME B&PV Code Section XI [2] requirements. Any components that are classified as core support structures as defined in ASME B&PV Code Section XI IWB 2500 IWA 9000, and listed in Table IWB 2500-1. Category B-N-3 [2] have requirements that remain in effect and may only be altered as allowed by 10CFR50.55a [4].

3.3.2 Results of Categorization and Aging Management Strategy Development

The results of this process are described below and shown in Tables 3-1 through 3-3. In these tables, the right-hand column characterizes the final group: "P" corresponds to Primary components, "E" corresponds to Expansion components, "X" to Existing Programs components and "N" refers to No Additional Measures components. "A", "B" and "C" refers to the categories after the initial screening and FMECA.

- Of the total components identified for the B&W-designed PWR internals [8], the 39 components listed in Table 3-1 were determined to require further evaluation. Of these, 14 are Primary components and 12 are Expansion components, with the remaining 13 requiring No Additional Measures. There are no Existing Programs components for the B&W-designed PWR internals.

- Of the total components identified for the CE-designed PWR internals [10], the 28 components listed in Table 3-2 were determined to require further evaluation. Of these, 14 are Primary components, 8 are Expansion components, 3 are Existing Programs components, with the remaining 3 requiring No Additional Measures.
- Of the total components identified for the Westinghouse-designed PWR internals [10], the 32 components listed in Table 3-3 were determined to require further evaluation. Of these, 9 are Primary components, 9 are Expansion components, 8 are Existing Programs components, with the remaining 6 requiring No Additional Measures.

Table 3-1
Final disposition of B&W internals

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR and IC	Final Group
Plenum Cover Assembly											
Plenum Cover Weldment Rib Pads	304 SS	C	A	A	P	A	A	A	A	A	P
Plenum Cover Support Flange	304 SS	C	A	A	P	A	A	A	A	A	P
Alloy X-750 Dowels-to-Plenum Cover Bottom Flange Welds	Alloy 82 Weld	B	N	A	A	A	A	A	A	A	N
Control Rod Guide Tube (CRGT) Assembly											
CRGT Spacer Castings	CF3M	B	A	A	A	A	P Note 1	A	A	A	P Note 1
CRGT Rod Guide Tubes	304L SS	B	A	A	N	A	A	A	A	A	N
CRGT Rod Guide Sectors	304L SS	B	A	A	N	A	A	A	A	A	N
Core Support Shield Assembly											
CSS Top Flange	304 SS	C	A	A	P	A	A	A	A	A	P
UCB Bolts	Alloy A-286 or Alloy X-750	C	P	A	A	A	A	A	A	A	P
CSS Cast Outlet Nozzles (ONS-3, DB)	CF8	B	A	A	A	A	A Note 2	A	A	A	N Note 2
CSS Vent Valve Top Retaining Ring	15-5PH	B	A	A	A	A	P	A	A	A	P
CSS Vent Valve Bottom Retaining Ring	15-5PH	B	A	A	A	A	P	A	A	A	P

Table 3-1
Final disposition of B&W internals (continued)

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR and IC	Final Group
Core Barrel Assembly											
Core Barrel Cylinder (Including Vertical and Circumferential Seam Welds)	304 SS, 308L SS Welds	B	A	A	A	A	A	E	A	A	E
Alloy X-750 Core Barrel-to-Former Plate Dowel	Alloy X-750	B	N	A	A	A	A	N	A	A	N
Alloy X-750 Dowel-to-Core Barrel Cylinder Fillet Welds	Alloy 82 Weld	B	N	A	A	A	A	A	A	A	N
Thermal Shield Upper Restraint Cap Screws (Not Exposed)	304 SS	B	A	A	N	N	A	A	A	N	N
Baffle Plates	304 SS	C	A	N	A	A	A	P	N	A	P
Former Plates	304 SS	C	A	N	A	A	A	E	N	A	E
CB Bolts	304 SS	C	A	E	E	E	A	E	N	E	E
FB Bolts (Note 3)	304 SS	C	A	P	P	P	A	P	N	P	P
Internal BB Bolts (Note 3)	304 SS	C	A	N	E	E	A	E	N	E	E
External BB Bolts	304 SS	C	A	E	E	E	A	E	N	E	E
Accessible Locking Device and Locking Weld (FB Bolts and Internal BB Bolts)	304 SS Locking Device, 308L SS Locking Weld	B	A	P	A	A	A	P	A	A	P
Inaccessible Locking Device and Locking Weld (CB Bolts and External BB Bolts)	304 SS Locking Device, 308L SS Locking Weld	B	A	E	A	A	A	E	A	A	E

Table 3-1
Final disposition of B&W internals (continued)

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR and IC	Final Group
LCB Bolts	Alloy A-286 or Alloy X-750	C	P	A	A	A	A	A	A	A	P
UTS Bolts	Alloy A-286 or Alloy X-750	B	E	A	A	A	A	A	A	A	E
SSHT Studs/Nuts (CR-3) or Bolts (DB)	Alloy X-750	B	E	A	A	A	A	A	A	A	E
Upper Grid Assembly											
Alloy X-750 Dowel-to-Upper Grid Rib Section Bottom Flange Welds	Alloy 82 Weld	B	N	A	A	A	A	A	A	A	N
Upper Fuel Assembly Support Pads: Alloy X-750 Dowel Locking Weld	Alloy 82 Weld	B	E	A	A	A	A	A	A	A	E
Lower Grid Assembly											
Lower Fuel Assembly Support Pads: Pad, Pad-to-Rib Section Weld, Alloy X-750 Dowel, Cap Screw, Their Locking Welds	304SS with 308L SS Weld, Except Alloy X-750 Dowel with Alloy 69 Weld	B	A or E Note 4	A	A	A	A	E	A	A	E
Lower Grid Assembly Alloy X-750 Dowel-to-Guide Block Welds	Alloy 82 Weld	B	P	A	A	A	A	A	A	A	P
Alloy X-750 Bolts for Lower Grid Shock Pads (TMI-1 only)	Alloy X-750	B	E	A	A	A	A	A	A	A	E
Alloy X-750 Dowel-to-Lower Grid Shell Forging Welds	Alloy 82 Weld	B	N	A	A	A	A	A	A	A	N
Alloy X-750 Dowel-to-Lower Grid Rib Section Welds	Alloy 69 Weld	B	N	N	A	A	A	N	A	A	N

Table 3-1
Final disposition of B&W internals (continued)

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE	IE	VS	ISR and IC	Final Group
Lower Grid Rib-to-Shell Forging Cap Screws	304 SS	B	A	A	N	N	A	A	A	N	N
Lower Grid Support Post Pipe Cap Screws	304 SS	B	A	A	N	N	A	A	A	N	N
LTS Studs/Nuts or Bolts	Alloy A-286 or Alloy X-750	B	E	A	A	A	A	A	A	A	E
Flow Distributor Assembly											
FD Bolts	Alloy A-286 or Alloy X-750	C	P Note 5	A	A	A	A	A	A	A	P Note 5
Alloy X-750 Dowel-to-Flow Distributor Flange Welds	Alloy 82 Weld	B	N	A	A	A	A	A	A	A	N
IMI Guide Tube Assembly											
IMI Guide Tube Spiders and Spider-to-Lower Grid Rib Section Welds	CF8, 308L SS Weld	B	A	A	A	A	P	P	A	A	P

Notes to Table 3-1:

1. The CRGT spacer castings upgraded to "Primary" from "Expansion" due to deletion of previously linked "Primary" components (CSS vent valve discs and CSS vent valve shaft/hinge pins) which are active components and thus not subject to aging management requirements.
2. Thermal Embrittlement (TE) revised from "P" to "A" based on review of material data (ferrite content) from ONS-3 and DB; final grouping accordingly changed from "Primary" to "No Additional Measures"
3. Bolt overload after hard contact with the baffle and former plates is identified in Reference 13. This mechanism is only applicable to the FB bolts and internal BB bolts and their locking devices; "Primary" for the FB bolts and their locking devices, and "Expansion" for the internal BB bolts and their locking devices.
4. Only the Alloy X-750 dowel locking weld in the listed items for the lower fuel assembly support pads is susceptible to SCC and categorized as Expansion for SCC. Other items are Category A for SCC.
5. Upgraded to "Primary" from "Expansion" in accordance with the NRC SER [27].

Table 3-2
Final disposition of CE internals

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE (Note 1)	IE (Note 1)	VS	ISR and IC	Final Group
Upper Internals Assembly											
Fuel Alignment Plate (Core Shrouds with Full-Height Shroud Plates)	304 SS	B	N	A	N	P	A	A	A	A	P
Lower Support Structure											
Core Support Plate	304 SS 304L SS	C	N	N	N	P	A	P	A	A	P
Fuel Alignment Pins (Core Shrouds with Full-Height Shroud Plates)	A286 SS	C	A	X	X	X	A	X	A	X	X
Core Support Columns	304 SS	B	P Note 4	P Note 4	A	P Note 4	A	P Note 4	A	A	P Note 4
Core Support Columns	CF8	B	P Note 4	P Note 4	A	P Note 4	P Note 4	P Note 4	A	A	P Note 4
Core Support Deep Beams (Core Shrouds with Full-Height Shroud Plates)	304 SS	C	X	X	A	P	A	P	A	A	P
Core Support Column Bolts	316 SS	B	A	E	N	E	A	E	A	N	E
Lower Core Support Beams	304 SS	A	E Note 5	A	A	E Note 5	A	A	A	A	E Note 5
Control Element Assembly (CEA)-Shroud Assemblies											
Instrument Tubes	304 SS	B	P	A	A	P	A	A	A	A	P
Core Support Barrel Assembly											
Upper Cylinder (including girth welds)	304 SS	B	E	A	A	A	A	A	A	A	E

Table 3-2
Final disposition of CE internals (continued)

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE (Note 1)	IE (Note 1)	VS	ISR and IC	Final Group
Lower Cylinder Girth Welds	304 SS	C	P Note 4	P Note 6	A	A	A	P Note 4	A	A	P Note 4
Lower Cylinder Axial Welds	304 SS	C	E	E	A	A	A	E	A	A	E
Upper Core Barrel Flange Weld	304 SS	B	P	A	X	A	A	A	A	A	P
Upper Core Barrel Flange	304 SS	A	E Note 5	N	A	N	N	N	N	N	E Note 5
Lower Core Barrel Flange	304 SS	B	E	A	A	E	A	A	A	A	E
Lower Core Barrel Flange Weld	304 SS	B	E	A	A	P	A	A	A	A	P
Thermal Shield Positioning Pins (Note 2)	UNS S21800	B	A	A	N	N	A	A	A	N	N
Core Shroud Assembly											
Shroud Plates (Bolted) (Entire Assembly)	304 SS	C	N	E	A	A	A	P	P	A	P
Shroud Plates (Welded)	304 SS	C	N	P	A	A	A	P	P	A	P
Former Plates (Bolted) (Entire Assembly)	304 SS	B	N	E	A	A	A	P	P	A	P
Former Plates (Welded)	304 SS	B	N	P	A	A	A	P	P	A	P
Ribs	304 SS	B	N	E	A	A	A	E	N	A	E
Rings (Core Shrouds with Full-Height Shroud Plates)	304 SS	B	N	E	A	A	A	E	N	A	E
Core Shroud Bolts	316 SS	B	A	P	N	N	A	P	P	P	P
Barrel-Core Shroud Bolts	316 SS	B	A	E	N	N	A	E	A	E	E
Core Shroud Tie Rods	348 SS	B	A	A	N	N	A	N	A	N	N
Core Shroud Tie Rod Nuts	316 SS	B	A	A	N	N	A	N	A	N	N

Table 3-2
Final disposition of CE internals (continued)

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE (Note 1)	IE (Note 1)	VS	ISR and IC	Final Group
Guide Lug Insert Bolts (Note 3)	A286 SS	B	A	A	X	X	A	A	A	X	X
In-Core Instrumentation (ICI)											
ICI Thimble Tubes-Lower	Zircaloy-4	C	A	A	X	A	A	A	A	A	X

Notes to Table 3-2:

1. The significance of thermal and irradiation embrittlement is directly related to the probability of a flaw existing in the component. There are no recommendations for inspection to determine embrittlement level because these mechanisms cannot be directly observed. However, potential embrittlement must be considered in flaw tolerance evaluations.
2. One plant has an existing program for this item.
3. Bolt deterioration may lead to degradation in lug fixtures. Inspection recommendations relate to the entire guide lug fixture.
4. Upgraded to “Primary” from “Expansion” in accordance with the NRC SER [27].
5. Upgraded to “Expansion” from “No Additional Measures” in accordance with the NRC SER [27].
6. Mechanism “IASCC” upgraded from “No Additional Measures” to “Primary” for lower cylinder welds in accordance with the NRC SER [27].

