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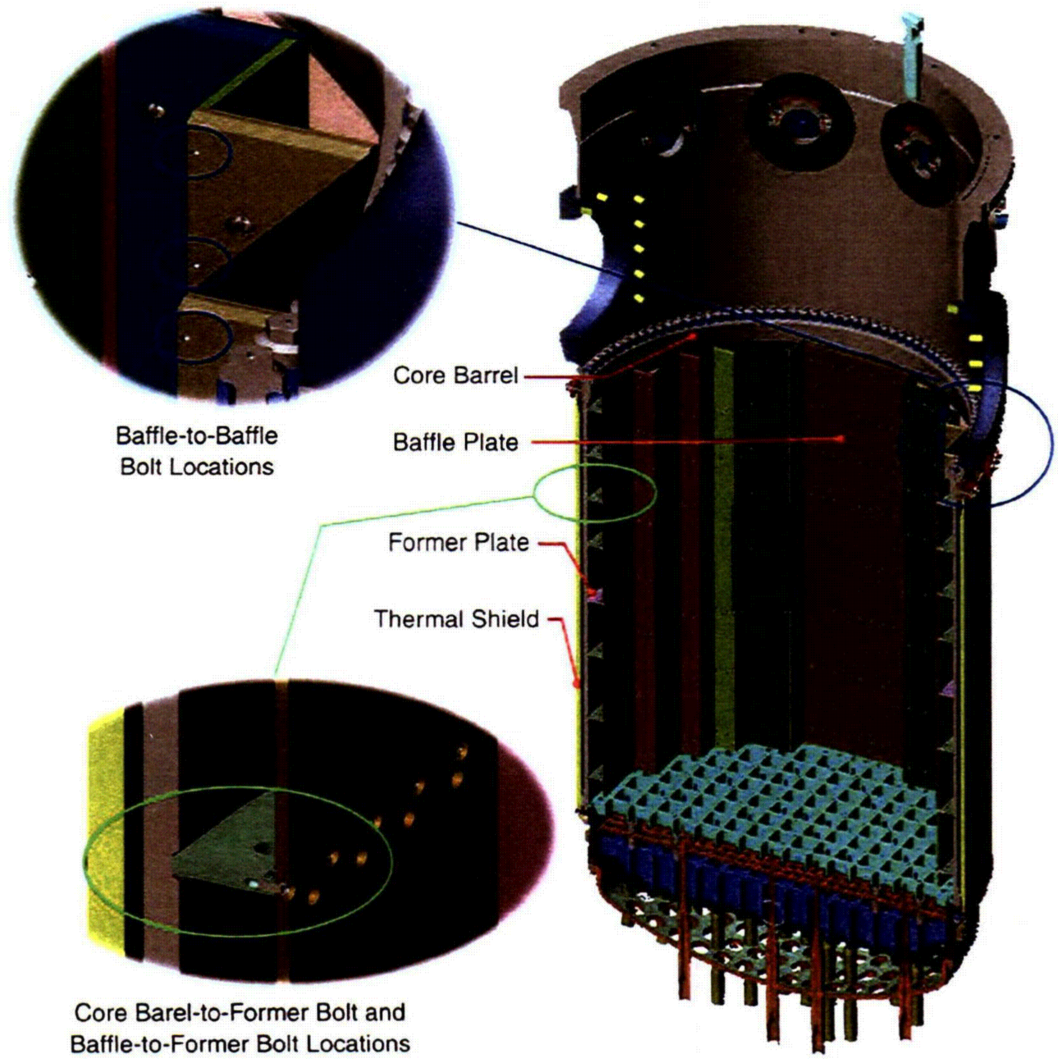
Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)



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REPORT SUMMARY

The Materials Reliability Program (MRP) developed inspection and evaluation (I&E) guidelines for managing long-term aging of pressurized water reactor (PWR) reactor internals. Specifically, the guidelines are applicable to reactor internal structural components; they do not address fuel assemblies, reactivity control assemblies, or welded attachments to the reactor vessel.

Background

Demonstrating that effects of aging degradation in PWR internals are adequately managed is essential for maintaining a healthy fleet and assuring continued functionality of reactor internals. As a work product of the MRP, these I&E guidelines are intended to support that demonstration, with requirements for inspections to detect effects of aging degradation. The program to develop these guidelines has been underway for almost a decade, organized around a framework and strategy for managing effects of aging in PWR internals, dependent on a substantial database of material data and supporting evaluation results. The goal of this development was primarily to support license renewal, but the guidelines are intended to apply to the current license period as well.

Objectives

To provide generic I&E guidelines for each PWR design for use by individual plant owners in preparing and executing their PWR internals aging management programs (AMPs).

Approach

An experienced team consisting of utility and nuclear steam supply system (NSSS) vendors and EPRI experts, representing a broad spectrum of reactor design, operations, and materials expertise, worked on the project. The team reviewed available data and industry experience on materials aging to develop a systematic approach for identifying and prioritizing inspection requirements for internals. The key sequential steps in the process included the following:

- development of screening criteria, with susceptibility levels for the eight postulated aging mechanisms relevant to reactor internals and their effects;
- initial component screening and categorization, using susceptibility levels and FMECA (failure modes, effects, and criticality assessment) to identify the relative ranking of components;
- functionality assessment of degradation for components and assemblies of components; and
- aging management strategy development combining results of the functionality assessment with component accessibility, operating experience, existing evaluations, and prior examination results to determine the appropriate aging management methodology, baseline examination timing, and the need for and the timing of subsequent inspections.

Through this process, reactor internals for all three PWR designs were evaluated, and appropriate recommendations for aging management actions specific to each component were provided.

Results

One "mandatory," three "needed," and one "good practice" implementation requirements have been developed. These requirements provide the framework and details for individual utility reactor internals AMPs.

EPRI Perspective

The guidelines are based on a broad set of assumptions about plant operation, which encompass the range of current plant conditions for the U.S. fleet of PWRs. The aging management strategies reports (MRP-231 and MRP-232) provide the basis for these guidelines. The functional evaluations that support the guidelines were based on representative configurations and operational histories, which were generally conservative, but not necessarily bounding in every parameter. These assumptions are a conservative representation of U.S. PWR operating plants, all of which implemented low-leakage core-loading patterns early in their operating life. The recommendations are, thus, applicable to all U.S. PWR operating plants as of May 2007 for the three designs identified. These guidelines also are considered applicable to plants that have replaced components or component assemblies; however, alternatives can be technically justified.

The Inspection Standard for PWR internals (MRP-228) is the companion document to these I&E guidelines and provides examination requirement standards for components listed in the guidelines.

Keywords

Pressurized water reactor
Reactor internals
Inspection guidelines
Aging management
License renewal
Material reliability program

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LIST OF ACRONYMS

AMP	Aging Management Program
ASME	American Society of Mechanical Engineers
B&PV	Boiler & Pressure Vessel
B&W	Babcock & Wilcox
BB	Baffle-to-Baffle
BMI	Bottom Mounted Instrumentation
BWR	Boiling Water Reactor
BWRVIP	Boiling Water Reactor Vessel & Internals Project
CAP	Corrective Action Program
CASS	Cast Austenitic Stainless Steel
CB	Core Barrel
CBF	Core Barrel-to-Former
CE	Combustion Engineering
CEA	Control Element Assembly
CFR	Code of Federal Regulations
CR-3	Crystal River Unit 3
CRGT	Control Rod Guide Tube
CSA	Core Support Assembly
CSS	Core Support Shield
DB	Davis-Besse
E	Expansion, I&E Guidelines Component Group
ECP	Electro-Chemical Potential
EFPY	Effective Full Power Years
EPFM	Elastic-Plastic Fracture Mechanics
EPRI	Electric Power Research Institute
ET	Electromagnetic Testing (Eddy Current)
EVT	Enhanced Visual Testing (a Visual NDE Method that includes EVT-1)

FB	Baffle-to-Former
FD	Flow Distributor
FMECA	Failure Mode, Effects, and Criticality Analysis
GALL	Generic Aging Lessons Learned
HWC	Hydrogen Water Chemistry
I&E	Inspection and Evaluation
IASCC	Irradiation-Assisted Stress Corrosion Cracking
ICI	In-Core Instrumentation
IGSCC	Intergranular SCC
IMI	Incore Monitoring Instrumentation
IP	Issue Program
ISI	Inservice Inspection
ISR	Irradiation-Enhanced Stress Relaxation
ITG	Issue Task Group
JOBB	Joint Owners Baffle Bolt
LCB	Lower Core Barrel
LCP	Lower Core Plate
LEFM	Linear Elastic Fracture Mechanics
LTS	Lower Thermal Shield
LOCA	Loss-of-Coolant-Accident
MRP	Materials Reliability Program
N	No Additional Measures, I&E Guidelines Component Group
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NRC	U. S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
ONS	Oconee Nuclear Station (ONS-1, ONS-2, and ONS-3)
P	Primary, I&E Guidelines Component Group
PH	Precipitation-Hardenable (Heat Treatment)
PMMP	Preventive Maintenance Management Program
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group

PWSCC	Primary Water SCC
QA	Quality Assurance
RCS	Reactor Coolant System
RI-FG	Reactor Internals Focus Group
RI-ITG	Reactor Internals Issue Task Group
SCC	Stress Corrosion Cracking
SS	Stainless Steel
SSE	Safe Shutdown Earthquake
SSHT	Surveillance Specimen Holder Tube
TLAA	Time-Limited Aging Analysis
TMI-1	Three Mile Island Unit 1
UCB	Upper Core Barrel
UCP	Upper Core Plate
USP	Upper Support Plate
UT	Ultrasonic Testing (a Volumetric NDE Method)
UTS	Upper Thermal Shield
VT	Visual Testing (a Visual NDE Method that Includes VT-1 and VT-3)
X	Existing, I&E Guidelines Component Group
XL	Extra-Long Westinghouse Fuel

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1 EXECUTIVE SUMMARY

Demonstration that the effects of aging degradation in pressurized water reactor (PWR) internals are adequately managed is essential for maintaining a healthy fleet and assuring continued functionality of the reactor internals. As a work product of the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) Reactor Internals Focus Group (RI-FG), these Inspection & Evaluation (I&E) guidelines are intended to support that demonstration, with requirements for inspection to detect the effects of aging degradation. These guidelines are provided to individual plant owners for use in preparing and executing their PWR internals aging management programs (AMPs). These guidelines contain **Mandatory, Needed,** and **Good Practice** requirements that must be implemented per the Materials Initiative [1]. Section 7 describes all of the requirements of the guidelines, including an implementation schedule based on NRC approval. The requirements contained in this document are applicable to Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse Nuclear Steam Supply System (NSSS) PWR designs currently operating in the United States.

These guidelines do not reduce, alter, or otherwise affect current American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI [2] or plant-specific licensing inservice inspection requirements.

The program to develop these guidelines has been underway for almost a decade, organized around a framework and strategy for managing the effects of aging in PWR internals, dependent on a substantial database of material data and supporting evaluation results. The key sequential steps included the following:

- development of screening criteria, with susceptibility levels for the eight postulated aging mechanisms relevant to reactor internals and their effects;
- initial component screening and categorization, using the susceptibility levels to identify the relative susceptibility of the components;
- functionality assessment of degradation for components and assemblies of components;
- aging management strategy development combining the results of functionality assessment with component accessibility, operating experience, existing evaluations, and prior examination results to determine the appropriate aging management methodology, baseline examination timing, and the need for and the timing of subsequent inspections.

Through this process, the reactor internals for all three PWR designs were assigned to one of the following four groups: Primary, Expansion, Existing Programs, and No Additional Measures components. Definitions and recommendations for aging management actions specific to each group are provided in Sections 3 and 4.

The aging management elements needed for Primary and Expansion components were selected from existing, well-proven visual, surface, and volumetric examination methodologies that have been subject to widespread, relevant application. Each component in the Primary and Expansion groups was then assessed in terms of the degradation effect (e.g., cracking caused by particular mechanisms, loss of material caused by wear), appropriate examination methodology for detection of that effect, accessibility of that component for the examination method selected, and industry experience with those examinations. The Inspection Standard for PWR internals (MRP-228) [3] is the companion document to these I&E guidelines and provides the examination requirement standards for the components listed herein.

The Primary components requirements are listed in Tables 4-1, 4-2, and 4-3 of Section 4 for the Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse designs, respectively. The Expansion components requirements are listed in Tables 4-4, 4-5, and 4-6 for the B&W, CE, and Westinghouse designs, respectively. These tables provide the assembly/sub-assembly/component description, the relevant degradation effect and associated degradation mechanism, any link between a Primary component and a related Expansion component, the examination method, and examination coverage.

The Existing Programs components requirements are listed in Tables 4-7, 4-8, and 4-9 for the B&W, CE, and Westinghouse designs, respectively. These tables and the supporting text identify the components and the references to the existing programs.

Tables are not provided for the No Additional Measures components. This group of components has been determined to need no additional aging management. However, for those components in the No Additional Measures group that are classified as core support structures in plant-specific documentation, the inservice inspection requirements of the ASME Code Section XI, Subsection IWB, Examination Category B-N-3 [2] must continue to be met, unless specific relief is granted as allowed by Title 10 Part 50.55a [4] of the Code of Federal Regulations (10CFR50.55a) or plant-specific licensing documentation.

The examination acceptance criteria and the expansion criteria for the primary/expansion links are described in Tables 5-1, 5-2, and 5-3 for the B&W, CE, and Westinghouse designs, respectively. These examination acceptance criteria include visual examination relevant conditions that require disposition by additional examinations, engineering evaluation, or repair/replacement.

Section 6 is for information and contains various options that are available for the disposition of conditions detected during examinations (Section 4) that are unable to satisfy the examination acceptance criteria (Section 5).

2 INTRODUCTION

2.1 Background

This document provides inspection and evaluation (I&E) guidelines for use by the industry in developing an aging management program (AMP) for Pressurized Water Reactor (PWR) internals. The elements of an AMP are defined in Appendix A, where the extent to which each element is met by these guidelines is discussed. The goal is to ensure the long-term safety, integrity, and reliability of PWR internals using proven and familiar methods for inspection, monitoring, surveillance, and reporting. The guidelines are based on work performed over the past decade by the commercial nuclear power industry, first through the Joint Owners Baffle Bolt (JOBB) Program, then through the EPRI Materials Reliability Program (MRP) Reactor Internals Issue Task Group (RI-ITG) and, later, by the MRP Reactor Internals Focus Group (RI-FG). This program is organized around a framework and strategy [5] for managing the effects of aging in PWR internals, together with a substantial database of material data and supporting results (e.g., see [6]). The key steps in the framework and strategy process are shown in the flowchart of Figure 2-1.

Based upon the framework and strategy, and on the accumulated data, three important precursor elements to these I&E guidelines were then developed:

- screening criteria, considering chemical composition, neutron fluence exposure, temperature history, and representative stress levels, for determining the relative susceptibility of PWR internals to the eight postulated aging mechanisms [7] – stress corrosion cracking (SCC), irradiation-assisted stress corrosion cracking (IASCC), wear, fatigue, thermal aging embrittlement, irradiation embrittlement, irradiation-enhanced stress relaxation and creep, and void swelling;
- categorization of PWR internals, based on the screening criteria and the likelihood and severity of safety and economic consequences, into categories that range from those components for which these issues are insignificant (Category A) to those components that are potentially moderately significant (Category B) to those components that are potentially significantly affected (Category C) [8, 9, and 10]; and
- functionality assessment of components and assemblies of components based on representative plant designs using irradiated and aged material properties to determine the effects of the degradation mechanisms on functionality [11 and 12].

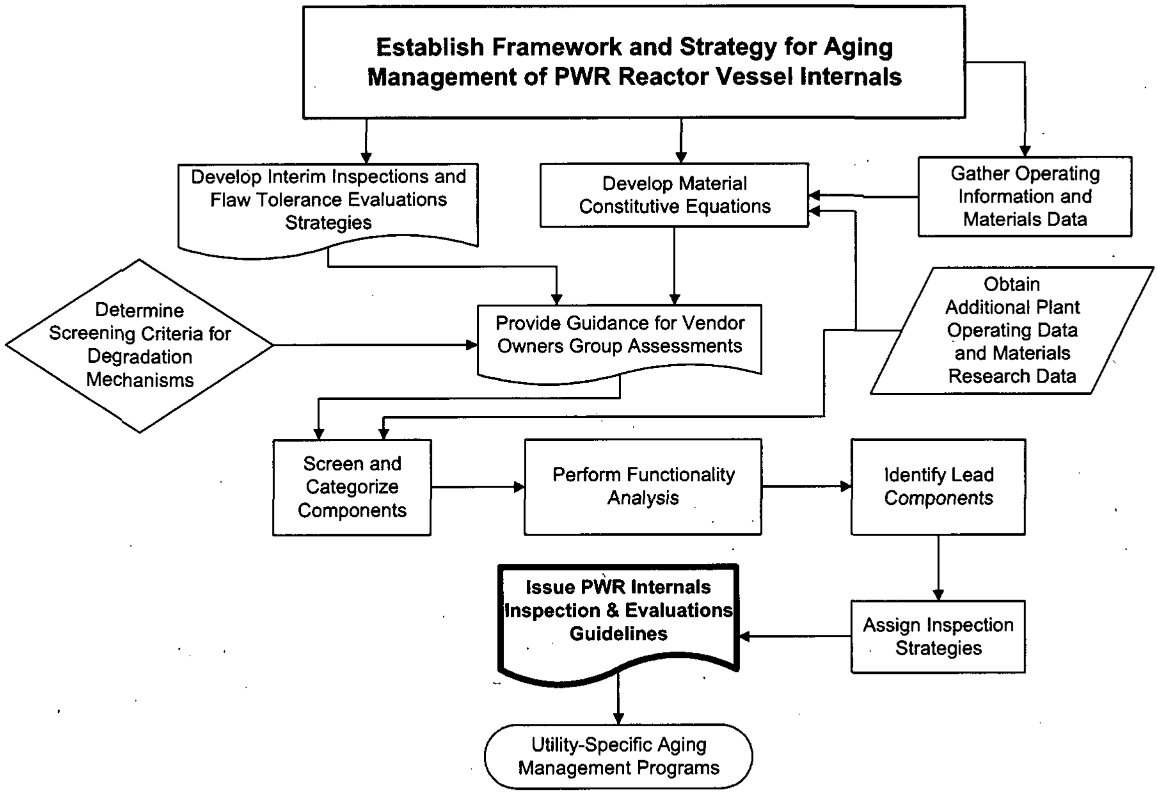


Figure 2-1
MRP framework and strategy for aging management of PWR internals

2.2 Aging Management Strategy Development

The aging management strategy development combined the results of functionality assessment with component accessibility, operating experience, existing evaluations, and prior examination results to determine the appropriate methodologies for maintaining the long-term functions of PWR internals safely and economically [13 and 14]. This process permitted further categorization of PWR internals into functional groups. Figure 2-2 shows the links between the categorization based on screening criteria, the functionality analysis, the aging management strategy development, and the I&E guidelines. The ultimate result of the process was to assign the components into Primary, Expansion, Existing Programs, and No Additional Measures groups, with appropriate recommendations to support AMP development. Complete definitions of these four groups are provided in Section 3.3.1.

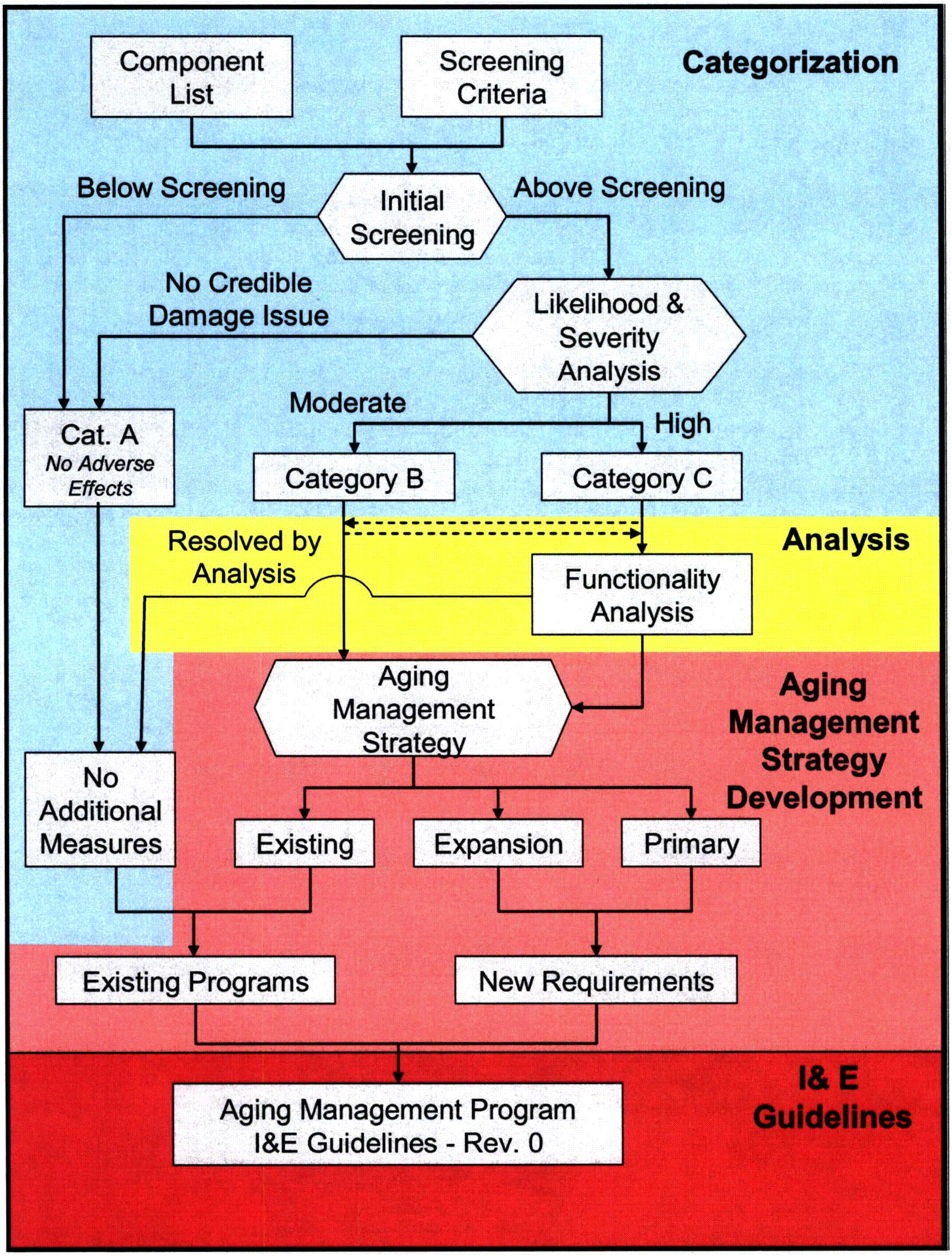


Figure 2-2
Links between categorization, functionality analysis, aging management strategy development and the I&E guidelines

2.3 Scope

These guidelines are intended to prescribe programs and activities that will assure the long-term safe and reliable operation of PWR internals as they age. As appropriately noted, the guidelines have requirements for both the original and the renewed licensing term (60-year plant life).

These guidelines are applicable to the reactor internal structural components; they do not address fuel assemblies, reactivity control assemblies, or welded attachments to the reactor vessel. They are intended for operating commercial pressurized water reactors in the U.S., operated as base load generation units. These guidelines do not supersede or modify any plant specific commitments without specific approval to do so by the regulatory body.

Section 3 provides a brief overview of currently licensed U.S. PWR internals – B&W, CE, and Westinghouse – that further defines the scope of these I&E guidelines. Section 4 identifies the components and inspection requirements. The examination acceptance criteria and the expansion criteria for the primary/expansion links are described in Section 5. Section 6 is for information and contains various options that are available for the disposition of conditions detected during examinations (Section 4) that are unable to satisfy the examination acceptance criteria (Section 5).

The implementation of these guidelines is governed by the Materials Guidelines Implementation Protocol (Appendix D) of NEI 03-08 [1]. The Mandatory, Needed, and Good Practice requirements are summarized in Section 7.

2.4 Guidelines Applicability

The guidelines are intended to serve as the primary basis for owner preparation of a reactor internals AMP in accordance with the requirement cited in Section 7. It is beyond the scope of the guidelines, however, to ensure the satisfaction of every plant-specific license renewal or power uprate commitment. Plant-specific commitments remain the responsibility of the owner.

The guidelines are based on a broad set of assumptions about plant operation, which encompass the range of current plant conditions for the U.S. domestic fleet of PWRs. The functionality analyses and supporting aging management strategies in MRP-231 [13] and MRP-232 [14] provide the basis for these guidelines. These evaluations were based on representative configurations and operational histories, which were generally conservative, but not necessarily bounding in every parameter.

General assumptions used in the analysis include:

- 30 years of operation with high leakage core loading patterns (fresh fuel assemblies loaded in peripheral locations) followed by implementation of a low-leakage fuel management strategy for the remaining 30 years of operation;
- base load operation, i.e., typically operates at fixed power levels and does not usually vary power on a calendar or load demand schedule; and
- no design changes beyond those identified in general industry guidance or recommended by the original vendors.

These assumptions are a conservative representation of U.S. PWR operating plants, all of which implemented low leakage core loading patterns early in operating life. The recommendations are thus applicable to all U.S. PWR operating plants as of May 2007 for the three designs identified. These guidelines are also considered applicable to plants that have replaced components or component assemblies; however, alternatives can be technically justified.

Plant modifications made or considered after this date should be reviewed to assess impacts on strategies contained in these guidelines.

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.

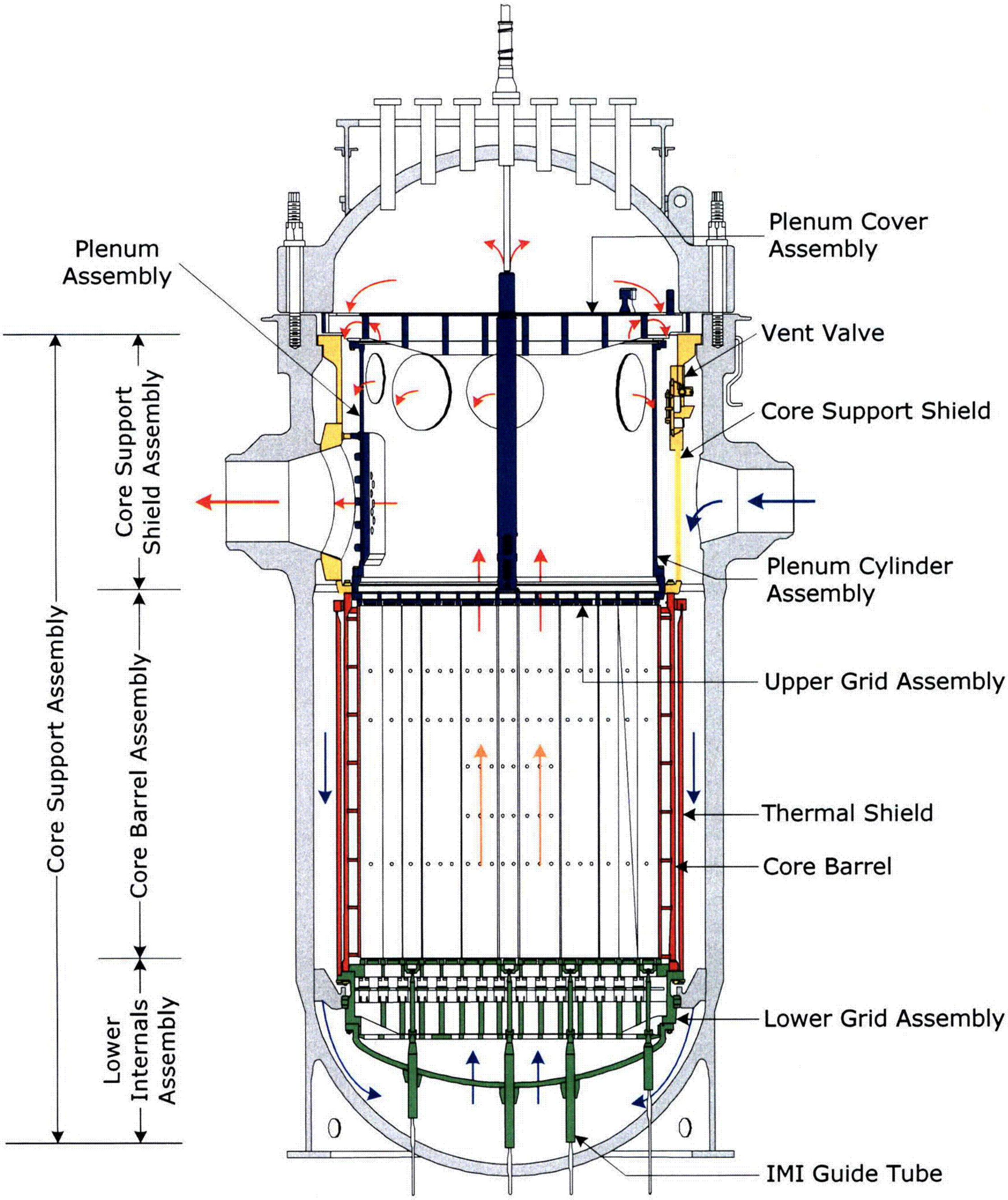


Figure 3-1
Overview of typical B&W internals

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.

