Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228)

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PRODUCT DESCRIPTION

Demonstration that the effects of aging degradation in PWR internals are adequately managed is essential for maintaining a healthy fleet and ensuring continued functionality of the reactor internals components. As a work product of the EPRI Materials Reliability Program (MRP) Inspection Issues Task Group in cooperation with the Reactor Internals Focus Group (RI-FG), these inspection standards are intended to support the PWR Internals Inspection and Evaluation Guidelines (MRP-227) to detect the effects of aging degradation. The requirements contained in this report are applicable to Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse nuclear steam supply system PWR designs currently operating in the United States. This report defines inspection standards and documented inspection techniques for PWR vessel internal components.

Results and Findings

Procedure standards have been developed for ultrasonic and visual inspection. Many of the inspection techniques have been demonstrated and documented previously to meet other inspection requirements and will be adopted to meet these requirements. In support of the bolting ultrasonic demonstrations, realistic bolt mockups have been manufactured with controlled flaws. The visual inspections reference the requirements of ASME Section XI or the requirements of this standard. PWR owners can use these mockups and documented techniques to inspect their vessel internal components in compliance with MRP guidance.

Challenges and Objectives

Inservice inspection program managers and PWR vessel internal program managers can use this report along with the PWR Internals I&E Guidelines to ensure that their components are inspected in compliance with MRP guidance. This standard and the guidelines will be updated as additional information is obtained from these inspections during the life of the plants.

Applications, Value, and Use

The information contained in this report is applied by all PWR owners in preparation for managing aging degradation of vessel internal components during refueling outages.

EPRI Perspective

This report provides the PWR fleet with inspection procedure requirements for all of the primary and expansion vessel internal components included in the I&E Guidelines and offers a stable mechanism for documenting the capability of the evolving inspection technology. The program to develop the guidelines and this inspection standard 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 inspection techniques.

Approach

The MRP strives to make effective inspection techniques available by developing inspection standards to ensure the structural integrity of the components and providing demonstrated, documented techniques for effectively examining susceptible components.

Keywords

NDE
Internals
Inspection
Ultrasound
Visual

ABSTRACT

This report is a work product of EPRI's Materials Reliability Program (MRP) and has been prepared to implement the reactor internals inspection program described in EPRI report 1016596, *PWR Internals Inspection and Evaluation Guidelines (MRP-227)*. It contains requirements specific to the inspection methodologies involved as well as requirements for qualification of the NDE systems used to perform those inspections. This report is intended for use by individual plant owners in preparing inspection procedures and qualifying NDE systems—that is, the combinations of equipment, procedure, and personnel—used to perform inspections needed for their PWR internals aging management programs. As a companion to MRP-227, this standard is used to support compliance with the mandatory requirements that must be implemented as required by NEI 03-08 as explained in Section 7 of MRP-227.

LIST OF ACRONYMS

ASME	American Society of Mechanical Engineers
B&PV	boiler and pressure vessel
B&W	Babcock & Wilcox
BMI	bottom mounted instrumentation
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CASS	cast austenitic stainless steel
CE	Combustion Engineering
CEA	control element assembly
CFR	Code of Federal Regulations
CRGT	control rod guide tube
CSA	core support assembly
CSS	core support shield
EPRI	Electric Power Research Institute
EVT	enhanced visual testing (a visual NDE method that includes EVT-1)
IASCC	irradiation-assisted stress corrosion cracking
ICI	in-core instrumentation
IGSCC	intergranular SCC
ISI	inservice inspection
ITG	issue task group
LCB	lower core barrel
MRP	Materials Reliability Program
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
PH	precipitation-hardenable (heat treatment)
PWR	pressurized water reactor

PWSCC	primary water SCC
QA	quality assurance
SCC	stress corrosion cracking
SS	stainless steel
TJ	Technical Justification
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)

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1 INTRODUCTION

This report is a work product of the EPRI Materials Reliability Program (MRP) and has been prepared to implement the reactor internals inspection program described in the EPRI report *PWR Internals Inspection and Evaluation Guidelines (MRP-227)* [1]. It contains requirements specific to the inspection methodologies involved as well as requirements for qualification of the NDE systems used to perform those inspections.

1.1 Applicability

This document is intended for the use of individual plant owners in preparing inspection procedures and qualifying NDE systems—that is, the combinations of equipment, procedure, and personnel—used to perform inspections needed for their PWR internals aging management programs. As a companion to MRP-227, this standard is required for compliance with the mandatory requirements in Section 7 of MRP-227. This standard is applicable to internals in the three nuclear steam supply system (NSSS) PWR designs currently operating in the United States: Babcock & Wilcox (B&W), Combustion Engineering (CE), and Westinghouse Nuclear.

1.2 Background

The process for NDE system qualification described in this report is based on the Technical Justification in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section V, Article 14, which has been used for guidance. The Technical Justification is a detailed explanation of the procedure, including the method, and any laboratory or field experience that supports the procedure capabilities. The Technical Justification provides the technical basis and rationale for the qualification. This qualification process builds on the field experience and previous work performed by the NSSS vendors and other inspection organizations to develop the examination systems already in use throughout industry and does not mandate any new performance demonstrations. Where there is a need to simulate performance-specific examinations for the purpose of preparing the Technical Justifications required by this document—or for other reasons, such as to qualify improved techniques—new or existing vendor mockups may be used or, where mockups of such components are not otherwise available, mockups developed by the MRP may be used.

1.3 Organization

This report is organized into six sections, including this introduction. Section 2 contains the general procedures for preparing the Technical Justifications for NDE systems, the formal process for the control and use of mockups developed by the MRP, and specific requirements for inspections. Section 3 addresses flaw measurement for the visual inspection process as may be required based on the results of those inspections to support condition assessments for specific components within the scope of the Inspection and Enforcement (I&E) Guideline. Sections 4, 5, and 6 deal with the inspections and Technical Justifications required for reactor internals of each of the three primary NSSS designs operating in the United States. These sections summarize and supplement the information presented in the I&E Guideline that is necessary for the development of the Technical Justifications required by this standard. Section 7 summarizes the implementation requirements for this standard.

2
GENERAL PROCEDURES

The general procedures describe the requirements for NDE system qualification, use of mockups, and general requirements for visual and ultrasonic examination for reactor vessel internal components. An NDE system is generally defined as the procedure, personnel, and equipment used to perform an NDE examination. The term *NDE system* is applicable for any method of examination, including visual inspection.

The general procedures provide a uniform approach for NDE system qualification and consistency in the application of examination methods.

2.1 Technical Justifications for NDE System Qualification

2.1.1 Purpose

2.1.1.1 Technical Justifications as described herein are required for qualification of NDE systems other than visual examinations. Generic standards for visual examinations are provided in Section 2.3.

2.1.1.2 This section outlines the requirements for Technical Justifications so that they are prepared and documented in a consistent and formal manner.

2.1.2 References

2.1.2.1 American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section V (ASME Section V), Nondestructive Examination [2].

2.1.2.2 American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI (ASME Section XI), Rules for Inservice Inspection of Nuclear Power Plant Components [3].

2.1.3 Description and Requirement for the Technical Justification

2.1.3.1 Content deleted – EPRI/MRP Proprietary Information

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2.1.4 Content of Technical Justification

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2.2 Use of MRP Mockups

2.2.1 Purpose

2.2.1.1 MRP mockups are developed based on an identified need where such mockups are not otherwise available or where available mockups are determined to be insufficient. The process for MRP-member utilities and their vendors to use mockups developed by the MRP Inspection Issue Task Group (ITG) is outlined in 2.2.2.

2.2.1.2 This document also serves as a guide to schedule the use of these mockups to prevent schedule conflicts and to ensure that personnel in the EPRI NDE Program are available to support the use of the mockups, as needed.

2.2.2 Mockup Usage Requirements

2.2.2.1 Content deleted – EPRI/MRP Proprietary Information

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2.2.3 Mockup Use Prioritization

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2.3 Generic Standards for Visual Inspection of Reactor Pressure Vessel Internals, Components, and Associated Repairs

2.3.1 Purpose

This document describes requirements and recommendations for the performance of underwater remote visual examination of reactor pressure vessel (RPV) internals, components, and associated repairs.

2.3.2 Scope

2.3.2.1 Content deleted – EPRI/MRP Proprietary Information

2.3.2.2 Content deleted – EPRI/MRP Proprietary Information

2.3.2.3 Content deleted – EPRI/MRP Proprietary Information

2.3.2.4 Content deleted – EPRI/MRP Proprietary Information

2.3.3 Definitions

2.3.3.1 Indication

Evidence of an apparent interruption in the normal structure of a material or product. Indications may be relevant or non-relevant.