Table 3-3
Final disposition of Westinghouse internals

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE (Note 1)	IE (Note 1)	VS	ISR and IC	Final Group
Control Rod Guide Tube Assembly											
Lower Flanges	CF8	B	P	A	A	P	P	P	A	A	P
Guide Plates (Cards)	304 SS	C	N	A	P	N	A	A	A	A	P
C-Tubes (Note 2)	304 SS	C	A	A	P	A	A	A	A	A	N
Sheaths (Note 2)	304 SS	C	A	A	P	A	A	A	A	A	N
Guide Tube Support Pins	Alloy X-750	C	X	A	X	X	A	A	A	N	X
Upper Internals Assembly											
Upper Support Ring or Skirt	304 SS	B	E	A	A	X	A	A	A	A	X
Upper Core Plate	304 SS	A	A	A	E Note 5	E Note 5	A	A	A	A	E Note 5
Baffle-Former Assembly											
Baffle-Edge Bolts	316 SS, 347 SS	C	A	P	N	P	A	P	P	P	P
Baffle Plates and Former Plates (Note 3)	304 SS	B	A	N	A	A	A	N	P	A	P
Baffle-Former Bolts	316 SS, 347 SS	C	A	P	N	P	A	P	P	P	P
Barrel-Former Bolts	316 SS, 347 SS	C	A	E	N	E	A	E	E	E	E
Bottom Mounted Instrumentation System											
BMI Column Bodies	304 SS	B	N	N	A	E	A	E	N	A	E
BMI Column Collars	304 SS	B	A	N	A	A	A	N	N	A	N
BMI Column Cruciforms	CF8	B	A	N	A	A	N	N	N	A	N
BMI Column Extension Tubes	304 SS	B	N	N	A	A	A	N	N	A	N
Flux Thimble Tube Plugs	304 SS	B	N	N	A	A	A	N	N	A	N
Flux Thimbles (Tubes)	316 SS	C	N	N	X	A	A	N	N	A	X

Table 3-3
Final disposition of Westinghouse internals (continued)

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE (Note 1)	IE (Note 1)	VS	ISR and IC	Final Group
Core Barrel Assembly											
Core Barrel Flange	304 SS	B	E	A	X	A	A	A	A	A	X
Core Barrel Outlet Nozzle Welds	304 SS	B	E	A	A	E	A	A	A	A	E
Core Barrel Girth Welds	304 SS	C	P Note 6	P Note 6	A	A	A	P Note 6	A	A	P Note 6
Core Barrel Axial Welds	304 SS	C	E	E	A	A	A	E	A	A	E
Upper Core Barrel Flange Weld	304 SS	C	P	E	A	A	A	A	A	A	P
Lower Internals Assembly											
Lower Core Plate	304 SS	C	N	X	X	X	A	X	N	A	X
XL Lower Core Plate	304 SS	C	N	X	X	X	A	X	A	A	X
Lower Support Casting	CF8	A	A	A	A	A	E Note 5	A	A	A	E Note 5
Lower Support Forging	304 SS	A	A	A	A	A	A	A	A	A	E Note 5
Lower Support Assembly											
Lower Support Column Bodies	CF8	B	A	E	A	A	N	E	N	A	E
Lower Support Column Bodies	304 SS	B	A	E	A	A	A	E	N	A	E
Lower Support Column Bolts	304 SS	B	A	E	N	E	A	E	N	E	E
Thermal Shield Assembly											
Thermal Shield Flexures	304 SS	B	A	N	P	P	A	N	A	N	P
Alignment and Interfacing Components											
Clevis Insert Bolts	Alloy X-750	B	A	A	X	A	A	A	A	A	X
Internals Hold Down Spring (Note 4)	304 SS	B	A	A	P	A	A	A	A	A	P
Upper Core Plate Alignment Pins	304 SS	B	X	A	X	A	A	A	A	A	X

Notes to Table 3-3:

1. The significance of thermal and irradiation embrittlement is directly related to the probability of a flaw existing in the component. There are no recommendations for inspection to determine embrittlement level because these mechanisms cannot be directly observed. However, potential embrittlement must be considered in flaw tolerance evaluations.
2. Some of the items in the control rod guide tube (CRGT) assembly, namely the C-tubes and sheaths, have been placed in the No Additional Measures group, because decisions on remediation of wear and degradation in the CRGT assembly will be based only on the conditions detected in the Primary CRGT item, the guide tubes (cards).
3. The concern is a result of the collective interaction of all components that comprise the assembly and not strictly focused on the plates.
4. The hold-down spring does not directly degrade by wear. It first degrades by loss in preload, which leads to wear when an inadequate preload remains.
5. Upgraded to "Expansion" from "No Additional Measures" in accordance with the NRC SER [27].
6. Upgraded to "Primary" from "Expansion" in accordance with the NRC SER [27].

4 AGING MANAGEMENT REQUIREMENTS

The ultimate goal of an aging management program (AMP) is to monitor the condition of the internals to maintain appropriate levels of plant safety and reliability. Properly managed, the plants will fulfill their license renewal commitments.

Inspection and evaluation in support of aging management requirements typically consists of the following:

- selection of items for aging management;
- selection of the type of examination or other methodologies appropriate for each applicable degradation mechanism;
- specification of the required level of examination qualification;
- schedule of first and frequency of any subsequent examinations;
- sampling and coverage;
- expansion of scope if sufficient evidence of degradation is observed;
- examination acceptance criteria;
- methods for evaluating examination results not meeting the examination acceptance criteria;
- updating the program based on industry-wide results; and
- contingency measure to repair, replace, or mitigate.

The listed elements of inspection and evaluation interrelate. For example, the particulars of the examination acceptance criteria may affect the rules for sampling or frequency of examination.

This section of the guidelines specifies aging management requirements that are appropriate to detect the expected effects of the degradation mechanisms, and are considered acceptable for the development of an AMP. The criterion for acceptability of an aging management requirement is that it accomplishes the AMP goal, namely, ensuring the continued achievement of safety related and economically important functions of the internals. The technical bases used to develop these aging management requirements are documented [13, 14].

Some of the aging management requirements listed, for example, examination acceptance criteria, deserve greater elaboration and are therefore discussed in Section 5.

Section 4.1 describes the overall aging management approach. Then, Section 4.2 describes the various examination methodologies, ranging from general condition visual examinations to more rigorous visual, surface, and volumetric examinations, with a final sub-section that describes

physical measurement. Section 4.3 summarizes the examination requirements that are recommended for two groups of PWR internals – Primary and Expansion.

The requirements stated within this section may revert to those required by ASME Code Section XI [2] if components are repaired, modified or replaced such that the effects of aging are fully mitigated. Demonstration of the adequacy of repair, replacement, or modification activities to fully mitigate the effect of aging is the responsibility of the owner. In addition, repair, replacement or modification activities may also warrant revision to the scope and/or frequency of the generic requirements stated in these guidelines. This includes re-establishing the technical basis for the replaced components (if not fully mitigated) and the technical basis for examination of any linked Expansion components, which was developed on the basis of expert panel solicitation [15]. Individual utilities will be responsible for the technical justification of such activities to demonstrate their acceptability for different requirements than those stated in these guidelines.

The requirements for the PWR internals in the Existing Programs group are described in Section 4.4. As described in Section 4.5, those PWR internals in the No Additional Measures group require no further actions with respect to management of aging degradation, other than to continue any existing requirements that affect these components.

4.1 Aging Management Approach

The aging management approach for PWR internals consists of four major elements: (1) component categorization and aging management strategy development; (2) selection of aging management methodologies for PWR internals that are both appropriate and based on an adequate level of applicable experience; (3) qualification of the recommended methodologies that is based on adequate technical justification; and (4) implementation of the recommendations based on the Guideline for the Management of Materials Issues [1]. Each element in the approach is described in greater detail in the following paragraphs.

4.1.1 PWR Internals Categorization and Aging Management Strategy Development

The PWR internals categorization and aging management strategy development were summarized in Section 3.

4.1.2 Selection of Established Aging Management Methodologies

The second part of the aging management approach involved the selection of aging management methodologies for the PWR internals. The criteria for selection were based on:

- the methodologies should be appropriate for the characterization of particular age-related degradation effects; and
- the aging management methodologies should concentrate on techniques that have been subject to widespread application.

For these two reasons, the selected aging management methodologies emphasize existing, well-proven techniques that have been subject to widespread, relevant application. These methodologies are described in Section 4.2.

4.1.3 Aging Management Methodology Qualification

An extensive experience base for the aging management methodologies described in this section of the I&E guidelines permits selection of known aging management methodologies. Many inspections specified herein are remote visual examinations, whether visual VT-1, EVT-1 or VT-3. For remote visual examinations, no procedural qualifications are required beyond ASME B&PV Code Section XI requirements. Remote visual examinations must meet the additional generic requirements of the Inspection Standard [3] for equipment and training of personnel, and in the case of visual EVT-1, a surface condition assessment and limitations on camera angle and scan speed. All other methodologies specified herein already have well established procedural qualifications, such as volumetric examination of bolting [16]. Thus the level of procedural qualification for examinations other than remote visual is limited to technical justification. This level of qualification is appropriate. Failures of internals do not result in pressure boundary failures. Internals are either of robust design resulting in flaw tolerance well above the detection level that can be established via technical justification or consist of assemblies for which a single (or often multiple) component item failure does not prevent the assembly from performing its function.

The Inspection Standard [3] provides detailed guidance for conducting and justifying the selected examination techniques and the technical justifications required for different examination methodologies and component configurations.

4.1.4 Implementation of Aging Management Requirements

Information on the implementation of the aging management requirements is provided in Section 7 of these I&E guidelines.

4.2 Aging Management Methodologies

The aging management methodologies described in these guidelines include visual examinations, surface examinations, volumetric examinations, and physical measurements. Each of these methodologies is suitable for managing the effects of one or more aging degradation mechanisms for PWR internals, depending upon:

- tolerance of the component functionality to the progression of particular effects;
- accessibility of the component by the equipment needed for the examination; and
- suitability of the equipment for detecting the particular effect.

Where appropriate the examination methodologies selected for use in these guidelines are as specified in the latest U.S. Nuclear Regulatory Commission (NRC) approved edition and addenda of ASME B&PV Code Section XI [2], including those discussed in 4.2.1 and 4.2.2.

These methodologies are described in the following sub-sections.

4.2.1 Visual (VT-3) Examination

One examination methodology selected for use in these guidelines, which has an extensive history of use for PWR internals, is visual (VT-3) examination. Such visual examinations are exclusively relied upon for detection of general degradation of PWR internals subject to Table IWB-2500-1 B-N-3 [2] requirements. Visual (VT-3) examinations are conducted to determine

the general mechanical and structural condition of components by detecting discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion; and by identifying conditions that could affect operational or functional adequacy of components. This type of examination has been determined to be acceptable for the continued monitoring of many of the internals within the scope of these guidelines.

When specified in these guidelines, a visual (VT-3) examination is conducted in accordance with the requirements of the Inspection Standard [3]. Visual (VT-3) examinations of internals are conducted using remote examination techniques, because of personnel radiation exposure issues.

A large amount of industry experience is available relative to the application of visual (VT-3) examination procedures for examining PWR internals; however, implementation of character height requirements for VT-3 is relatively new. Thus the VT-3 required by these guidelines has greater detection capability than most of the Table IWB-2500-1 B-N-3 [2] examinations previously conducted.

4.2.2 Visual (VT-1 and EVT-1) Examinations

Other examination methodologies selected for use in these guidelines are visual (VT-1 and EVT-1) examinations. The visual (VT-1) examination and the enhanced visual (EVT-1) examination were selected where a greater degree of detection capability than visual (VT-3) examination was needed to manage the aging effect. Unlike the detection of general degradation conditions by visual (VT-3) examination, visual (VT-1) and enhanced visual (EVT-1) examinations are conducted to detect discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, corrosion, or erosion. Specifically, VT-1 is used for the detection of surface discontinuities such as gaps, while EVT-1 is used for the detection of surface breaking flaws.

When specified in these guidelines, a visual (VT-1) examination is conducted in accordance with the requirements of the Inspection Standard [3]. Enhanced visual (EVT-1) examination is also conducted in accordance with the requirements described for visual (VT-1) examination with additional requirements (such as camera scanning speed) as specified in the Inspection Standard [3].

As with visual (VT-3) examination, the current ASME B&PV Code [2] requirements for visual (VT-1) examination became more rigorous than the previous ASME B&PV Code versions. Many previous VT-1 examinations were only required to discern a 1/32" black line on a gray background. These limitations led the NRC and industry to adopt modified visual examinations for use in detecting flaws discovered in boiling water reactor (BWR) internals. The most recent research conducted by the EPRI Non-Destructive Examination (NDE) Center established the VT-1 character heights specified in Reference 2 as equally or better able to detect the degradation effects than the modified visual examination requirements developed previously [17].

4.2.3 Surface Examination

In order to further characterize discontinuities on the surface of components, surface examination can supplement either visual (VT-3) or (VT-1/EVT-1) examinations specified in these guidelines. This supplemental examination may thus be used to reject or accept relevant indications. A surface examination is an examination that indicates the presence of surface

discontinuities, and the ASME B&PV Code [2] lists magnetic particle, liquid penetrant, eddy current, and ultrasonic examination methods as surface examination alternatives. Here, only the electromagnetic testing (ET), also called eddy current surface examination method, is covered.

When selected for use as a supplemental examination to examinations performed in these guidelines, an ET examination is conducted in accordance with the requirements of the Inspection Standard [3].

ET examination is widely used for heat exchanger tubing inspections. Eddy currents are induced in the inspected object by electromagnetic coils, with disruptions in the eddy current flow caused by surface or near-surface anomalies detected by suitable instrumentation. Industry experience with ET examination is relatively robust, especially in the aerospace and petroleum refinery industries. The experience base for PWR nuclear systems is moderately robust, in particular for examination of steam generator, flux thimble, and heat exchanger tubing.

4.2.4 Volumetric Examination

Another methodology selected for use in these guidelines is volumetric examination. An ultrasonic examination (UT) was selected where visual or surface examination is unable to detect the effect of the age-related degradation for some PWR internals. For example, irradiation-assisted stress corrosion cracking (IASCC) in baffle/former bolts may occur under the bolt head – in the shank or threaded region – and will be undetectable by visual or surface examination unless the bolt is removed and subject to examination over its entire length.

When specified in these guidelines, an ultrasonic examination (UT) is conducted in accordance with the requirements of the Inspection Standard [3].

While UT has only been selected for use in these guidelines for detection of aging effects in bolting, UT is also permissible as an alternative or supplement to the specified visual examinations for other configurations such as plates and welds. This is consistent with Reference 2.

The industry has had extensive experience with the application of ultrasonic examination (UT) to PWR internals bolts, pins, and fasteners, in particular with baffle/former bolting examinations. The industry also has extensive experience in applying UT to BWR internals to detect intergranular stress corrosion cracking (IGSCC) in stainless steel and nickel-base welded plates, stainless steel internals piping, and nickel-base forgings and bolting.