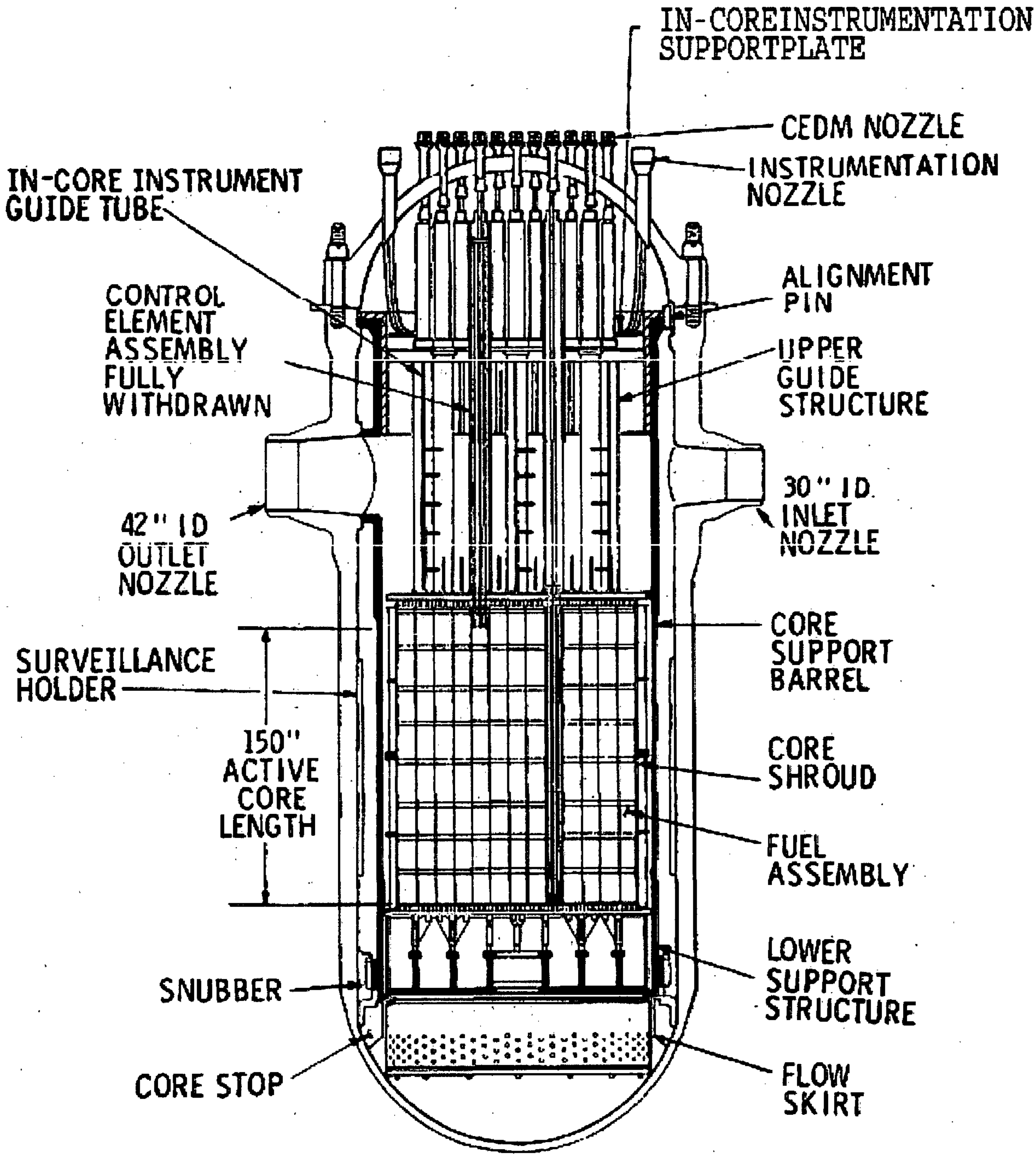


Figure 3-2
Overview of typical CE internals

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.

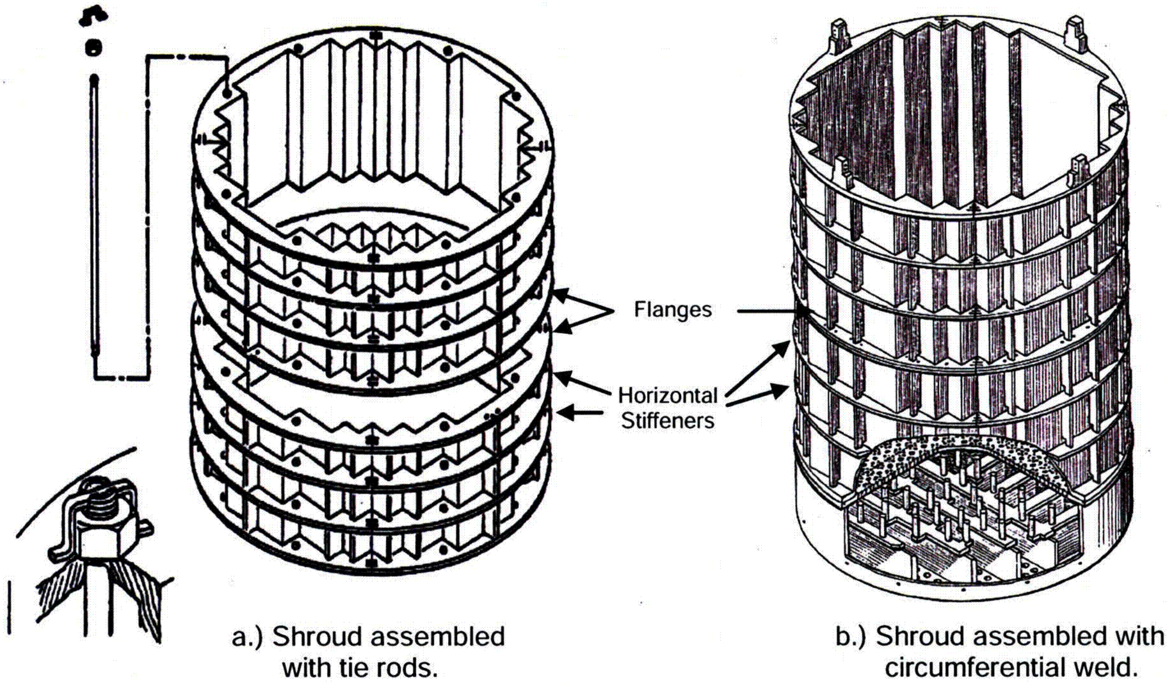


Figure 3-3
CE welded core shroud designs assembled in two vertical sections (with top-mounted ICI)

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.

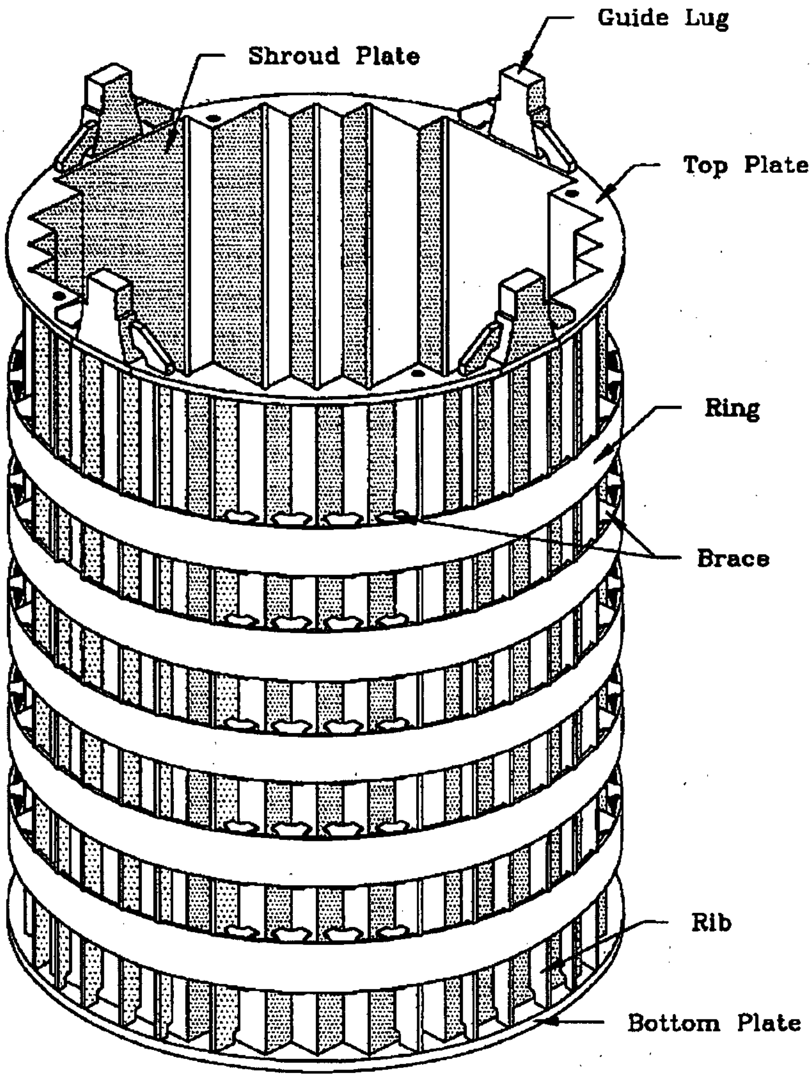


Figure 3-4
CE welded core shroud with full height panels (with bottom-mounted ICI)

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.

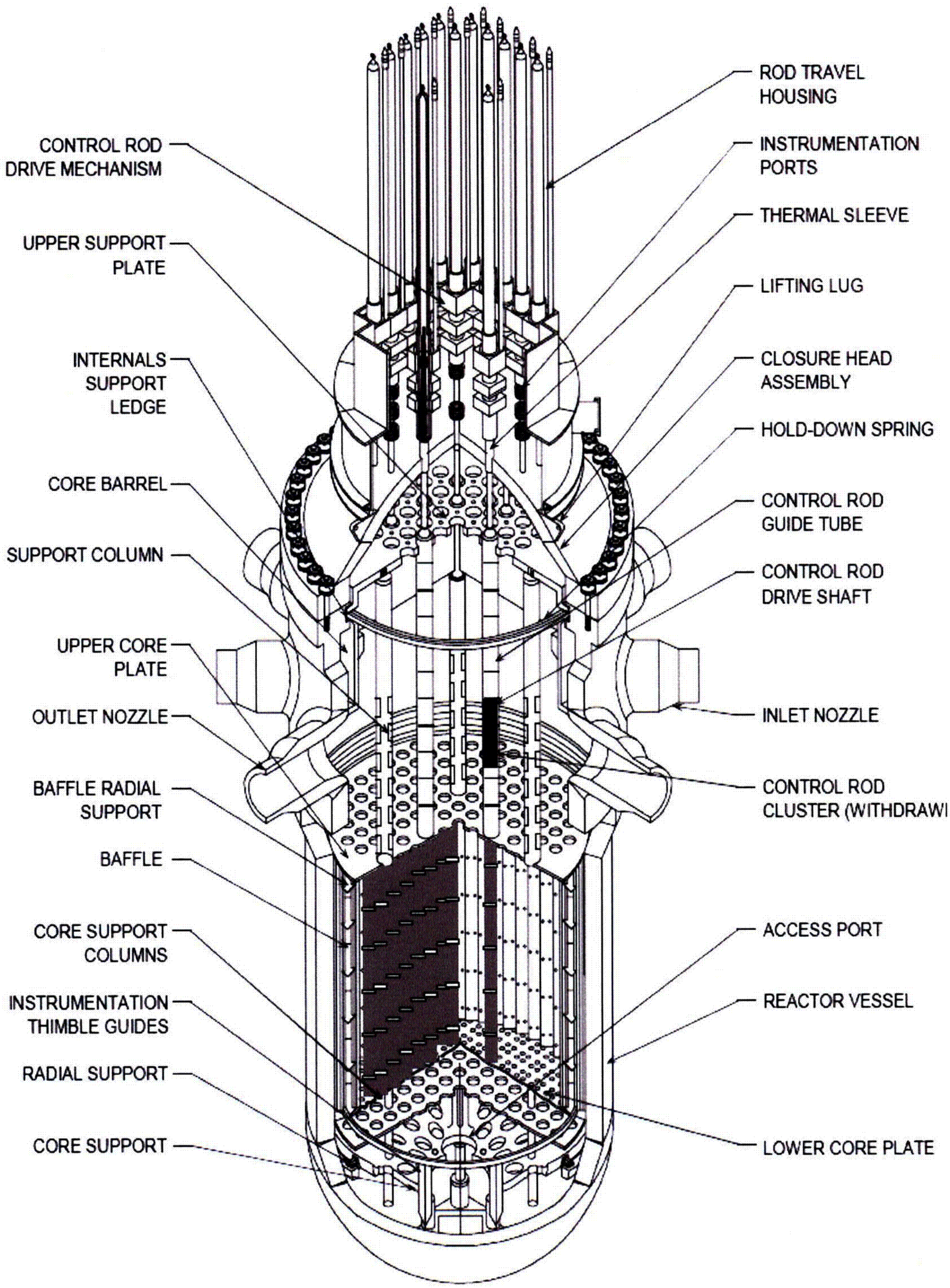


Figure 3-5
Overview of typical Westinghouse internals

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 (< 100 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 analysis 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 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 41 components listed in Table 3-1 were determined to require further evaluation (Category B and C components). Of these, 15 are Primary components and 14 are Expansion components, with the remaining 12 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 26 components listed in Table 3-2 were determined to require further evaluation (Category B and C components). Of these, 11 are Primary components, 9 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 29 components listed in Table 3-3 were determined to require further evaluation (Category B and C components). Of these, 8 are Primary components, 7 are Expansion components, 8 are Existing Programs components, with the remaining 6 requiring No Additional Measures.

Table 3-1
Final disposition of category B and C 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	E	A	A	A	E
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	P	A	A	A	P
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
CSS Vent Valve Discs	CF8	B	A	A	A	A	P	A	A	A	P
CSS Vent Valve Disc Shaft or Hinge Pin	431 SS	B	A	A	A	A	P	A	A	A	P

Table 3-1
Final disposition of category B and C 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 1)	304 SS	C	A	P	P	P	A	P	N	P	P
Internal BB Bolts (Note 1)	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 category B and C 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 Bolts (CR-3, 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 2)	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 category B and C 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 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	E	A	A	A	A	A	A	A	E
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:

1. 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; "Primary" for the FB bolts, and "Expansion" for the internal BB bolts.
2. 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.

Table 3-2
Final disposition of category B and C 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	E	E	A	E	A	E	A	A	E
Core Support Columns	CF8	B	E	E	A	E	E	E	A	A	E
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
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 Welds	304 SS	B	E	A	A	A	A	A	A	A	E
Lower Cylinder Welds	304 SS	C	E	N	A	A	A	E	A	A	E
Upper Core Barrel Flange Weld	304 SS	B	P	A	X	A	A	A	A	A	P
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

Table 3-2
Final disposition of category B and C CE internals (continued)

Component	Material	Initial Category	SCC	IASCC	Wear	Fatigue	TE (Note 1)	IE (Note 1)	VS	ISR and IC	Final Group
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
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:

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.

Table 3-3
Final disposition of category B and C 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
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 category B and C 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 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 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:

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.

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. Appendix A identifies the elements of a complete AMP that these guidelines support.

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 Industry Initiative 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. Some of these methodologies already have well established procedural qualifications, such as volumetric examination of bolting. For those requiring additional procedural qualification, Article 14 of Section 5 of the ASME Code [16] provides the criteria for the possible levels of rigor that can be selected for the qualification of examination methodologies. For example, the level of procedural qualification for volumetric (UT) examination of bolting 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 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 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 IWB-2500 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 IWB-2500 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 Code [2] requirements for visual (VT-1) examination became more rigorous than the previous ASME 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 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 even though the locking devices for the external baffle-to-baffle bolts and barrel-to-former bolts are Expansion components, they 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 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 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:

  - CRGT spacer castings
  - Lower 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, the CSS cast outlet nozzles, the CSS vent valve discs, the control rod guide tube (CRGT) spacer castings, 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 CRGT spacer castings (Figure 4-5) are an expansion item for thermal embrittlement. The primary items are the IMI guide tube spiders, the CSS cast outlet nozzles, and the CSS vent valve discs (see following primary item sub-section). 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.

The lower 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.

Primary:

- CSS cast outlet nozzles (applicable to Oconee Nuclear Station Unit 3 (ONS-3) and Davis-Besse (DB) only)
- CSS vent valve discs (applicable to all plants)

  Expand to:

  - CRGT spacer castings

The CSS outlet nozzles are currently inspected by the existing ASME Section XI [2] 10-year ISI program, while the CSS vent valve discs are inspected every refueling outage as part of the existing vent valve inspection program defined in BAW-2248A, Page 4.3 and Table 4-1 [18]. The lower grid fuel pads and their associated welds are already part of the ASME Section XI [2] 10-year ISI program and are inspected via visual (VT-3) examination.

The expansion item is covered in the previous primary component sub-section.

Primary (applicable to all plants):

- CSS vent valve top retaining ring
- CSS vent valve bottom retaining ring
- CSS vent valve disc shaft or hinge pin

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:

  - Lower core barrel (LCB) bolt locking devices (Expansion to LCB applies if the required Primary examination of LCB bolt locking devices has not been performed as scheduled in Table 4-1)
  - Upper thermal shield (UTS) bolt locking devices (applicable to all plants)
  - Lower thermal shield (LTS) bolt locking devices (applicable to all plants)
  - Flow distributor (FD) bolt locking devices (applicable to all plants)
  - Surveillance specimen holder tube (SSHT) bolt 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 locking devices (applicable to all plants)
  - Flow distributor (FD) bolt locking devices (applicable to all plants)
  - Surveillance specimen holder tube (SSHT) bolt 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 these bolts are also examined by volumetric (UT) examination.

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

  Primary:

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

    Expand to:

    - Lower Core Barrel (LCB) bolts (Expansion to LCB applies if the required Primary examination of LCB bolts has not been performed as scheduled in Table 4-1)

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

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

The potential degradation mechanism for the structural bolting rings is stress corrosion cracking. For bolting, 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 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 of 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 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:

    - Remaining core barrel assembly welds, starting with the lower core barrel flange weld
    - Core support column welds (these components receive a visual (VT-3) examination)

  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:

    - Remaining core barrel welds
    - Lower support column bodies (non cast)

  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)

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.
Core Support Shield Assembly CSS cast outlet nozzles	ONS-3, DB	Cracking (TE), including the detection of surface irregularities, such as damaged or fractured material	CRGT spacer castings	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible surfaces. See Figure 4-9.
Core Support Shield Assembly CSS vent valve discs (Note 1)	All plants				100% of accessible surfaces (see BAW-2248A, page 4.3 and Table 4-1). See Figures 4-10 and 4-11.
Core Support Shield Assembly CSS vent valve top retaining ring CSS vent valve bottom retaining ring CSS vent valve disc shaft or hinge pin (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 Figures 4-10 and 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	Cracking (SCC)	LCB bolts (Note 3) UTS, LTS, and FD bolts SSHT bolts (CR-3 and DB only) Lower grid shock pad bolts (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 to be determined after evaluating the baseline results. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts. See Figure 4-7.
Core Barrel Assembly Lower core barrel (LCB) bolts and their locking devices	All plants	Cracking (SCC)	UTS, LTS, and FD bolts SSHT bolts (CR-3 and DB only) Lower grid shock pad bolts (TMI-1 only)	Volumetric examination (UT) of the bolts during the next 10-year ISI interval from 1/1/2006. Subsequent examination to be determined after evaluating the baseline results. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts. 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, IC/ISR/Fatigue/Wear, Overload)	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 to 15 additional years.	100% of accessible bolts. 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 for the external baffle-to-baffle bolts and 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. See Figure 4-2.
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 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 ten-year interval.	100% of accessible locking welds of the 24 dowel-to-guide block welds. See Figure 4-4.

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

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
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 separation of spider arms from the lower grid rib section at the weld	CRGT spacer castings Lower 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 TE/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 ten-year interval.	100% of accessible top surfaces of 52 spider castings and welds to the adjacent lower grid rib section. See Figures 4-3 and 4-6.

Notes:

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, variation in coloration 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. Expansion to LCB applies if the required Primary examination of LCB has not been performed as scheduled in this table.

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)	Core support column bolts, Barrel-shroud bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination after 10 to 15 additional EFPY to confirm stability of bolting pattern. Re-examination for high-leakage core designs requires continuing inspections on a ten-year interval.	100% of accessible bolts, or as supported by plant-specific justification. 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)	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)	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	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	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)	Remaining core barrel assembly welds, Core support column 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 upper flange weld. See Figure 4-15.
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 plant-specific fatigue analysis. See Figure 4-15.
Lower Support Structure Core support plate	All plants with a core support plate	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 plant-specific fatigue analysis. See Figure 4-16.

Table 4-2
CE plants Primary components (continued)

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
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 plant-specific fatigue analysis. See Figure 4-17.
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	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:

1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-2.

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)	Bottom-mounted instrumentation (BMI) column bodies, Lower support column bodies (cast)	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. See Figure 4-21.
Core Barrel Assembly Upper core barrel flange weld	All plants	Cracking (SCC)	Remaining core barrel welds, Lower support column bodies (non cast)	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. See Figure 4-22.
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	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. See Figure 4-23.

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 Baffle-former bolts	All plants	Cracking (IASCC, Fatigue)	Lower support column bolts, Barrel-former bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination after 10 to 15 additional EFPY to confirm stability of bolting pattern. Re-examination for high-leakage core designs requires continuing examinations on a ten-year interval.	100% of accessible bolts or as supported by plant-specific justification. 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.
Baffle-Former Assembly Assembly	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.

Table 4-3
Westinghouse plants Primary components (continued)

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
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. Replacement of 304 springs by 403 springs is required when the spring stiffness is determined to relax beyond design tolerance. See Figure 4-28.
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:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 5-3.

Table 4-4
B&W plants Expansion components

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Upper Grid Assembly Alloy X-750 dowel-to-upper 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.	100% of accessible dowel locking welds. See Figure 4-6 (i.e., these are similar to the lower fuel assembly support pads).
Control Rod Guide Tube Assembly CRGT spacer castings	All plants	Cracking (TE), including the detection of fractured spacers or missing screws	CSS cast outlet nozzle, CSS vent valve disks, or IMI guide tube spiders	Visual (VT-3) examination.	100% of accessible surfaces at the 4 screw locations (at every 90°) (limited accessibility). See Figure 4-5.
Core Barrel Assembly Upper thermal shield bolts (UTS)	All plants	Cracking (SCC)	UCB and LCB bolts	Volumetric examination (UT).	100% of accessible bolts. See Figure 4-7.
Core Barrel Assembly Surveillance specimen holder tube (SSHT) studs/nuts (CR-3) or bolts (DB)	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	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 (Note 1)	Examination Coverage
Core Barrel Assembly Baffle-to-baffle bolts Core barrel-to-former bolts	All plants	Cracking (IASCC, IE, IC/ISR/Fatigue/ Wear, Overload)	Baffle-to-former bolts	Internal baffle-to-baffle bolts: No examination requirements, Justify by evaluation or by replacement.	N/A. See Figure 4-2.
				External baffle-to-baffle bolts, 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	Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.
Lower Grid Assembly Lower 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 TE/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.	100% of accessible pads, dowels, and cap screws, and associated welds. See Figure 4-6.

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

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Lower Grid Assembly Alloy X-750 dowel-to-lower 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.	100% of accessible dowels welds. See Figure 4-6.
Lower Grid Assembly Lower grid shock pad bolts	TMI-1	Cracking (SCC)	UCB and LCB bolts	Volumetric examination (UT).	100% of accessible bolts. See Figure 4-4.
Lower Grid Assembly Lower thermal shield (LTS) bolts **Flow Distributor Assembly** Flow distributor (FD) bolts	All plants	Cracking (SCC)	UCB and LCB bolts	Volumetric examination (UT).	100% of accessible bolts. See Figure 4-8.

Note:
1. Examination acceptance criteria and expansion criteria for the B&W components are in Table 5-1.

Table 4-5
CE plants Expansion components

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Core Shroud Assembly (Bolted) Barrel-shroud bolts	Bolted plant designs	Cracking (IASCC, Fatigue)	Core shroud bolts	Volumetric (UT) examination, with initial and subsequent examination frequencies dependent on the results of core shroud bolt examinations.	100% (or as supported by plant-specific justification) of barrel-shroud and guide lug insert bolts with neutron fluence exposures > 3 displacements per atom (dpa). See Westinghouse design 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, with initial and subsequent examinations dependent on the results of the upper (core support barrel) flange weld examinations.	100% of accessible welds and adjacent base metal. See Figure 4-15.
Core Support Barrel Assembly Remaining core barrel assembly welds	All plants	Cracking (SCC)	Upper (core support barrel) flange weld	Enhanced visual (EVT-1) examination, with initial and subsequent examinations dependent on the results of core barrel assembly upper flange 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.

Table 4-5
CE plants Expansion components (continued)

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Lower Support Structure Core support column welds	All plants except those with core shrouds assembled with full-height shroud plates	Cracking (SCC, IASCC, Fatigue) including damaged or fractured material	Upper (core support barrel) flange weld	Visual (VT-3) examination, with initial and subsequent examinations based on plant evaluation of SCC susceptibility and demonstration of remaining fatigue life.	Examination coverage determined by plant-specific analysis. See Figures 4-16 and 4-31.
Core Shroud Assembly (Bolted) Core support column bolts	Bolted plant designs	Cracking (IASCC, Fatigue)	Core shroud bolts	Ultrasonic (UT) examination, with initial and subsequent examination frequencies dependent on the results of core shroud bolt examinations.	100% (or as supported by plant-specific analysis) of core support column bolts with neutron fluence exposures > 3 dpa. See Figures 4-16 and 4-33.
Core Shroud Assembly (Welded) Remaining axial welds	Plant designs with core shrouds assembled in two vertical sections	Cracking (IASCC)	Core shroud plate-former plate weld	Enhanced visual (EVT-1) examination, with initial and subsequent examination frequencies dependent on the results of the core shroud weld examinations.	Axial weld seams other than the core shroud re-entrant corner welds at the core mid-plane. See Figure 4-12.

Table 4-5
CE plants Expansion components (continued)

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (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)	Shroud plates of welded core shroud assemblies	Enhanced visual (EVT-1) examination, with initial and subsequent examination frequencies dependent on the results of the core shroud weld examinations.	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, with initial and subsequent examinations dependent on the results of the instrument guide tubes examinations.	100% of tubes in CEA shroud assemblies. See Figure 4-18.

Note:

1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-2.

Table 4-6
Westinghouse plants Expansion components

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Core Barrel Assembly Barrel-former bolts	All plants	Cracking (IASCC, Fatigue)	Baffle-former bolts	Volumetric (UT) examination, with initial and subsequent examinations dependent on results of baffle-former bolt examinations.	100% of accessible bolts. Accessibility may be limited by presence of thermal shields or neutron pads. See Figure 4-23.
Lower Support Assembly Lower support column bolts	All plants	Cracking (IASCC, Fatigue)	Baffle-former bolts	Volumetric (UT) examination, with initial and subsequent examinations dependent on results of baffle-former bolt examinations.	100% of accessible bolts or as supported by plant-specific justification. See Figures 4-32 and 4-33.
Core Barrel Assembly Core barrel flange, Core barrel outlet nozzles, Lower core barrel flange weld	All plants	Cracking (SCC, Fatigue)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination, with initial examination and re-examination frequency dependent on the examination results for upper core barrel flange.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal. See Figure 4-22.
Lower Support Assembly Lower support column bodies (non cast)	All plants	Cracking (IASCC)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination, with initial examination and re-examination frequency dependent on the examination results for upper core barrel flange weld.	100% of accessible surfaces. See Figure 4-34.

Table 4-6
Westinghouse plants Expansion components (continued)

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Lower Support Assembly Lower support column bodies (cast)	All plants	Cracking (IASCC) including the detection of fractured support columns	Control rod guide tube (CRGT) lower flanges	Visual (EVT-1) examination.	100% of accessible support columns. See Figure 4-34.
Bottom Mounted Instrumentation System Bottom-mounted instrumentation (BMI) column bodies	All plants	Cracking (Fatigue) including the detection of completely fractured column bodies	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. 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.

Note:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 5-3.