2.3.3.2 Relevant indication

a. Welds. Cracks, or indications that exhibit characteristics of cracking, are considered relevant.

b. Components. Cracking or other significant degradation that could impair the ability of the component to perform its design function is considered relevant.

2.3.3.3 Non-relevant indication

An indication that is evaluated as not being relevant according to paragraph 2.3.3.2. Non-relevant indications include fabrication marks, material roughness, and other conditions acceptable by material, design, and manufacturing specifications of the component.

2.3.3.4 Resolution card

A card bearing the characters described in paragraphs 2.3.6.3.b.1 and 2.3.6.3.b.2 used to verify the adequacy of lighting and/or remote camera resolution.

2.3.3.5 Essential variable

Any element, component, or combination of the equipment used for the inspection that, if changed, could affect the ability of the inspection equipment to detect indications or an evaluator's ability to evaluate indications. The specific essential variables will be as described in the inspection procedure. These include the camera, camera tube or board, and camera lens; video processor, monitor, and recording device (if evaluations will be performed from recordings); and inspection conditions, such as lens-to-subject distance.

2.3.3.6 Resolution demonstration

The process of demonstrating the ability of the remote visual inspection equipment, equipment setup, inspection area environment, and inspection technique to resolve the required characters on the resolution card.

2.3.4 Personnel Training/Experience

2.3.4.1 Personnel evaluating inspection data shall be certified as a Level II or Level III examiner in the VT-1 and/or VT-3 method, as appropriate, to a written practice meeting the requirements of ASME, Section XI.

2.3.4.2 **Content deleted – EPRI/MRP Proprietary Information**

2.3.5 Equipment Requirements

2.3.5.1 Resolution standards for demonstrating system performance

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2.3.5.2 Underwater cameras

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2.3.5.3 Camera lenses

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2.3.5.4 Lighting

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2.3.5.5 Viewing monitors

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2.3.5.6 Video data recording

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2.3.5.7 Hard copy processors

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2.3.5.8 Video review equipment

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2.3.6 Inspection Requirements

2.3.6.1 Surface conditions

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2.3.6.2 Environmental conditions

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2.3.6.3 Equipment resolution demonstration requirements

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2.3.6.4 Area(s) of interest

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2.3.6.5 Inspection technique

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2.3.6.6 Inspection planning

In order to optimize inspection time, consideration should be given to organizing the inspections to be performed in specific areas of the vessel coincidentally, such as performing a VT-3 of the locking device while performing the ultrasonic examination of the associated bolt.

2.3.6.7 Classification of indications

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2.3.6.8 Measurement of relevant indications

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2.3.7 Documentation of Results

2.3.7.1 Content deleted – EPRI/MRP Proprietary Information

2.4 Ultrasonic Examination of Bolting in Reactor Pressure Vessel Internals

2.4.1 Purpose

This section describes requirements and recommendations for the performance of ultrasonic examination (UT) of bolting in reactor vessel internal components.

2.4.2 Scope

2.4.2.1 This section is to be used by PWR utilities when performing UT of RPV internal bolting to meet the inspection and evaluation guidelines contained in MRP-227.

2.4.2.2 The MRP-227 references for the required bolting UT examinations of both primary and expansion bolting are provided for information only in Sections 4, 5, and 6 of this report. The bolting identified as *primary* or *expansion* may change as further field experience accumulates, so the current examination requirements must be verified to the latest version of MRP-227 or the version referenced by the approved license renewal application.

2.4.3 Definitions

2.4.3.1 Indication

Evidence of an apparent interruption in the normal structure of a material or product. Indications may be relevant or non-relevant.

2.4.3.2 Relevant indication

An ultrasonic indication evaluated as cracking or other significant degradation that could impair the ability of the bolting to perform its design function is considered relevant.

2.4.3.3 Non-relevant indication

An indication that is evaluated as not being relevant, according to 2.4.3.1 and 2.4.3.2 above, such as a mode-converted response from the sides of component.

2.4.3.4 Essential variables

Any element, setting, component, or combination of the equipment used for the inspection that, if changed, could affect either the ability of the inspection equipment to detect indications or an evaluator's ability to evaluate indications. The specific essential variables will be as described in the inspection procedure. For ultrasonic examinations, this includes the transducer size, transducer frequency, transducer arrangement, transducer shape, cable length, number of intermediate connectors, UT scope/system, UT scope/system settings, pulse repetition rate, calibration, and data analysis software. Changes in the essential variables of the procedure require revision to the Technical Justification.

2.4.4 Personnel Training/Experience

2.4.4.1 Content deleted – EPRI/MRP Proprietary Information

2.4.4.2 Content deleted – EPRI/MRP Proprietary Information

2.4.5 Equipment Requirements

The equipment included in the UT procedure as an essential variable or range of variables as described in this section must be used for the examinations and be included in the Technical Justification and demonstration, as applicable.

2.4.6 Examination Requirements

2.4.6.1 Surface conditions

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2.4.6.2 Area(s) of interest

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2.4.6.3 The examination procedure shall specify the following essential variables:

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2.4.7 Classification of Indications

Indications shall be classified as either *relevant* or *non-relevant*. If an indication cannot be classified during initial inspections, additional re-looks shall be performed. The UT procedure shall contain instructions describing how the data will be interpreted and analyzed. All relevant indications shall be reviewed by another UT Level II or Level III examiner.

2.4.8 Documentation of Results

Documentation should, as a minimum, include the following: 1) the inspection procedure number and revision, date(s) of examination(s) and evaluation(s), and names of personnel performing the examinations and data review and analysis, 2) the location and extent of the bolts examined, 3) a calculation of the percentage or number of bolts that were successfully examined compared to the number scheduled for examination, and 4) the number and location of bolts with flaw indications, including their unique identification, if available.

2.4.9 Technical Justifications and Demonstrations

2.4.9.1 Ultrasonic examination systems are required to be qualified by a Technical Justification as described in Section 2.1 of this document. The organization that prepared the procedure will also be responsible for preparing the Technical Justification. The utilities are responsible for approving the Technical Justification and may request that the MRP assist by reviewing the Technical Justification.

2.4.9.2 Information on Technical Justifications completed by two inspection vendors is included in Appendix A.

3
FLAW LENGTH MEASUREMENT BY VISUAL EXAMINATION

3.1 Background

Studies by other industry programs and the NRC [4] have identified the limitations of remote visual examination systems used for underwater visual examinations, including flaw length sizing accuracy associated with measurement techniques used. Although this standard does not restrict the flaw measurement techniques that may be used, the applicability of the measurement techniques for the components and for the damage mechanisms on which they will be used must be supported by the Technical Justification for the NDE system.

3.2 Techniques

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3.3 Determining Uncertainty

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4
B&W REACTOR INTERNALS INSPECTION

4.1 Summary Reactor Internals Description

The internals of B&W reactors have two major structural components: the plenum assembly and the core support assembly (CSA). The CSA is composed of three main subassemblies: the core support shield (CSS), the core barrel, and the lower internals.

The major components of the core support assembly—the core support shield assembly, the core barrel assembly, and the lower internals assembly—are joined together by bolting. During refueling, the core support assembly remains in place in the reactor vessel. The core support assembly is removed only for inspection or to facilitate scheduled inspections of the reactor vessel interior surfaces. The plenum assembly sits inside the core support shield and includes the plenum cover assembly (which is bolted to the top of the plenum cylinder), the upper grid assembly, and the control rod guide tubes. The lower internals assembly includes the lower grid assembly, flow distributor assembly, and in-core monitoring instrumentation (IMI) guide tube assemblies. The general arrangement of the B&W-designed PWR internal components is shown in Figure 4-1.