4.2.5 Physical Measurements

The effects of loss of material caused by wear, the loss of pre-load or clamping force caused by such mechanisms as thermal and irradiation-enhanced stress relaxation, and excessive distortion or deflection caused by void swelling can be managed in some cases by physical measurements. Satisfaction of prescribed limits on these physical measurements (see Section 5.2) is intended to demonstrate that the affected components remain functional and can continue in service for a determined period until the next set of physical measurements. If the prescribed limits are exceeded, corrective action or evaluation for continued service is required.

In some cases, these effects may involve changes in clearances, settings, and physical displacements that can be monitored by visual means, supplemented by physical measurements that characterize the magnitude of the effects. This methodology may be used in conjunction with visual (VT-3) examination, which includes "verifying parameters, such as clearances,

settings, and physical displacements." The measurement of these parameters and their comparison to prescribed limits extends beyond visual (VT-3) examination, and will be referred to as "physical measurement of the effects of degradation."

4.3 Primary and Expansion Component Requirements

The aging management requirements for Primary and Expansion PWR internals are covered in this section. As described in Section 3.3, Primary components are those for which the effects of at least one of the eight aging mechanisms is above the screening criteria, and for which additional aging management is needed to manage those effects. The particular additional aging management methodologies were selected from the methodologies described in Section 4.2. The implementation schedule for the Expansion components will depend on the findings from the application of the additional aging management methodologies to the Primary components. The expansion criteria are defined in Section 5.

Sections 4.3.1, 4.3.2, and 4.3.3 identify and discuss the aging management methodologies for the Primary and Expansion components for B&W, CE, and Westinghouse plants, respectively. The requirements for these components are listed in Tables 4-1 through 4-6. For example, the Primary and Expansion requirements for Westinghouse internals are listed in Tables 4-3 and 4-6. These tables contain columns describing the component; any particular applicability requirement for that component; the degradation effect to be detected; the examination method; the examination coverage; and any linkage between the Primary and Expansion components. The technical bases for the examination requirements are contained in the aging management strategy reports [13, 14].

There are no specified examinations where inadequate coverage is anticipated to be an issue. However if a utility determines that the examination coverage is questionable with respect to meeting the intent of the guidelines, the condition should be entered in the utility's corrective action program for disposition.

The term "accessible" as used in Tables 4-1 through 4-6 is defined as a component surface or volume for which an examination is specified in accordance with MRP-228 that can be examined with the technologies specified in MRP-228. This accessibility is consistent with current ASME Section XI practices.

4.3.1 B&W Components

Tables 4-1 and 4-4 describe the examination requirements for PWR internals Primary and Expansion components for B&W plants.

The following is a list of the B&W Primary and Expansion components by examination technique.

- ***Visual (VT-3) Examination***

  Primary (applicable to all plants):

  - Baffle plates

    Expand to:

    - Core barrel cylinder (including vertical and circumferential seam welds)
    - Former plates

Since the regions around flow or bolt holes are preferential crack initiation sites, the surface area within one inch of the flow and bolt hole edges represents the required examination coverage.

Note that even though the core barrel cylinder and the former plates are Expansion components, they require an evaluation and not an inspection.

  Primary (applicable to all plants):

  - Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts

    Expand to:

    - Locking devices for the external baffle-to-baffle bolts and barrel-to-former bolts

Note that the bolts associated with the baffle-to-former bolt locking devices are also examined by volumetric (UT) examination. Note also that the locking devices for the internal and external baffle-to-baffle bolts and barrel-to-former bolts require an evaluation and not an inspection.

  Primary (applicable to all plants):

  - Alloy X-750 dowel-to-guide block welds

    Expand to:

    - Alloy X-750 dowel locking welds to the upper and lower grid fuel assembly support pads

The locking welds may be susceptible to cracking as a result of stress corrosion cracking (i.e., primary water stress corrosion cracking (PWSCC)). The recommended program to manage cracking of the locking welds is in conjunction with the existing Examination Category B-N-3 of the ASME B&PV Code Section XI [2] ISI program. The guide block area is accessible when the core support assembly is removed from the vessel. The 10-year interval is considered adequate due to the low consequences of failure. Due to weld residual stresses and the constrained

geometry, it is anticipated that significant cracks will be accompanied by locking device/weld separation and therefore be detectable by the visual (VT-3) examination method.

Primary (applicable to all plants):

- IMI guide tube spiders
- IMI guide tube spider-to-lower grid rib section welds

  Expand to:

  - Lower grid fuel assembly support pad items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screw, and their locking welds

The IMI guide tube spiders and their associated welds, and the lower grid fuel assembly support pads and their associated welds may have degradation by thermal or irradiation embrittlement. The effects of thermal and irradiation embrittlement can be detected by inspection to detect fracture in the items.

The IMI guide tube spiders are primary items for thermal embrittlement.

The lower grid fuel assembly support pad items (Figure 4-6) consist of the stainless steel block, Alloy X-750 dowels, and stainless steel cap screws, all susceptible to irradiation embrittlement. The primary item is the IMI guide tube spider and associated fillet welds. Cracking of the dowel or cap screw tack weld may be observed, but more likely, the aging mechanism will be detected by the grid pad not being properly located. The lower grid fuel assembly support pads and their associated welds are part of the ASME B&PV Code Section XI [2] 10-year ISI program and are inspected via visual (VT-3) examination.

Primary (applicable to all plants):

- CRGT spacer castings

There are no Expansion items for these components.

The control rod guide tube (CRGT) spacer castings (Figure 4-5) are primary items for thermal embrittlement. The effects of thermal embrittlement can be detected by inspection to detect fracture in the CRGT spacer castings. The spacer castings are a part of the CRGT structure. The spacer castings do have limited accessibility from the top or bottom of the CRGT through a center free path. This of course presumes that the plenum assembly is removed from the vessel. Remote video can be used to perform a visual (VT-3) examination at the quarter points where the threaded connections are present. These lanes are not blocked by the rod guide tubes. The examination would look for fracture of the spacer surface or evidence that the spacer is not approximately centered. The threaded fasteners are welded to the OD of the pipe column, so it is possible that a degraded threaded location would not be detected. In this case, it is assumed that the redundant support is acceptable for continued operation.

Primary (applicable to all plants):

- CSS vent valve top retaining ring
- CSS vent valve bottom retaining ring

There are no expansion items for these components.

The vent valves are contained in the core support shield assembly where the plenum assembly resides. These valves are check valves meant to relieve pressure in the interior of the core support assembly during a large break LOCA, preventing backpressure from reversing coolant flow through the core. These vent valves can be damaged due to mishandling when inserting and removing the plenum. The vent valve components listed above were identified as being susceptible to thermal aging embrittlement, which may lead to cracking. An existing program is in place at each of the B&W-designed units that requires testing and inspection of the vent valve assemblies each refueling outage. The aging management measures provided in these requirements include a provision to visually inspect the valve body and disc seating surfaces. Continuation of the existing vent valve testing and inspection requirements will manage cracking of the vent valve component items that could cause loss of the vent valve function.

Primary (applicable to all plants):

- Plenum cover weldment rib pads
- Plenum cover support flange
- CSS top flange

There are no expansion items for these components.

The potential age-related degradation mechanism for the core clamp region is wear. The purpose of the clamping is to stabilize and significantly restrict rigid body pendulum motion of the core support assembly. Wear at these locations will progress from motions generated by fluid flow once the loss of core clamping is initiated. Note that a one-time physical measurement is to be performed prior to subsequent visual (VT-3) examination.

Primary:

- Upper core barrel (UCB) bolt locking devices (applicable to all plants)

  Expand to:

  - Upper thermal shield (UTS) bolt locking devices (applicable to all plants)
  - Lower thermal shield (LTS) bolt or stud/nut locking devices (applicable to all plants)
  - Surveillance specimen holder tube (SSHT) bolt or stud/nut locking devices (Crystal River Unit 3 (CR-3) and Davis-Besse (DB) only)
  - Lower grid shock pad bolt locking devices (TMI-1 only)
- Lower core barrel (LCB) bolt locking devices (applicable to all plants)

  Expand to:

  - Upper thermal shield (UTS) bolt locking devices (applicable to all plants)
  - Lower thermal shield (LTS) bolt or stud/nut locking devices (applicable to all plants)
  - Surveillance specimen holder tube (SSHT) bolt or stud/nut locking devices (Crystal River Unit 3 (CR-3) and Davis-Besse (DB) only)
  - Lower grid shock pad bolt locking devices (TMI-1 only)

- Flow Distributor (FD) bolt locking devices (applicable to all plants)

  Expand to:

  - Upper thermal shield (UTS) bolt locking devices (applicable to all plants)
  - Lower thermal shield (LTS) bolt or stud/nut locking devices (applicable to all plants)
  - Surveillance specimen holder tube (SSHT) bolt or stud/nut locking devices (Crystal River Unit 3 (CR-3) and Davis-Besse (DB) only)
  - Lower grid shock pad bolt locking devices (TMI-1 only)

Note that the bolts or stud/nuts associated with these locking devices are also examined by volumetric (UT) examination.

- ***Volumetric (UT) Examination***

  Primary:

  - Upper core barrel (UCB) bolts (applicable to all plants)

    Expand to:

    - Upper thermal shield (UTS) bolts (applicable to all plants)
    - Lower thermal shield (LTS) bolts or stud/nuts (applicable to all plants)
    - Surveillance specimen holder tube (SSHT) bolts or stud/nuts (Crystal River Unit 3 (CR-3) and Davis-Besse (DB) only)
    - Lower grid shock pad bolts (TMI-1 only)

  - Lower core barrel (LCB) bolts (applicable to all plants)

    Expand to:

    - Upper thermal shield (UTS) bolts (applicable to all plants)
    - Lower thermal shield (LTS) bolts or stud/nuts (applicable to all plants)
    - Surveillance specimen holder tube (SSHT) bolts or stud/nuts (Crystal River Unit 3 (CR-3) and Davis-Besse (DB) only)
    - Lower grid shock pad bolts (TMI-1 only)

  - Flow Distributor (FD) bolts (applicable to all plants)

    Expand to:

    - Upper thermal shield (UTS) bolts (applicable to all plants)
    - Lower thermal shield (LTS) bolt or stud/nuts (applicable to all plants)
    - Surveillance specimen holder tube (SSHT) bolt or stud/nuts (Crystal River Unit 3 (CR-3) and Davis-Besse (DB) only)
    - Lower grid shock pad bolts (TMI-1 only)

Note that the locking devices associated with these bolts or stud/nuts are also examined by visual (VT-3) examination.

The potential degradation mechanism for the high-strength bolting rings is stress corrosion cracking. For bolting or stud/nuts, this mechanism is best detected using ultrasonic examination techniques.

The upper core barrel bolts are accessible for ultrasonic examination while the core support shield assembly is in the reactor vessel and the plenum is removed. Ultrasonic examination of the upper core barrel bolts can be performed during a normal refueling outage. The lower core barrel bolts and flow distributor bolts are only accessible when the core support shield assembly is removed from the reactor vessel. Some lower core barrel bolts are more difficult to examine and are inaccessible for replacement due to the presence of the core guide blocks mounted on the side of the lower grid assembly.

Primary (applicable to all plants):

- Baffle-to-former (FB) bolts

  Expand to:

  - Baffle-to-baffle (BB) bolts
  - Core barrel-to-former (CBF) bolts

Note that the locking devices associated with these bolts are also examined by visual (VT-3) examination.

Note that even though the baffle-to-baffle (BB) bolts and core barrel-to-former (CBF) bolts are Expansion components, they require an evaluation and not an inspection.

- ***Physical Measurement***

  Primary (applicable to all plants):

  - Plenum cover weldment rib pads
  - Plenum cover support flange
  - CSS top flange

There are no expansion items for these components.

Note: the measurement is performed to determine the differential height of top of the plenum rib pads to the reactor vessel seating surface with all three items inside the reactor vessel, but with the fuel assemblies removed.

Note that these components are subsequently examined by visual (VT-3) examination.

4.3.2 CE Components

Tables 4-2 and 4-5 describe the examination requirements for the PWR internals Primary and Expansion components for CE plants.

The following is a list of the CE Primary and Expansion components by examination technique.

- ***Visual (VT-3) Examination***

  Primary (applicable to all plants):

  - Core support column welds

  There are no expansion items for this component.

  Primary (applicable to bolted plant designs):

  - Core shroud assembly (bolted)

There are no expansion items for this component.

Note that the core shroud assembly (bolted) is examined in order to detect void swelling effects as evidenced by abnormal interaction with fuel assemblies, gaps along high fluence shroud plate joints, vertical displacement of shroud plates near high fluence joint.

  Primary (applicable to all plants with instrument guide tubes in the control element assembly (CEA) shroud assembly):

  - Instrument guide tubes (peripheral)

    Expand to:

    - Remaining instrument guide tubes within the CEA shroud assemblies

- ***Visual (VT-1 and EVT-1) Examinations***

  Primary (applicable to plant designs with core shrouds assembled in two vertical sections):

  - Core shroud assembly (welded)

There are no expansion items for this component.

Note that the core shroud assembly (welded) is examined in order to detect void swelling effects as evidenced by separation between the upper and lower core shroud segments.

  Primary (applicable to plant designs with core shrouds assembled in two vertical sections):

  - Core shroud plate-former plate weld

    Expands to:

    - Remaining axial welds

  Primary (applicable to plant designs with core shrouds assembled with full-height shroud plates)

  - Shroud plates

    Expand to:

    - Remaining axial welds
    - Ribs and rings

  Primary (applicable to all plants):

  - Upper (core support barrel) flange weld

Expands to:

- Lower core support beams
- Core support barrel assembly upper cylinder (including welds)

Primary (applicable to all plants):

- Core support barrel assembly lower cylinder girth welds

  Expands to:

  - Core support barrel assembly lower cylinder axial welds

Note that the core support barrel lower cylinder axial welds are not included as Primary components since they are subject to lower stresses than the girth welds and are thus less susceptible to stress corrosion cracking (SCC or IASCC).