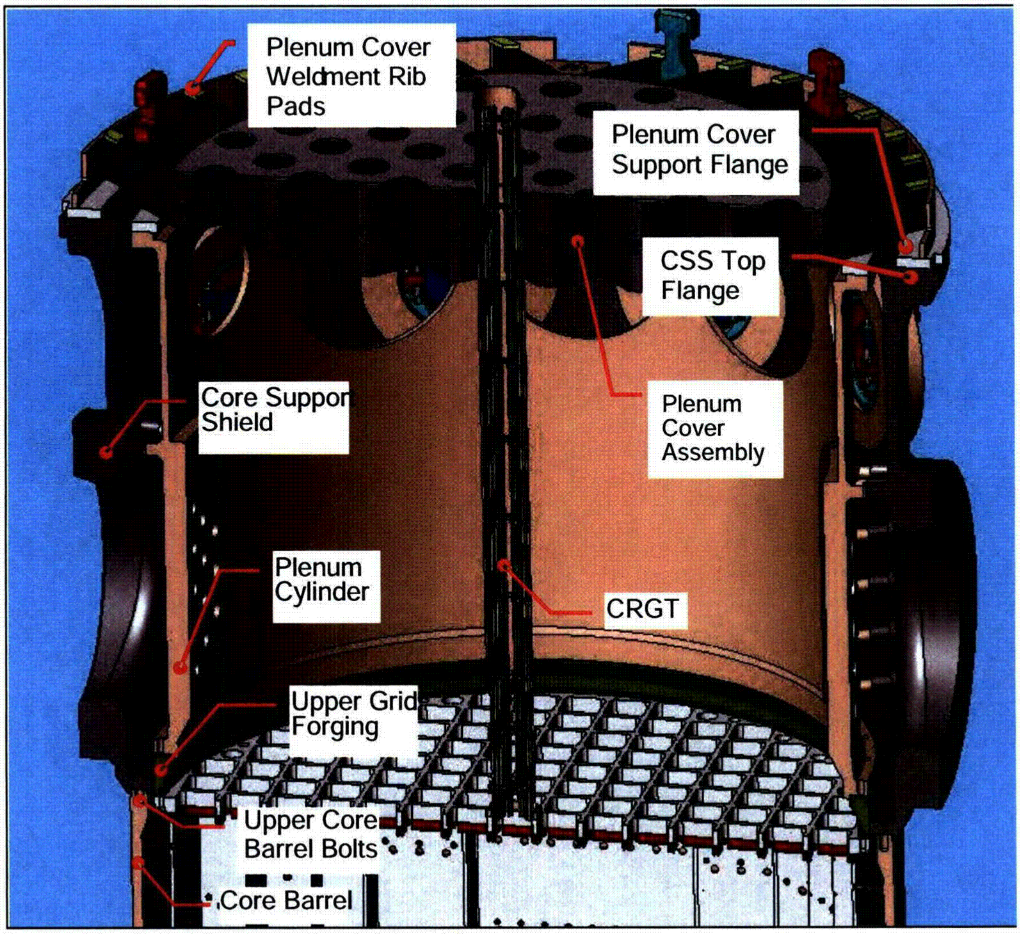


Figure 4-1
Typical upper internals arrangement for B&W-designed PWRs

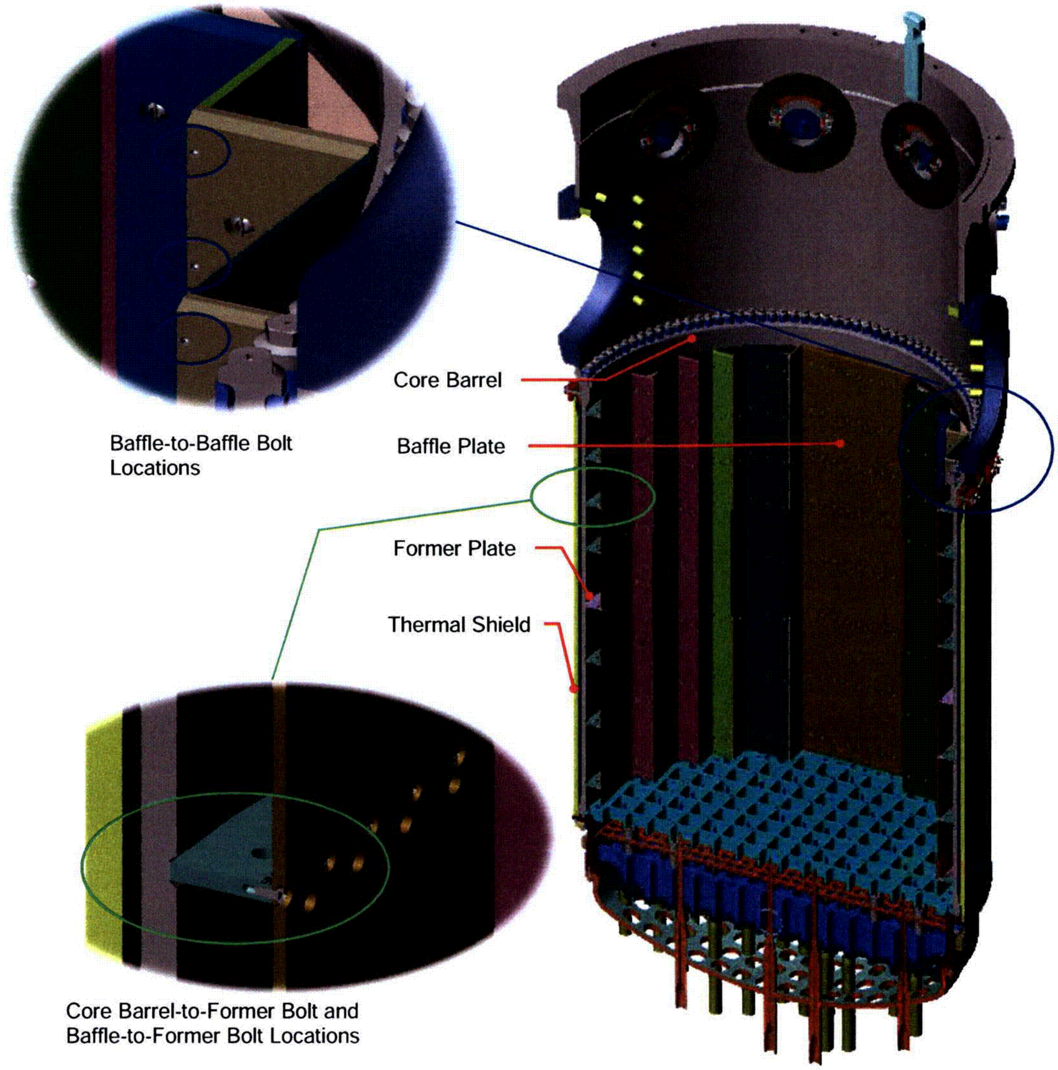


Figure 4-2
Typical internals core barrel assembly for B&W-designed PWRs

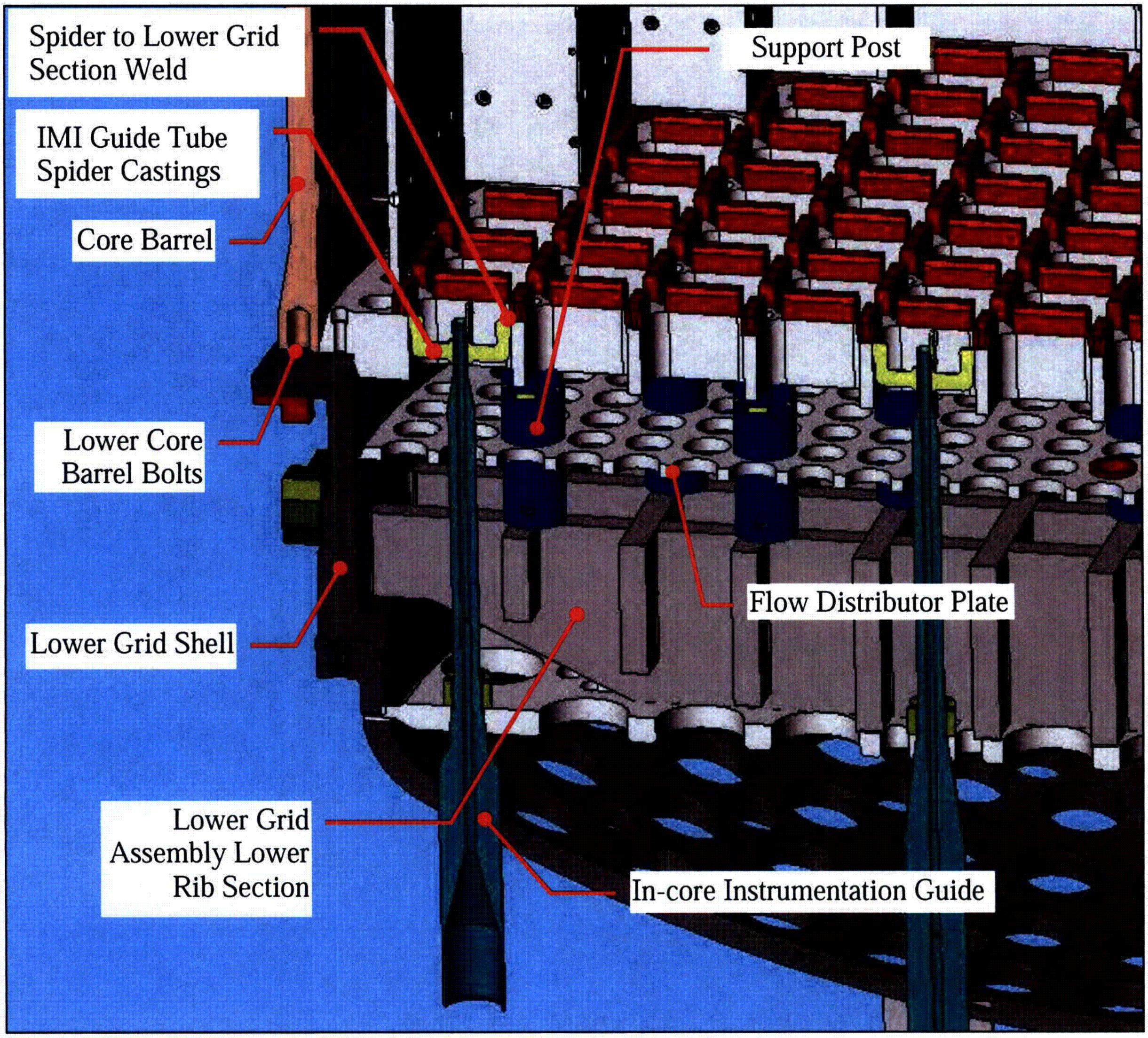


Figure 4-3
Typical lower internals arrangement for B&W-designed PWRs

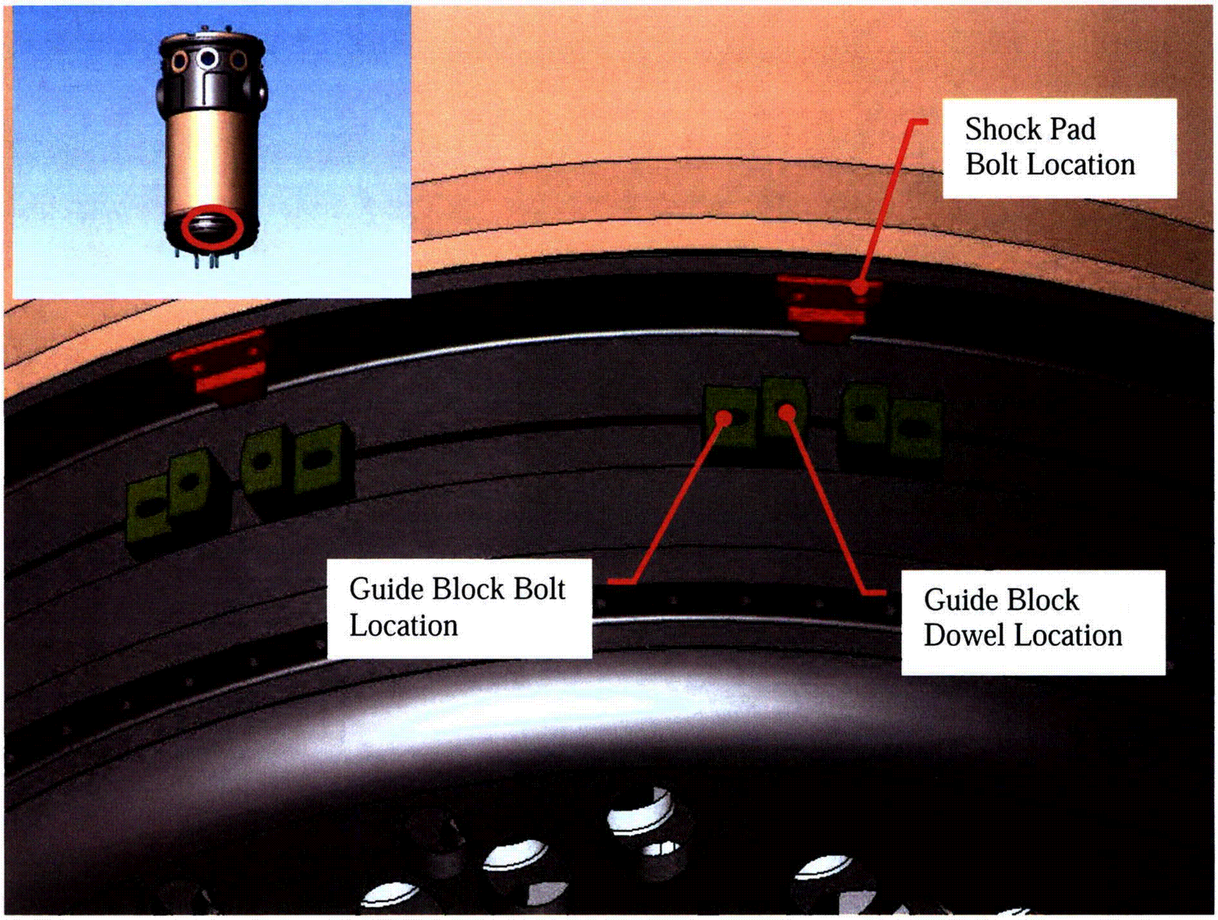


Figure 4-4
Typical guide block and shock pad locations for B&W-designed PWRs

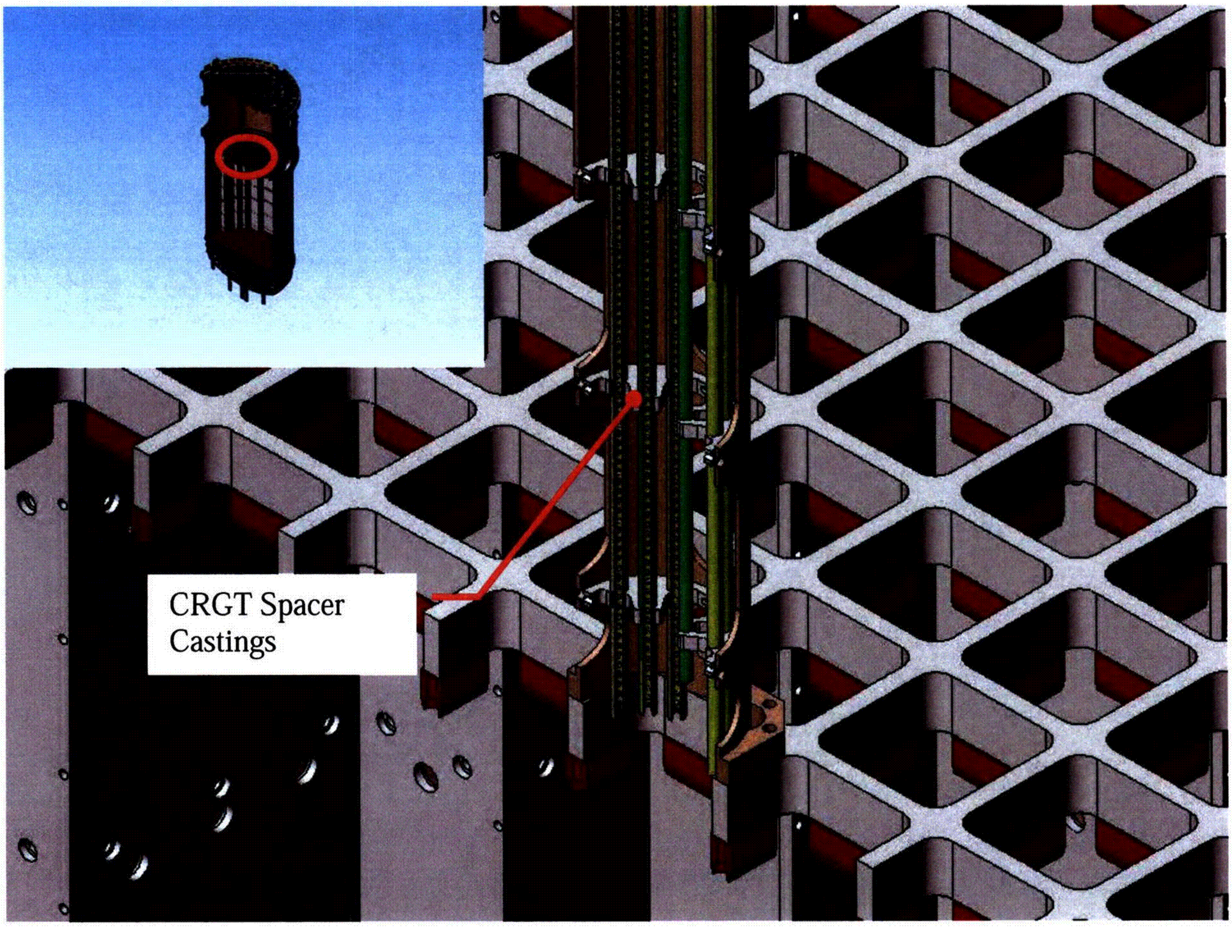


Figure 4-5
Typical control rod guide tube (CRGT) for B&W-designed PWRs (one of 69 CRGTs shown)

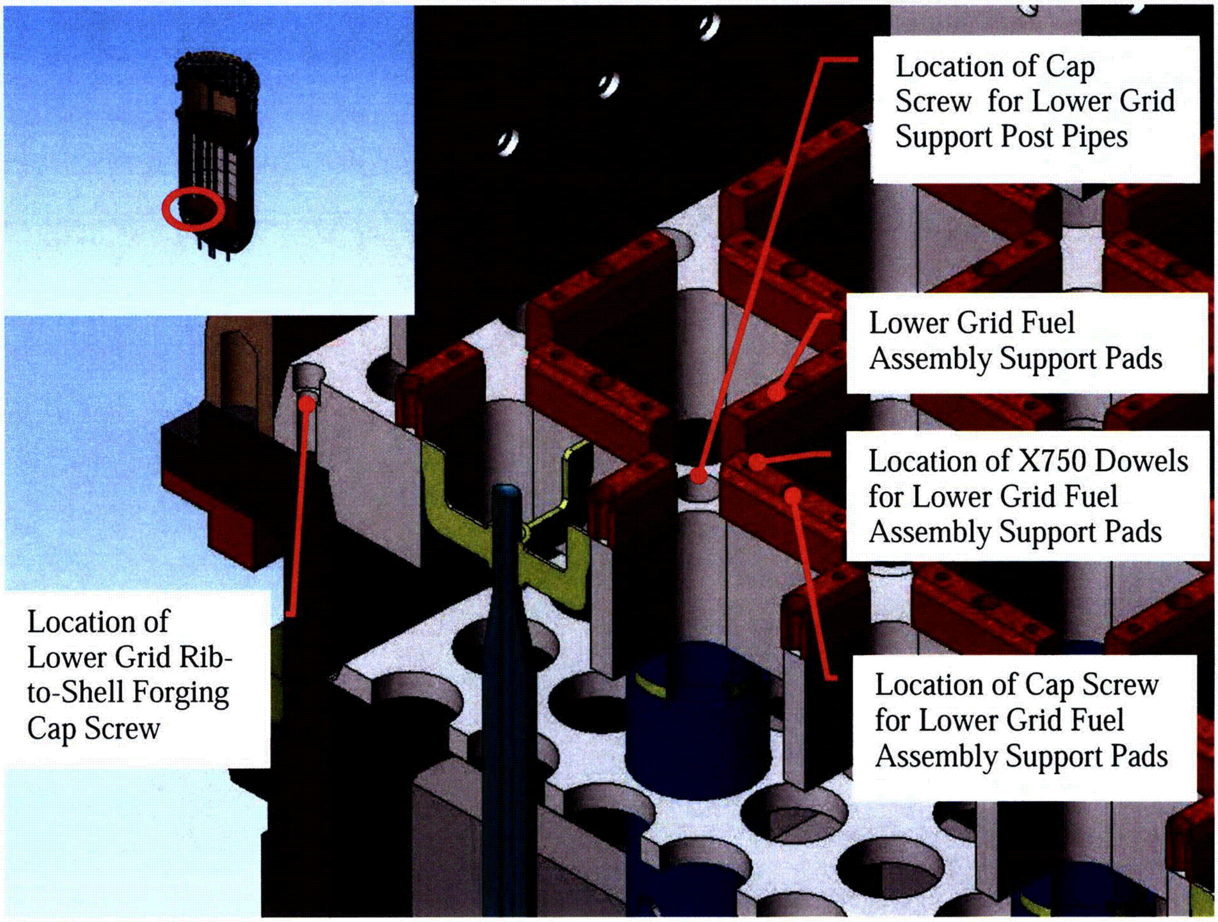


Figure 4-6
Typical lower grid assembly and fuel assembly support pads for B&W-designed PWRs

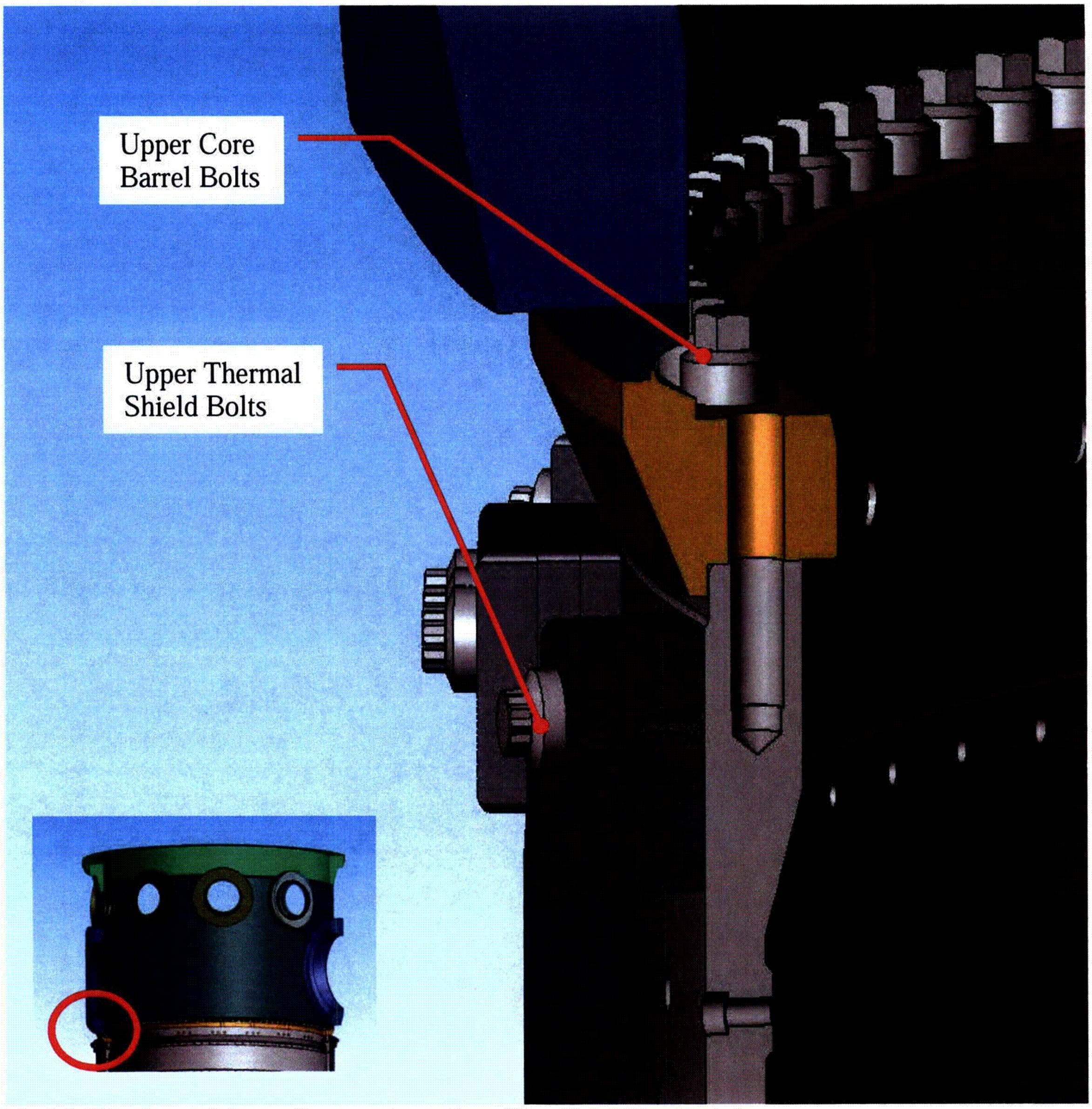


Figure 4-7
Typical upper thermal shield bolts and upper core barrel bolts for B&W-designed PWRs

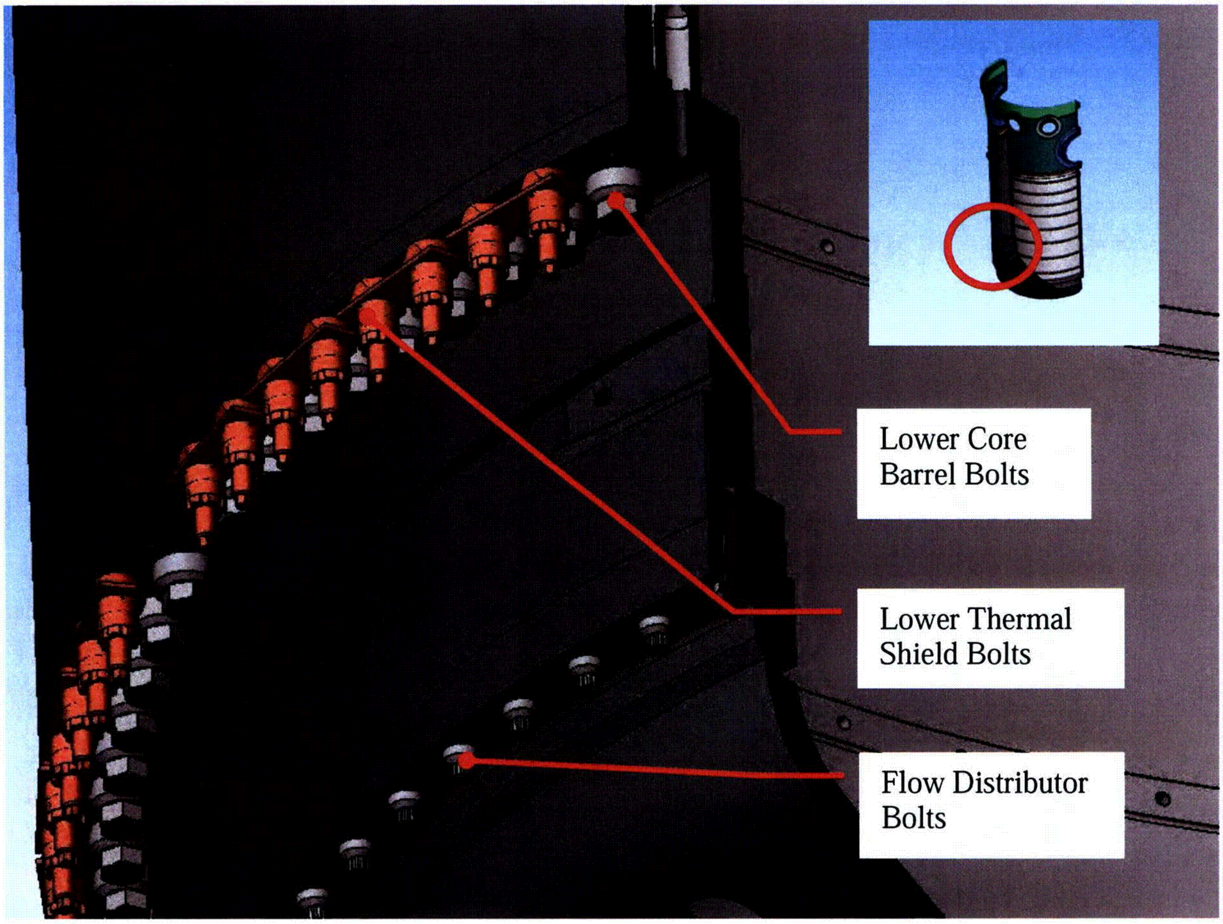


Figure 4-8
Typical lower thermal shield bolts, lower core barrel bolts, and flow distributor bolts for the B&W-designed PWRs

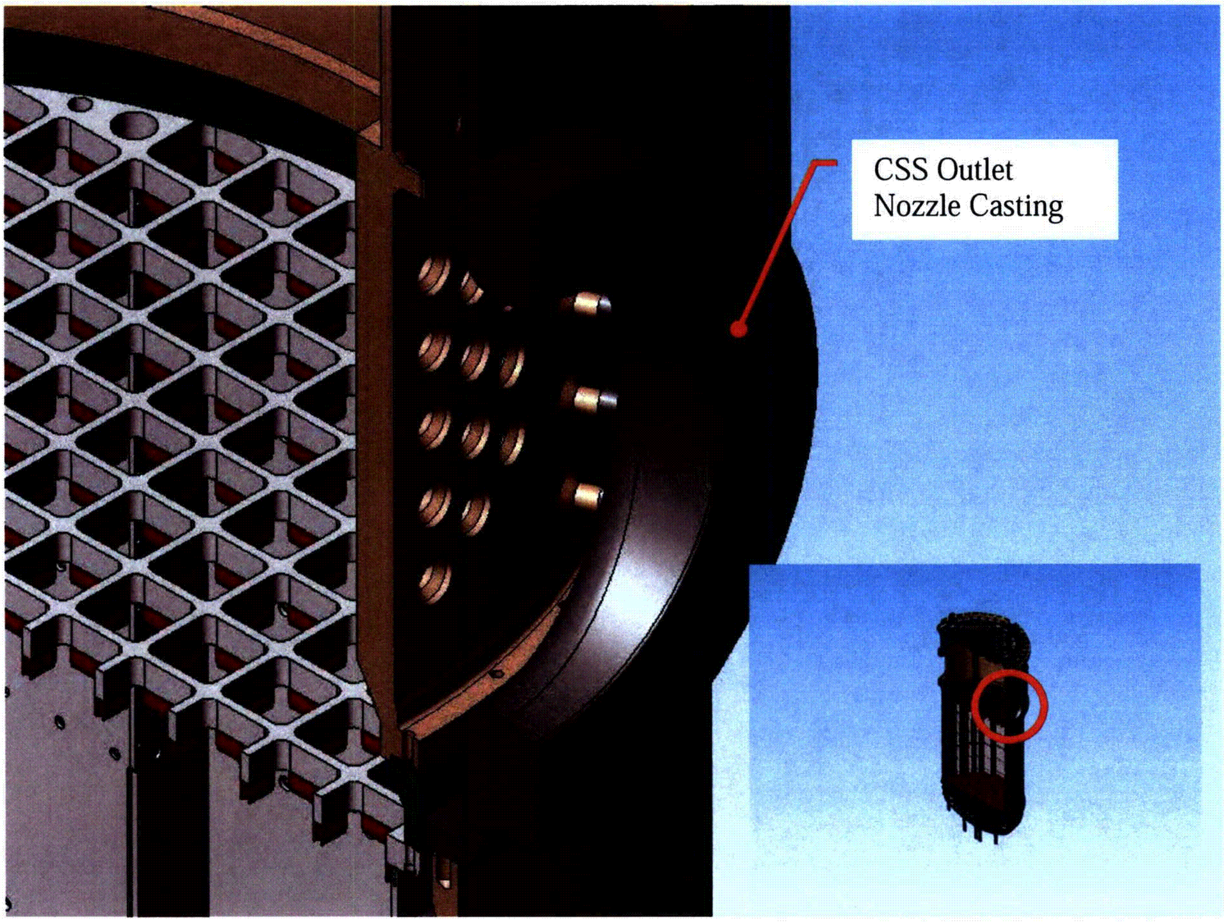


Figure 4-9
Typical core support shield (CSS) outlet nozzle for the B&W-designed PWRs