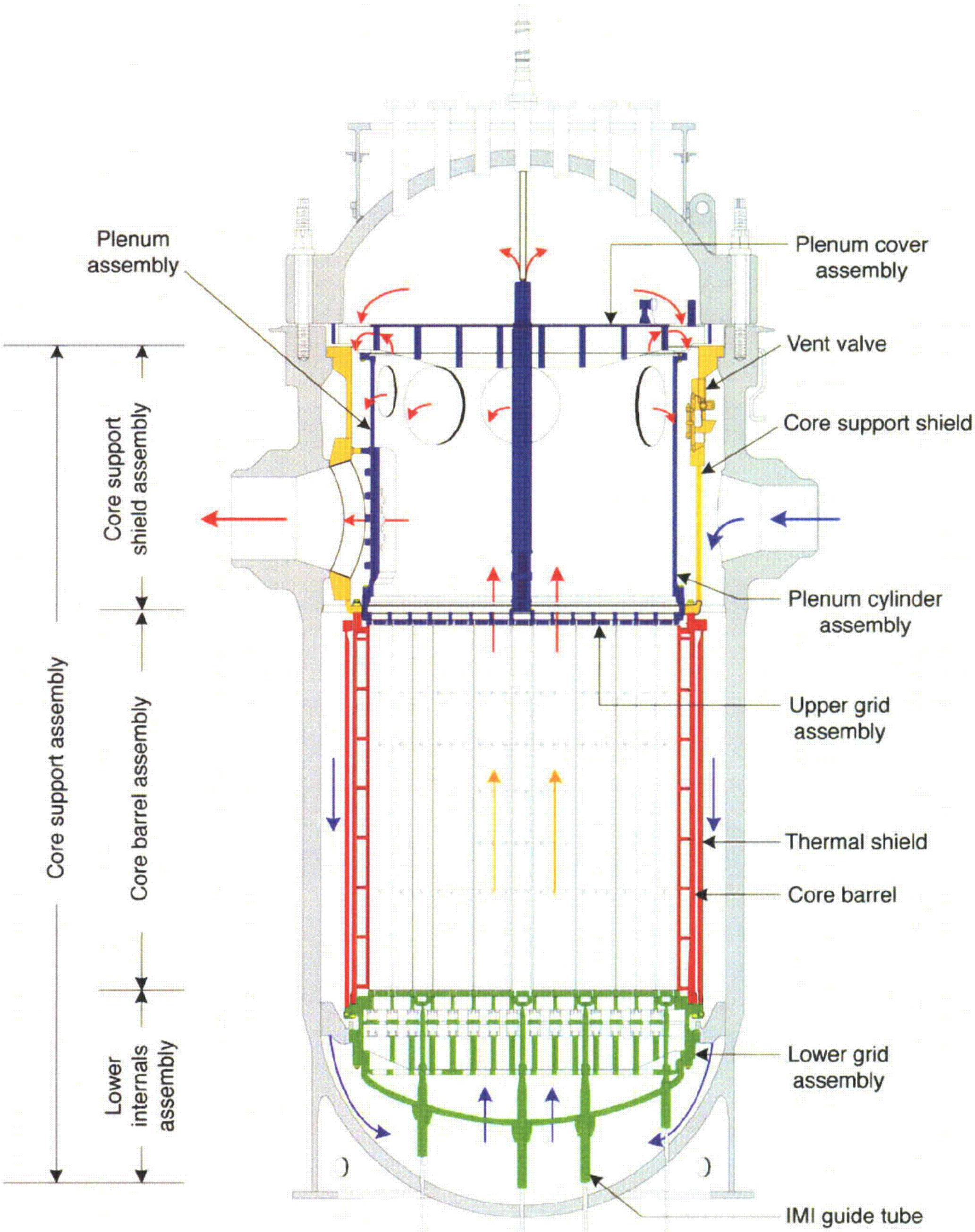


Figure 4-1
Typical Arrangement of B&W Core Support and Lower Internals

(Note: some component items are rotated for clarity)

4.2 Technical Justifications

The qualifying Technical Justifications are required for ultrasonic examinations of B&W internal components and bolting, respectively. As described in Section 2, the Technical Justification shall include descriptions of the bolting designs and damage mechanisms, a general description of the examination system, a description of inspection parameters including procedure and personnel requirements, and a description of procedure experience. The damage mechanisms are required to be addressed in the Technical Justification for each primary inspection item on the basis of susceptibility. The primary components as described in Section 4 of the Inspection and Evaluation Guidelines (MRP-227) have at least one damage mechanism above the screening criteria and require additional aging management program elements to manage the effects. The expansion components are scheduled for examination only on the basis of the findings from examination of the primary examination; however, many B&W expansion components are inaccessible and are identified for evaluation and replacement rather than inspection. The examination requirements identified in MRP-227 for both primary and expansion components are summarized for information in Tables 4-1 and 4-2. The damage mechanisms for the primary components identified in MRP-227 are described for information in Sections 4.2.1 and 4.2.2. Typical drawings for these components are shown for information only in Figures 4-2 through 4-12.

Table 4-1
Required Volumetric Examinations for Babcock & Wilcox Reactor Internals (Information)

<table>
<tr><th colspan="7">Volumetric (UT) Examinations</th></tr>
<tr><th>Primary examination item (See MRP-227, Table 4-1)</th><th>Material (See MRP-227, Table 3-1)</th><th>Plant (See MRP-227, Table 3-1)</th><th>Figure</th><th>Damage Mechanism (See MRP-227, Table 3-1)</th><th>Expansion Components (See MRP-227, Table 4-4)</th><th>Requirements</th></tr>
<tr><td rowspan="2">Core support shield assembly
Upper core barrel (UCB) bolts</td><td rowspan="2">Alloy A-286 or Alloy X-750</td><td rowspan="2">All</td><td rowspan="2">4-8</td><td rowspan="2">Cracking (SCC)</td><td rowspan="4">All plants: bolts on upper thermal shield, lower thermal shield, and flow distributor (Fig. 4-8)
Surveillance sample holder tubes: bolts for Davis Besse and studs/nuts (Fig. 4-8)
For Three Mile Island 1, bolts on the lower grid shock pad (Fig 4-5)</td><td rowspan="6">See MRP-227, Tables 4-1 and 4-4</td></tr>
<tr></tr>
<tr><td rowspan="2">Core barrel assembly
Lower core barrel (LCB) bolts</td><td rowspan="2">Alloy A-286 or Alloy X-750</td><td rowspan="2">All</td><td rowspan="2">4-9</td><td rowspan="2">Cracking (SCC)</td></tr>
<tr></tr>
<tr><td rowspan="2">Core barrel assembly
Baffle-to-former bolts</td><td rowspan="2">304 SS</td><td rowspan="2">All</td><td rowspan="2">4-3</td><td rowspan="2">Cracking (IASCC, IE, IC/ISR/fatigue/wear, overload)</td><td>All plants: internal baffle-to-baffle bolts (Fig. 4-3)</td></tr>
<tr><td>All plants: external baffle-to-baffle bolts and core barrel-to-former bolts (Fig. 4-3)</td></tr>
</table>

Table 4-1 (continued)
Required Volumetric Examinations for Babcock & Wilcox Reactor Internals (Information)

Volumetric (UT) Examinations						
Primary examination item (See MRP-227, Table 4-1)	**Material (See MRP-227, Table 3-1)**	**Plant (See MRP-227, Table 3-1)**	**Figure**	**Damage Mechanism (See MRP-227, Table 3-1)**	**Expansion Components (See MRP-227, Table 4-4)**	**Requirements**
Core support shield assembly CSS cast outlet nozzles	Nozzles: CF8 Spacers: CF3M	ONS-3, DB	4-10	Cracking (TE)	CRGT spacer castings (Fig. 4-6)	See MRP-227: Tables 4-1 and 4-3
Core support shield assembly CSS vent valve disk (Note 1)	Disk: CF8 Spacers: CF3M	All	4-11	Cracking (TE)	CRGT spacer castings (Fig. 4-6)	
Core support shield assembly CSS vent valve top retaining ring CSS vent valve bottom retaining ring CSS vent valve disk shaft or hinge pin (Note 1)	Rings: 15-5PH Shaft/Pin: 431	All	4-12	Cracking (TE)	None	
Core barrel assembly Baffle plates	304 SS	All	4-3	Cracking (IE)	Core barrel cylinder and former plates (Fig. 4-3)	

Table 4-2
Required Visual Examinations for Babcock & Wilcox Reactor Internals (Information)

Visual (VT-3) Examinations						
Primary Examination Item (See MRP-227, Table 4-1)	**Material (See MRP-227, Table 3-1)**	**Plant (See MRP-227, Table 3-1)**	**Figure**	**Damage Mechanism (See MRP-227, Table 3-1)**	**Expansion Component (See MRP-227, Table 4-3)**	**Requirements**
Core barrel assembly Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	304 SS	All	4-3	Cracking (IASCC, IE, overload)	Locking devices, including locking welds, of the baffle-to-baffle bolts and barrel-to-former bolts (Fig. 4-3)	See MRP-227, Tables 4-1 and 4-3
Lower grid assembly Alloy X-750 dowel-to-guide block welds		All	4-5	Cracking (SCC)	Alloy X-750 dowel locking welds to the lower fuel assembly support pads and, for all plants except Davis Besse, upper fuel assembly support pads (Fig. 4-7)	
Incore monitoring Instrumentation (IMI) guide tube assembly IMI guide tube spider IMI guide tube spider-to-lower grid rib section welds	Spider: CF8 Weld: 308L Spacers: CF3M Pad Items: 304 w/308L weld, except Alloy X-750 dowel with Alloy 69 weld	All	4-4 and 4-7	Cracking (TE/IE)	CRGT spacer castings (Fig. 4-6) Lower fuel assembly support pad items: Pad, Alloy X-750 dowel, cap screw, and their locking welds (Fig. 4-7)	