Primary (applicable to all plants with core shrouds assembled with full-height shroud plates):

- Deep beams

There are no expansion items for this component.

Primary (depends on time-limited aging analysis [TLAA]):

- Core support barrel assembly lower flange weld (applicable to all plants)
- Core support plate (applicable to all plants with a core support plate)
- Fuel alignment plate (applicable to all plants with core shrouds assembled with full-height shroud plates)

There are no expansion items for these components.

- ***Volumetric (UT) Examination***

  Primary (applicable to bolted plant designs):

  - Core shroud bolts

    Expand to:

    - Core support column bolts
    - Barrel-shroud bolts

4.3.3 Westinghouse Components

Tables 4-3 and 4-6 describe the examination requirements for the PWR internals Primary and Expansion components for Westinghouse plants.

The following is a list of the Westinghouse Primary and Expansion components by examination technique.

- ***Visual (VT-3) Examination***

  Primary:

  - Baffle-former assembly (applicable to all plants)

- Thermal shield flexures (applicable to all plants with thermal shields)
- Guide plates (cards) (applicable to all plants)

There are no expansion items for these components.

Note that the baffle-former assembly is examined in order to detect void swelling effects as evidenced by abnormal interaction with fuel assemblies, gaps along high fluence baffle joint, vertical displacement of baffle plates near high fluence joint, or broken or damaged edge bolt locking systems along high fluence baffle joint. Also note that the PWROG is conducting a guide card wear project.

Primary:

- Baffle-edge bolts (applicable to all plants with baffle-edge bolts)

There are no expansion items for these components.

Note that the baffle-edge bolts are examined in order to detect lost or broken locking devices, failed or missing bolts, or protrusion of bolt heads.

- ***Visual (VT-1 and EVT-1) Examinations***

  Primary (applicable to all plants):

  - Upper core barrel flange weld

    Expands to:

    - Lower support column bodies (non cast)

  Primary (applicable to all plants):

  - Lower core barrel flange weld (alternatively designated as core barrel-to-support plate weld in some plant designs)
  - Upper and lower core barrel girth welds

    Expand to:

    - Upper and lower core barrel axial welds

  Note that the upper and lower core barrel axial welds are not included as Primary components since they are subject to lower stresses than the girth welds and are thus less susceptible to stress corrosion cracking (SCC or IASCC).

  Primary (applicable to all plants):

  - Control rod guide tube (CRGT) assembly lower flange welds

    Expand to:

    - Bottom-mounted instrumentation (BMI) column bodies (these components receive a visual (VT-3) examination)
    - Lower support column bodies (cast)
    - Upper core plate

  - Lower support forging or casting

Note that the examination coverage is 100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal.

- ***Volumetric (UT) Examination***

  Primary (applicable to all plants):

  – Baffle-former bolts

    Expand to:

    - Lower support column bolts
    - Barrel-former bolts

- ***Physical Measurement***

  Primary (applicable to all plants with 304 stainless steel hold down springs):

  – Internals hold down spring

There are no expansion items for this component.

Table 4-1
B&W plants Primary components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Plenum Cover Assembly & Core Support Shield Assembly Plenum cover weldment rib pads Plenum cover support flange CSS top flange	All plants	Loss of material and associated loss of core clamping pre-load (Wear)	None	One-time physical measurement no later than two refueling outages from the beginning of the license renewal period. Perform subsequent visual (VT-3) examination on the 10-year ISI interval.	Determination of differential height of top of plenum rib pads to reactor vessel seating surface, with plenum in reactor vessel. See Figure 4-1.
Control Rod Guide Tube Assembly CRGT spacer castings	All plants	Cracking (TE), including the detection of fractured spacers or missing screws	None	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	Accessible surfaces at each of the 4 screw locations (at every 90°) of 100% of the CRGT spacer castings (limited accessibility). See Figure 4-5.
Core Support Shield Assembly CSS vent valve top retaining ring CSS vent valve bottom retaining ring (Note 1)	All plants	Cracking (TE), including the detection of surface irregularities, such as damaged, fractured material, or missing items	None	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible surfaces (see BAW-2248A, page 4.3 and Table 4-1). See Figure 4-11.

Table 4-1
B&W plants Primary components (continued)

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Core Support Shield Assembly Upper core barrel (UCB) bolts and their locking devices	All plants	Bolts: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UTS bolts and LTS studs/nuts or bolts and their locking devices SSHT studs/nuts or bolts and their locking devices (CR-3 and DB only) Lower grid shock pad bolts and their locking devices (TMI-1 only)	Volumetric examination (UT) of the bolts within two refueling outages from 1/1/2006 or next 10-year ISI interval, whichever is first. Subsequent examination on the 10-year ISI interval unless an evaluation of the baseline results submitted for NRC staff approval justifies a longer interval between examinations. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts and their locking devices. (Note 3) See Figure 4-7.
Core Barrel Assembly Lower core barrel (LCB) bolts and their locking devices	All plants	Bolt: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UTS bolts and LTS studs/nuts or bolts and their locking devices SSHT studs/nuts or bolts and their locking devices (CR-3 and DB only) Lower grid shock pad bolts and their locking devices (TMI-1 only)	Volumetric examination (UT) of the bolts during the next 10-year ISI interval from 1/1/2006. Subsequent examination on the 10-year ISI interval unless an evaluation of the baseline results submitted for NRC staff approval justifies a longer interval between examinations. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts and their locking devices (Note 3) See Figure 4-8.

Table 4-1
B&W plants Primary components (continued)

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Core Barrel Assembly Baffle-to-former bolts	All plants	Cracking (IASCC, IE, Overload) (Note 4)	Baffle-to-baffle bolts, Core barrel-to-former bolts	Baseline volumetric examination (UT) no later than two refueling outages from the beginning of the license renewal period with subsequent examination after 10 additional years.	100% of accessible bolts. (Note 3) See Figure 4-2.
Core Barrel Assembly Baffle plates	All plants	Cracking (IE), including the detection of readily detectable cracking in the baffle plates	Core barrel cylinder (including vertical and circumferential seam welds), Former plates	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of the accessible surface within 1 inch around each flow and bolt hole. See Figure 4-2.
Core Barrel Assembly Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	All plants	Cracking (IASCC, IE, Overload), including the detection of missing, non-functional, or removed locking devices or welds	Locking devices, including locking welds, for the external baffle-to-baffle bolts and Core barrel-to-former bolts	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible baffle-to-former and internal baffle-to-baffle bolt locking devices. (Note 3) See Figure 4-2.

Table 4-1
B&W plants Primary components (continued)

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Flow Distributor Assembly Flow distributor (FD) bolts and their locking devices	All plants	Bolt: Cracking (SCC) Locking Devices: Loss of material, damaged or distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UTS bolts and LTS studs/nuts or bolts and their locking devices. SSHT studs/nuts or bolts and their locking devices (CR-3 and DB only) Lower grid shock pad bolts and their locking devices (TMI-1 only)	Volumetric examination (UT) of the bolts during the next 10-year ISI interval from 1/1/2006. Subsequent examination on the 10-year ISI interval unless an evaluation of the baseline results, submitted for NRC staff approval, justifies a longer interval between examinations. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts and their locking devices. (Note 3) See Figure 4-8.
Lower Grid Assembly Alloy X-750 dowel-to-guide block welds	All plants	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	Alloy X-750 dowel locking welds to the upper and lower grid fuel assembly support pads.	Initial visual (VT-3) examination no later than two refueling outages from the beginning of the license renewal period. Subsequent examinations on the 10-year ISI interval.	Accessible surfaces of 100% of the 24 dowel-to-guide block welds. See Figure 4-4.
Incore Monitoring Instrumentation (IMI) Guide Tube Assembly IMI guide tube spiders IMI guide tube spider-to-lower grid rib section welds	All plants	Cracking (TE/IE), including the detection of fractured or missing spider arms or, Cracking (IE), including separation of spider arms from the lower grid rib section at the weld	Lower grid fuel assembly support pad items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screw, and their locking welds (Note: the pads, dowels, and cap screws are included because of IE of the welds)	Initial visual (VT-3) examination no later than two refueling outages from the beginning of the license renewal period. Subsequent examinations on the 10-year ISI interval.	100% of top surfaces of 52 spider castings and welds to the adjacent lower grid rib section. See Figures 4-3 and 4-6.

Notes to Table 4-1:

1. A verification of the operation of each vent valve shall also be performed through manual actuation of the valve. Verify that the valves are not stuck in the open position and that no abnormal degradation has occurred. Examine the valves for evidence of scratches, pitting, embedded particles, leakage of the seating surfaces, cracking of lock welds and locking cups, jack screws for proper position, and wear. The frequency is defined in each unit's technical specifications or in their pump and valve inservice test programs (see BAW-2248A, page 4-3 and Table 4-1[18]).
2. Examination acceptance criteria and expansion criteria for the B&W components are in Table 5-1.
3. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-1, must be examined for inspection credit.
4. The primary aging degradation mechanisms for loss of joint tightness for this item are IC and ISR. Fatigue and Wear, which can also lead to cracking, are secondary aging degradation mechanisms after significant stress relaxation and loss of preload has occurred due to IC/ISR. Bolt stress relaxation cannot readily be inspected by NDE. Only bolt cracking is inspected by UT inspection. The effect of loss of joint tightness on the functionality will be addressed by analysis of the core barrel assembly, which will be performed to address Applicant/Licensee action item 6 in the SE [27].

Table 4-2
CE plants Primary components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Shroud Assembly (Bolted) Core shroud bolts	Bolted plant designs	Cracking (IASCC, Fatigue) Aging Management (IE and ISR) (Note 2)	Core support column bolts, Barrel-shroud bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a ten-year interval.	100% of accessible bolts (see Note 3). Heads are accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. See Figure 4-24.
Core Shroud Assembly (Welded) Core shroud plate-former plate weld	Plant designs with core shrouds assembled in two vertical sections	Cracking (IASCC) Aging Management (IE) (Note 2)	Remaining axial welds	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	Axial and horizontal weld seams at the core shroud re-entrant corners as visible from the core side of the shroud, within six inches of central flange and horizontal stiffeners. See Figures 4-12 and 4-14.
Core Shroud Assembly (Welded) Shroud plates	Plant designs with core shrouds assembled with full-height shroud plates	Cracking (IASCC) Aging Management (IE) (Note 2)	Remaining axial welds, Ribs and rings	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	Axial weld seams at the core shroud re-entrant corners, at the core mid-plane (± three feet in height) as visible from the core side of the shroud. See Figure 4-13.

Table 4-2
CE plants Primary components (continued)

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Shroud Assembly (Bolted) Assembly	Bolted plant designs	Distortion (Void Swelling), including: • Abnormal interaction with fuel assemblies • Gaps along high fluence shroud plate joints • Vertical displacement of shroud plates near high fluence joint Aging Management (IE)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	Core side surfaces as indicated. See Figures 4-25 and 4-26.
Core Shroud Assembly (Welded) Assembly	Plant designs with core shrouds assembled in two vertical sections	Distortion (Void Swelling), as evidenced by separation between the upper and lower core shroud segments Aging Management (IE)	None	Visual (VT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	If a gap exists, make three to five measurements of gap opening from the core side at the core shroud re-entrant corners. Then, evaluate the swelling on a plant-specific basis to determine frequency and method for additional examinations. See Figures 4-12 and 4-14.

Table 4-2
CE plants Primary components (continued)

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Support Barrel Assembly Upper (core support barrel) flange weld	All plants	Cracking (SCC)	Lower core support beams Core support barrel assembly upper cylinder Upper core barrel flange	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible surfaces of the upper flange weld (Note 4). See Figure 4-15.
Core Support Barrel Assembly Lower cylinder girth welds	All plants	Cracking (SCC, IASCC) Aging Management (IE)	Lower cylinder axial welds	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible surfaces of the lower cylinder welds (Note 4). See Figure 4-15
Lower Support Structure Core support column welds	All plants	Cracking (SCC, IASCC, Fatigue including damaged or fractured material) Aging Management (IE, TE)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible surfaces of the core support column welds (Note 5). See Figures 4-16 and 4-31

Table 4-2
CE plants Primary components (continued)

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Support Barrel Assembly Lower flange weld	All plants	Cracking (Fatigue)	None	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.	Examination coverage to be defined by evaluation to determine the potential location and extent of fatigue cracking. See Figures 4-15 and 4-16.
Lower Support Structure Core support plate	All plants with a core support plate	Cracking (Fatigue) Aging Management (IE)	None	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.	Examination coverage to be defined by evaluation to determine the potential location and extent of fatigue cracking. See Figure 4-16.
Upper Internals Assembly Fuel alignment plate	All plants with core shrouds assembled with full-height shroud plates	Cracking (Fatigue)	None	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.	Examination coverage to be defined by evaluation to determine the potential location and extent of fatigue cracking. See Figure 4-17.

Table 4-2
CE plants Primary components (continued)

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Control Element Assembly Instrument guide tubes	All plants with instrument guide tubes in the CEA shroud assembly	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports	Remaining instrument guide tubes within the CEA shroud assemblies	Visual (VT-3) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval. Plant-specific component integrity assessments may be required if degradation is detected and remedial action is needed.	100% of tubes in peripheral CEA shroud assemblies (i.e., those adjacent to the perimeter of the fuel alignment plate). See Figure 4-18.
Lower Support Structure Deep beams	All plants with core shrouds assembled with full-height shroud plates	Cracking (Fatigue) that results in a detectable surface-breaking indication in the welds or beams Aging Management (IE)	None	Enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval, if adequacy of remaining fatigue life cannot be demonstrated.	Examine beam-to-beam welds, in the axial elevation from the beam top surface to four inches below. See Figure 4-19.