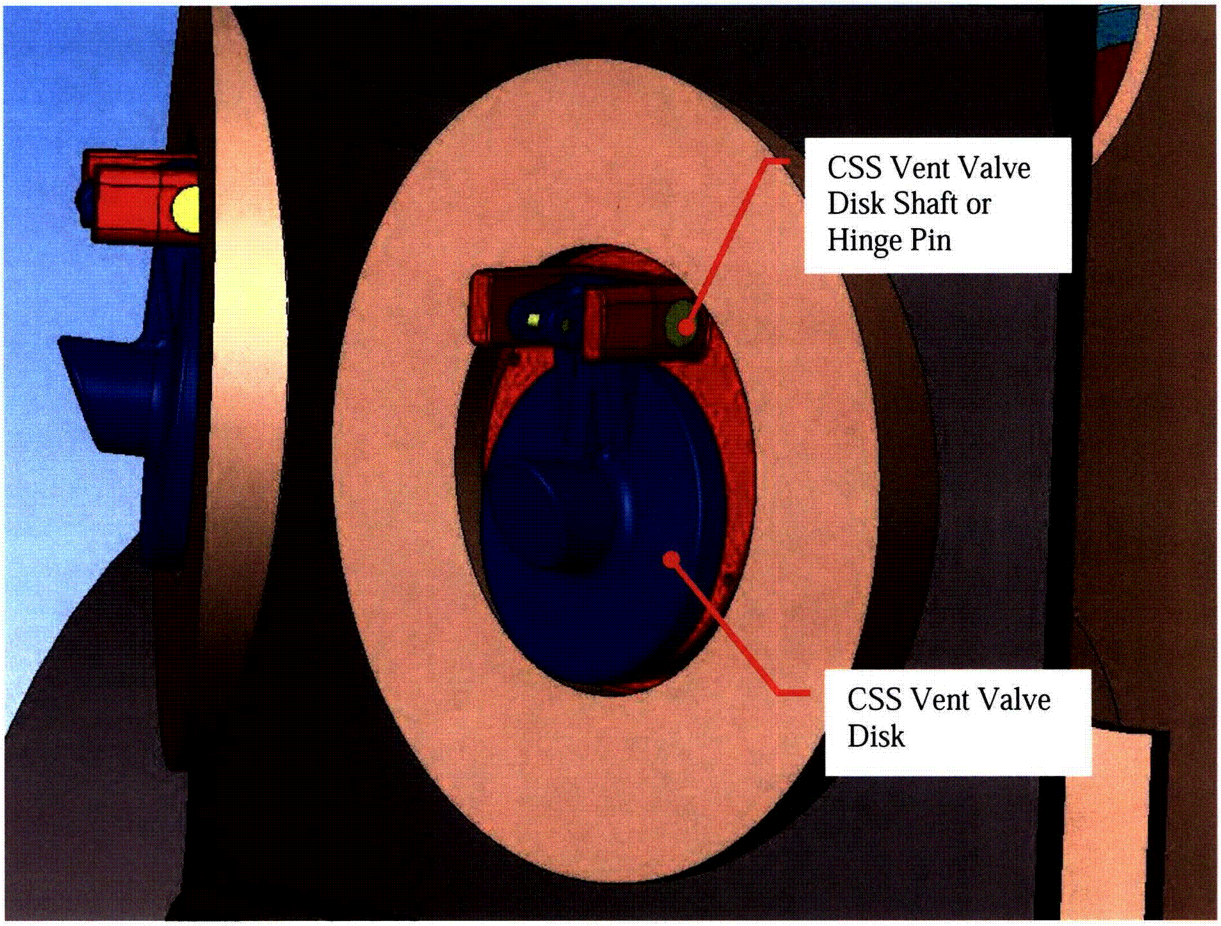


Figure 4-10
Typical core support shield (CSS) vent valve – outside view – for the B&W-designed PWRs

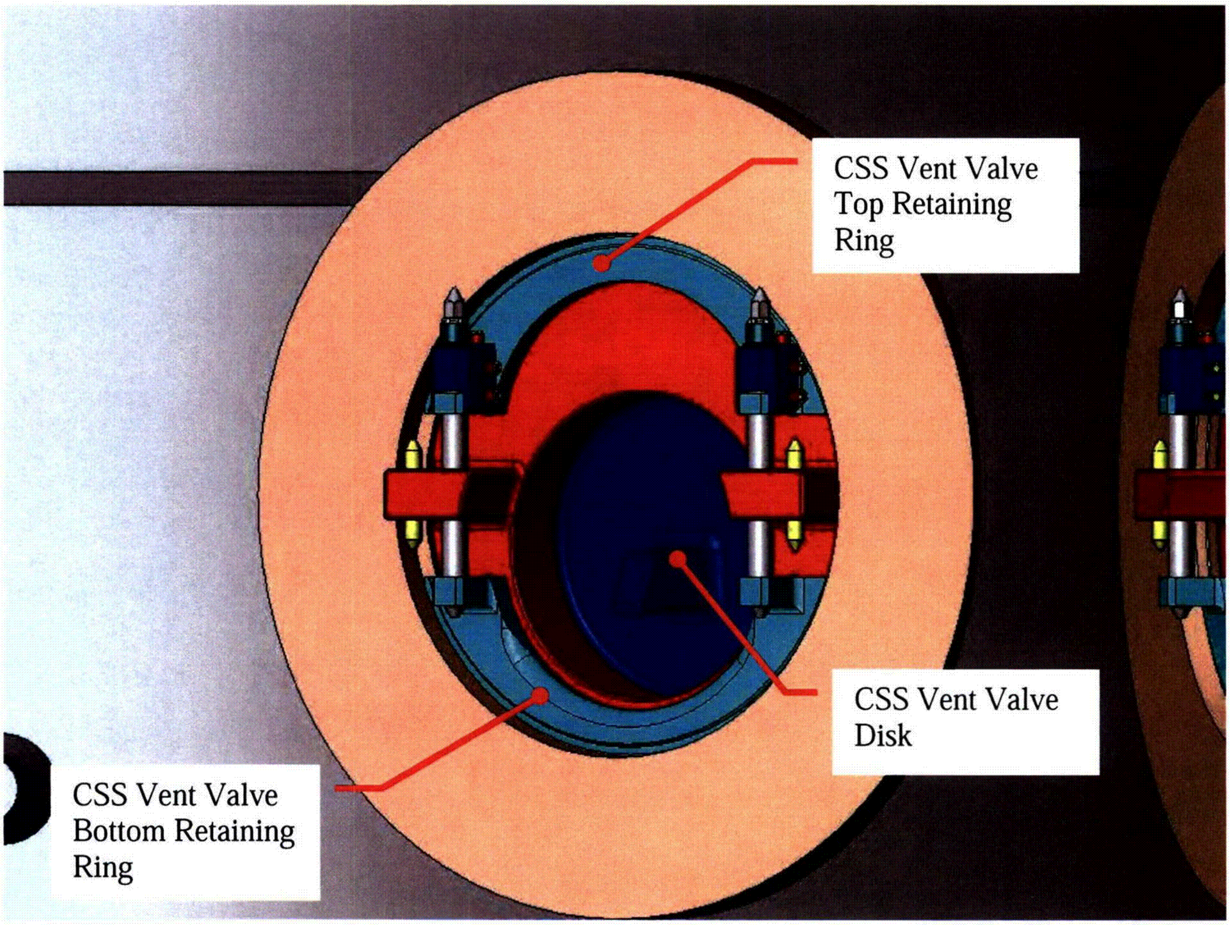


Figure 4-11
Typical core support shield (CSS) vent valve – inside view – for the B&W-designed PWRs

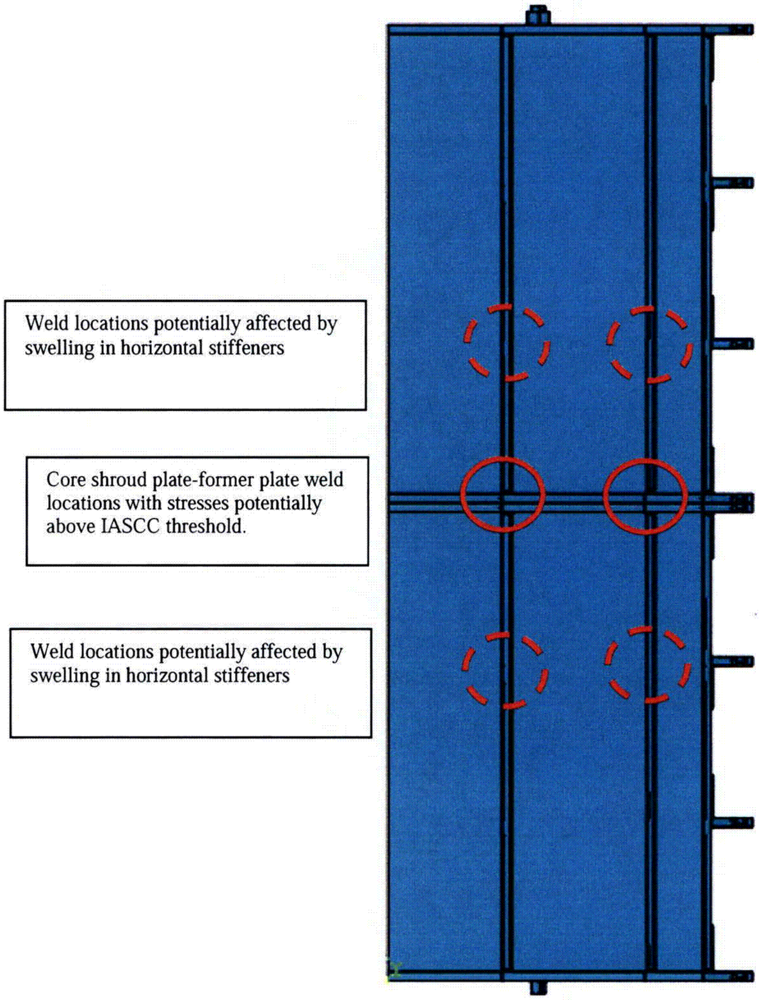


Figure 4-12
Potential crack locations for CE welded core shroud assembled in stacked sections

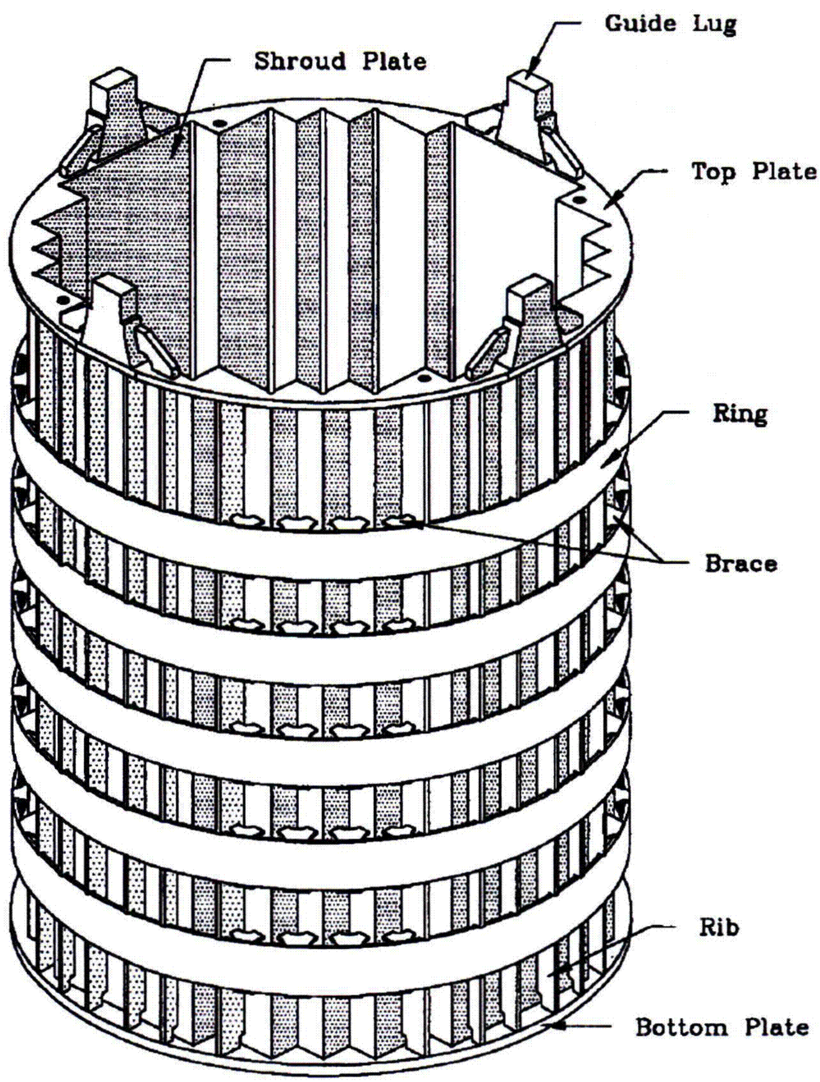


Figure 4-13
CE welded core shroud with full height panels

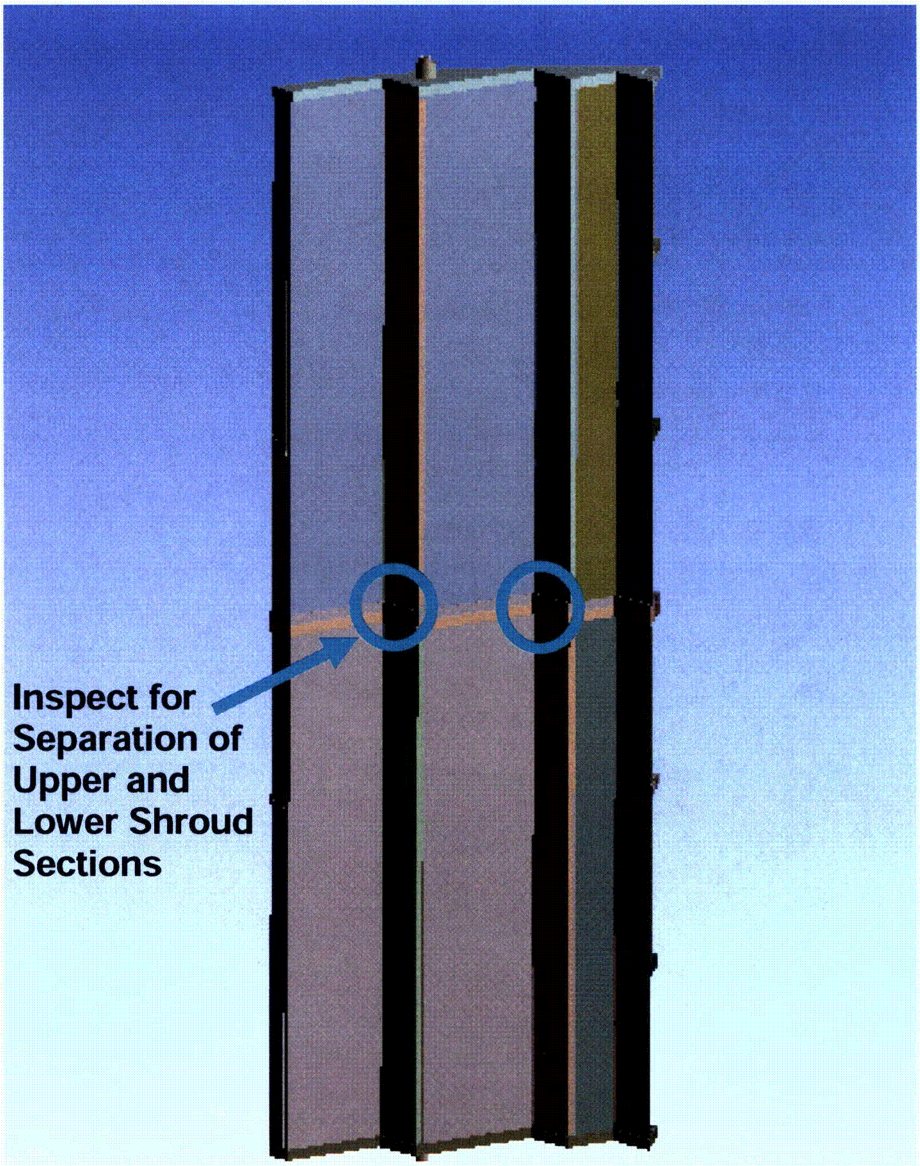


Figure 4-14
Locations of potential separation between core shroud sections caused by swelling induced warping of thick flange plates in CE welded core shroud assembled in stacked sections

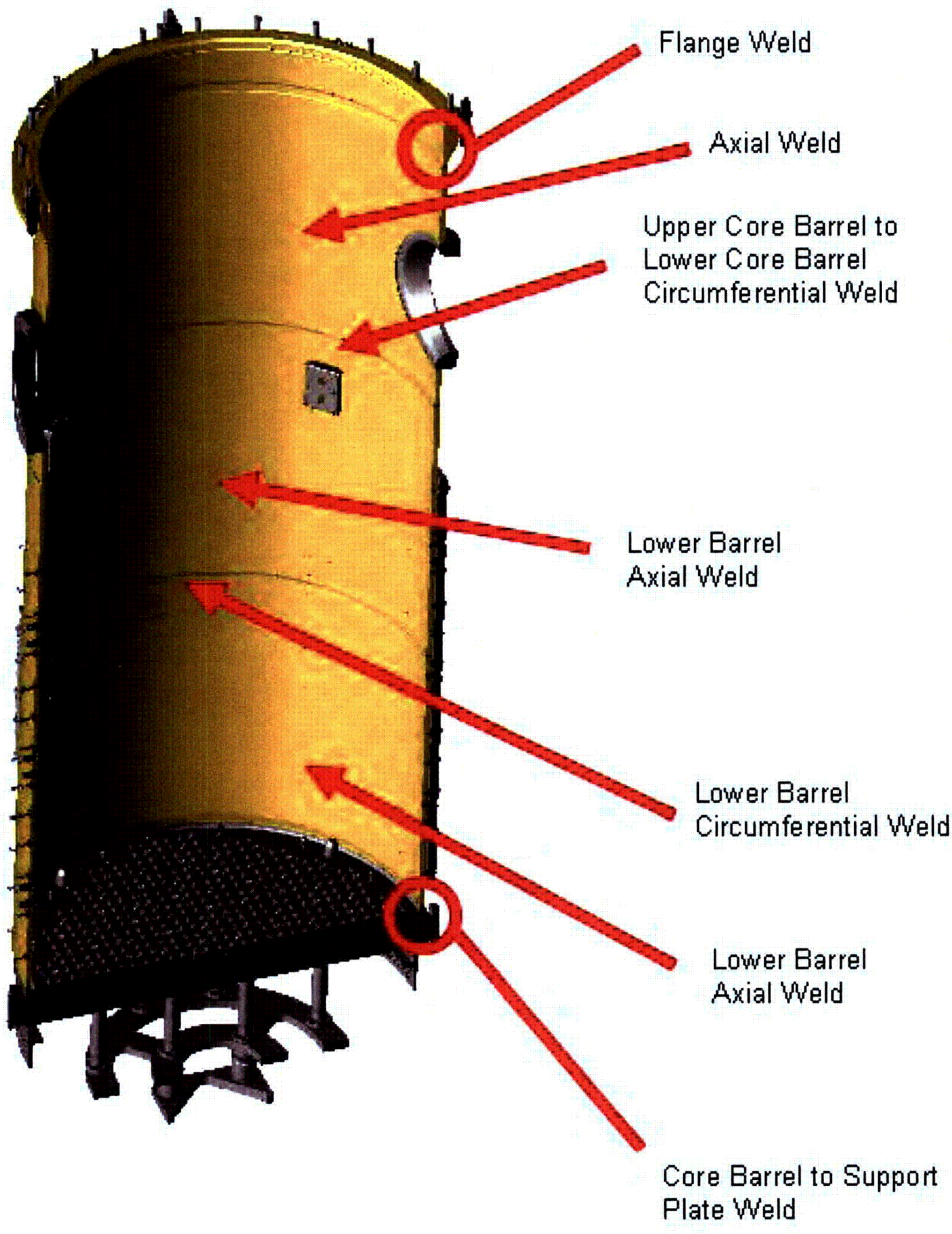
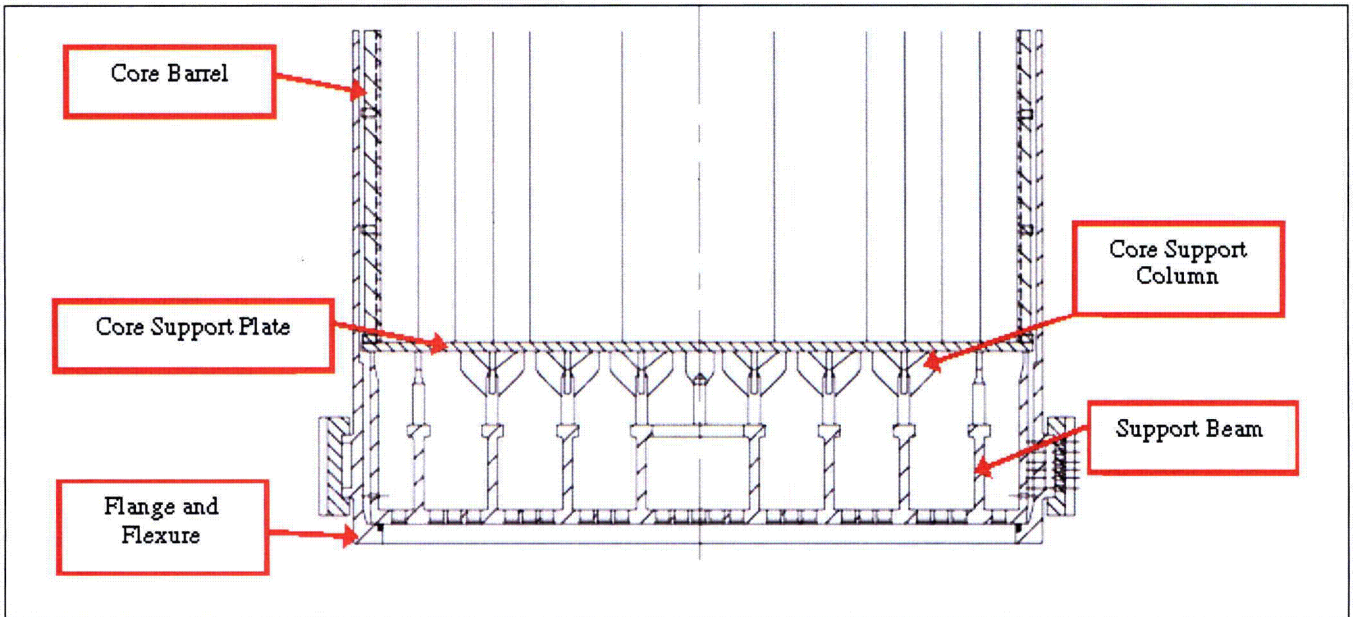
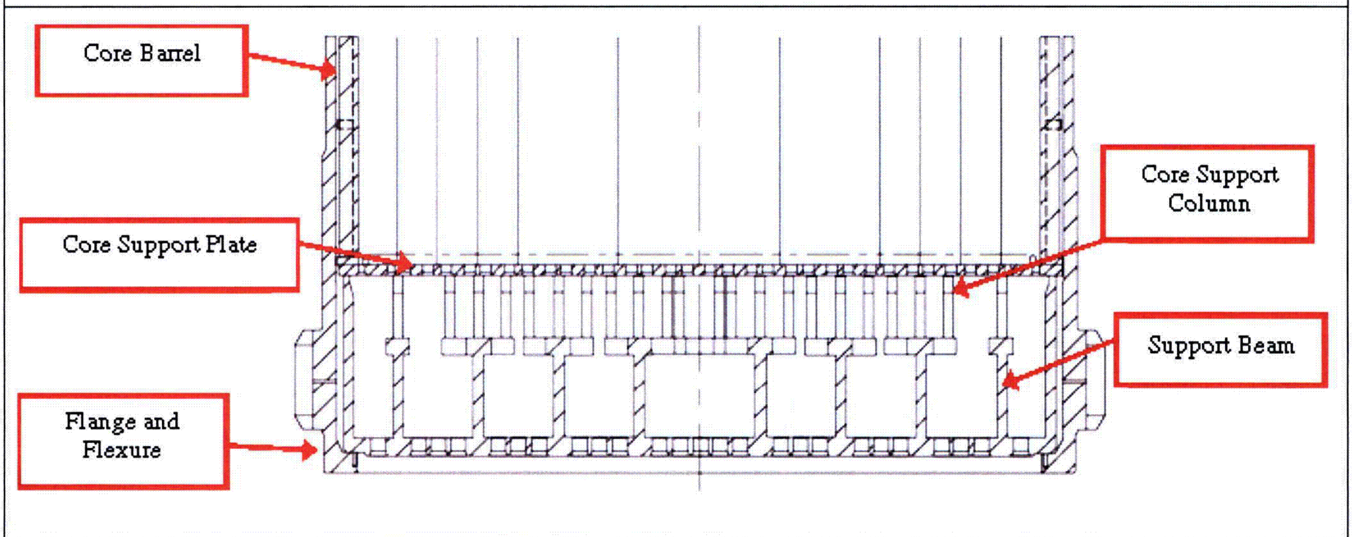


Figure 4-15
Typical CE core support barrel structure

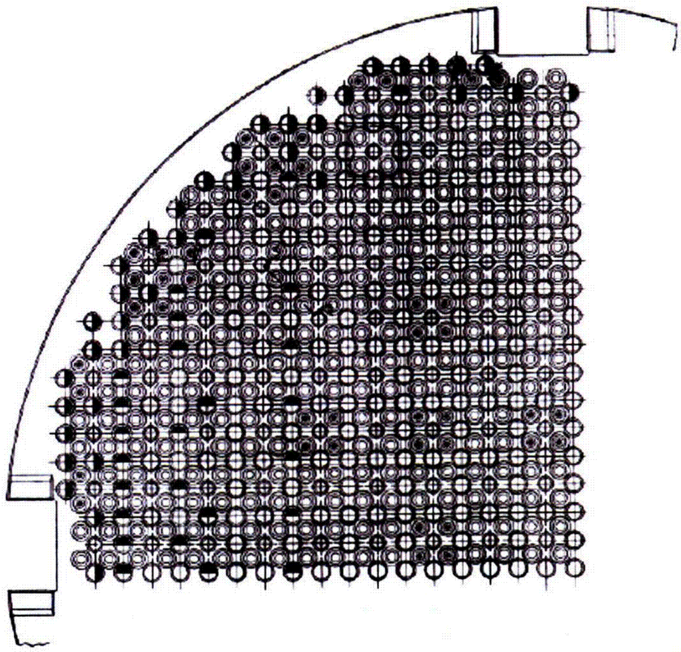


"Winged" Core Support Columns

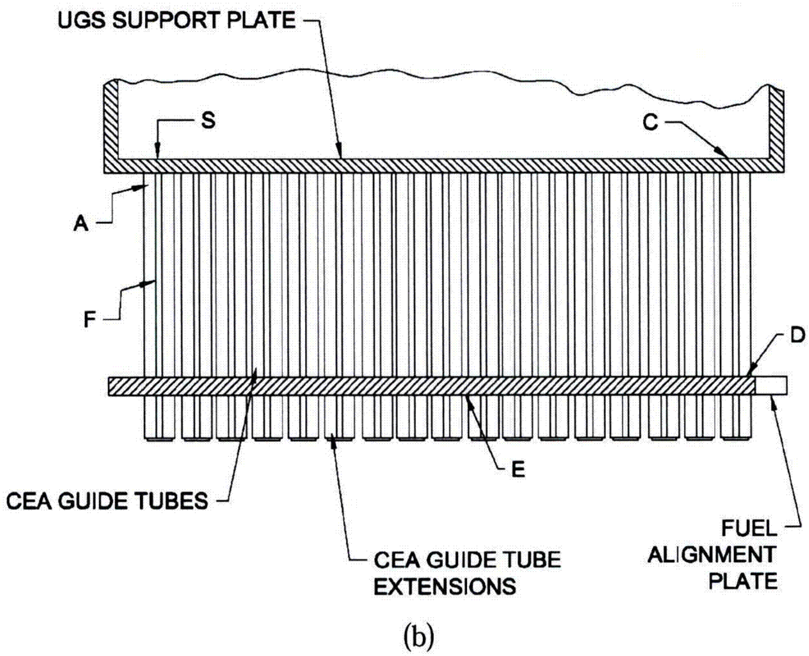


Core Support Columns without "Winged" Design

Figure 4-16
CE lower support structures for welded core shrouds: separate core barrel and lower support structure assembly with lower flange and core support plate



(a)



(b)

Figure 4-17
(a) Schematic illustration of a portion of the fuel alignment plate, and (b) Radial-view schematic illustration of the guide tubes protruding through the plate in upper internals assembly of CE core shrouds with full-height shroud plates