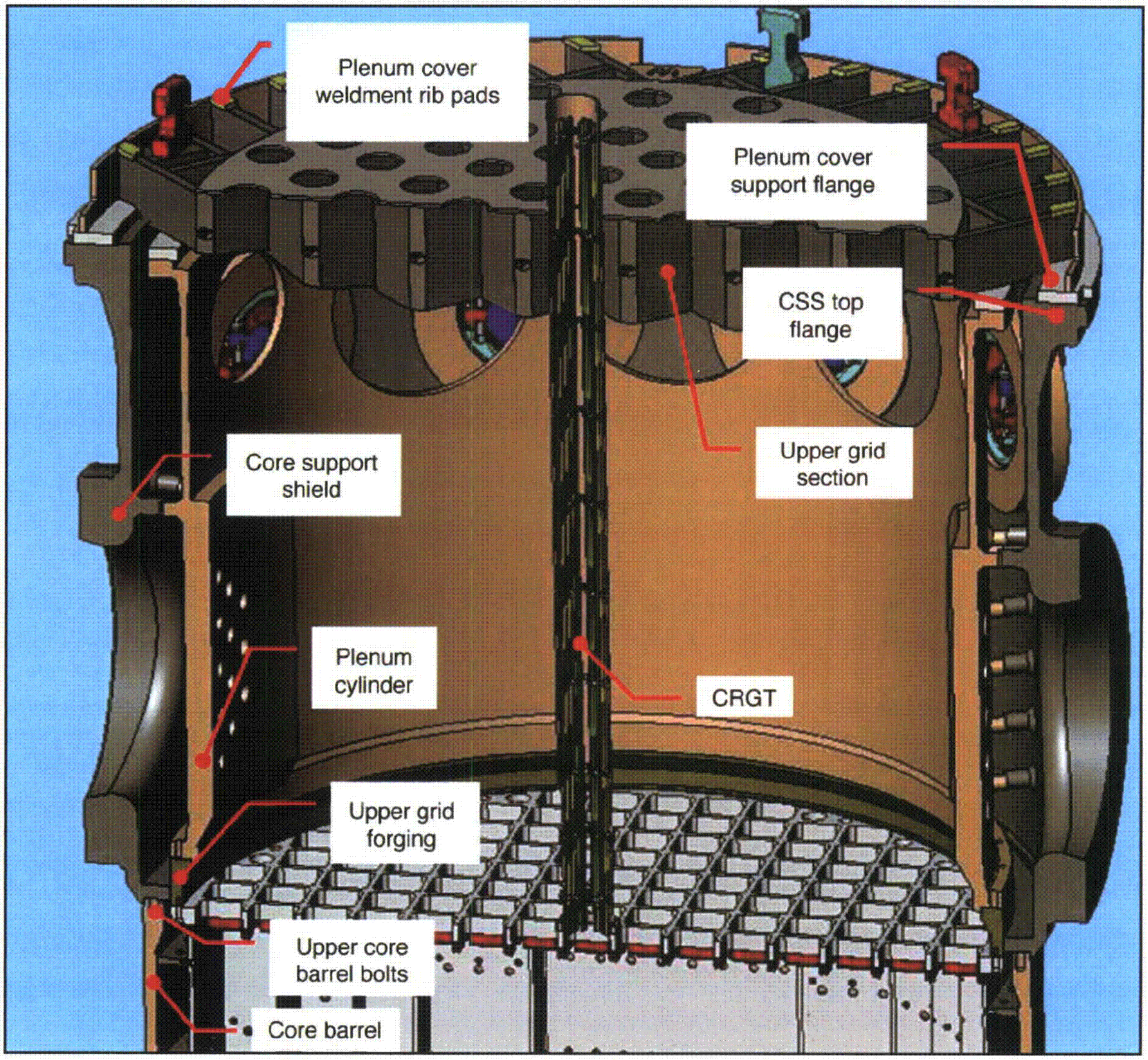


Figure 4-2
B&W Upper Internal Components

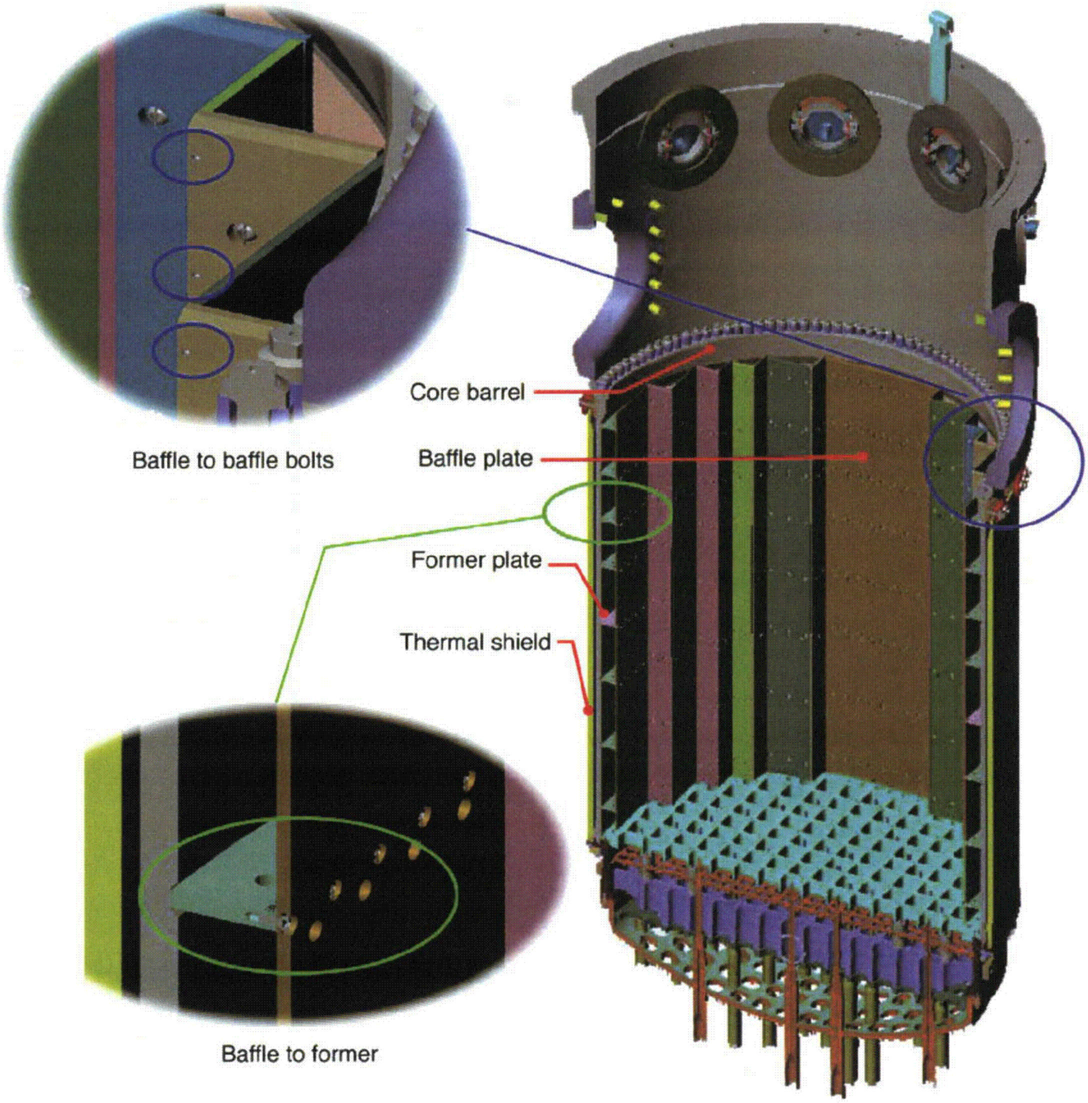


Figure 4-3
Configuration of B&W Internals Showing Locations of Baffle-to-Baffle and Baffle-to-Former Bolted Connections