Note to Table 4-2:

1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-2.
2. Void swelling effects on this component is managed through management of void swelling on the entire core shroud assembly.
3. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-2, must be examined for inspection credit.
4. A minimum of 75% of the total weld length (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-2, must be examined from either the inner or outer diameter for inspection credit.
5. A minimum of 75% of the total population of core support column welds.

Table 4-3
Westinghouse plants Primary components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Control Rod Guide Tube Assembly Guide plates (cards)	All plants	Loss of Material (Wear)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period, and no earlier than two refueling outages prior to the start of the license renewal period. Subsequent examinations are required on a ten-year interval.	20% examination of the number of CRGT assemblies, with all guide cards within each selected CRGT assembly examined. See Figure 4-20
Control Rod Guide Tube Assembly Lower flange welds	All plants	Cracking (SCC, Fatigue) Aging Management (IE and TE)	Bottom-mounted instrumentation (BMI) column bodies, Lower support column bodies (cast) Upper core plate Lower support forging/casting	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal on the individual periphery CRGT assemblies. (Note 2) See Figure 4-21.
Core Barrel Assembly Upper core barrel flange weld	All plants	Cracking (SCC)	Lower support column bodies (non cast) Core barrel outlet nozzle welds	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4). See Figure 4-22.
Core Barrel Assembly Upper and lower core barrel cylinder girth welds	All plants	Cracking (SCC, IASCC, Fatigue)	Upper and lower core barrel cylinder axial welds	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4). See Figure 4-22

Table 4-3
Westinghouse plants Primary components (continued)

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Barrel Assembly Lower core barrel flange weld (Note 5)	All plants	Cracking (SCC, Fatigue)	None	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4).
Baffle-Former Assembly Baffle-edge bolts	All plants with baffle-edge bolts	Cracking (IASCC, Fatigue) that results in • Lost or broken locking devices • Failed or missing bolts • Protrusion of bolt heads Aging Management (IE and ISR) (Note 6)	None	Visual (VT-3) examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Bolts and locking devices on high fluence seams. 100% of components accessible from core side (Note 3). See Figure 4-23.
Baffle-Former Assembly Baffle-former bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR) (Note 6)	Lower support column bolts, Barrel-former bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a ten-year interval.	100% of accessible bolts (Note 3). Heads accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. See Figures 4-23 and 4-24.

Table 4-3
Westinghouse plants Primary components (continued)

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Baffle-Former Assembly Assembly (Includes: Baffle plates, baffle edge bolts and indirect effects of void swelling in former plates)	All plants	Distortion (Void Swelling), or Cracking (IASCC) that results in • Abnormal interaction with fuel assemblies • Gaps along high fluence baffle joint • Vertical displacement of baffle plates near high fluence joint • Broken or damaged edge bolt locking systems along high fluence baffle joint	None	Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Core side surface as indicated. See Figures 4-24, 4-25, 4-26 and 4-27.
Alignment and Interfacing Components Internals hold down spring	All plants with 304 stainless steel hold down springs	Distortion (Loss of Load) Note: This mechanism was not strictly identified in the original list of age-related degradation mechanisms [7].	None	Direct measurement of spring height within three cycles of the beginning of the license renewal period. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years.	Measurements should be taken at several points around the circumference of the spring, with a statistically adequate number of measurements at each point to minimize uncertainty. See Figure 4-28.

Table 4-3
Westinghouse plants Primary components (continued)

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields	Cracking (Fatigue) or Loss of Material (Wear) that results in thermal shield flexures excessive wear, fracture, or complete separation	None	Visual (VT-3) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of thermal shield flexures. See Figures 4-29 and 4-36.

Notes to Table 4-3:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 5-3.
2. A minimum of 75% of the total identified sample population must be examined.
3. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined for inspection credit.
4. A minimum of 75% of the total weld length (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, must be examined from either the inner or outer diameter for inspection credit.
5. The lower core barrel flange weld may be alternatively designated as the core barrel-to-support plate weld in some Westinghouse plant designs.
6. Void swelling effects on this component is managed through management of void swelling on the entire baffle-former assembly.

Table 4-4
B&W plants Expansion components

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Upper Grid Assembly Alloy X-750 dowel-to-upper grid fuel assembly support pad welds	All plants (except DB)	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	Alloy X-750 dowel-to-guide block welds	Visual (VT-3) examination. Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of 100% of the dowel locking welds. See Figure 4-6 (i.e., these are similar to the lower grid fuel assembly support pads).
Core Barrel Assembly Upper thermal shield (UTS) bolts and their locking devices	All plants	Bolt or Stud/Nut: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UCB, LCB or FD bolts and their locking devices	Bolt or Stud/Nut: Volumetric examination (UT). Locking Devices: Visual (VT-3) examination Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	100% of accessible bolts or studs/nuts and their locking devices (Note 2). See Figure 4-7.
Core Barrel Assembly Surveillance specimen holder tube (SSHT) studs/nuts (CR-3) or bolts (DB) and their locking devices	CR-3, DB				
Core Barrel Assembly Core barrel cylinder (including vertical and circumferential seam welds) Former plates	All plants	Cracking (IE), including readily detectable cracking	Baffle plates	No examination requirements. Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.

Table 4-4
B&W plants Expansion components (continued)

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Barrel Assembly Baffle-to-baffle bolts Core barrel-to-former bolts	All plants	Cracking (IASCC, IE, Overload) (Note 3)	Baffle-to-former bolts	Internal baffle-to-baffle bolts: No examination requirements, Justify by evaluation or by replacement.	An acceptable examination technique currently not available. See Figure 4-2.
				External baffle-to-baffle bolts, core barrel-to-former bolts: No examination requirements. Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.
Core Barrel Assembly Locking devices, including locking welds, for the external baffle-to-baffle bolts and core barrel-to-former bolts	All plants	Cracking (IASCC, IE)	Locking devices, including locking welds, of baffle-to-former bolts or internal baffle-to-baffle bolts	No examination requirements. Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.
Lower Grid Assembly Lower grid fuel assembly support pad items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screw, and their locking welds (Note: the pads, dowels and cap screws are included because of IE of the welds)	All plants	Cracking (IE), including the detection of separated or missing welds, missing support pads, dowels, cap screws and locking welds, or misalignment of the support pads	IMI guide tube spiders and spider-to-lower grid rib section welds	Visual (VT-3) examination. Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of the pads, dowels, and cap screws, and associated welds in 100% of the lower grid fuel assembly support pads. See Figure 4-6.

Table 4-4
B&W plants Expansion components (continued)

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Lower Grid Assembly Alloy X-750 dowel-to-lower grid fuel assembly support pad welds	All plants	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	Alloy X-750 dowel-to-guide block welds	Visual (VT-3) examination. Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of 100% of the support pad dowel locking welds. See Figure 4-6.
Lower Grid Assembly Lower grid shock pad bolts and their locking devices	TMI-1	Bolts: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UCB, LCB or FD bolts and their locking devices	Bolt: Volumetric examination (UT). Locking Devices: Visual (VT-3) examination. Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	100% of accessible bolts and their locking devices. (Note 2) See Figure 4-4.
Lower Grid Assembly Lower thermal shield (LTS) bolts (ANO-1, DB and TMI-1) or studs/nuts (ONS, CR-3) and their locking devices	All plants	Bolts or Studs/Nuts: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UCB,LCB or FD bolts and their locking devices	Bolt or Stud/Nut: Volumetric examination (UT). Locking Devices: Visual (VT-3) examination. Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	100% of accessible bolts and their locking devices. (Note 2) See Figure 4-8.

Notes to Table 4-4:

1. Examination acceptance criteria and expansion criteria for the B&W components are in Table 5-1.
2. A minimum of 75% of the total population (examined + unexamined) must be examined for inspection credit.
3. The primary aging degradation mechanisms for loss of joint tightness for these items are IC and ISR. Fatigue and Wear, which can also lead to cracking, are secondary aging degradation mechanisms after significant stress relaxation and loss of preload has occurred due to IC/ISR. Bolt stress relaxation cannot readily be inspected by NDE. Only bolt cracking could be inspected by UT inspection if it were possible for these bolts. Therefore, the effects of loss of joint tightness and/or cracking on the functionality of these bolts relative to the entire core barrel assembly will be addressed by analysis of the core barrel assembly, which will be performed to address Applicant/Licensee action item 6 of the SE [27].

Table 4-5
CE plants Expansion components

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Shroud Assembly (Bolted) Barrel-shroud bolts	Bolted plant designs	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	Core shroud bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100% (or as supported by plant-specific justification; Note 2) of barrel-shroud and guide lug insert bolts with neutron fluence exposures > 3 displacements per atom (dpa). See Figure 4-23.
Core Support Barrel Assembly Lower core barrel flange	All plants	Cracking (SCC, Fatigue)	Upper (core support barrel) flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible welds and adjacent base metal (Note 2). See Figure 4-15.
Core Support Barrel Assembly Upper cylinder (including welds)	All plants	Cracking (SCC) Aging Management (IE)	Upper (core support barrel) flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces of the welds and adjacent base metal (Note 2). See Figure 4-15.
Core Support Barrel Assembly Upper core barrel flange	All plants	Cracking (SCC)	Upper (core support barrel) flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible bottom surface of the flange (Note 2). See Figure 4-15.

Table 4-5
CE plants Expansion components (continued)

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Support Barrel Assembly Core barrel assembly axial welds	All plants	Cracking (SCC)	Core barrel assembly girth welds	Enhanced visual (EVT-1) examination, with initial and subsequent examinations dependent on the results of core barrel assembly girth weld examinations.	100% of one side of the accessible weld and adjacent base metal surfaces for the weld with the highest calculated operating stress. See Figure 4-15.
Lower Support Structure Lower core support beams	All plants except those with core shrouds assembled with full-height shroud plates	Cracking (SCC, Fatigue) including damaged or fractured material	Upper (core support barrel) flange weld	Visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2). See Figures 4-16 and 4-31.
Core Shroud Assembly (Bolted) Core support column bolts	Bolted plant designs	Cracking (IASCC, Fatigue) Aging Management (IE)	Core shroud bolts	Ultrasonic (UT) examination. Re-inspection every 10 years following initial inspection.	100% (or as supported by plant-specific analysis) of core support column bolts with neutron fluence exposures > 3 dpa (Note 2). See Figures 4-16 and 4-33.

Table 4-5
CE plants Expansion components (continued)

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Shroud Assembly (Welded) Remaining axial welds, Ribs and rings	Plant designs with core shrouds assembled with full-height shroud plates	Cracking (IASCC) Aging Management (IE)	Shroud plates of welded core shroud assemblies	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	Axial weld seams other than the core shroud re-entrant corner welds at the core mid-plane, plus ribs and rings. See Figure 4-13.
Control Element Assembly Remaining instrument guide tubes	All plants with instrument guide tubes in the CEA shroud assembly	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports.	Peripheral instrument guide tubes within the CEA shroud assemblies	Visual (VT-3) examination. Re-inspection every 10 years following initial inspection.	100% of tubes in CEA shroud assemblies (Note 2). See Figure 4-18.

Notes to Table 4-5:

1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-2.
2. A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).

Table 4-6
Westinghouse plants Expansion components

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Upper Internals Assembly Upper core plate	All plants	Cracking (Fatigue, Wear)	CRGT lower flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2).
Lower Internals Assembly Lower support forging or castings	All plants	Cracking Aging Management (TE in Casting)	CRGT lower flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2). See Figure 4-33.
Core Barrel Assembly Barrel-former bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE, Void Swelling and ISR)	Baffle-former bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100% of accessible bolts. Accessibility may be limited by presence of thermal shields or neutron pads (Note 2). See Figure 4-23.
Lower Support Assembly Lower support column bolts	All plants	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	Baffle-former bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100% of accessible bolts or as supported by plant-specific justification (Note 2). See Figures 4-32 and 4-33.

Table 4-6
Westinghouse plants Expansion components (continued)

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Barrel Assembly Core barrel outlet nozzle welds	All plants	Cracking (SCC, Fatigue) Aging Management (IE of lower sections)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 2). See Figure 4-22.
Core Barrel Assembly Upper and lower core barrel cylinder axial welds	All plants	Cracking (SCC, IASCC) Aging Management (IE)	Upper and lower core barrel cylinder girth welds	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 2). See Figure 4-22.
Lower Support Assembly Lower support column bodies (non cast)	All plants	Cracking (IASCC) Aging Management (IE)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2). See Figure 4-34.
Lower Support Assembly Lower support column bodies (cast)	All plants	Cracking (IASCC) including the detection of fractured support columns Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible support columns (Note 2). See Figure 4-34.

Table 4-6
Westinghouse plants Expansion components (continued)

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Bottom Mounted Instrumentation System Bottom-mounted instrumentation (BMI) column bodies	All plants	Cracking (Fatigue) including the detection of completely fractured column bodies Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (VT-3) examination of BMI column bodies as indicated by difficulty of insertion/withdrawal of flux thimbles. Re-inspection every 10 years following initial inspection. Flux thimble insertion/withdrawal to be monitored at each inspection interval.	100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal. See Figure 4-35.

Notes to Table 4-6:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 5-3.
2. A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).