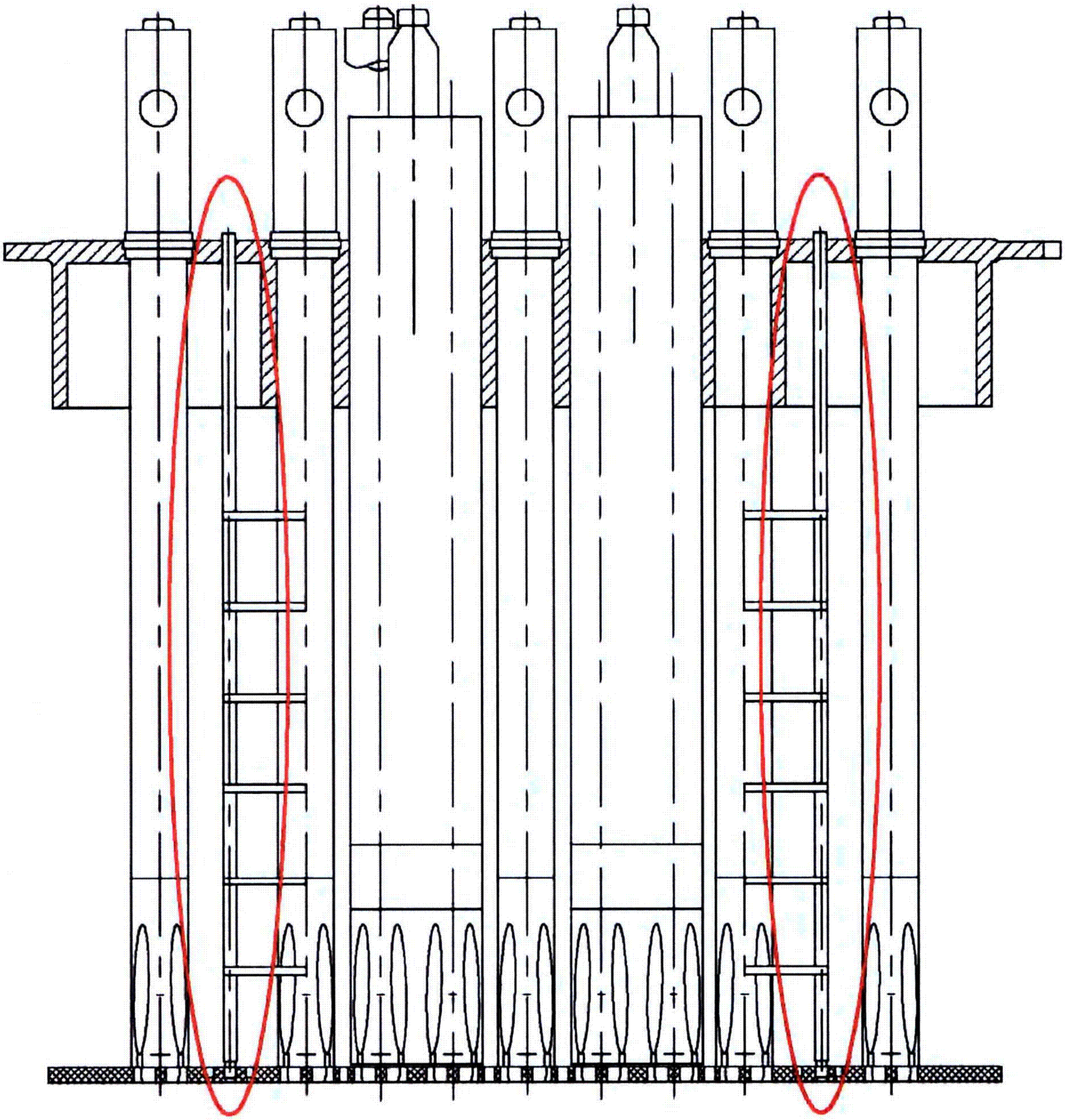
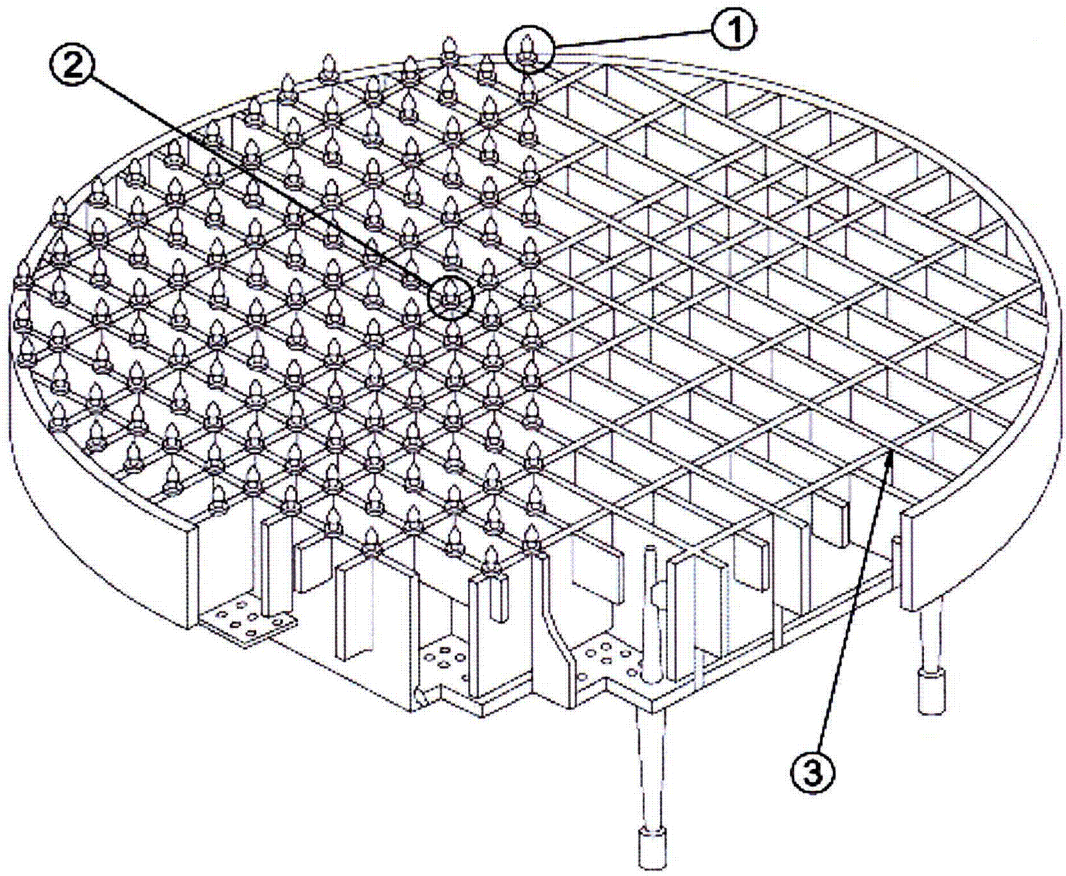


Figure 4-18
CE control element assembly (CEA) shroud instrument tubes (circled in red) are shown, along with the welded supports attaching them to the CEA shroud tube, in this schematic illustration



Illustrates the deep beam grid structure (number 3), as well as the fuel alignment pins (numbers 1 and 2)

Figure 4-19
Isometric view of the lower support structure in the CE core shrouds with full-height shroud plates units. Fuel rests on alignment pins

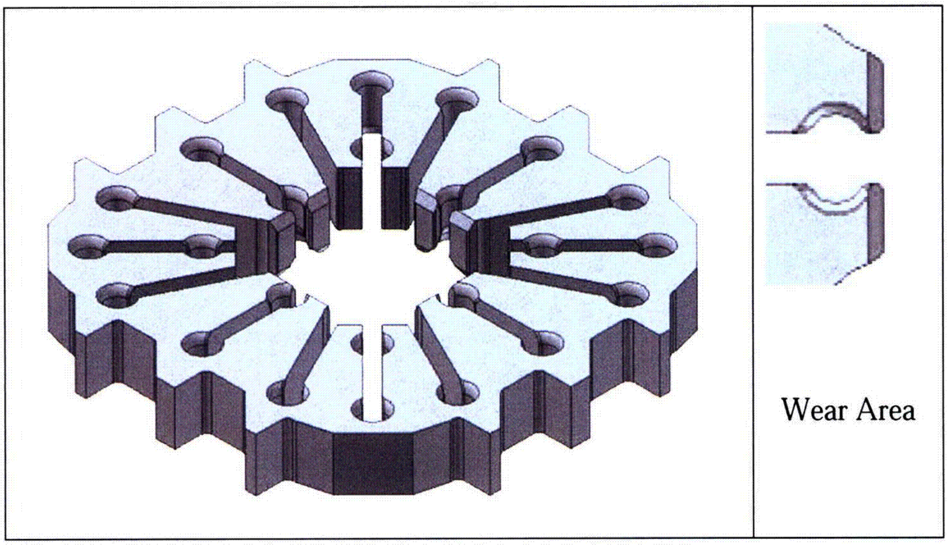


Figure 4-20
Typical Westinghouse control rod guide card (17x17 fuel assembly)

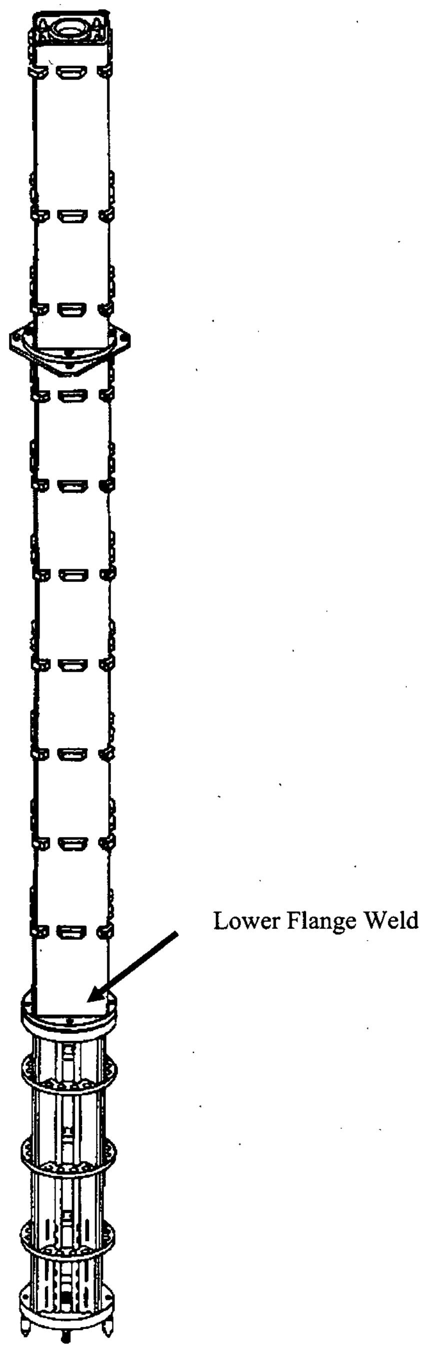


Figure 4-21
Typical Westinghouse control rod guide tube assembly

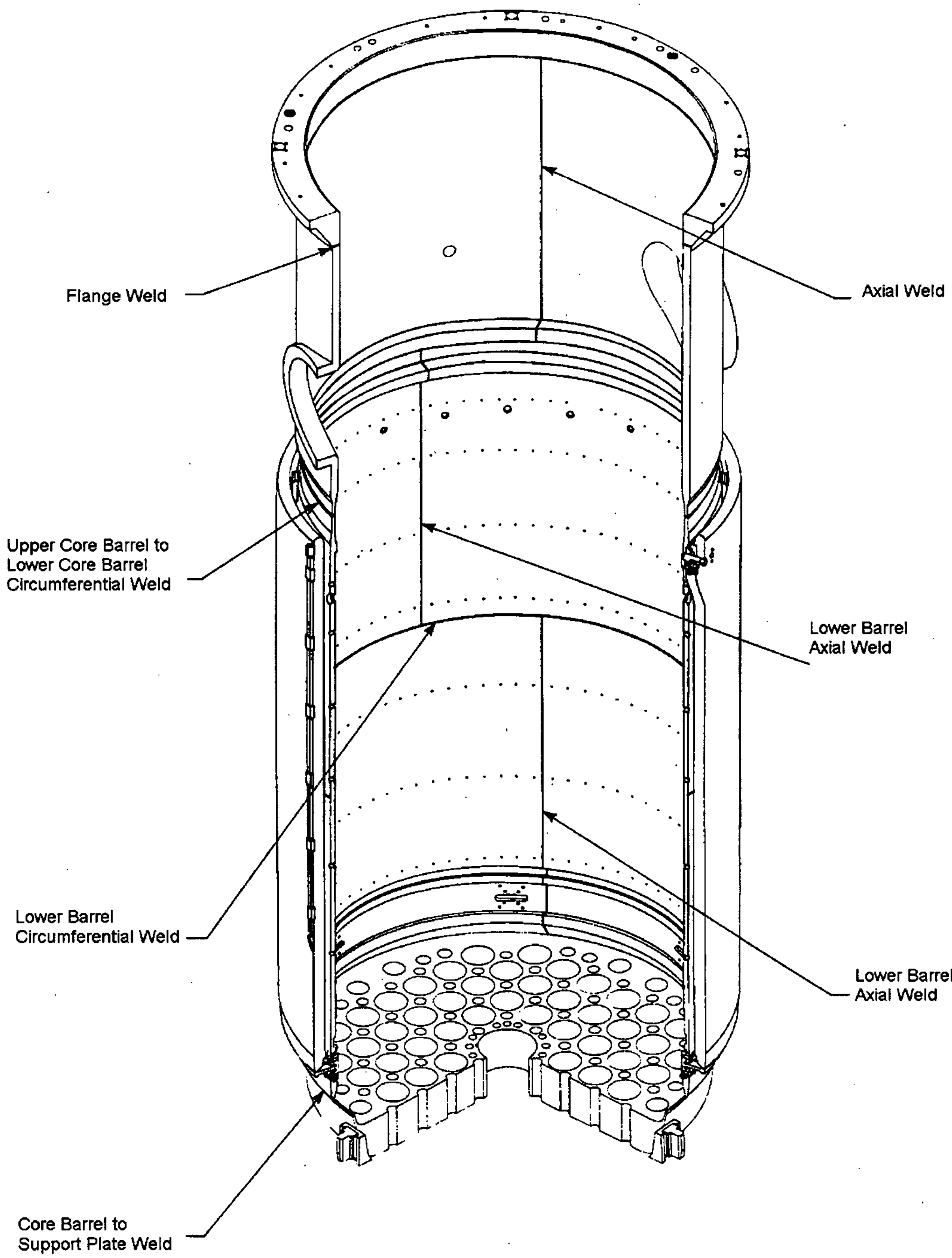


Figure 4-22
Major fabrication welds in typical Westinghouse core barrel

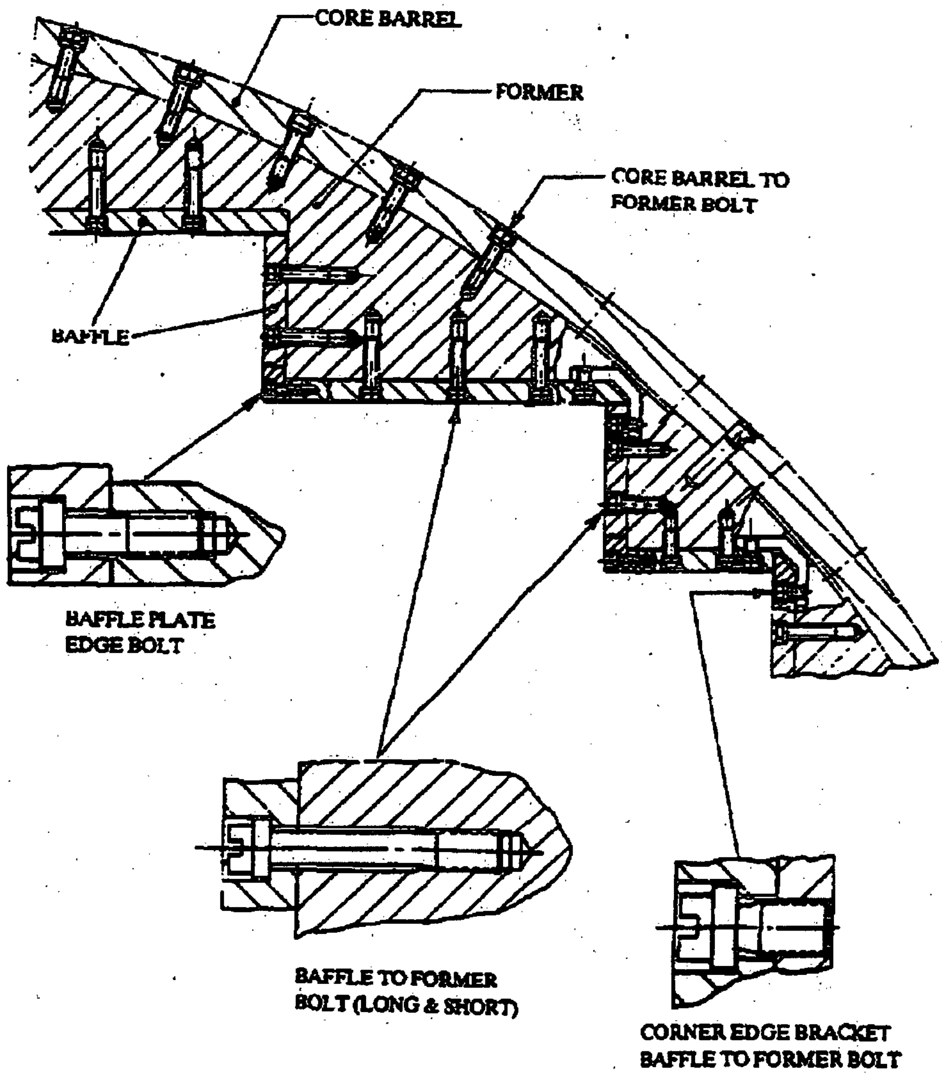


Figure 4-23
Bolt locations in typical Westinghouse baffle-former-barrel structure. In CE plants with bolted shrouds, the core shroud bolts are equivalent to baffle-former bolts and barrel-shroud bolts are equivalent to barrel-former bolts

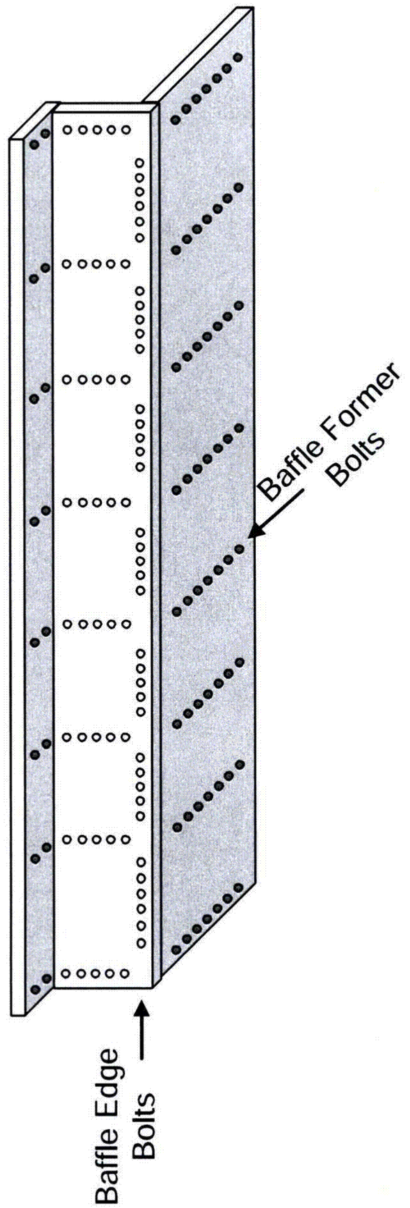


Figure 4-24
Baffle-edge bolt and baffle-former bolt locations at high fluence seams in bolted baffle-former assembly (note: equivalent baffle-former bolt locations in bolted CE shroud designs are core shroud bolts)

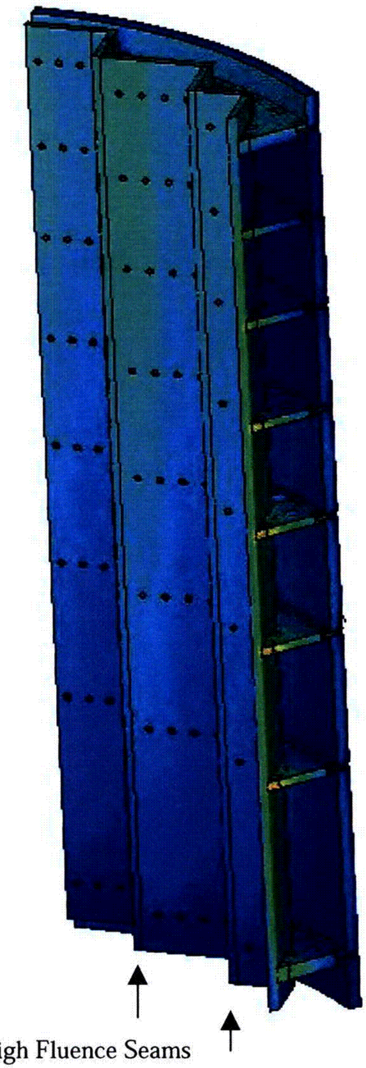


Figure 4-25
High fluence seam locations in Westinghouse baffle-former assembly

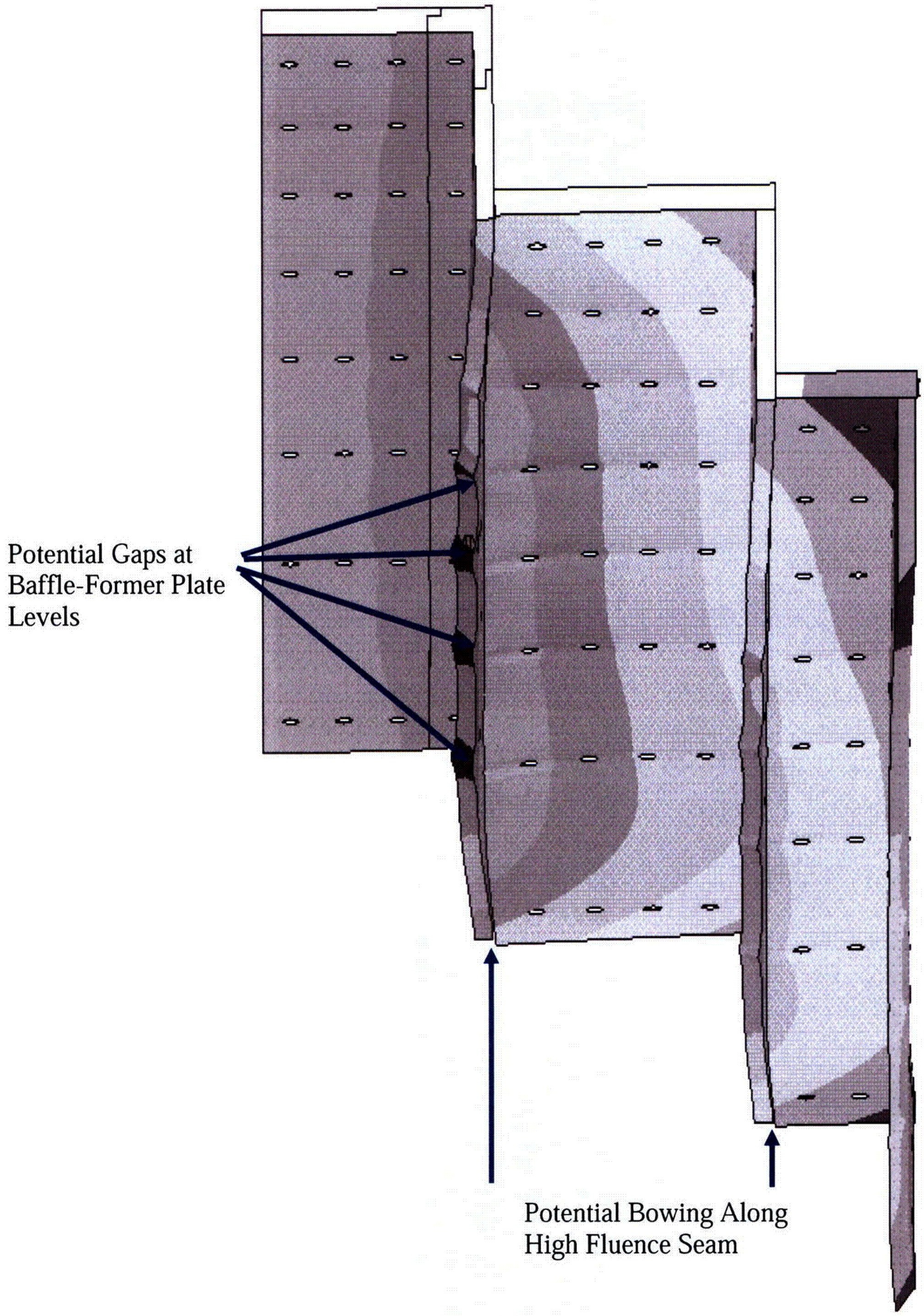


Figure 4-26
Exaggerated view of void swelling induced distortion in Westinghouse baffle-former assembly. This figure also applies to bolted CE shroud designs

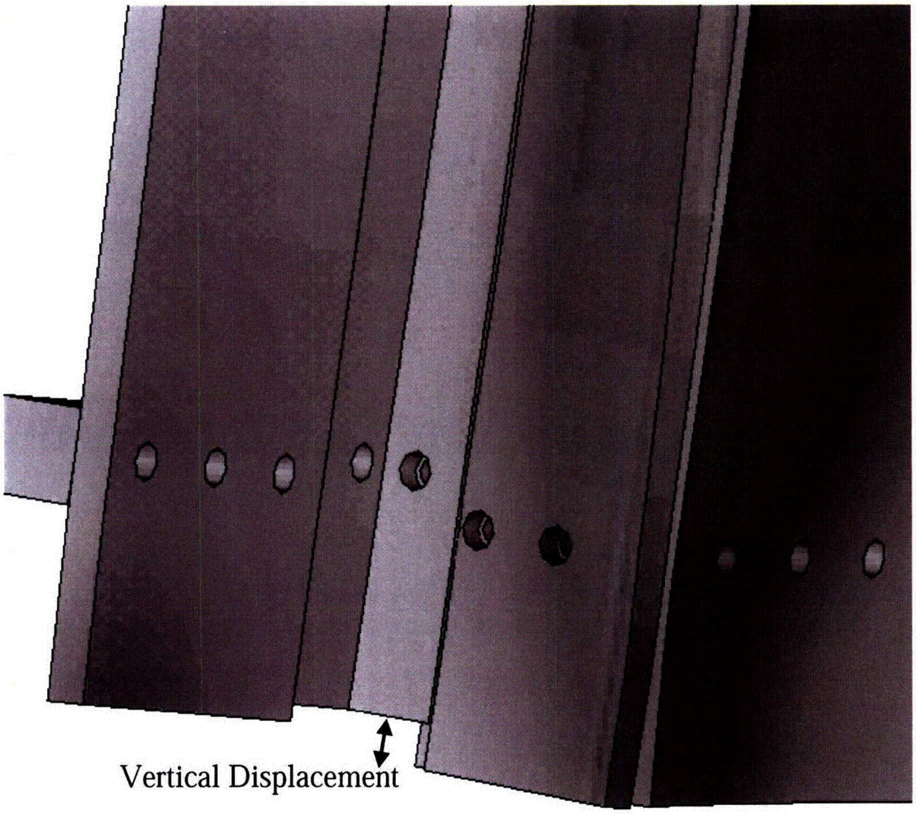


Figure 4-27
Vertical displacement of Westinghouse baffle plates caused by void swelling. This figure also applies to bolted CE shroud designs