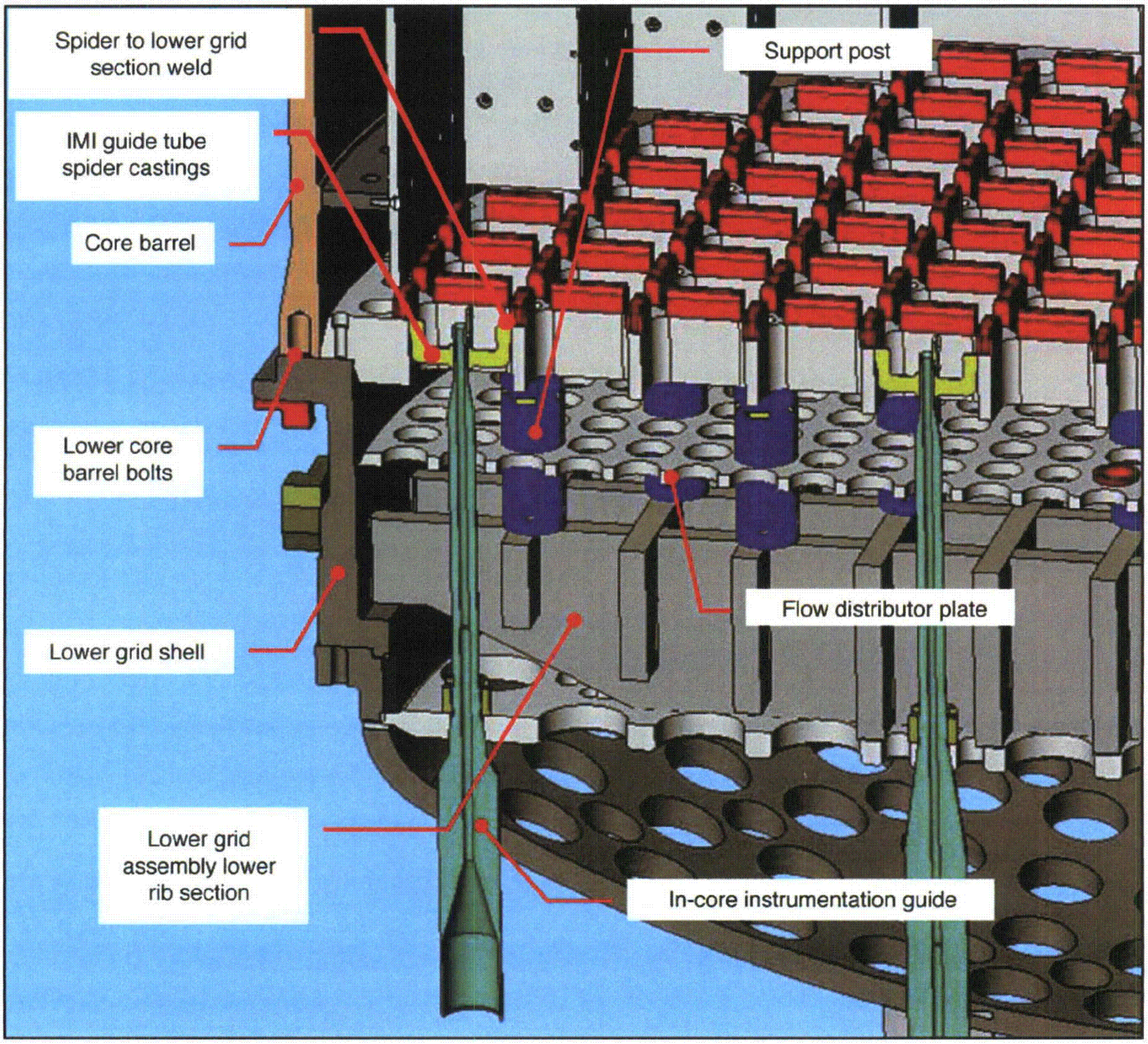


Figure 4-4
B&W Lower Internals and Bolting Locations

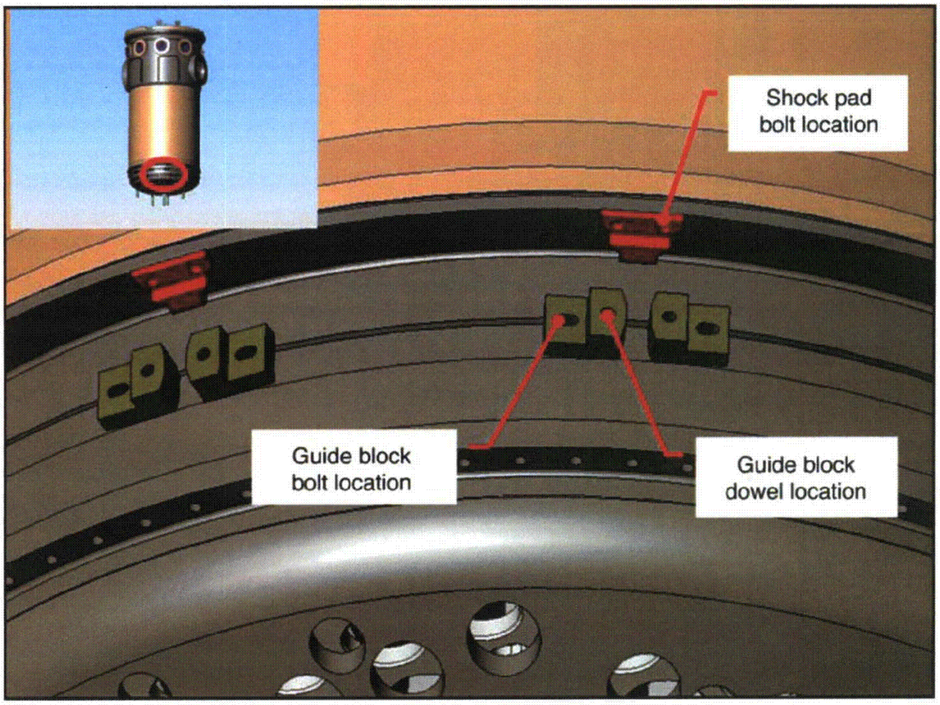


Figure 4-5
B&W Lower Internals Showing Dowel-to-Guide Block Locations (Alloy X-750 Locking Welds)

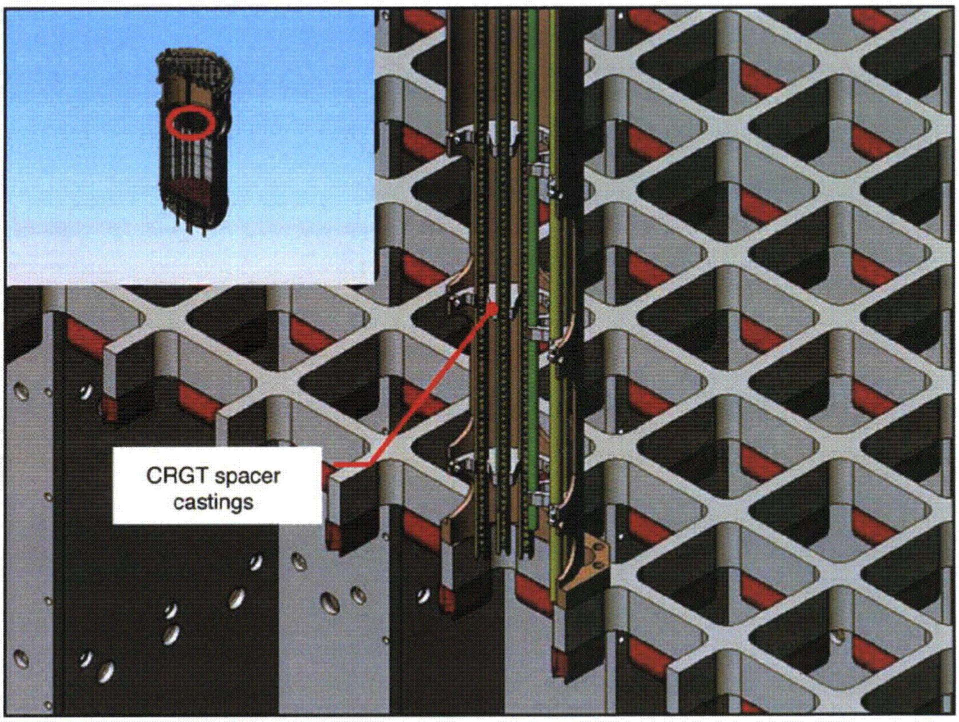


Figure 4-6
B&W Control Rod Guide Tube Configuration Showing Spacer Casting Location