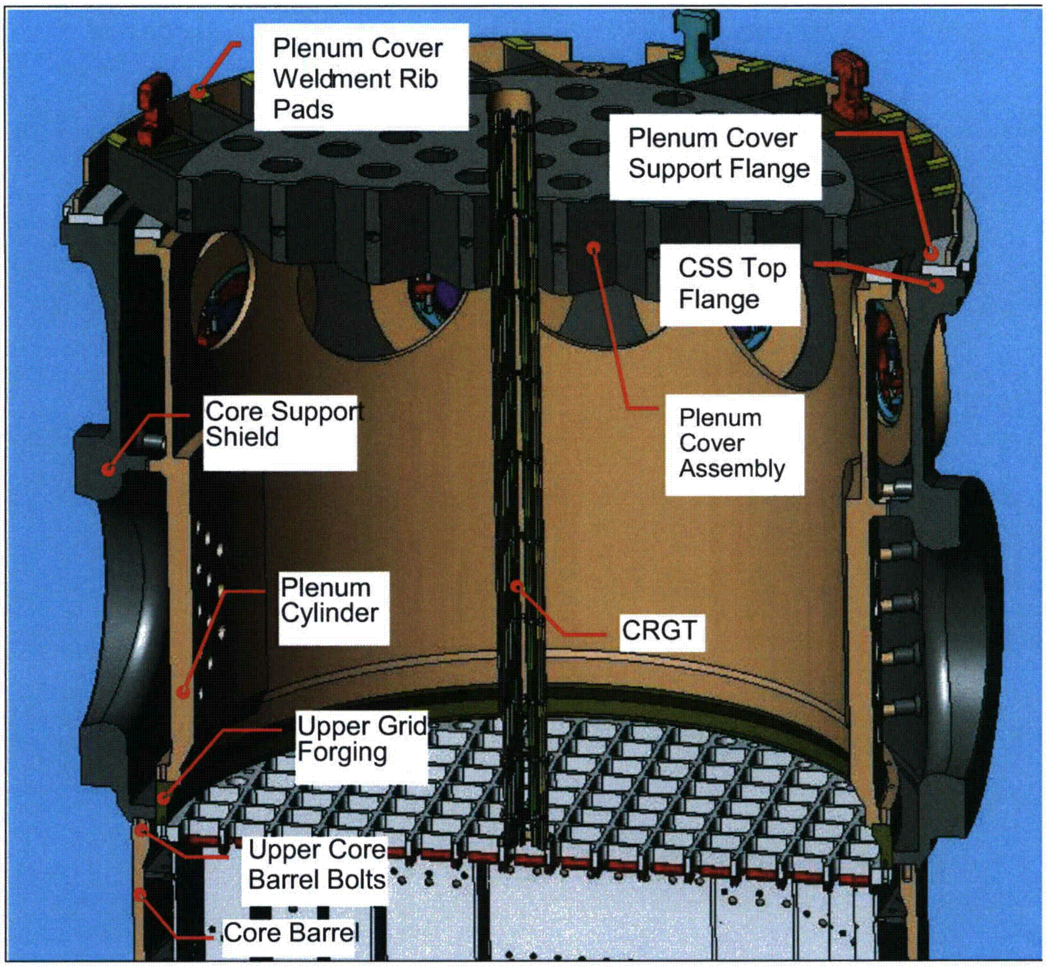

Figure 4-1
Typical upper internals arrangement for B&W-designed PWRs

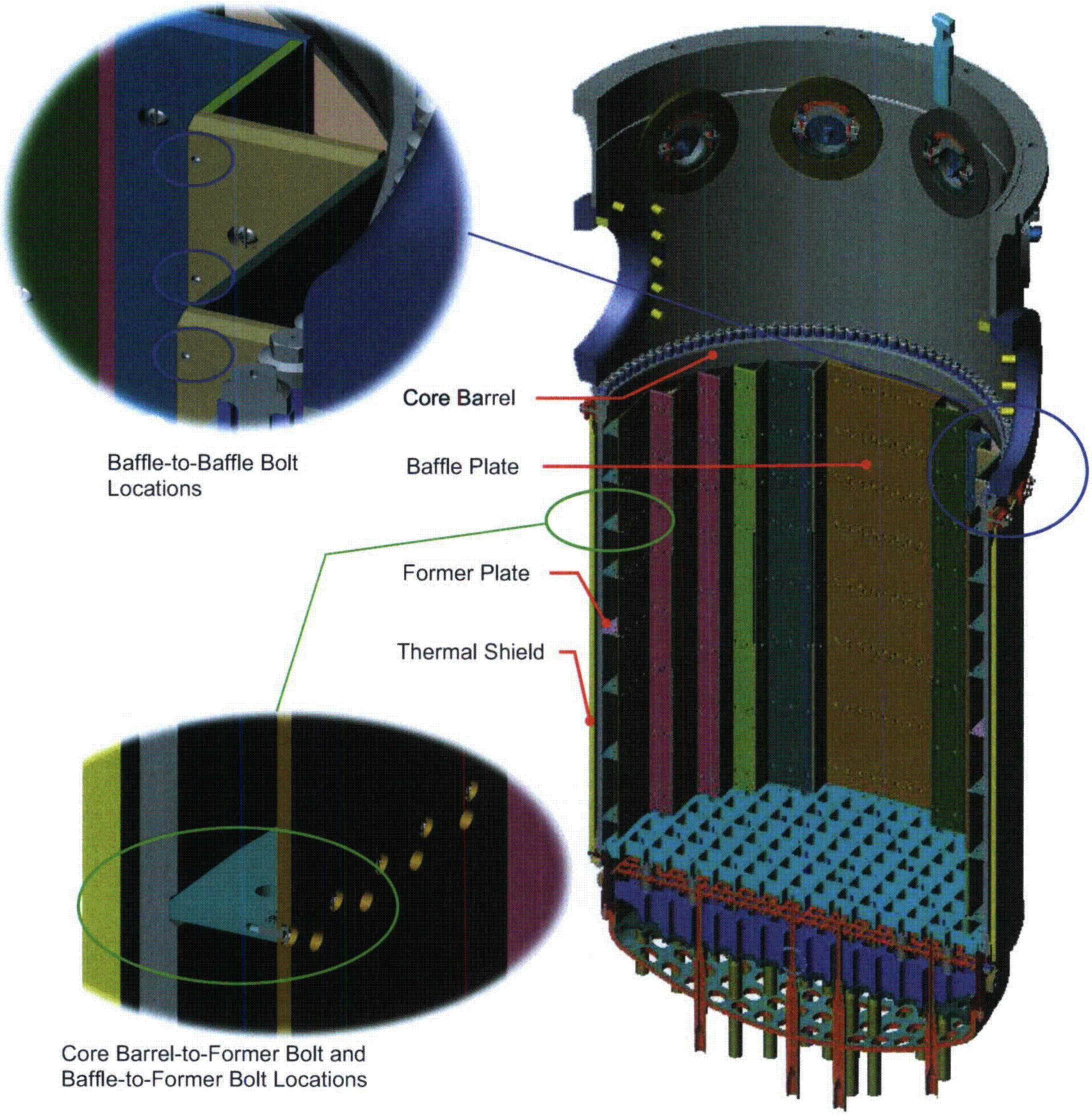

Figure 4-2
Typical internals core barrel assembly for B&W-designed PWRs

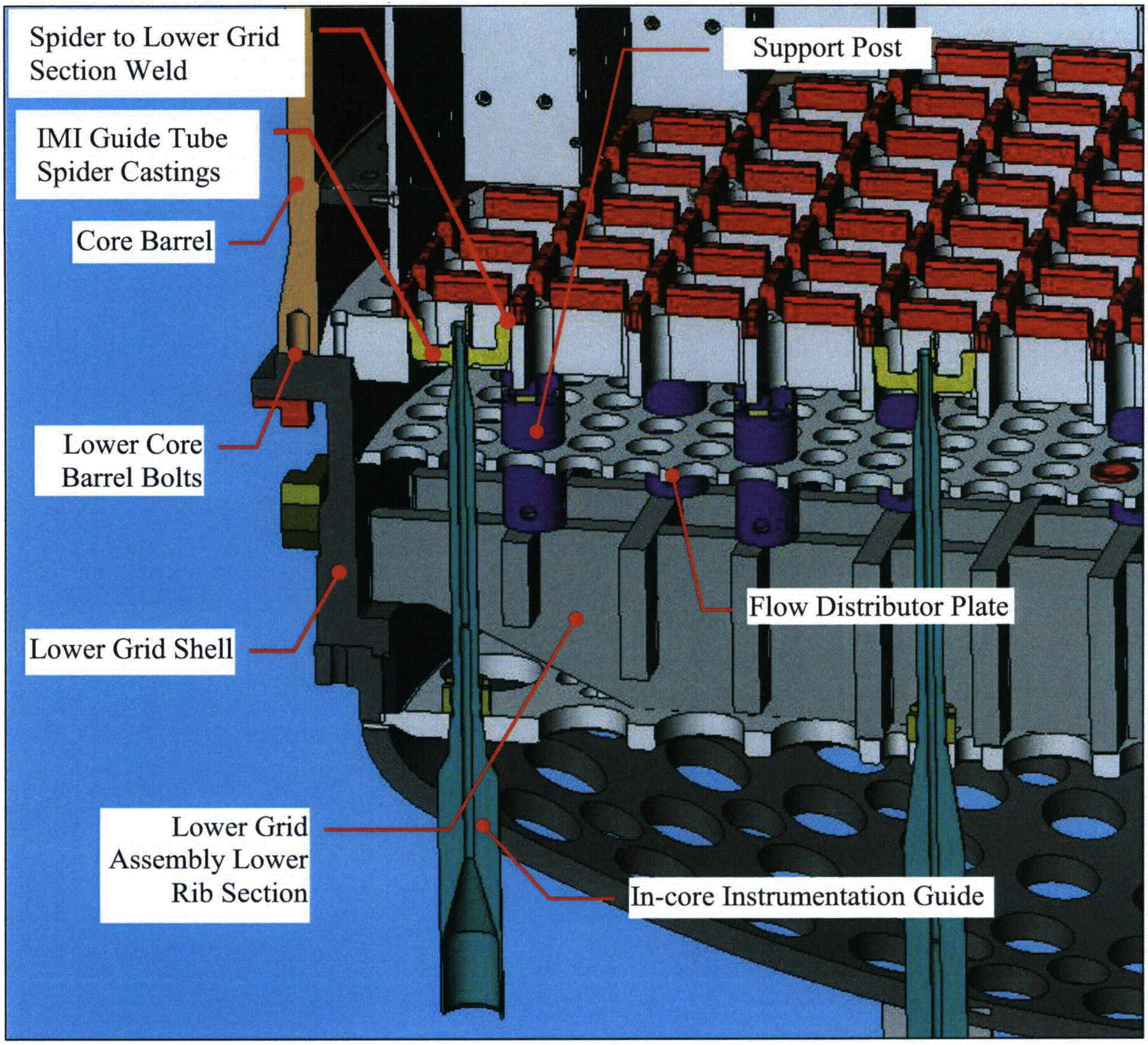

Figure 4-3
Typical lower internals arrangement for B&W-designed PWRs

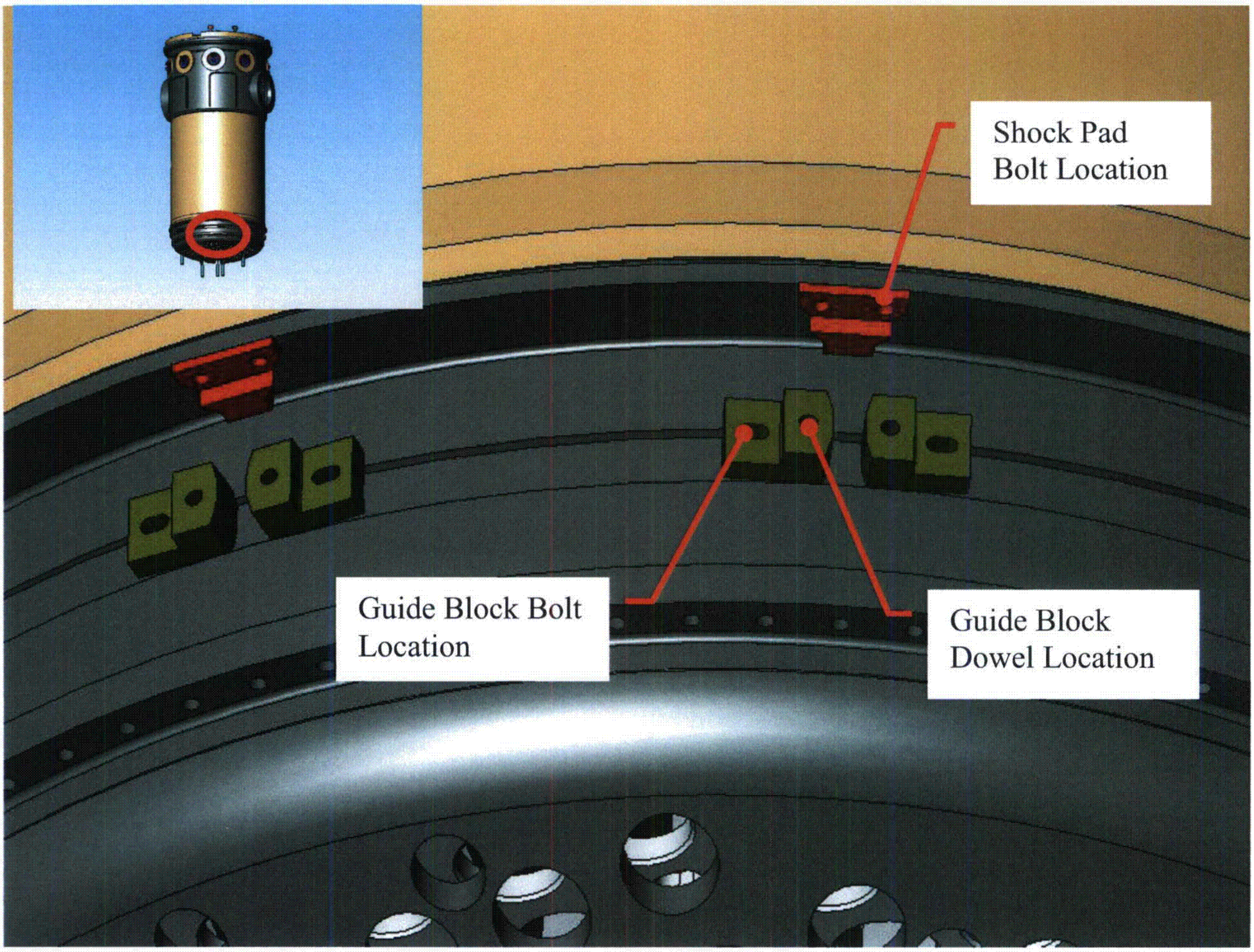

Figure 4-4
Typical guide block and shock pad locations for B&W-designed PWRs