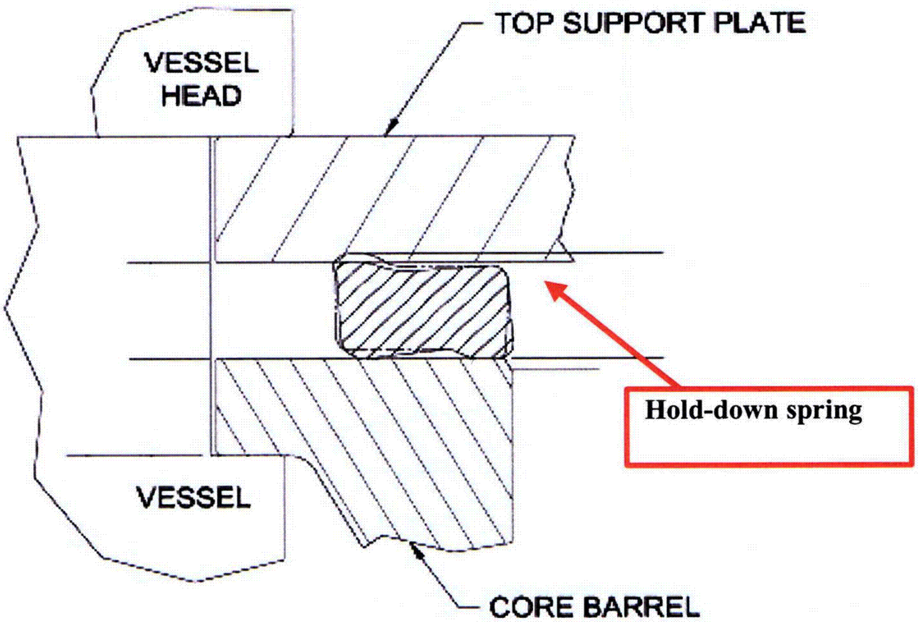


Figure 4-28
Schematic cross-sections of the Westinghouse hold-down springs

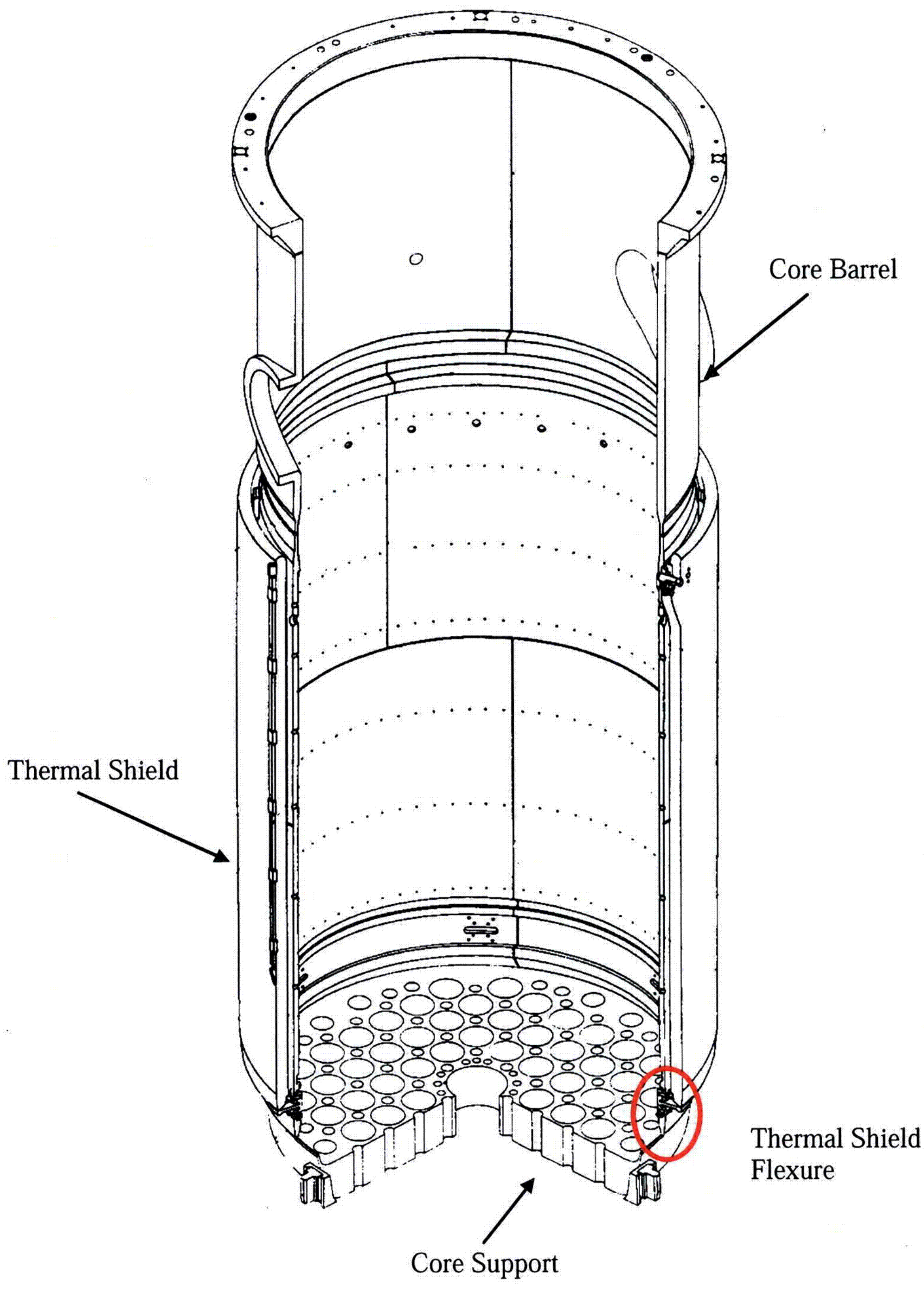


Figure 4-29
Location of Westinghouse thermal shield flexures

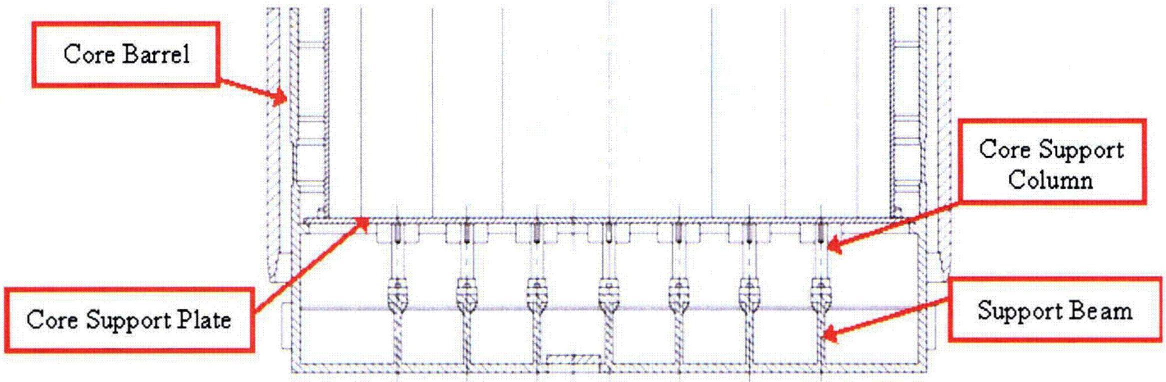
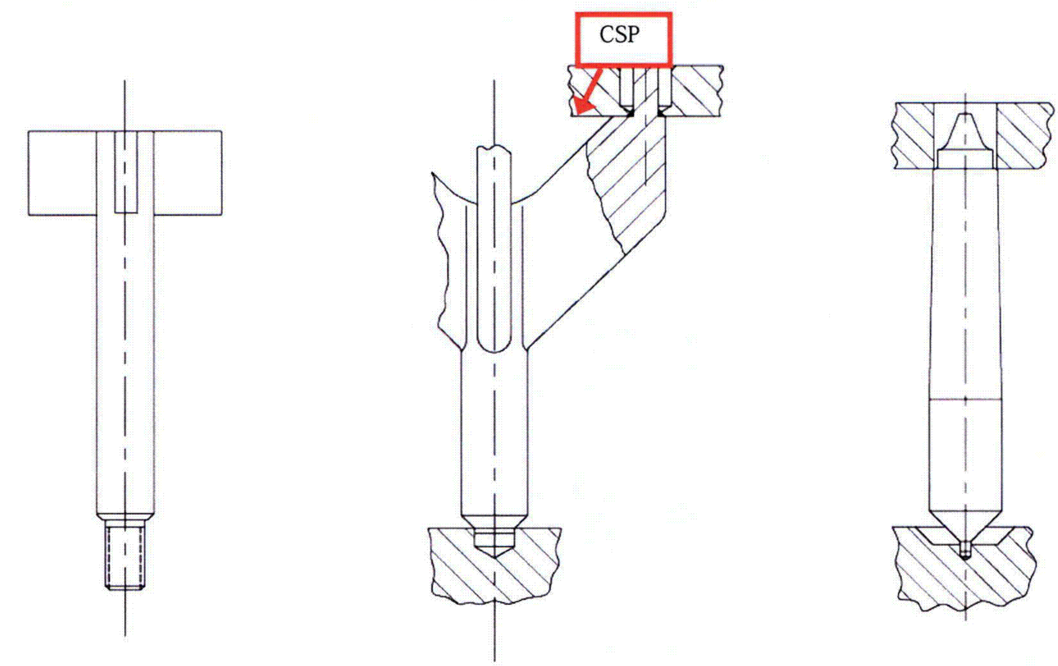


Figure 4-30
CE lower support structure assembly for plants with integrated core barrel and lower support structure with a core support plate (this design does not contain a lower core barrel flange)



a)	b)	c)

a) Early support column design
b) "Winged" support column design and plants with second generation core support assemblies
c) Later support column design used in plants with second generation core support assemblies

Figure 4-31
CE core support columns

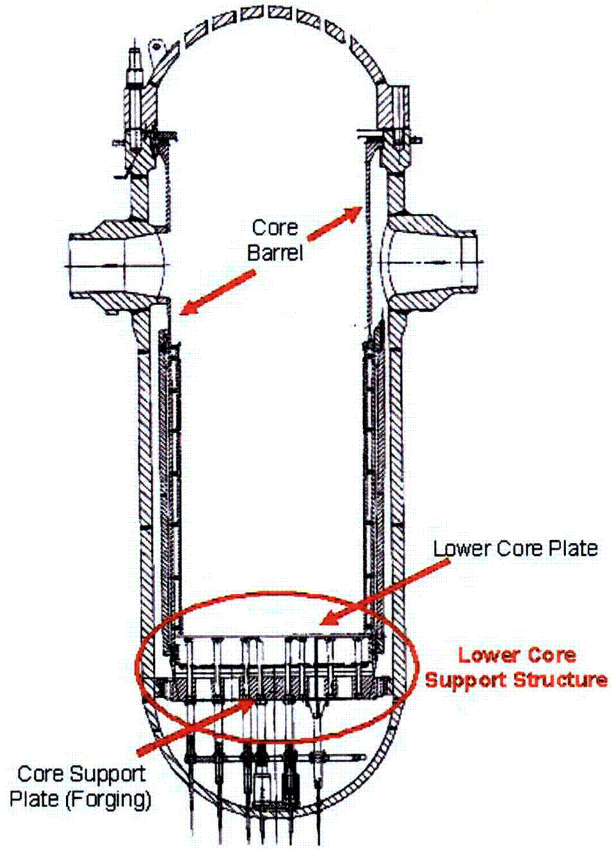


Figure 4-32
Schematic indicating location of Westinghouse lower core support structure. Additional details shown in Figure 4-33

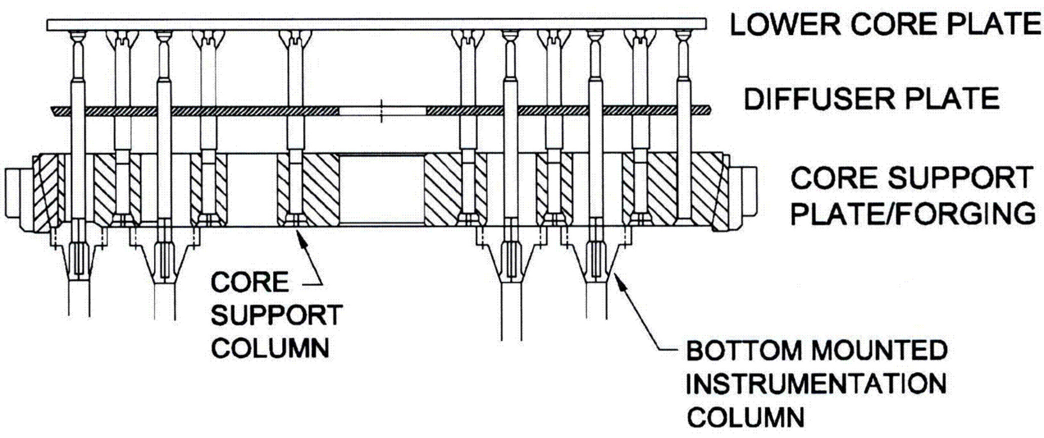


Figure 4-33
Westinghouse lower core support structure and bottom mounted instrumentation columns. Core support column bolts fasten the core support columns to the lower core plate

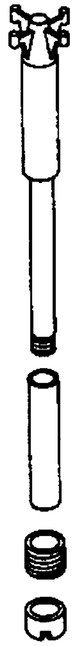


Figure 4-34
Typical Westinghouse core support column. Core support column bolts fasten the top of the support column to the lower core plate

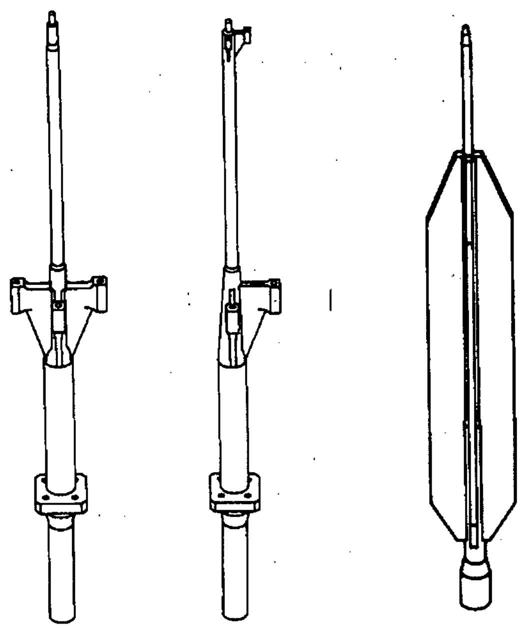


Figure 4-35
Examples of Westinghouse bottom mounted instrumentation column designs

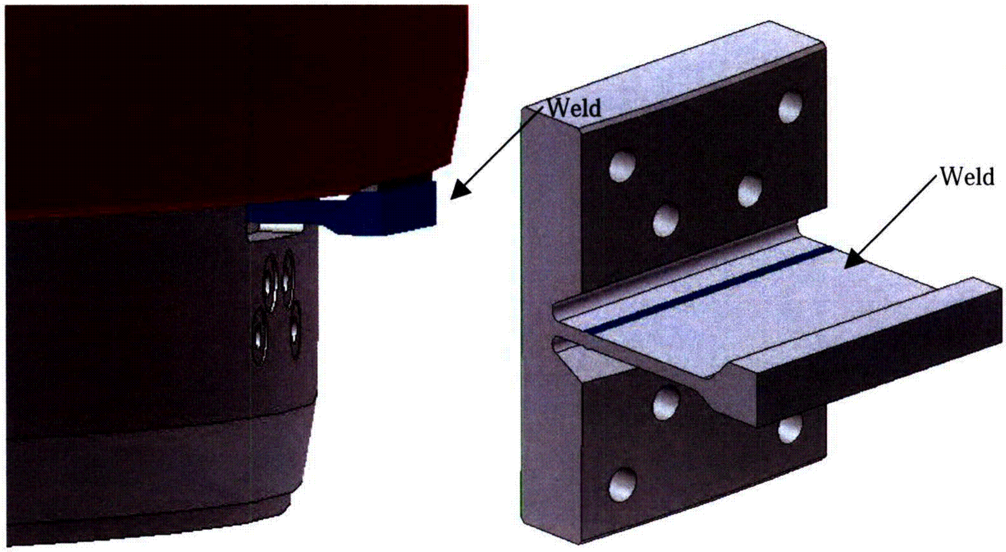


Figure 4-36
Typical Westinghouse thermal shield flexure

4.4 Existing Programs Component Requirements

Existing Programs components are those PWR internals for which current aging management activities required to maintain functionality are being implemented. The continuation of these activities is credited within these guidelines for adequate aging management for specific components.

Included in the Existing Programs are PWR internals that are classified as removable core support structures. ASME Section XI, IWB-2500, Examination Category B-N-3 [2] does not list component specific examination requirements for removable core support structures. Accordingly, factors such as original design, licensing and code of construction variability could result in significant differences in an individual plant's current B-N-3 requirements. These guidelines credit specific components contained within the general B-N-3 classification for maintaining functionality.

These examination requirements, as applied to the components designated in Tables 4-7, 4-8, and 4-9, have been determined to provide sufficient aging management for these components.

Table 4-7
B&W plants Existing Programs components

No existing generic industry programs were considered sufficient for monitoring the aging effects addressed by these guidelines for B&W plants. Therefore, no components for B&W plants were placed into the Existing Programs group.

Table 4-8
CE plants Existing Programs components

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Core Shroud Assembly Guide lugs Guide lug inserts and bolts	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination, general condition examination for detection of excessive or asymmetrical wear.	First 10-year ISI after 40 years of operation, and at each subsequent inspection interval.
Lower Support Structure Fuel alignment pins	All plants with core shrouds assembled with full-height shroud plates	Cracking (SCC, IASCC, Fatigue)	ASME Code Section XI	Visual (VT-3) examination to detect severed fuel alignment pins, missing locking tabs, or excessive wear on the fuel alignment pin nose or flange.	Accessible surfaces at specified frequency.
Lower Support Structure Fuel alignment pins	All plants with core shrouds assembled in two vertical sections	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	Accessible surfaces at specified frequency.
Core Barrel Assembly Upper flange	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	Area of the upper flange potentially susceptible to wear.

Table 4-9
Westinghouse plants Existing Programs components

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Core Barrel Assembly Core barrel flange	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination to determine general condition for excessive wear.	All accessible surfaces at specified frequency.
Upper Internals Assembly Upper support ring or skirt	All plants	Cracking (SCC, Fatigue)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
Lower Internals Assembly Lower core plate XL lower core plate (Note 1)	All plants	Cracking (IASCC, Fatigue)	ASME Code Section XI	Visual (VT-3) examination of the lower core plates to detect evidence of distortion and/or loss of bolt integrity.	All accessible surfaces at specified frequency.
Lower Internals Assembly Lower core plate XL lower core plate (Note 1)	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
Bottom Mounted Instrumentation System Flux thimble tubes	All plants	Loss of material (Wear)	NUREG-1801 Rev. 1	Surface (ET) examination.	Eddy current surface examination as defined in plant response to IEB 88-09.
Alignment and Interfacing Components Clevis insert bolts	All plants	Loss of material (Wear) (Note 2)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.
Alignment and Interfacing Components Upper core plate alignment pins	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	All accessible surfaces at specified frequency.

Notes:

1. XL = "Extra Long" referring to Westinghouse plants with 14-foot cores.
2. Bolt was screened in because of stress relaxation and associated cracking; however, wear of the clevis/insert is the issue.

Also included in Existing Programs are those components for which existing guidance has been issued (e.g., from the nuclear steam supply system (NSSS) vendors or Owners Groups) to address degradation that manifested itself during the current operational life of the PWR fleet. The continued implementation of this guidance has been determined to adequately manage the aging effects for these components.

4.4.1 B&W Components

Table 4-7 describes the PWR internals in the Existing Programs for B&W plants.

No existing generic industry programs contain the specificity considered sufficient for monitoring the aging effects addressed by these guidelines for B&W plants. Therefore, no components for B&W plants were placed into the Existing Programs group.

4.4.2 CE Components

Table 4-8 describes the PWR internals in the Existing Programs for CE plants.

The following is a list of the CE Existing Programs Components.

- *ASME Section XI*

  Existing:

  - Guide lugs and guide lug inserts and bolts (applicable to all plants)
  - Fuel alignment pins (applicable to all plants with core shrouds assembled with full-height shroud plates and all plants with core shrouds assembled in two vertical sections)
  - Upper flange (applicable to all plants)

These component items may be considered core support structures listings that are typically examined during the 10-year inservice inspection per ASME Code Section XI Table IWB-2510, B-N-3 [2]. For these component items, the requirements of B-N-3 (visual VT-3) are considered sufficient to monitor for the aging effects addressed by these guidelines.

- *Plant-specific*

The guidance for ICI thimble tubes and thermal shield positioning pins is limited to plant specific recommendations and thus have no generic reference, nor are they included in Table 4-8. The owner should review their specific design, upgrade status, and plant commitments for CE ICI thimble tubes.

4.4.3 Westinghouse Components

Table 4-9 describes the PWR internals in the Existing Programs for Westinghouse plants.

The following is a list of the Westinghouse Existing Programs Components.

- *ASME Section XI*

  Existing:

  - Core barrel flange (applicable to all plants)
  - Upper support ring or skirt (applicable to all plants)
  - Lower core plate and XL lower core plate (applicable to all plants)
  - Clevis insert bolts (applicable to all plants)
  - Upper core plate alignment pins (applicable to all plants)

These component items are considered core support structures that are typically examined during the 10-year inservice inspection per ASME Code Section XI Table IWB-2510, B-N-3 [2]. For these component items, the requirements of B-N-3 (visual VT-3) are considered sufficient to monitor for the aging effects addressed by these guidelines.

- *Plant-specific*

The guidance for flux thimble tubes is included in Table 4-9 and is based on owner commitments.

The guidance for guide tube support pins (split pins) is limited to plant specific recommendations and thus have no generic reference. Subsequent performance monitoring should follow the supplier recommendations. They thus are not included in Table 4-9. The owner should review their specific design, upgrade status, and asset management plans for Westinghouse guide tube support pins (split pins).

4.5 No Additional Measures Components

It has been determined that no additional aging management is necessary for components in this group. In no case does this determination relieve utilities of the ASME Code Section XI [2] IWB Examination Category B-N-3 inservice inspection requirements for components from this group classified as core support structures unless specific relief is granted as allowed by 10CFR50.55a [4].

5 EXAMINATION ACCEPTANCE CRITERIA AND EXPANSION CRITERIA

The purpose of this section is to provide both examination acceptance criteria for conditions detected as a result of the examination requirements in Section 4, Tables 4-1 through 4-6, as well as criteria for expanding examinations to the Expansion components when warranted by the level of degradation detected in the Primary components.

Examination acceptance criteria identify the visual examination relevant condition(s) or signal-based level or relevance of an indication that requires formal disposition for acceptability. Based on the identified condition, and supplemental examinations if required, the disposition process results in an evaluation and determination of whether to accept the condition until the next examination or repair or replace the item. An acceptable disposition process is described in Section 6. Section 5.1 provides a discussion of relevant conditions applicable to the visual examination methods and of relevant indications applicable to the volumetric examinations employed in the guidelines. Section 5.2 provides examination acceptance criteria for physical measurements. These criteria are contained in Tables 5-1, 5-2, and 5-3 for B&W, CE, and Westinghouse plants, respectively.

Additionally, Tables 5-1, 5-2, and 5-3 contain expansion criteria for B&W, CE, and Westinghouse plants, respectively. Expansion criteria are intended to form the basis for decisions about expanding the set of components selected for examination or other aging management activity, in order to determine whether the level of degradation represented by the detected conditions has extended to other components judged to be less affected by the degradation.

Table 5-1
B&W plants examination acceptance and expansion criteria

Item	*Applicability*	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	*Expansion Criteria*	Additional Examination Acceptance Criteria
Plenum Cover Assembly & Core Support Shield Assembly Plenum cover weldment rib pads Plenum cover support flange CSS top flange	All plants	One-time physical measurement. In addition, a visual (VT-3) examination is conducted for these items. The measured differential height from the top of the plenum rib pads to the vessel seating surface shall average less than 0.004 inches compared to the as-built condition. The specific relevant condition for these items is wear that may lead to a loss of function.	None	N/A	N/A
Core Support Shield Assembly CSS cast outlet nozzles	ONS-3, DB	Visual (VT-3) examination. The specific relevant condition is evidence of surface irregularities, such as damaged or fractured nozzle material.	CRGT spacer castings	Confirmed evidence of relevant conditions for a single CSS cast outlet nozzle shall require that the VT-3 examination be expanded to include 100% of the accessible surfaces at the 4 screw locations (at every 90°) of the CRGT spacer castings by the completion of the next refueling outage.	The specific relevant condition is evidence of fractured spacers or missing screws.

Table 5-1
B&W plants examination acceptance and expansion criteria (continued)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Support Shield Assembly CSS vent valve discs	All plants	Visual (VT-3) examination. The specific relevant condition is evidence of surface irregularities, such as damaged or fractured disc material.	CRGT spacer castings	Confirmed evidence of relevant conditions in two or more CSS vent valve discs shall require that the VT-3 examination be expanded to include 100% of the accessible surfaces at the 4 screw locations (at every 90°) of the CRGT spacer castings by the completion of the next refueling outage.	The specific relevant condition is evidence of fractured spacers or missing screws.
Core Support Shield Assembly CSS vent valve top retaining ring CSS vent valve bottom retaining ring CSS vent valve disc shaft or hinge pin	All plants	Visual (VT-3) examination. The specific relevant condition is evidence of damaged or fractured material, and missing items.	None	N/A	N/A

Table 5-1
B&W plants examination acceptance and expansion criteria (continued)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Support Shield Assembly Upper core barrel (UCB) bolts and their locking devices	All plants	1) Volumetric (UT) examination of the UCB bolts. The examination acceptance criteria for the UT of the UCB bolts shall be established as part of the examination technical justification. 2) Visual (VT-3) examination of the UCB bolt locking devices. The specific relevant condition for the VT-3 of the UCB bolt locking devices is evidence of broken or missing bolt locking devices.	LCB bolts (Note 2) UTS, LTS, and FD bolts SSHT bolts (CR-3 and DB only) Lower grid shock pad bolts (TMI-1 only)	1) Confirmed unacceptable indications exceeding 10% of the UCB bolts shall require that the UT examination be expanded by the completion of the next refueling outage to include: *For all plants* 100% of the accessible UTS, LTS, and FD bolts, *Additionally for TMI-1* UT examination to include 100% of the accessible lower grid shock pad bolts, *Additionally for CR-3 and DB* UT examination to include 100% of the accessible SSHT bolts. 2) Confirmed evidence of relevant conditions exceeding 10% of the UCB bolt locking devices shall require that the VT-3 examination be expanded by the completion of the next refueling outage to include: *For all plants* 100% of the accessible UTS, LTS, and FD bolt locking devices, *Additionally for TMI-1* 100% of the accessible lower grid shock pad bolt locking devices, *Additionally for CR-3 and DB* 100% of the accessible SSHT bolt locking devices.	1) The examination acceptance criteria for the UT of the expansion bolting shall be established as part of the examination technical justification. 2) The specific relevant condition for the expansion of the VT-3 locking devices is evidence of broken or missing bolt locking devices.

Table 5-1
B&W plants examination acceptance and expansion criteria (continued)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Lower core barrel (LCB) bolts and their locking devices	All plants	1) Volumetric (UT) examination of the LCB bolts. The examination acceptance criteria for the UT of the LCB bolts shall be established as part of the examination technical justification. 2) Visual (VT-3) examination of the LCB bolt locking devices. The specific relevant condition for the VT-3 of the LCB bolt locking devices is evidence of broken or missing bolt locking devices.	UTS, LTS, and FD bolts SSHT bolts (CR-3 and DB only) Lower grid shock pad bolts (TMI-1 only)	1) Confirmed unacceptable indications exceeding 10% of the LCB bolts shall require that the UT examination be expanded by the completion of the next refueling outage to include: *For all plants* 100% of the accessible UTS, LTS, and FD bolts, *Additionally for TMI-1* 100% of the accessible lower grid shock pad bolts, *Additionally for CR-3 and DB* 100% of the accessible SSHT bolts. 2) Confirmed evidence of relevant conditions exceeding 10% of the LCB bolt locking devices shall require that the VT-3 examination be expanded by the completion of the next refueling outage to include: *For all plants* 100% of the accessible UTS, LTS, and FD bolt locking devices, *Additionally for TMI-1* 100% of the accessible lower grid shock pad bolt locking devices, *Additionally for CR-3 and DB,* 100% of the accessible SSHT bolt locking devices.	1) The examination acceptance criteria for the UT of the expansion bolting shall be established as part of the examination technical justification. 2) The specific relevant condition for the expansion of the VT-3 of the locking devices is evidence of broken or missing bolt locking devices.

Table 5-1
B&W plants examination acceptance and expansion criteria (continued)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Baffle-to-former bolts	All plants	Baseline volumetric (UT) examination of the baffle-to-former bolts. The examination acceptance criteria for the UT of the baffle-to-former bolts shall be established as part of the examination technical justification.	Baffle-to-baffle bolts, Core barrel-to-former bolts	Confirmed unacceptable indications in greater than or equal to 5% (or 43) of the baffle-to-former bolts, provided that none of the unacceptable bolts are on former elevations 3, 4, and 5, or greater than 25% of the bolts on a single former plate, shall require an evaluation of the internal baffle-to-baffle bolts for the purpose of determining whether to examine or replace the internal baffle-to-baffle bolts. The evaluation may include external baffle-to-baffle bolts and core barrel-to-former bolts for the purpose of determining whether to replace them.	N/A
Core Barrel Assembly Baffle plates	All plants	Visual (VT-3) examination. The specific relevant condition is readily detectable cracking in the baffle plates.	a. Former plates b. Core barrel cylinder (including vertical and circumferential seam welds)	a and b. Confirmed cracking in multiple (2 or more) locations in the baffle plates shall require expansion, with continued operation of former plates and the core barrel cylinder justified by evaluation or by replacement by the completion of the next refueling outage.	a and b. N/A

Table 5-1
B&W plants examination acceptance and expansion criteria (continued)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	All plants	Visual (VT-3) examination. The specific relevant condition is missing, non-functional, or removed locking devices.	Locking devices for the external baffle-to-baffle bolts and barrel-to-former bolts	Confirmed relevant conditions in greater than or equal to 1% (or 11) of the baffle-to-former or internal baffle-to-baffle bolt locking devices shall require an evaluation of the external baffle-to-baffle and core barrel-to-former bolt locking devices for the purpose of determining continued operation or replacement.	N/A
Lower Grid Assembly Alloy X-750 dowel-to-guide block welds	All plants	Initial visual (VT-3) examination. The specific relevant condition is separated or missing locking weld, or missing dowel.	Alloy X-750 dowel locking welds to the upper and lower fuel assembly support pads	Confirmed evidence of relevant conditions at two or more locations shall require that the VT-3 examination be expanded to include the Alloy X-750 dowel locking welds to the upper and lower fuel assembly support pads by the completion of the next refueling outage.	The specific relevant condition is separated or missing locking weld, or missing dowel.

Table 5-1
B&W plants examination acceptance and expansion criteria (continued)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Incore Monitoring Instrumentation (IMI) Guide Tube Assembly IMI guide tube spiders IMI guide tube spider-to-lower grid rib section welds	All plants	Initial visual (VT-3) examination. The specific relevant conditions for the IMI guide tube spiders are fractured or missing spider arms. The specific relevant conditions for the IMI spider-to-lower grid rib section welds are separated or missing welds.	a. CRGT spacer castings b. Lower fuel assembly support pad items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screw, and their locking welds	a. Confirmed evidence of relevant conditions for two or more IMI guide tube spider locations shall require that the VT-3 examination be expanded to include 100% of the accessible surfaces at the 4 screw locations (at every 90°) of the CRGT spacer castings by the completion of the next refueling outage. b. Confirmed evidence of relevant conditions at two or more IMI guide tube spider locations or IMI guide tube spider-to-lower grid rib section welds shall require that the VT-3 examination be expanded to include lower fuel assembly support pad items by the completion of the next refueling outage.	a. For the CRGT spacer castings, the specific relevant conditions are fractured spacers or missing screws. b. For the lower fuel assembly support pad items (pads, pad-to-rib section welds, Alloy X-750 dowels, cap screws, and their locking welds), the specific relevant conditions are separated or missing welds, missing support pads, dowels, cap screws and locking welds, or misalignment of the support pads.

Notes:

1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s).
2. Expansion to LCB applies if the required Primary examination of LCB has not been performed as scheduled in Table 4-1.

Table 5-2
CE plants examination acceptance and expansion criteria

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Shroud Assembly (Bolted) Core shroud bolts	Bolted plant designs	Volumetric (UT) examination. The examination acceptance criteria for the UT of the core shroud bolts shall be established as part of the examination technical justification.	a. Core support column bolts b. Barrel-shroud bolts	a. Confirmation that >5% of the core shroud bolts in the four plates at the largest distance from the core contain unacceptable indications shall require UT examination of the lower support column bolts barrel within the next 3 refueling cycles. b. Confirmation that >5% of the core support column bolts contain unacceptable indications shall require UT examination of the barrel-shroud bolts within the next 3 refueling cycles.	a and b. The examination acceptance criteria for the UT of the core support column bolts and barrel-shroud bolts shall be established as part of the examination technical justification.
Core Shroud Assembly (Welded) Core shroud plate-former plate weld	Plant designs with core shrouds assembled in two vertical sections	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Remaining axial welds	Confirmation that a surface-breaking indication > 2 inches in length has been detected and sized in the core shroud plate-former plate weld at the core shroud re-entrant corners (as visible from the core side of the shroud), within 6 inches of the central flange and horizontal stiffeners, shall require EVT-1 examination of all remaining axial welds by the completion of the next refueling outage.	The specific relevant condition is a detectable crack-like surface indication.