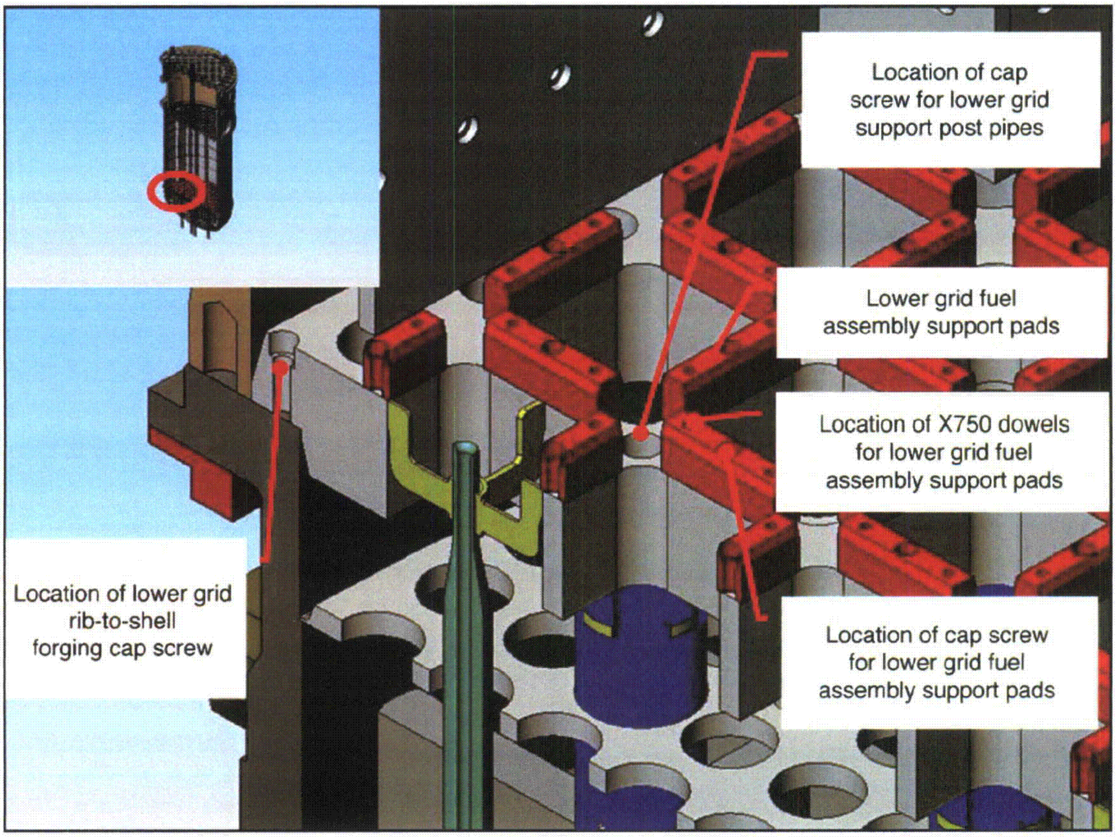


Figure 4-7
B&W Lower Grid Assembly Showing Fuel Assembly Support Pad Items

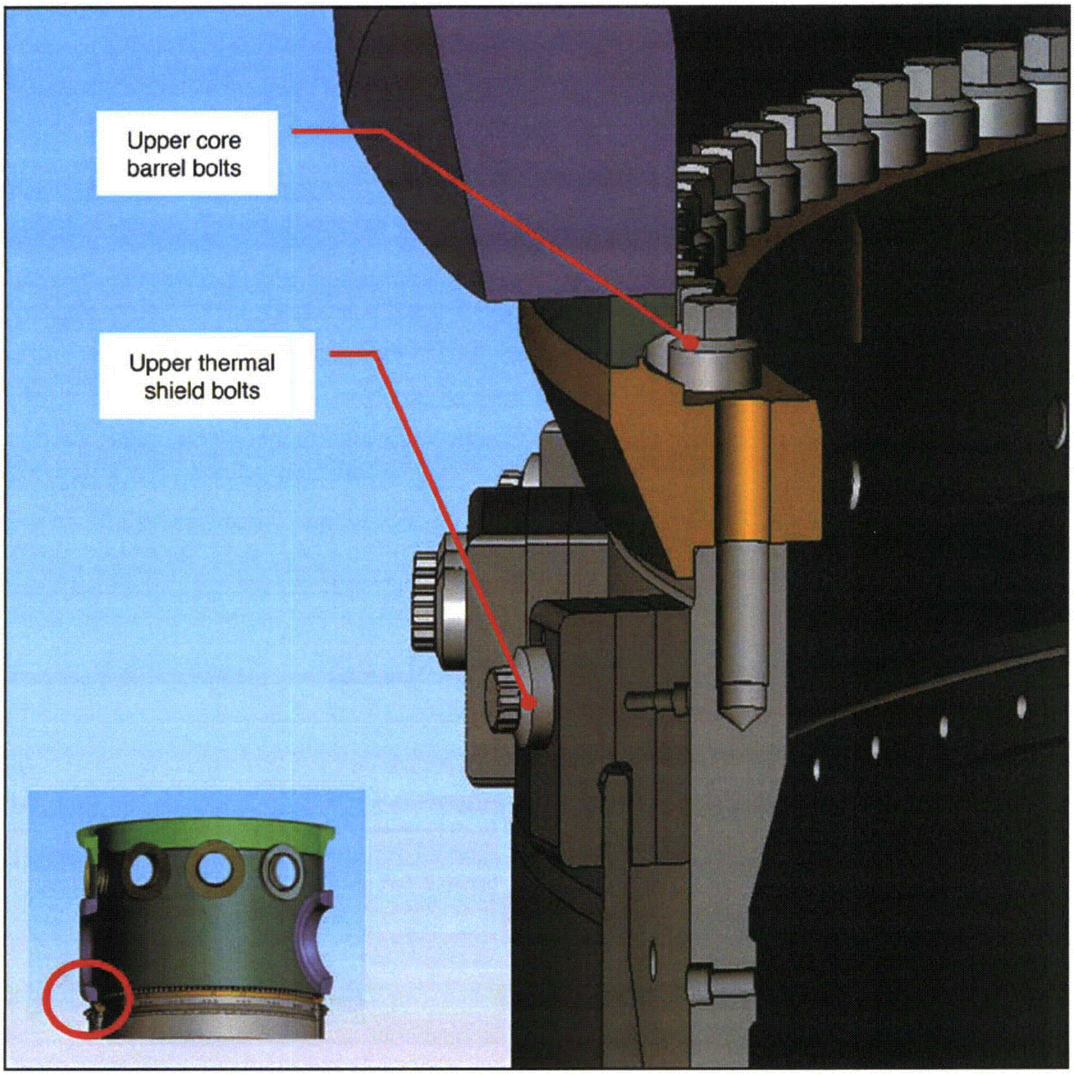


Figure 4-8
B&W Bolting Locations in Upper Thermal Shield and Upper Core Barrel

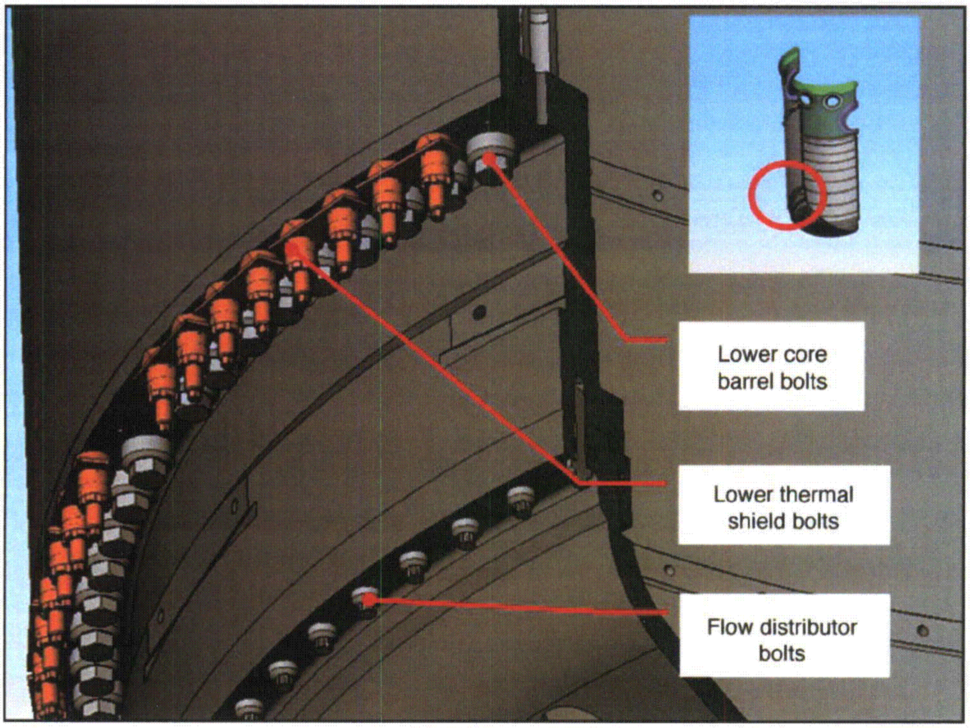


Figure 4-9
B&W Bolting Locations in Lower Thermal Shield, Lower Core Barrel, and Flow Distributor