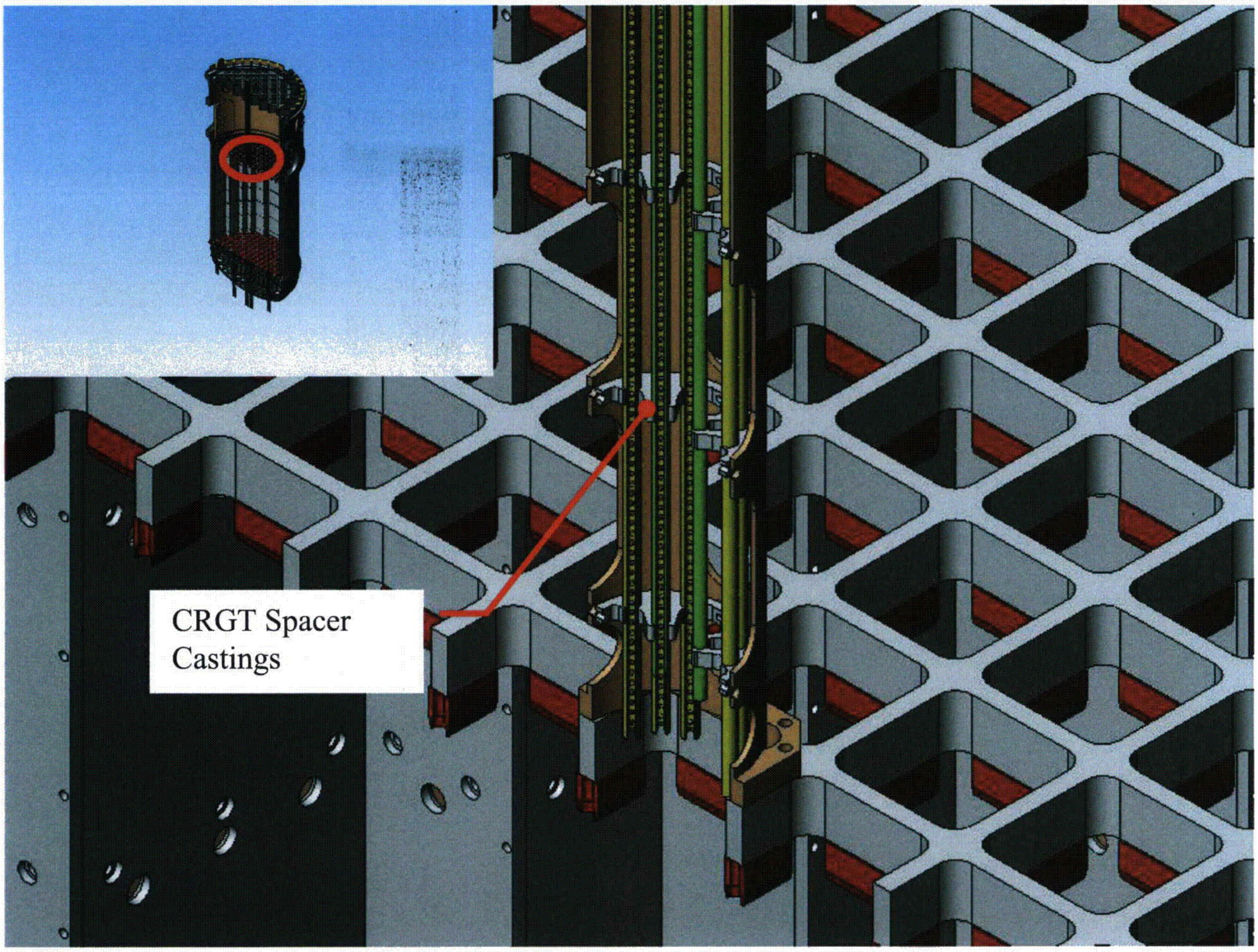

Figure 4-5
Typical control rod guide tube (CRGT) for B&W-designed PWRs (one of 69 CRGTs shown)

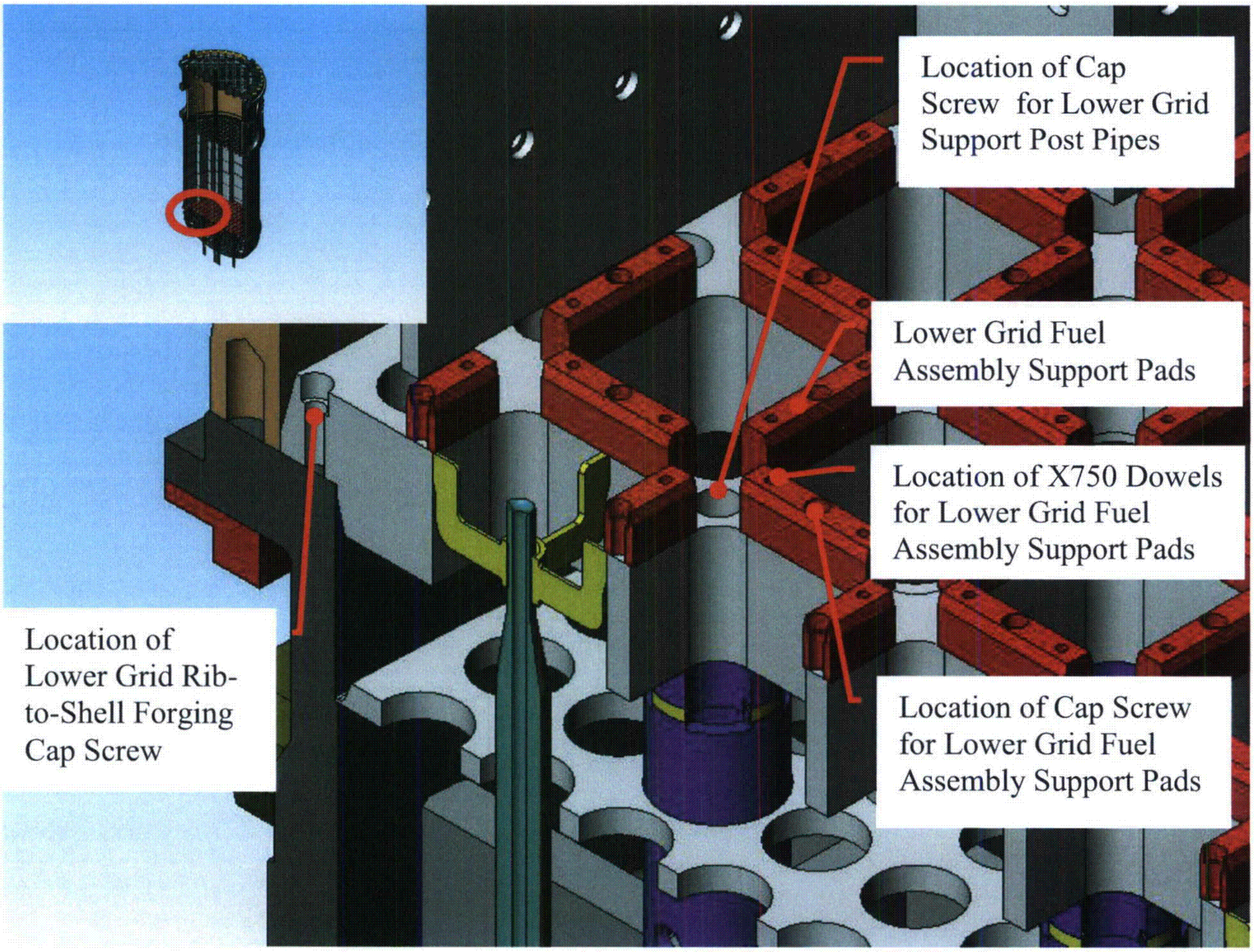

Figure 4-6
Typical lower grid assembly and fuel assembly support pads for B&W-designed PWRs

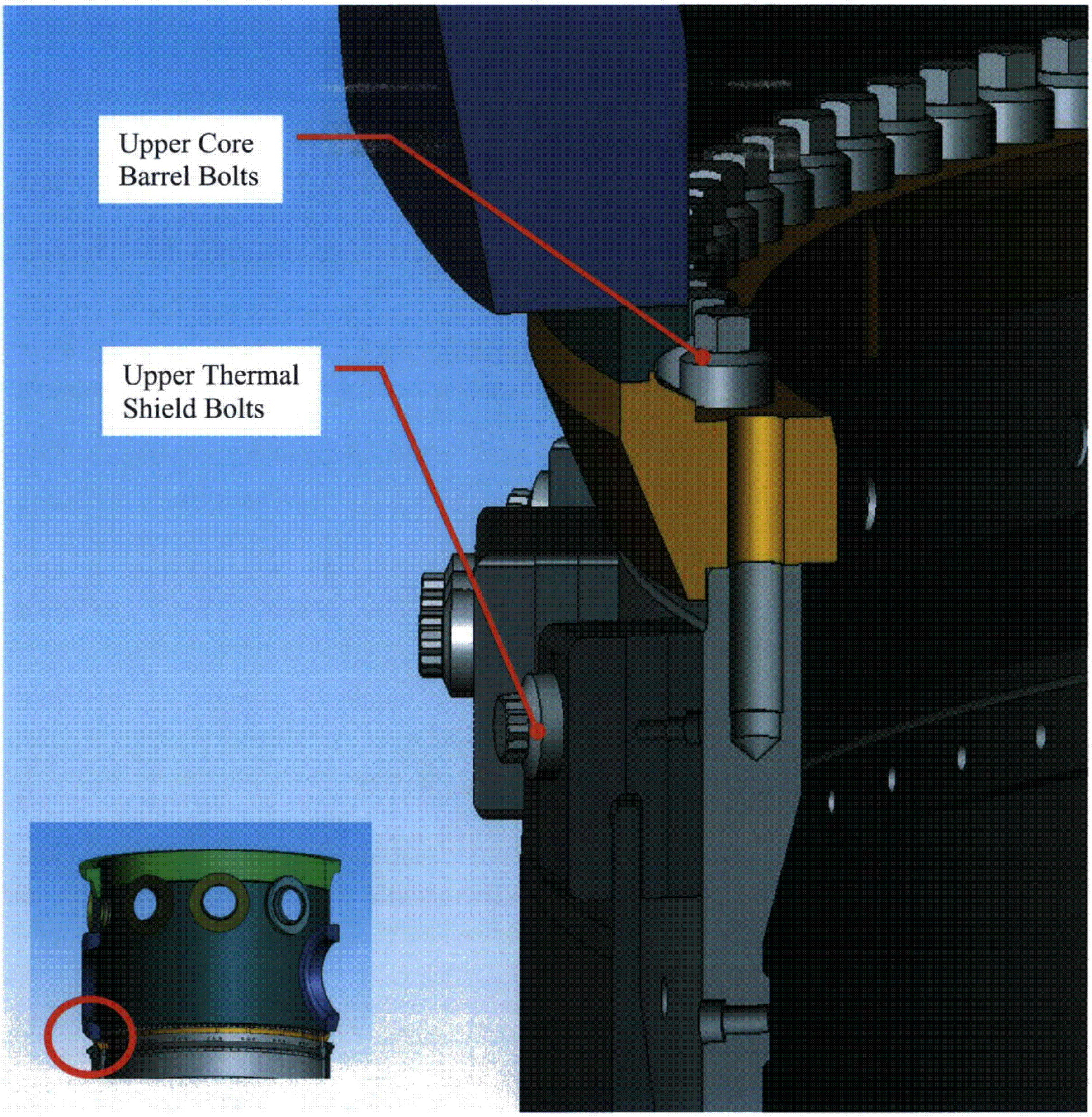

Figure 4-7
Typical upper thermal shield bolts and upper core barrel bolts for B&W-designed PWRs