Table 5-2
CE plants examination acceptance and expansion criteria (continued)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Shroud Assembly (Welded) Shroud plates	Plant designs with core shrouds assembled with full-height shroud plates	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	a. Remaining axial welds b. Ribs and rings	a. Confirmation that a surface-breaking indication > 2 inches in length has been detected and sized in the axial weld seams at the core shroud re-entrant corners at the core mid-plane shall require EVT-1 or UT examination of all remaining axial welds by the completion of the next refueling outage. b. If extensive cracking is detected in the remaining axial welds, an EVT-1 examination shall be required of all accessible rib and ring welds by the completion of the next refueling outage.	The specific relevant condition is a detectable crack-like surface indication.

Table 5-2
CE plants examination acceptance and expansion criteria (continued)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Shroud Assembly (Bolted) Assembly	Bolted plant designs	Visual (VT-3) examination. The specific relevant conditions are evidence of abnormal interaction with fuel assemblies, gaps along high fluence shroud plate joints, and vertical displacement of shroud plates near high fluence joints.	None	N/A	N/A
Core Shroud Assembly (Welded) Assembly	Plant designs with core shrouds assembled in two vertical sections	Visual (VT-1) examination. The specific relevant condition is evidence of physical separation between the upper and lower core shroud sections.	None	N/A	N/A

Table 5-2
CE plants examination acceptance and expansion criteria (continued)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Support Barrel Assembly Upper (core support barrel) flange weld	All plants	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Remaining core barrel assembly welds beginning with: a. lower flange weld, followed by: b. remaining accessible core barrel assembly welds, and c. core support column welds (cast)	a. Confirmation that a surface-breaking indication >2 inches in length has been detected and sized in the upper flange weld shall require that an EVT-1 examination of the lower flange weld be performed by the completion of the next refueling outage. b. Confirmation that a surface-breaking indication >2 inches in length has been detected and sized in the lower flange weld shall require an EVT-1 examination of all remaining accessible core barrel assembly welds by the completion of the next refueling outage. c. Confirmation of cracking in any of the remaining accessible core barrel assembly welds shall require a VT-3 examination of cast core support column welds, taking into account the general compressive loading of these columns and the potential for thermal aging embrittlement of the castings.	a and b. The specific relevant condition is a detectable crack-like surface indication. c. The specific relevant condition is damaged or fractured material of the cast core support column welds.

Table 5-2
CE plants examination acceptance and expansion criteria (continued)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	*Additional Examination* Acceptance Criteria
Core Support Barrel Assembly Lower flange weld	All plants	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like indication.	None	N/A	N/A
Lower Support Structure Core support plate	All plants with a core support plate	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	None	N/A	N/A
Upper Internals Assembly Fuel alignment plate	All plants with core shrouds assembled with full-height shroud plates	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	None	N/A	N/A

Table 5-2
CE plants examination acceptance and expansion criteria (continued)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Control Element Assembly Instrument guide tubes	All plants with instruments tubes in the CEA shroud assembly	Visual (VT-3) examination. The specific relevant conditions are missing supports and separation at the welded joint between the tubes and the supports.	Remaining instrument tubes within the CEA shroud assemblies	Confirmed evidence of missing supports or separation at the welded joint between the tubes and supports shall require the visual (VT-3) examination to be expanded to the remaining instrument tubes within the CEA shroud assemblies by completion of the next refueling outage.	The specific relevant conditions are missing supports and separation at the welded joint between the tubes and the supports.
Lower Support Structure Deep beams	All plants with core shrouds assembled with full-height shroud plates	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like indication.	None	N/A	N/A

Notes:
1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s).

Table 5-3
Westinghouse plants examination acceptance and expansion criteria

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Control Rod Guide Tube Assembly Guide plates (cards)	All plants	Visual (VT-3) examination. The specific relevant condition is wear that could lead to loss of control rod alignment and impede control assembly insertion.	None	N/A	N/A
Control Rod Guide Tube Assembly Lower flange welds	All plants	Enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	a. Bottom-mounted instrumentation (BMI) column bodies b. Lower support column bodies (cast)	a. Confirmation of surface-breaking indications in two or more CRGT lower flange welds, combined with flux thimble insertion/withdrawal difficulty, shall require visual (VT-3) examination of BMI column bodies by the completion of the next refueling outage. b. Confirmation of surface-breaking indications in two or more CRGT lower flange welds shall require EVT-1 examination of cast lower support column bodies within three fuel cycles following the initial observation.	a. For BMI column bodies, the specific relevant condition for the VT-3 examination is completely fractured column bodies. b. For cast lower support column bodies, the specific relevant condition is a detectable crack-like surface indication.

Table 5-3
Westinghouse plants examination acceptance and expansion criteria (continued)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Upper core barrel flange weld	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	a. Remaining core barrel welds b. Lower support column bodies (non cast)	a. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel flange weld shall require that the EVT-1 examination, and any supplementary UT examination, be expanded to include the core barrel-to-support plate weld by the completion of the next refueling outage. If extensive confirmed indications in the core barrel-to-support plate weld are detected, further expansion of the EVT-1 examination shall include the remaining core barrel assembly welds. b. If extensive cracking in the remaining core barrel welds is detected, EVT-1 examination shall be expanded to include the upper six inches of the accessible surfaces of the non-cast lower support column bodies within three fuel cycles following the initial observation.	a and b. The specific relevant condition is a detectable crack-like surface indication.

Table 5-3
Westinghouse plants examination acceptance and expansion criteria (continued)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Baffle-Former Assembly Baffle-edge bolts	All plants with baffle-edge bolts	Visual (VT-3) examination. The specific relevant conditions are missing or broken locking devices, failed or missing bolts, and protrusion of bolt heads.	None	N/A	N/A
Baffle-Former Assembly Baffle-former bolts	All plants	Volumetric (UT) examination. The examination acceptance criteria for the UT of the baffle-former bolts shall be established as part of the examination technical justification.	a. Lower support column bolts b. Barrel-former bolts	a. Confirmation that more than 5% of the baffle-former bolts actually examined on the four baffle plates at the largest distance from the core (presumed to be the lowest dose locations) contain unacceptable indications shall require UT examination of the lower support column bolts within the next three fuel cycles. b. Confirmation that more than 5% of the lower support column bolts actually examined contain unacceptable indications shall require UT examination of the barrel-former bolts.	a and b. The examination acceptance criteria for the UT of the lower support column bolts and the barrel-former bolts shall be established as part of the examination technical justification.

Table 5-3
Westinghouse plants examination acceptance and expansion criteria (continued)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Baffle-Former Assembly Assembly	All plants	Visual (VT-3) examination. The specific relevant conditions are evidence of abnormal interaction with fuel assemblies, gaps along high fluence shroud plate joints, vertical displacement of shroud plates near high fluence joints, and broken or damaged edge bolt locking systems along high fluence baffle plate joints.	None	N/A	N/A
Alignment and Interfacing Components Internals hold down spring	All plants with 304 stainless steel hold down springs	Direct physical measurement of spring height. The examination acceptance criterion for this measurement is that the remaining compressible height of the spring shall provide hold-down forces within the plant-specific design tolerance.	None	N/A	N/A

Table 5-3
Westinghouse plants examination acceptance and expansion criteria (continued)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields	Visual (VT-3) examination. The specific relevant conditions for thermal shield flexures are excessive wear, fracture, or complete separation.	None	N/A	N/A

Notes:
1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s).

5.1 Examination Acceptance Criteria

5.1.1 Visual (VT-3) Examination

Visual (VT-3) examination has been determined to be an appropriate NDE method for the detection of general degradation conditions in many of the susceptible components. The ASME Code Section XI, Examination Category B-N-3 [2], provides a set of relevant conditions for the visual (VT-3) examination of removable core support structures in IWB-3520.2. These are:

1. structural distortion or displacement of parts to the extent that component function may be impaired;
2. loose, missing, cracked, or fractured parts, bolting, or fasteners;
3. corrosion or erosion that reduces the nominal section thickness by more than 5%;
4. wear of mating surfaces that may lead to loss of function; and
5. structural degradation of interior attachments such that the original cross-sectional area is reduced more than 5%.

For components in the Existing Programs group, these general relevant conditions are sufficient. However, for components where visual (VT-3) is specified in the Primary or the Expansion group, more specific descriptions of the relevant conditions are provided in Tables 5-1, 5-2, and 5-3 for the benefit of the examiners. Typical examples are "fractured material" and "completely separated material." One or more of these specific relevant condition descriptions may be applicable to the Primary and Expansion components listed in Tables 5-1, 5-2, and 5-3.

The examination acceptance criteria for components requiring visual (VT-3) examination is thus the absence of the relevant condition(s) specified in Tables 5-1, 5-2, and 5-3.

The disposition can include a supplementary examination to further characterize the relevant condition, an engineering evaluation to show that the component is capable of continued operation with a known relevant condition, or repair/replacement to remediate the relevant condition.

5.1.2 Visual (VT-1) Examination

Visual (VT-1) examination is defined in the ASME Code Section XI [2] as an examination "conducted to detect discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, corrosion, or erosion." For these guidelines VT-1 has only been selected to detect distortion as evidenced by small gaps between the upper-to-lower mating surfaces of CE welded core shrouds assembled in two vertical sections.

The examination acceptance criterion is thus the absence of the relevant condition of gaps that would be indicative of distortion from void swelling.

5.1.3 Enhanced Visual (EVT-1) Examination

Enhanced visual (EVT-1) examination has the same requirements as the ASME Code Section XI [2] visual (VT-1) examination, with additional requirements given in the Inspection Standard [3]. These enhancements are intended to improve the detection and characterization of discontinuities taking into account the remote visual aspect of reactor internals examinations. As a result, EVT-1 examinations are capable of detecting small surface breaking cracks and surface crack length sizing when used in conjunction with sizing aids (e.g. landmarks, ruler, and tape measure). EVT-1 examination has been selected to be the appropriate NDE method for detection of cracking in plates or their welded joints. Thus the relevant condition applied for EVT-1 examination is the same as found for cracking in Reference 2 which is crack-like surface breaking indications.

Therefore, until such time as generic engineering studies develop the basis by which a quantitative amount of degradation can be shown to be tolerable for the specific component, any relevant condition is to be dispositioned. In the interim, the examination acceptance criterion is thus the absence of any detectable surface breaking indication.

5.1.4 Surface Examination

Surface ET (eddy current) examination is specified as an alternative or as a supplement to visual examinations. No specific acceptance criteria for surface (ET) examination of PWR internals locations are provided in the ASME Code Section XI [2]. Since surface ET is employed as a signal-based examination, a technical justification per the Inspection Standard [3] provides the basis for detection and length sizing of surface-breaking or near-surface cracks. The signal-based relevant indication for surface (ET) is thus the same as the relevant condition for enhanced visual (EVT-1) examination. The acceptance criteria for enhanced visual (EVT-1) examinations in 5.1.3 (and accompanying entries in Tables 5-1, 5-2, and 5-3) are therefore applied when this method is used as an alternative or supplement to visual examination.

5.1.5 Volumetric Examination

The intent of volumetric examinations specified for bolts in Section 4.3 of these I&E guidelines is to detect planar defects. No flaw sizing measurements are recorded or assumed in the acceptance or rejection of individual bolts or pins. Individual bolts or pins are accepted based on the detection of relevant indications established as part of the examination technical justification. When a relevant indication is detected in the cross-sectional area of the bolt or pin, it is assumed to be non-functional and the indication is recorded. A bolt or pin that passes the criterion of the examination is assumed to be functional.

Because of this pass/fail acceptance of individual bolts or pins, the examination acceptance criterion for volumetric (UT) examination of bolts and pins is based on a reliable detection of indications as established by the individual technical justification for the proposed examination. This is in keeping with current industry practice. For example, planar flaws on the order of 30% of the cross-sectional area have been demonstrated to be reliably detectable in previous bolt NDE technical justifications for baffle-former bolting.

Bolted and pinned assemblies are evaluated for acceptance based on meeting a specified number and distribution of functional bolts and pins. As discussed in Section 6.4, criteria for this evaluation can be: 1) found in previous Owners Group reports, 2) developed for use by the PWROG or 3) developed on a plant-specific basis by the applicable NSSS vendor.

5.2 Physical Measurements Examination Acceptance Criteria

Continued functionality can be confirmed by physical measurements where, for example, loss of material caused by wear, loss of pre-load of clamping force caused by various degradation mechanisms, or distortion/deflection caused by void swelling may occur. Where appropriate, these physical measurements are described in Section 4.3, with limits applicable to the various designs. For B&W designs, the acceptable tolerance for the measured differential height from the top of the plenum rib pads to the vessel seating surface has been generically established and is provided in Table 5-1. For Westinghouse designs, tolerances are available on a design or plant-specific basis and thus are not provided generically in these guidelines. For CE designs, no physical measurements are specified.

5.3 Expansion Criteria

The criteria for expanding the scope of examination from the Primary components to their linked Expansion components is contained in Tables 5-1, 5-2, and 5-3 for B&W, CE, and Westinghouse plants, respectively. The logic and basis for the levels of degradation warranting expansion is documented in an MRP letter [15].

6 EVALUATION METHODOLOGIES

There are various options that are available for the disposition of conditions detected during examinations (Section 4) that are unable to satisfy the examination acceptance criteria (Section 5). These options include, but are not limited to: (1) supplemental examinations, such as a surface examination, to supplement a visual (VT-1) or an enhanced visual (EVT-1) examination, to further characterize and potentially dispose of a detected condition; (2) engineering evaluation that demonstrates the acceptability of a detected condition; (3) repair, in order to restore a component with a detected condition to acceptable status; or (4) replacement of a component with an unacceptable detected condition.

The first option involves the re-examination of a component with an unacceptable detected condition with an alternative examination method that has the potential capability to further define or confirm with greater precision the component physical condition. This additional characterization may enable the more precise character of that detected condition to be found acceptable for continued service. An example would be the volumetric (UT) examination to depth size a surface-breaking flaw detected by either visual (VT-1) or enhanced visual (EVT-1) examination.

Section 6 concentrates on the second option, evaluation methodologies that can be used for evaluating flaws detected during the examinations described in Section 4 that exceed the examination acceptance criteria described in Section 5. The evaluation process depends upon the loading applied to the component, assembly, or system. Typical loading information to be considered is provided in Section 6.1 and evaluation methodology options are described in subsequent sections. These methodologies range from the satisfaction of limit load requirements for the internals assembly or component cross section to the satisfaction of flaw stability requirements using either linear elastic fracture mechanics (LEFM) or elastic-plastic fracture mechanics (EPFM), depending upon applicability. In addition, recommendations for flaw depth assumptions, in the absence of flaw depth sizing during examination, and flaw growth assumptions for subsequent operation until the next examination, are described. Justification for flaw evaluation fracture toughness limits is also provided. Design-specific or fleet-specific flaw handbooks may be used as an engineering evaluation tool.

6.1 Loading Conditions

The purpose of this section is to describe the typical loading conditions that govern the evaluation of flaws exceeding the examination acceptance criteria of Section 5.

Core support structures are designed to a set of defined loading conditions that typically include deadweight, such as the weight of the structure itself and an assigned portion of the weight of the fuel assemblies; mechanical loads, such as fuel assembly spring forces and control rod actuation loads; hydraulic loads; loadings caused by flow-induced vibration; loss-of-coolant accident (LOCA) loads; thermal loads, such as those from both normal operation thermal transients and upset condition thermal transients, as well as gamma heating; operating basis earthquake (OBE) and safe shutdown earthquake (SSE) seismic loads; handling loads that might occur during refueling and internals removal for inservice examinations; and interference conditions, friction forces, and dynamic insertion loads. Confirmation of required loading and combination requirements on an individual plant basis is essential prior to conducting any assessment.

For the case of many bolts and pins, the defined loading conditions include interference conditions, friction forces due to differential thermal growth, and dynamic insertion loads, in addition to dead weight, seismic, and vibration loadings.

The loading conditions for internal structures that are not core support structures are less well documented publicly. However, should an engineering evaluation be required for any internals structure (both core support structures and other internals), the original design basis should be examined, in order to determine the availability of actual or potential loading conditions.

6.2 Evaluation Requirements

The evaluation of component conditions that do not satisfy the examination acceptance criteria of Section 5 must be performed for a future state that corresponds to the next required examination or later. This future state should be determined based on the observed condition and a projection of future condition based on progressing degradation. The progressing degradation estimate should be based on a combination of operating experience (bolt failure histories), applicable testing data (crack growth rates in plate material), and available analytical results for that component. Uncertainties in predictive measures should be considered where applicable. Options for performing evaluations are contained in the following sub-sections.

6.2.1 Limit Load Evaluation

Evaluation Requirement

An assembly or component that cannot meet the examination acceptance criteria of Section 5 of these I&E guidelines may be subject to limit load requirements as an evaluation disposition option, in order to continue in service in the existing condition. For PWR internals, the threshold for limit load requirements only is based on the accumulated neutron fluence exposure identified in BWRVIP-100-A [19]. This requirement states that, for accumulated neutron fluence less than $3x10^{20}$ n/cm^2 (E > 1 MeV), or approximately 0.5 dpa, only a limit load evaluation requirement must be met for continued service of the internals assembly or individual component. A discussion and explanation of this requirement is contained in the following paragraphs.

Discussion and Explanation

Irrespective of the level of neutron irradiation exposure, limit load requirements can be satisfied for the affected assembly or component, in order to continue service until the end of the current inservice inspection interval. Therefore, the affected assembly or component can be shown to satisfy limit load requirements which may follow procedures similar to those given in the ASME Code Section XI, Appendix C [20]. The limit load calculation is carried out to find the critical degree of degradation within the elements of the assembly, or the progress of flaw parameters (location of the remaining cross section neutral axis and the effective flaw length) that cause the cross section to reach its limit load. For austenitic stainless steel, the stress limits for primary loading may be based on the irradiated mechanical strength properties for the minimum estimated fluence accumulated at the loaded section.

A safety factor of 2.77 on the limit load for expected loadings (ASME Service Loadings A and B) and a safety factor of 1.39 on the limit load for unexpected loadings (ASME Service Loadings C and D) must be met for the applied load on the assembly, or on the membrane and bending stresses in the component. The component analysis must demonstrate that a plastic hinge does not form in the remaining ligament of the cross section. For sections that have relatively uniform loss of material, and for unflawed sections that experience increased loading due to failure in other sections, the limiting primary stress and deflections for ASME Level C and D combinations should meet the plant design basis, or alternatively, meet the requirements of ASME Section III, Appendix F [21].

If the neutron fluence exposure is less than 3×10^{20} n/cm^2 (E > 1 MeV), or approximately 0.5 dpa, this is the only evaluation that needs to be met for acceptance of the PWR internals assembly or individual component. No fracture toughness requirements need to be met for neutron fluence exposures less than this value.

6.2.2 Fracture Mechanics Evaluation

For neutron fluence levels exceeding 0.5 dpa, either an elastic-plastic fracture mechanics (EPFM) evaluation or a linear elastic fracture mechanics (LEFM) evaluation must be performed to assure continued structural integrity in the presence of detected flaws that exceed the examination acceptance criteria of Section 5. For neutron fluence above 0.5 dpa and below 5 dpa, EPFM is the preferred method. For neutron fluence above 5 dpa, LEFM should be utilized. Non-mandatory Appendix C of the ASME Code Section XI [20] provides general guidance which may be followed for performing such evaluations. Although the appendix strictly applies to austenitic stainless steel piping, the discussion of flaw growth due to fatigue, or due to stress corrosion cracking (SCC), or due to a combination of the two is relevant. Note, however, that fatigue crack growth rates in Article C-8000 are limited to air environments only, and that fatigue crack growth in water environments and SCC crack growth rates are not available yet.

For the case of IASCC, considerable research has been conducted on the effects of various levels of irradiation exposure on crack growth resistance, primarily by the Boiling Water Reactor Vessel & Internals Project (BWRVIP) [19]. Reference 19 also provides the technical basis for the recommendation of either LEFM or EPFM. Figure 6-1, reproduced from Reference 19, shows the data that were used to produce a set of conservative J-R curves (crack growth resistance curves) for various exposure levels. Figures 6-2 and 6-3, also reproduced from Reference 19, show the lower bound for the power law parameter, C, and the upper bound for the power law parameter, n, in the curve fit to the crack growth resistance curve data given by

$$J_{mat} = C\,(\Delta a)^n$$ **Equation 6-1**

where J and C are in KJ/m^2 and Δa is in mm.

The lower bound expression for power law parameter C is given by

$$C = (1217.9*6.697*10^{10} + 0.3908*F^{0.5563})/(6.697*10^{10} + F^{0.5563})$$ **Equation 6-2**

The upper bound expression for power law parameter n is given by

$$n = 1/(4.962 - 0.02439*F^{0.09976})$$ **Equation 6-3**

The term F in the above expressions is the neutron fluence. At accumulated fluence values of approximately 1 dpa, the material has relatively high elastic-plastic crack growth resistance. For example, at 1 dpa, the upper bound power law parameter C equals 177 and the lower bound power law parameter n equals 0.492. Then, the crack growth resistance at 1.5 mm (0.059") of crack growth is 216 KJ/mm^2, or 1,609 $in\text{-}lb/in^2$. Elastic-plastic behavior would be expected at such a low fluence level.

At an accumulated fluence value of 10 dpa, C equals 55.2 and n equals 0.7833. Then, the crack growth resistance at 1.5 mm (0.059") of crack growth is 75.8 KJ/mm^2, or 565 $in\text{-}lb/in^2$. If the tangent to the crack growth resistance curve at 1.5 mm (0.059") is projected back to zero crack growth and converted to K_I through the expression

$$J_{IC} = (K_{IC})^2/E$$ **Equation 6-4**

where E is the elastic modulus, then K_{IC} equals 84 MPa√m, or about 76 ksi√in. This value of fracture toughness is in the range that would suggest that LEFM is perhaps more suitable than EPFM, even though some amount of plastic response remains.

However, at 15 dpa, C equals 44.54 and n equals 0.889, so that the crack growth resistance at 1.5 mm (0.059") of crack growth is only 64 KJ/mm^2, or 476 $in\text{-}lb/in^2$. Extrapolating the tangent of the crack growth resistance curve back to zero crack growth and converting gives K_{IC} = 55 MPa√m, or 50 ksi√in. Further analysis of more recent fracture toughness data at higher irradiation exposures for irradiated stainless steels has determined [25] that an appropriately conservative value for the fracture toughness of 38 MPa√m (34.6 ksi√in) should be used for high neutron fluence exposure.

Therefore, for fluence levels below 5 dpa, the elastic-plastic crack growth resistance curves based on Equations 6-1 to 6-3 should be used. For neutron fluence greater than 5 dpa, LEFM analyses should be used with a limiting fracture toughness K_{IC} = 55 MPa√m (50 ksi√in) for exposure levels between 5 and 15 dpa, and with a limiting fracture toughness K_{IC} = 38 MPa√m (34.6 ksi√in) for exposure levels greater than 15 dpa.

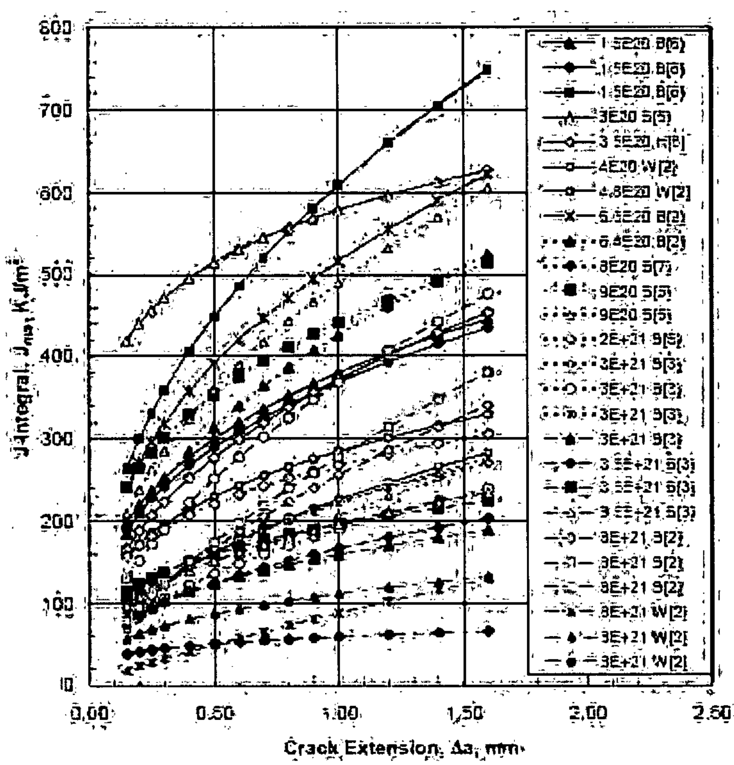


Figure 6-1
Experimental $J_{material}$ versus crack extension curves for stainless steel materials at various fluence levels [19]

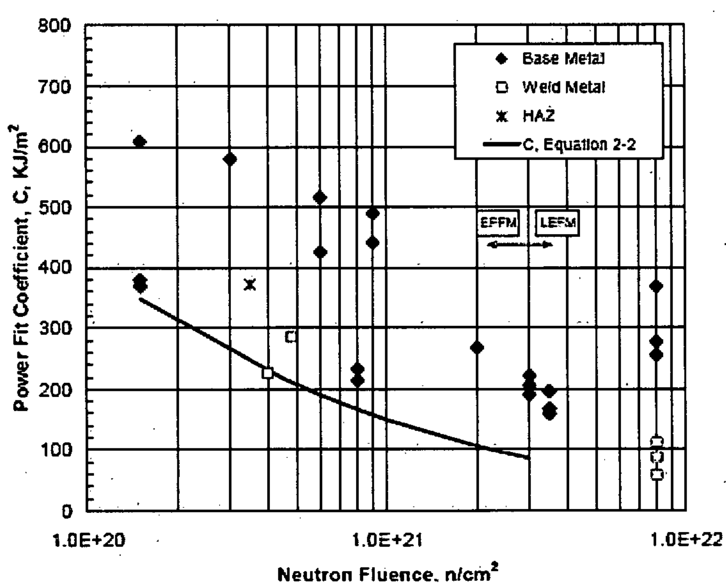


Figure 6-2
J-R curve power law parameter C as a function of neutron fluence for stainless steel, applicable for fluence less than $3x10^{21}$ n/cm^2 [19]

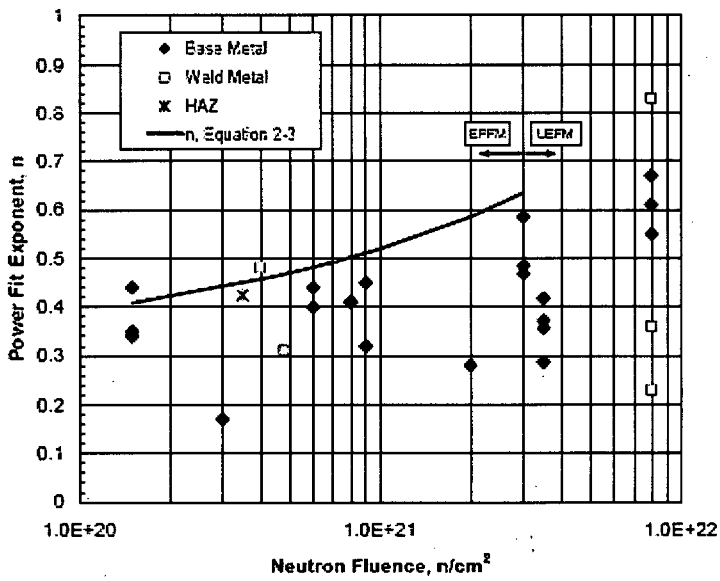


Figure 6-3
J-R curve power law parameter n as a function of neutron fluence for stainless steel, applicable for fluence less than $3x10^{21}$ n/cm^2 [19]

6.2.3 Flaw Depth Assumptions

If the flaw depth has been determined by either the primary examination or by a supplementary examination method, that flaw depth should be used in any subsequent flaw evaluation. If only the flaw length has been determined by the examination, the evaluation should be based on the assumption that the flaw extends completely through the cross section of the component. The evaluation may be based on an assumption of depth if justified by a sufficiently robust technical demonstration.

6.2.4 Crack Growth Assumptions

Prior to the limit load and fracture mechanics calculations, the cyclic and time-dependent flaw growth from the current time to the next examination must be calculated. For example, if the inservice inspection interval is ten years, the flaw growth must be calculated for a ten-year period. If the examination is a one-time examination only, the growth of the flaw to the end of component life must be calculated and shown to satisfy acceptable limits. If the end-of-period flaw exceeds limits, the inservice inspection interval should be adjusted and a subsequent inspection performed prior to exceeding the flaw limit.

In the absence of sufficient information on crack growth in relevant PWR environments, data from BWR hydrogen water chemistry (HWC) environments is the most electrochemically appropriate and readily available source. A crack growth rate of $1.1x10^{-5}$ inches per hour (2.5 mm/year) in the depth direction has been accepted by the NRC staff for BWR HWC environments in their safety evaluation of BWRVIP-14 [23]. This assumed flaw growth rate may be too conservative for a PWR water environment; therefore, the technical basis for reduced flaw growth rates is discussed in the following paragraphs.

The most recent information on flaw growth rates for irradiated austenitic stainless steels in BWR environments is provided in BWRVIP-99 [24]. The information in BWRVIP-99 is based on both laboratory data and on field measurements of crack growth rates in BWR core shroud beltline welds, as measured by ultrasonic testing. The data are considered proprietary. The major findings were that field-measured crack growth rates varied from $2x10^{-6}$ to $5.25x10^{-5}$ inches per hour (about 0.5 mm to 11 mm per year), with the crack growth rate as a function of depth much lower than the crack growth rate as a function of length. Laboratory crack growth rates depended upon electro-chemical potential (ECP), with the growth rates substantially lower in a HWC environment that is more typical of a PWR environment. The HWC crack growth rates varied from $1x10^{-7}$ to $4x10^{-5}$ inches per hour (0.02 mm to 9 mm per year). The nominal reduction in crack growth rate for the HWC environment was found to be approximately 20 times lower than the corresponding crack growth rates in nominal BWR environments. However, the scatter in the data is very large.