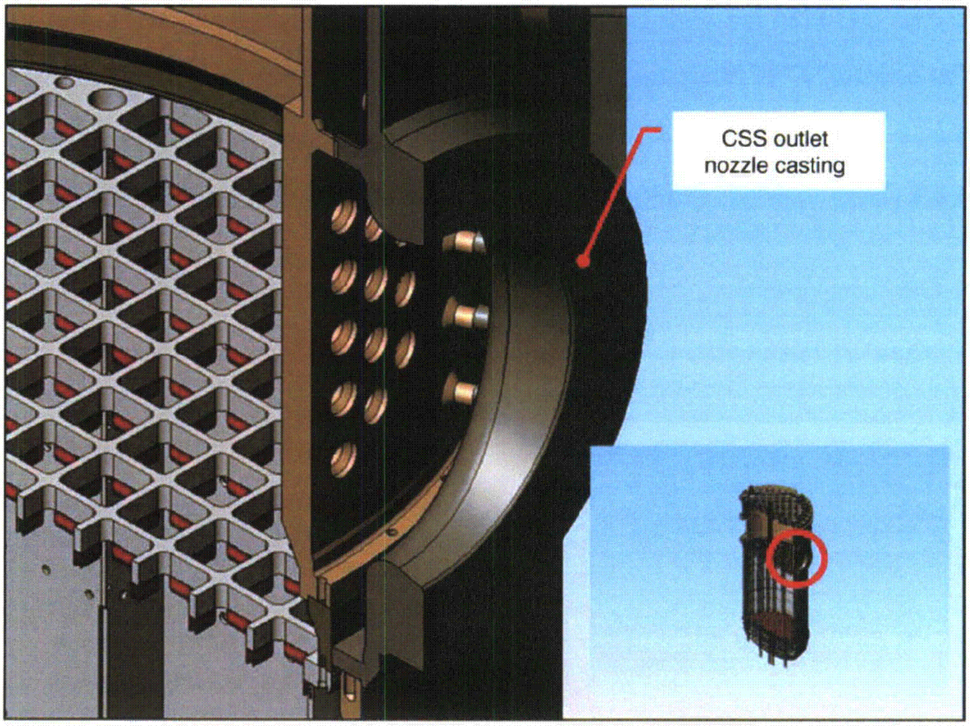


Figure 4-10
B&W Core Support Shield Cast Outlet Nozzle

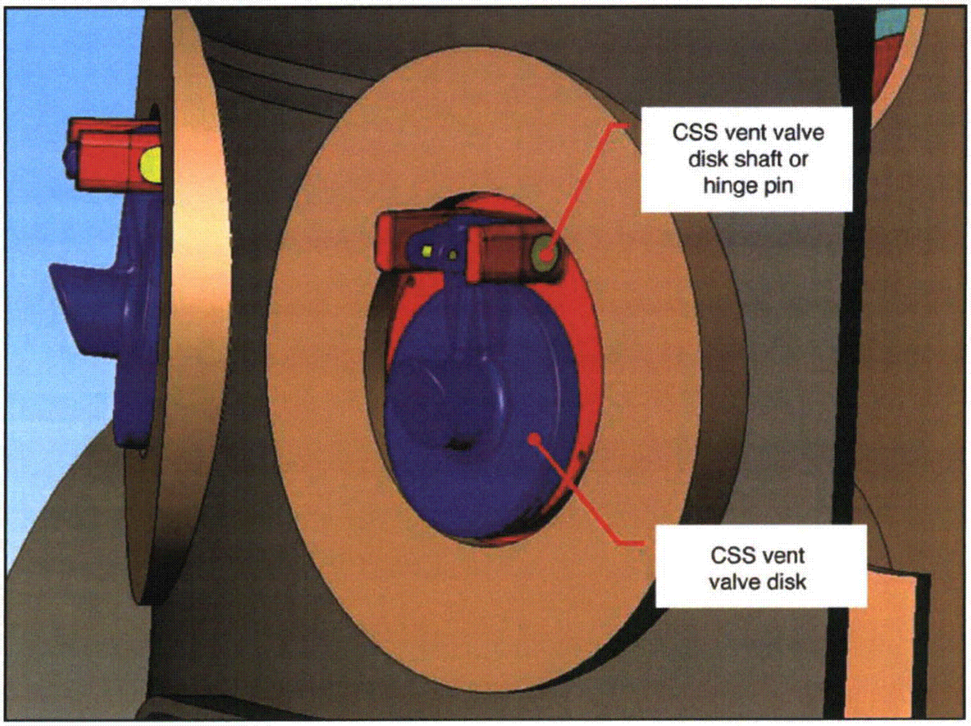


Figure 4-11
Outside View of B&W Core Support Shield Vent Valve

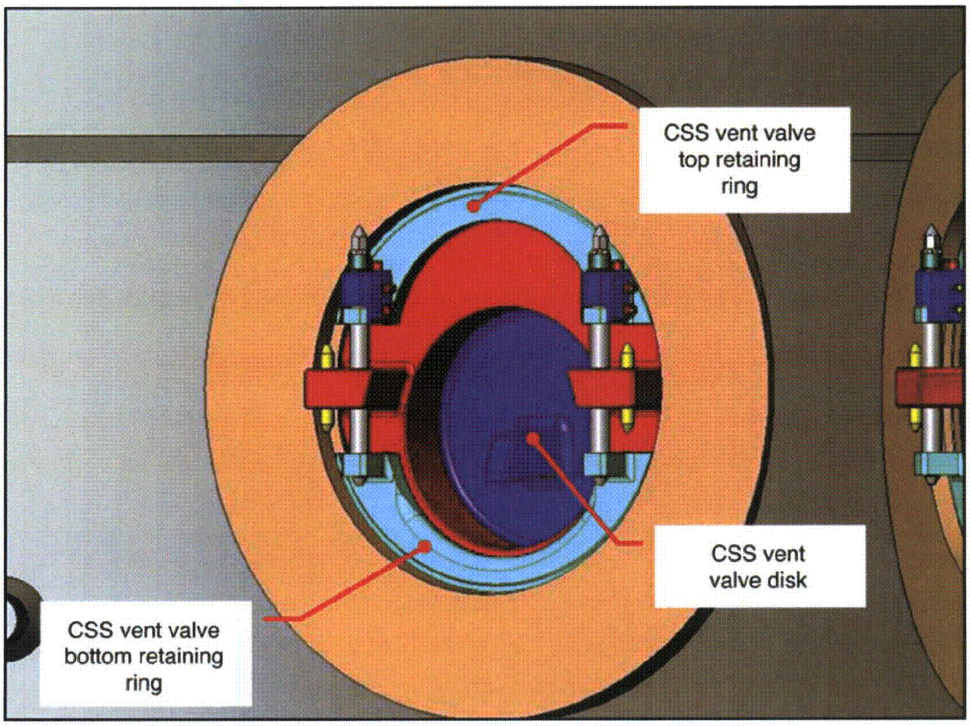


Figure 4-12
Inside View of B&W Typical Core Support Shield Vent Valve

4.2.1 Inspection and Damage Mechanisms for Bolting

The core support shield assembly, core barrel assembly, and core barrel baffle-to-former bolting require examination using the ultrasonic method. For upper and lower core barrel bolting, the potential damage mechanism is stress corrosion cracking (SCC). For baffle-to-former bolting, the potential damage is cracking related to various mechanisms, including irradiation-assisted stress corrosion cracking (IASCC), irradiation embrittlement (IE), irradiation-enhanced creep (IC), irradiation-enhanced stress relaxation (ISR), fatigue, wear, and overload. The MRP-227 references for required bolting examinations and damage mechanisms are summarized for information in Table 4-1.