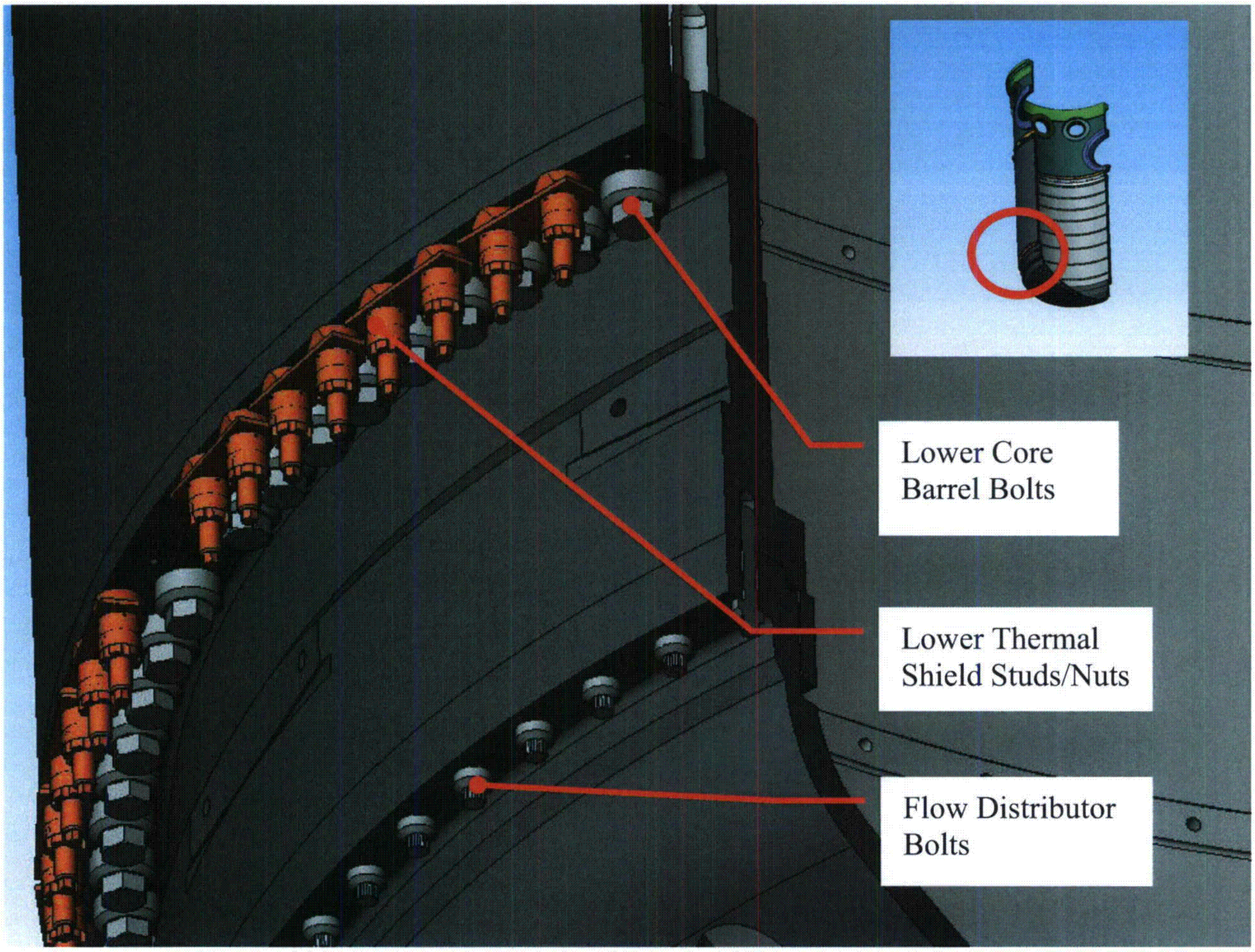

Figure 4-8
Typical lower thermal shield studs/nuts, lower core barrel bolts, and flow distributor bolts for the B&W-designed PWRs

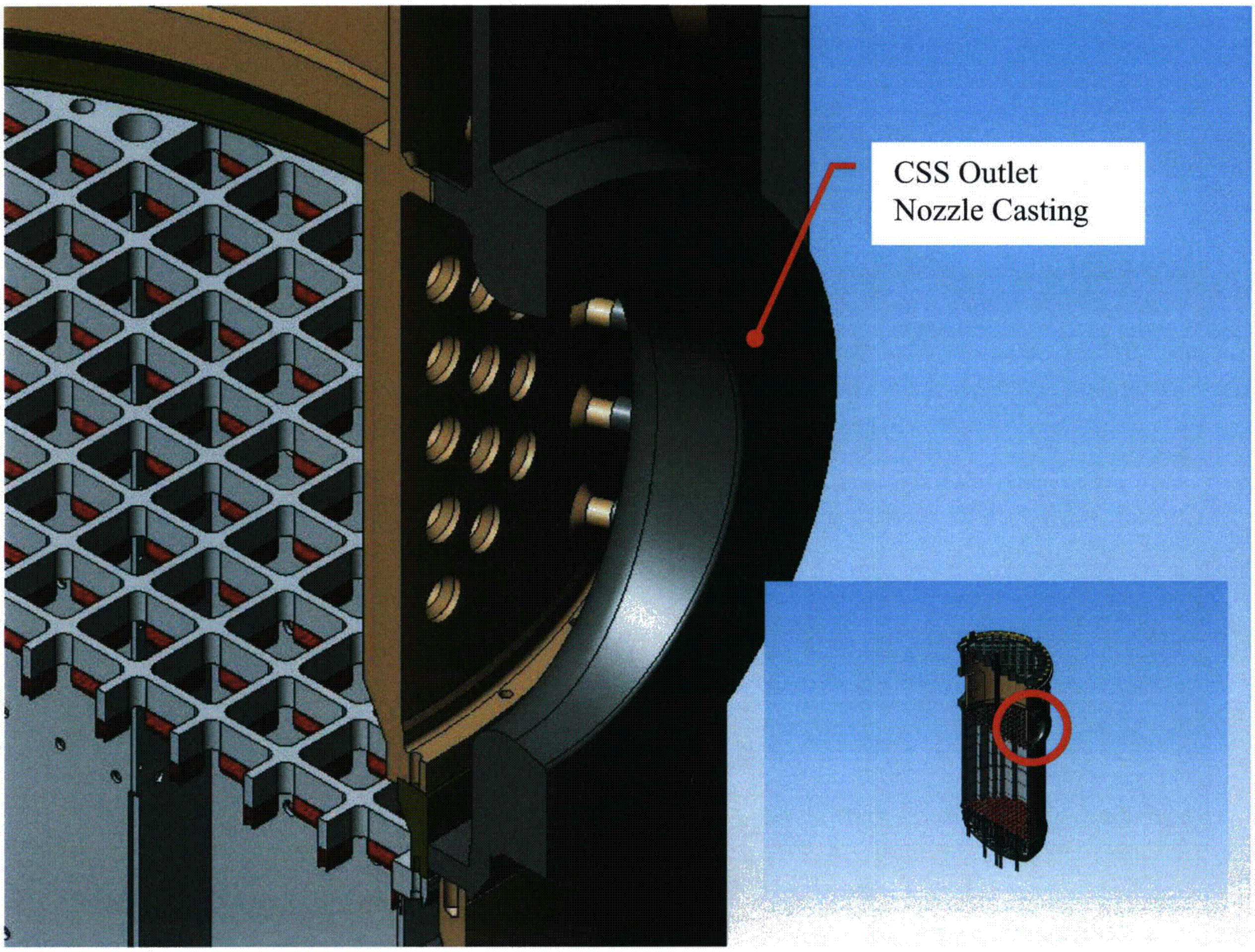

Figure 4-9
Typical core support shield (CSS) outlet nozzle for the B&W-designed PWRs

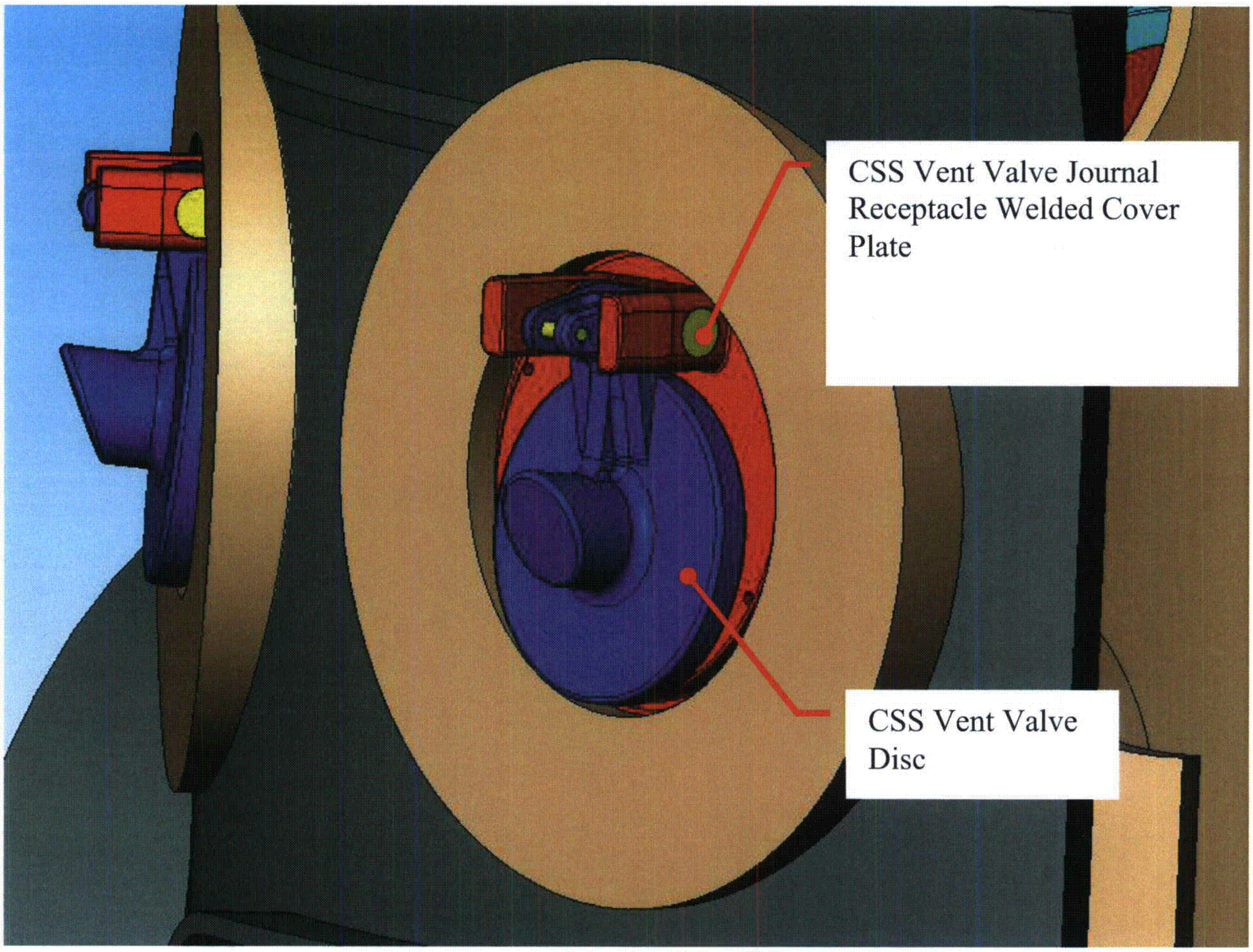

Figure 4-10
Typical core support shield (CSS) vent valve – outside view – for the B&W-designed PWRs

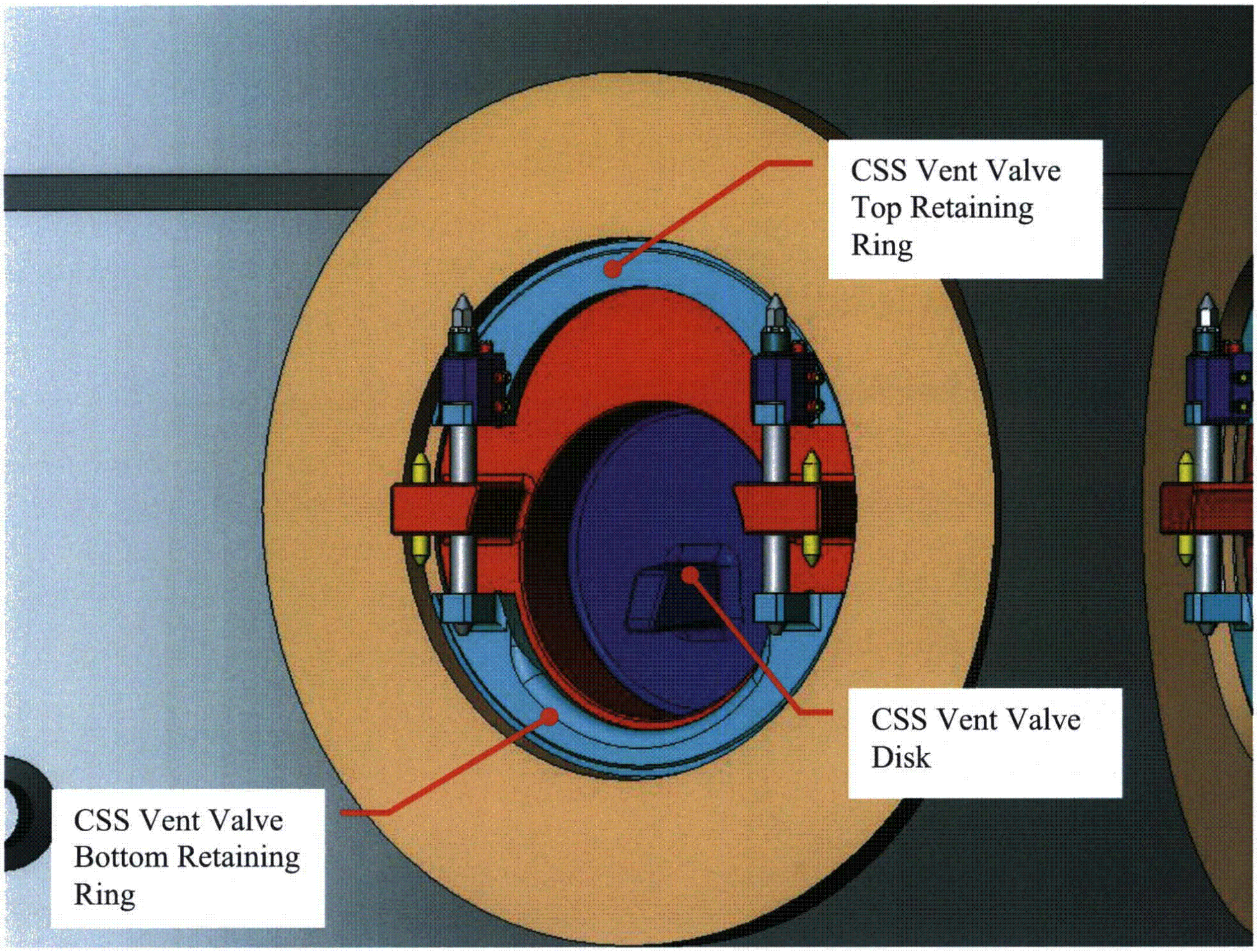

Figure 4-11
Typical core support shield (CSS) vent valve – inside view – for the B&W-designed PWRs