For HWC environments, the recommended curve is given by

$$da/dt = 2.72 \times 10^{-8} (K)^{2.5} \quad \textbf{Equation 6-5}$$

Figure 6-4 shows that this curve approximates an upper bound to the relevant laboratory HWC data.

The BWR HWC curve is seen to be representative for PWR water environments, compared to limited crack growth rate data in PWR environment shown in Figure 6-5 [25]. Therefore, the HWC curve may be used for all PWR IASCC and SCC analyses until generic curves are established for IASCC and SCC in PWR environment. The use of alternative crack growth rate correlations in any analysis must be accompanied by an appropriate technical justification.

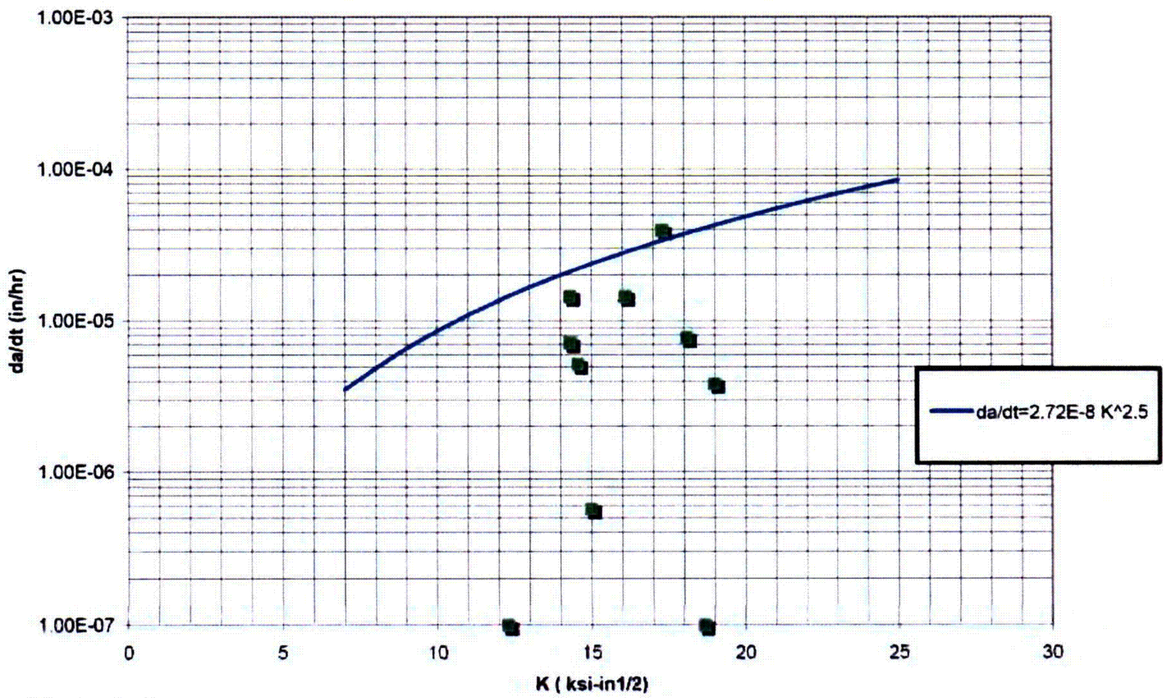


Figure 6-4
Proposed BWR hydrogen water chemistry crack growth curves for stainless steel irradiated between $5x10^{20}$ to $3x10^{21}$ n/cm^2 [24]

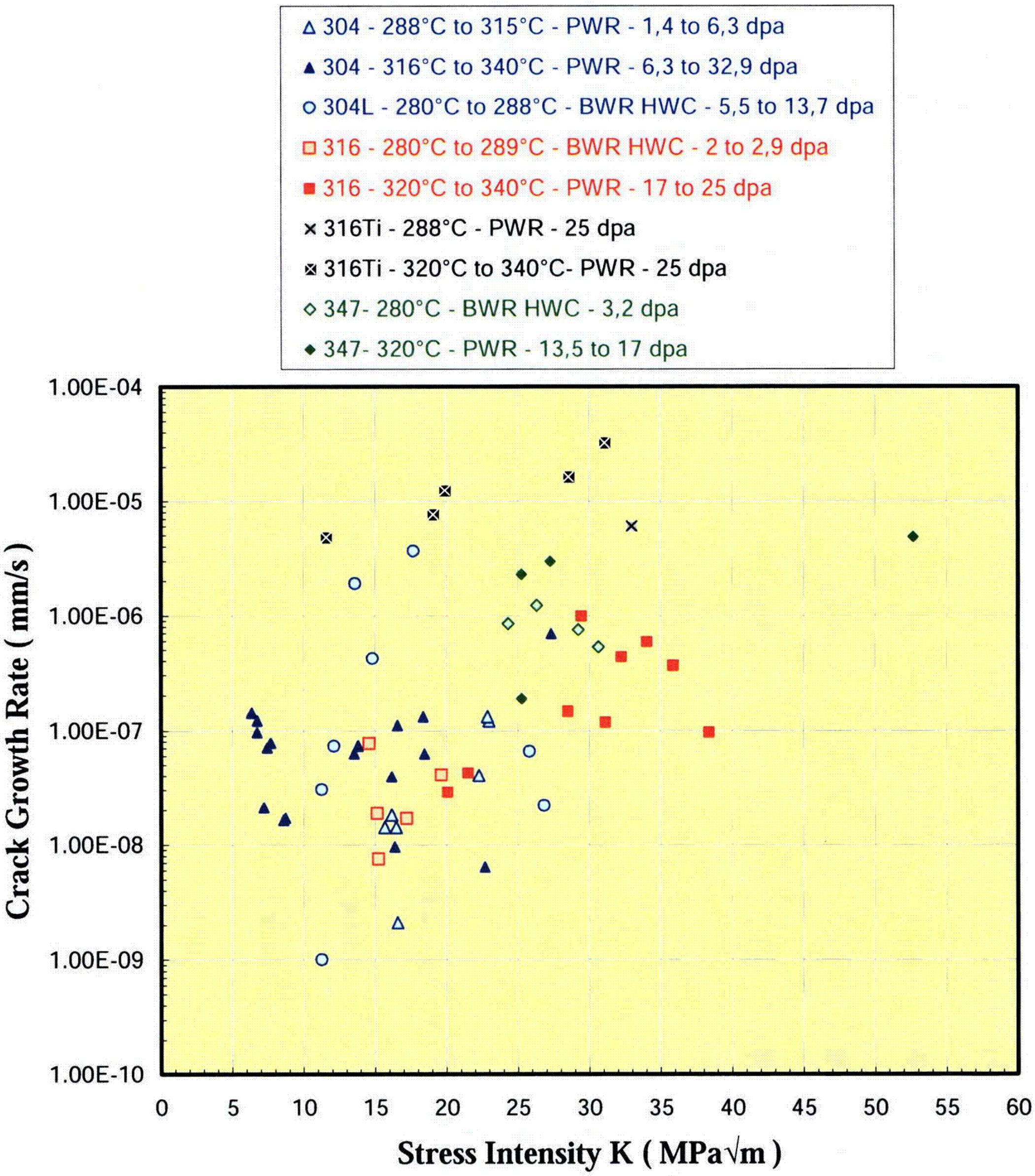


Figure 6-5
Effect of stress intensity on IASCC crack growth rate [25]

6.3 Evaluation of Flaws in Bolts and Pins

For bolts and pins, no evaluation of individual items is required. Individual bolts or pins that are found to be unacceptable during the UT examination should be assumed to be non-functional, and the acceptance criterion for continued operation of the assembly that contains one or more non-functional bolts or pins are based on the functioning of the assembly, not the individual bolt or pin. In addition, no evaluation of individual items is required where visual examinations are the basis for determining functionality of bolts, pins or locking devices. Assessments in cases where the assembly is found to be deficient are most often driven by loose parts or reassembly interference evaluations that may be resolved using standard processes to support continued operation. Typically these are part of existing plant corrective action programs and as such should be sufficient to disposition.

6.4 Assembly Level Evaluations

As indicated in Sections 5.1.5, bolts are not accepted or rejected based on flaw sizing but on flaw detection. Thus the bolted assembly must be evaluated based on the number of rejected bolts, the minimum number required for functionality and an assumed failure rate until the next examination. Assemblies that satisfy an evaluation criterion that has been established by the NSSS vendor may be dispositioned. Alternatively, an assembly level evaluation may be performed to ensure that required functionality is maintained through the period until the next examination. Essential features of this type of evaluation are described below.

A process that can be followed for those system level evaluations is provided in the following paragraphs. The process builds on the vendor functionality evaluations [11, 12]. Other approaches can also be used. The finite element models to be used for the system level evaluation could take advantage of geometric and loading symmetry. Examples of such models have been demonstrated for the B&W-designed and Westinghouse-designed baffle-former assemblies, the CE-designed core shroud assembly, and bottom core plate assemblies for different vendor designs. The bolts and pins that are elements of the assembly should be modeled in sufficient detail to capture the essential structural behavior needed to demonstrate function or the lack thereof. For example, the assumption that a particular bolt, pin, or fastener has failed can be accounted for by modeling the bolt or pin as a one-dimensional finite element with no axial or shear strength. If a particular bolt or pin is assumed to maintain at least some or most of its preload, then the representation of material strength must be appropriate. That material strength should account conservatively for the local fluence and temperature for particular bolts or pins. The geometric modeling of the bolts and pins for system level evaluations does not require the level of detail that would be needed to predict localized failure in a bolt or pin.

The number of bolts or pins that are assumed to be non-functional should bound the estimated number and pattern of non-functional bolts or pins at the end of the evaluation interval. The estimation process is beyond the scope of this document. A conservative pattern that differs from the actual observed pattern of non-functional bolts or pins may be used. The loads referred to in Section 6.1 should be applied to this assembly model, and the structural response determined. This structural response should then be compared to assembly functional requirements, and a determination should be made about the capability to continue to operate the assembly through the remainder of the inspection interval.

The precise functionality criteria for each assembly are beyond the scope of this document. Reference should be made to vendor-recommended criteria.

6.5 Evaluation of Flaws in Other Internals Structures

Reference 22 describes a methodology to be used to evaluate detected and sized flaws found in PWR internals – other than bolts or pins – that exceed the examination acceptance criteria in Section 5.1. This methodology is summarized in the following steps.

First, the neutron fluence for the component is calculated or derived from existing calculations.

Second, the applied stresses are found from either existing stress analyses or from a new stress analysis of the assembly containing the affected component location.

Third, the detected and sized flaw from the examination is applied to a representation of the geometry of interest. Reference 22 has provided a number of representative PWR internal core support geometries of interest.

Fourth, the growth of the flaw over the period of time until the next examination, or until the end of component life, as applicable, is calculated. The flaw growth calculation will depend on the active mechanism driving the flaw extension (i.e. IASCC, SCC, or fatigue). Reference 22 assumed that negligible flaw growth occurred prior to application of nominal, design-basis, and bounding loads.

Fifth, load evaluation requirements (for example, limit load) for the flawed geometry after flaw growth, subject to both expected and unexpected loads, should be met.

Sixth, applied fracture mechanics stress intensities or applied J-integrals are calculated from the combination of the stresses and the grown flaws for the representative core support geometry of interest, as applicable. LEFM solutions may be obtained from the literature, with a conversion to an elastic-plastic crack driving force valid for localized plasticity at the crack tip.

Finally, the applied fracture mechanics stress intensities or the applied J-integrals must be shown to meet the limits of Section 6.2.2. For LEFM calculations, the applied fracture mechanics stress intensity must be shown to be less than the material fracture toughness. For EPFM calculation, the evaluation procedure specified in ASME Section XI, non-mandatory Appendix K, Article K-4000, K-4220 [2], can be used to demonstrate flaw stability. Specifically, Paragraph K-4220 provides a flaw stability criterion that limits the elastic-plastic crack driving force to less than the material elastic-plastic crack growth resistance at a crack extension of 0.1 inches. The safety margin that is demonstrated in meeting the limits of Section 6.2.2 should be identified and justified for the classes of loading considered.

The methodology outlined above has been demonstrated in Reference 22, where five simple geometries were analyzed with assumed dimensions that represented a wide variety of PWR internals locations. Because of the uncertainty in the applied stresses and the conservatism of the bounding material fracture toughness, no safety margins were applied to the critical flaw size calculations. The five simple geometries analyzed are described below:

- A semi-elliptical surface crack in a flat plate that can represent: (i) a semi-elliptical surface crack at the inside or outside flat surface of baffle plates; (ii) a semi-elliptical surface crack at the inside or outside flat surface of a core support barrel; or (iii) a semi-elliptical surface crack at the inside or outside surface of a core barrel. The flaw can be either circumferential (e.g., in the circumferential weld seam of the core barrel) or longitudinal (e.g., in the vertical weld seam). A flat plate solution is adequate for these cylinders when the radius to thickness ratio (R/t) is greater than 36 and loading level is fairly low;

- A through-wall crack in the center of a plate that can represent: (i) a through-wall crack in baffle plates; (ii) a through-wall crack in the flat surface of a core support barrel; (iii) a circumferential through-wall crack (e.g. in the circumferential weld seam) in a core barrel; or (iv) a longitudinal through-wall crack (e.g. in the vertical weld seam) in a core barrel;
- A through-wall edge crack in a flat plate that can represent: (i) a through-wall crack emanating from the side edges of baffle plates; or (ii) a through-wall crack emanating from the edge of former plates;
- A through-wall edge crack emanating from a 1 and 3/8-inch diameter hole that can represent: (i) two through-wall edge cracks emanating from baffle-to-former bolt holes or cooling holes; or (ii) two through-wall edge cracks emanating from holes in former plates; and
- A quarter-circular corner crack in a rectangular bar that can represent: (i) a quarter-circular crack in the corner of baffle plates; or (ii) a quarter-circular crack at the inside corner of a core support barrel.

Although no detailed loading/stress information was available for the various geometries, limited information was used to estimate the maximum normal operating stress (2.5 ksi) and the maximum LOCA stress (10 ksi) in highly irradiated components. For completeness, however, remote tensile stress levels up to 50 ksi were analyzed.

For the three types of postulated through-wall flaws, the analyses showed that the critical flaw is more limiting for a through-wall edge crack or a through-wall edge crack emanating from a hole than for a through-wall centered crack. For a medium-width baffle plate (26-inch), the critical flaw length for a through-wall crack is 22.8 inches at 2.5 ksi and 7.62 inches at 10 ksi. For the same baffle plate, the critical flaw length for a through-wall edge crack is 11.3 inches at 2.5 ksi and 2.65 inches at 10 ksi.

7
IMPLEMENTATION REQUIREMENTS

The purpose of this section is to summarize the implementation requirements of these guidelines. These guidelines do not reduce, alter, or otherwise affect current ASME B&PV Code Section XI or plant-specific licensing inservice inspection requirements.

7.1 NEI 03-08 Implementation Protocol

These guidelines are a 'work product' of the EPRI MRP, an 'Issue Program (IP)' as defined in NEI 03-08 [1]. Addendum D to NEI 03-08, Implementation Protocol, defines the processes and expectations for implementing industry guidance issued under the Materials Initiative, and requires that IPs identify the specific implementation category for 'requirements' identified guideline-type work products.

The three implementation categories described in NEI 03-08 are as follows:

- Mandatory – to be implemented at all plants where applicable;
- Needed – to be implemented wherever possible, but alternative approaches are acceptable; and
- Good Practice – implementation is expected to provide significant operational and reliability benefits, but the extent of use is at the discretion of the individual utility.

Sections 7.2 through 7.6 list or summarize the requirements contained in this document. A failure to meet a Needed or a Mandatory requirement is a deviation from the guidelines and a written justification for the deviation must be prepared and approved as described in Addendum D to NEI 03-08 [1]. A copy of the deviation is sent to the MRP so that improvements to the guidelines can be developed.

7.2 Aging Management Program Requirement

Mandatory: *Each commercial U.S. PWR unit shall develop and document a PWR reactor internals aging management program (AMP) within thirty-six months following issuance of MRP-227-Rev. 0.*

Appendix A describes each of the attributes that comprise an acceptable AMP.

MRP-227-Rev. 0 is the first published version of these guidelines.

7.3 Reactor Internals Guidelines Implementation Requirement

Needed: *Each commercial U.S. PWR unit shall implement Tables 4-1 through 4-9 and Tables 5-1 through 5-3 for the applicable design within twenty-four months following issuance of MRP-227-A.*

Implementation of these guidelines is to take effect 24 months following issuance of MRP-227-A. MRP-227-A is the version that will have incorporated the changes proposed by the MRP in response to U.S. Nuclear Regulatory Commission (NRC) Requests for Additional Information, recommendations in the NRC Safety Evaluation and other necessary revisions identified since the previous publication of the report.

Earlier implementation may be required by plant-specific regulatory commitments (for example, license renewal approvals). Plants implementing these guidelines prior to the issuance of the "NRC-approved" version would thus implement the requirements in accordance with the current published version of these guidelines.

7.4 Examination Procedures Requirement

Needed: *Examinations specified in these guidelines shall be conducted in accordance with the Inspection Standard [3].*

7.5 Examination Results Requirement

Needed: *Examination results that do not meet the examination acceptance criteria defined in Section 5 of these guidelines shall be recorded and entered in the plant corrective action program and dispositioned.*

7.6 Aging Management Program Results Requirement

Good Practice: *Each commercial U.S. PWR unit should provide a summary report of all inspections and monitoring, items requiring evaluation, and new repairs to the MRP Program Manager within 120 days of the completion of an outage during which PWR internals are examined. The MRP template should be used for the report.*

This summary of the results will be compiled into an overall industry report which will track industry progress, aid in evaluation of significant issues, identification of fleet trends and determination of any needed revisions to these guidelines. The industry report will be updated biennially for the benefit of the fleet, the regulator, the PWROG and other industry stakeholders. This biennial report will serve to assist in review of operating experience, and required monitoring and trending for aging management programs established by the industry. In order to ensure completeness and consistency of reporting, the MRP will provide a template listing the requested information.

8 REFERENCES

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14. *Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals (MRP-232).* EPRI, Palo Alto, CA: 2008. 1016593.

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21. *ASME Boiler & Pressure Vessel Code, Section III, Division 1, Nonmandatory Appendices, Appendix F, "Rules for Evaluation of Service Loadings With Level D Service Limits,"* American Society of Mechanical Engineers, New York, NY, 2004 Edition, July 1, 2004.

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A
AGING MANAGEMENT PROGRAM ATTRIBUTES

A.1 Program Description

An aging management program based upon inspection and evaluation in conformance with the Materials Reliability Program (MRP) Inspection & Evaluation (I&E) guidelines (hereafter referred to as "the I&E guidelines") for PWR internals is intended to ensure the long-term integrity and safe operation of these components. In the following paragraphs, each of the ten attributes that comprise an acceptable aging management program are described, as given in NUREG-1801 [A1]. Some early license renewal plants have committed to an earlier version of the Generic Aging Lessons Learned (GALL) report (Revision 0) that describes the attributes for PWR internals specifically. Revision 1 of the GALL report [A1] does not specifically provide an AMP for PWR internals.

The attributes for an AMP are summarized in Table A-1 and meet the intent of either version of the GALL for PWR internals. The extent to which the requirements and information contained in the I&E guidelines constitute satisfaction of a particular attribute is then discussed. In some cases, such as Attribute 1 (Scope of the Program), Attribute 3 (Parameters Monitored/Inspected) and Attribute 4 (Detection of Aging Effects), the I&E guidelines provide complete satisfaction of the GALL requirements. In other cases, supplementary information must be assembled by the utility to satisfy all of the remaining GALL aging management program requirements. The supplementary information requirements are identified in this appendix. Additional information on some of the attributes is provided in the text of the I&E guidelines.

A.2 Evaluation and Technical Basis

1. Scope of Program

   The I&E guidelines for PWR internals provide generic requirements that help utilities assure functional integrity of safety-related PWR internals. The scope of the I&E guidelines covers internals in all commercial operating U.S. PWR nuclear power plants. The scope does not include consumable items such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation. The scope also does not include welded attachments to the reactor vessel.

The I&E guidelines include:

- summary descriptions of PWR internals and functions;
- summary of the categorization and aging management strategy development of potentially susceptible locations, based on the safety and economic consequences of aging degradation;
- guidance for methods, extent, and frequency of one-time, periodic, and conditional examinations and other aging management methodologies;
- acceptance criteria for the one-time, periodic, and conditional examinations and other aging management methodologies;
- methods for evaluation of aging effects detected by the aging management methodologies that exceed the examination acceptance criteria; and
- requirements for implementation of the I&E guidelines.

The PWR internals are separated into four groups: Primary components; Expansion components; Existing Programs components; and No Additional Measures components. Definitions of these groups are included in Section 3.

Details of the generic requirements for the PWR internals aging management program are provided in the I&E guidelines only for the Primary and Expansion groups. Existing Programs to be continued throughout the license renewal term are covered briefly.

2. Preventive Actions: The I&E guidelines do not specify any preventive actions other than their applicability limitations to base-loaded plants. However, the guidelines do rely on PWR water chemistry control to manage SCC and reduce the impact of IASCC. Therefore, an important adjunct to the aging management methodologies described by the I&E guidelines is PWR water chemistry control. The water chemistry program for PWRs relies on monitoring and control of reactor water chemistry as presented in Chapter XI.M2, "Water Chemistry," of NUREG-1801, Volume 2 [A1].

3. Parameters Monitored/Inspected: The program monitors the effects of eight aging degradation mechanisms on the intended function of PWR internals through one-time, periodic, and conditional examinations, and other aging management methodologies, as needed, in accordance with the ASME Code, Section XI [A2], which provides established criteria, and the Preventive Maintenance Management Program (PMMP) approved I&E guidelines for PWR internals. The eight aging degradation mechanisms, and their associated effects, are described in Section 3.

   The program contains elements that monitor and inspect for the parameters that govern the progress of each of these effects. Section 4 of the I&E guidelines for PWR internals describes the methodologies that provide the monitoring and inspection of these effects.

4. Detection of Aging Effects: The aging management methodologies described in Section 4 of the I&E guidelines for PWR internals are based on either existing inservice examinations required by the ASME Code, Section XI [A2], or on well-documented and well-demonstrated examination methods with which the industry has considerable experience. For example, the industry has considerable experience with the volumetric examination by

ultrasonic testing (UT) of PWR internals bolts, pins, and fasteners, such as baffle-former bolting in B&W and Westinghouse units. The extent of this experience and the capability of the UT methods used has been documented in the Inspection Standard [A3] that supports the implementation of the I&E guidelines for PWR internals. This document will be used by utilities to support the Technical Justifications that are needed for examination method demonstrations, in accordance with the requirements of the ASME Code, Section V, Article 14 [A4]. Based upon this supporting documentation, the methods, coverage, and schedule of the inspection and test techniques prescribed by the ASME Code, Section XI, and the PMMP approved I&E guidelines for PWR internals are intended to maintain structural integrity and ensure that the detection and correction of aging effects before the loss of intended function of PWR internals.

5. Monitoring and Trending: One-time, periodic, and conditional examinations and other aging management methodologies, scheduled in accordance with the ASME Code, Section XI [A2], which provides established criteria, and the I&E guidelines for PWR internals, provide timely detection of aging effects. In addition to the Primary components, Expansion components have been defined should the scope of examination and re-examination need to be expanded beyond the Primary group, should significant effects be detected.

6. Acceptance Criteria: Section 5 of the I&E guidelines for the PWR internals provides the examination acceptance criteria for the Primary and Expansion components. In addition, the criteria for expanding the examinations from the Primary components to include the Expansion components are provided. The examination acceptance criteria include: (i) specific, descriptive relevant conditions for the visual (VT-3) examinations; (ii) requirements for recording and dispositioning surface breaking indications that are detected and sized for length by the visual (VT-1/EVT-1) examinations; and (iii) requirements for system-level assessment of bolted or pinned assemblies with unacceptable volumetric (UT) examination indications that exceed specified limits. Any detected condition that does not satisfy these examination acceptance criteria must be dispositioned. Example methodologies that can be used to analytically disposition unacceptable conditions are discussed or referenced in Section 6 of the I&E guidelines. However, other demonstrated and verified alternatives to the Section 6 methodologies may be used.

7. Corrective Actions: Corrective actions following the detection of unacceptable conditions are fundamentally provided for in each plant's corrective action program (CAP). Additional guidance for disposition of unacceptable conditions for PWR internals may be found in the ASME Code, Section XI; in the PMMP-approved I&E guidelines for PWR internals; and in reports referenced therein or demonstrated through an appropriate technical justification. Section 6 of the I&E guidelines provides information on methodology that can be used for the evaluation of detected conditions that exceed the examination acceptance criteria of Section 5. In addition, the alternative of component repair and replacement procedures for PWR internals is subject to the requirements of the ASME Code Section XI. The implementation of the I&E guidelines for PWR internals, plus the implementation of any ASME Code requirements, is intended to provide an acceptable level of aging management of the safety-related components addressed in accordance with the corrective actions of 10 CFR Part 50, Appendix B [A5] or its equivalent, as applicable.

8. Confirmation Process: Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B [A5] or their equivalent, as applicable. It is expected that the implementation of the I&E guidelines for PWR internals will provide an acceptable level of quality for inspection, flaw evaluation, and other elements of aging management of the PWR internals that are addressed in accordance with the 10 CFR Part 50, Appendix B [A5] or their equivalent (as applicable), confirmation process, and administrative controls.

9. Administrative Controls: The administrative controls for such programs, including their implementing procedures and review and approval processes are under existing site 10 CFR 50 Appendix B [A5] Quality Assurance Programs or their equivalent, as applicable. Such a program is thus expected to be established with a sufficient level of documentation and administrative controls to ensure effective long term implementation.

10. Operating Experience: Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. However, a considerable amount of PWR internals aging degradation has been observed in European PWRs, with emphasis on cracking of baffle-former bolting. For this reason, the U.S. PWR owners and operators began a program a decade ago to inspect the baffle-former bolting in order to determine whether similar problems might be expected in U.S. plants. A benefit of this decision was the experience gained with the UT examination techniques used in the inspections. In addition, the industry began substantial laboratory testing projects in order to gather the materials data necessary to support future inspections and evaluations. Several other items with existing or suspected material degradation concerns that have been identified for PWR components are wear in thimble tubes and potentially in control guide cards and observed cracking in some high-strength bolting and in control rod guide tube alignment (split) pins. The latter are conditions that have been corrected primarily through bolt replacement with less susceptible material and improved control of pre-load. The PWR Internals Programs established per the I&E guidelines will be new programs. Accordingly, there is no direct programmatic history. The program is based upon industry operating experience, research data, and vendor evaluations. Development of the program relied upon the consensus review and inputs of the MRP Reactor Internals Core and Focus Groups, which include representatives from utilities, research scientists, and vendors. This program will continue to evolve as additional experience is gained. Reactor internals failures, both domestically and internationally have been considered in the development of the I&E guidelines.

Table A-1
Key elements of PWR internals aging management plan document

	NRC Evaluation Attribute	PWR Internals Aging Management Plan Elements
1	Scope of Program	The scope of the program covers the internals for all currently operating commercial U.S. PWRs.
2	Preventive Actions	Preventive actions include implementation of the PWR primary coolant chemistry program.
3	Parameters Monitored/Inspected	PWR internals in the Primary and Expansion groups may include additional AMP elements, such as UT or VT-1/EVT-1 examinations; PWR internals in the Existing Programs group rely on the requirements of the ASME Code Section XI examinations, as well as other applicable items; PWR internals for which the effects of all aging degradation effects are below screening criteria or tolerated with no loss of function, require no additional aging management elements; Section 4 of the guidelines defines the parameters to be monitored and/or inspected.
4	Detection of Aging Effects	The Inspection Standard [A3] provides evidence of the capability of additional AMP elements, such as UT or VT-1/EVT-1 examinations; existing AMP elements, such as the ASME Code Section XI VT-3 examinations, are based on codified examination standards.
5	Monitoring and Trending	Monitoring may be used to define the scope or schedule for some one-time or conditional examinations or inspections. Trending of inspection results, especially for early plant inspections, will be used to determine any needed modifications to the I&E guidelines; Section 7 of the I&E guidelines describes the monitoring and trending requirements.
6	Acceptance Criteria	Section 5 of the guidelines provides the examination acceptance criteria for the AMP elements described in Section 4, Tables 4-1 through 4-6; references to applicable supporting document, such as applicable ASME Code Section XI acceptance or industry-supplied criteria, are provided for the evaluation of AMP element application results.
7	Corrective Actions	Procedures for disposition of inspection findings that exceed examination acceptance criteria are provided in Section 6 of the guidelines, as implemented by individual plant corrective action programs.
8	Confirmation Process	Reference to site quality assurance procedures and associated regulations are provided.
9	Administrative Controls	Reference to mandatory element of a site-specific PWR internals program and need for compliance with 10CFR50 Appendix B.
10	Operating Experience	Industry PWR internals operating experience is described.

A.3 References

A1. Generic Aging Lessons Learned (GALL) Report, Volume 2, "Tabulation of Results," (NUREG-1801, Volume 2), Revision 1, U.S. Nuclear Regulatory Commission, Washington, DC: September 2005.

A2. ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components,* ASME Boiler and Pressure Vessel Code, 2001 Edition, including the 2002 and 2003 Addenda, American Society of Mechanical Engineers, New York, NY.

A3. *Materials Reliability Program: Inspection Standard for Reactor Internals (MRP-228).* EPRI, Palo Alto, CA: 2009. 1016609.

A4. *ASME Boiler & Pressure Vessel Code, Section V, Nondestructive Examination,* American Society of Mechanical Engineers, New York, NY, 2004 Edition, July 1, 2004.

A5. 10 CFR Part 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants,* Office of the Federal Register, National Archives and Records Administration, 2005.

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