4.2.2 Inspection and Damage Mechanisms for Components

Reactor internal components other than bolting are subject to VT-3 examination. For the core support shield assembly cast outlet nozzles, the potential damage mechanism is thermal aging embrittlement (TE). IE is the potential damage mechanism for baffle plates. The core barrel locking device assembly, including the locking welds, baffle-to-baffle bolts, and baffle-to-former bolts (see Section 4.2.1), are subject to the potential of cracking related to IASCC, IE, and overload. Cracking due to IE and TE is the potential damage mechanism for in-core monitoring instrumentation (IMI) guide tube spiders and IMI guide tube spider-to-lower grid rib section welds. SCC is the potential damage mechanism for Alloy X-750 dowel-to-guide block welds in the lower grid assembly. TE is also the potential cracking mechanisms in the core support shield vent valve disk, top and bottom retaining ring, and disk shaft or hinge pin. VT-3 examinations of the vent valves are also conducted in conjunction with the valve operational test described in the plant technical specification or pump and valve inservice test (IST) program. The VT-3 is used to check the jack screws for proper position and for wear to determine if the valves are stuck in the open position and to detect any abnormal degradation or other conditions (such as scratches, pitting, embedded particles, and variation in the coloration of seating surfaces) or potential cracking of the lock welds and locking cups. The required VT-3 examinations and damage mechanisms are summarized in Table 4-2. The MRP-227 references for the required VT-3 examinations and damage mechanisms for both primary and expansion components are provided for information in Table 4-2.

4.3 Mockups of B&W Internal Components and Bolting

Mockups may be used for developing and modifying procedures, determining examination system parameters, training and qualifying personnel, and supporting the Technical Justifications. Mockups shall be representative of the components or bolting to be examined, and discontinuities in mockups shall be sufficient in number, size, and orientation to demonstrate and evaluate the procedure and system variables relevant to the specific examination. The specific configuration and number of component mockups and bolting, including bolting sizes and types, shall be described in the Technical Justification.

Vendor and utility mockups can be used when available. Mockups of the core barrel assembly including baffle plates, locking devices, and locking device welds may be prepared by the MRP in the absence of vendor and utility mockups to simulate potential cracking caused by the damage mechanisms identified for these components: thermal aging and irradiation embrittlement. The mockups may be full size with crack locations and orientations as would be expected for those damage mechanisms.

4.4 Existing Programs

Existing programs for inspection of reactor internals, such as are required for compliance with ASME Section XI, Subsection IWB for examination categories B-N-1 and B-N-3, are discussed in the I&E Guideline (MRP-227) but are not in the scope of this standard. The inspection methodologies and techniques and the Technical Justifications may be used to enhance and support the existing programs.

5
COMBUSTION ENGINEERING REACTOR INTERNALS INSPECTION

5.1 Summary Reactor Internals Description

The internals of Combustion Engineering (CE) PWRs consist of three major structural components and three other major component groups. The structural components are the upper internals assembly, the core support barrel, and the lower internals assembly. The other major component groups are shroud assemblies for the control elements, the core shroud assembly, and the ICI support system. The general arrangement of the CE PWR internal components is shown in Figure 5-1.

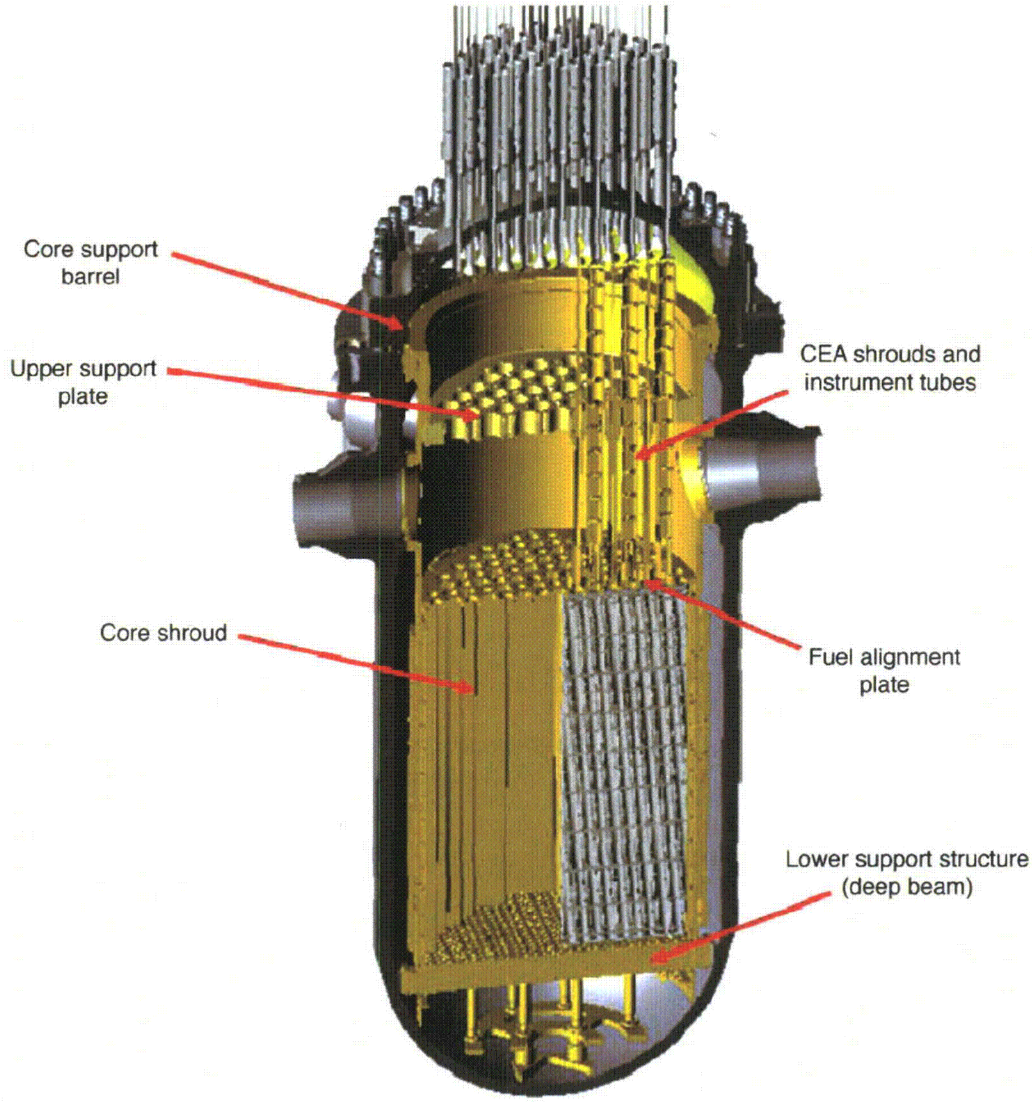


Figure 5-1
Typical Arrangement of CE Reactor Internals

(Note: some component items are rotated for clarity)

In the United States, CE reactors have three general shroud/ICI configurations: 1) bolted core shroud and top-mounted ICIs, 2) welded core shroud and top-mounted ICIs, and 3) welded core shroud and bottom-mounted ICIs.

The core support barrel consists of the core support barrel, the core support barrel upper flange, core support barrel alignment keys, and core support barrel snubbers. The upper flange ring of the core support barrel rests on a ledge in the reactor vessel.

The upper internals assembly is located above the reactor core, inside the core support barrel, and is removed during refueling. The flange on the upper end of the upper internals assembly rests on the core support barrel. The upper internals assembly includes the upper guide structure support plate, fuel assembly alignment plate, shroud assemblies for the control elements, upper guide structure grid assembly, upper guide structure cylinder, ICI supports, and the hold-down ring (or expansion compensating ring).

The core shroud is also located inside the core support barrel and is directly below the upper internals assembly. For reactors with a bolted core shroud and top-mounted ICI, the core shroud assembly is bolted to the core support barrel. For reactors with a welded core shroud and top-mounted ICI, the core shroud assembly is attached to the core support plate by tie rods or welds.

The lower internals assembly includes the core support plate, fuel alignment pins, core support columns, ICI supports, and the lower support structure beam assemblies. CE reactors 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.

The shroud assemblies for the control elements consist of control element shrouds, associated bolts, and extension shaft guides. The bottom part of the shrouds is bolted to the fuel assembly alignment plate at the lower end. The control element assembly extension shafts couple the control element drive mechanisms to the control element assemblies. The shroud assemblies for the control elements are attached by tie rods to the upper guide structure support plate.

For reactors with top-entry ICI assemblies, the ICI is inserted through the reactor vessel head through a nozzle into a guide tube. The guide tubes are connected to the top of the thimble support plate. The thimble tube support plate itself is attached to the upper guide structure support plate. As originally designed, the ICI thimble tubes extend downward from a flanged connection at the thimble support plate to the fuel alignment plate, and the lower zirconium alloy portion of the ICI thimble tube extends from the fuel alignment plate into the fuel assemblies. For plants with bottom-entry ICI assemblies, the ICI enters the core through the lower internals assembly, which also provides the ICI support.

5.2 Technical Justifications

The qualifying Technical Justifications are required for UT and examinations of CE internal components and bolting, respectively. As described in Section 2, the Technical Justification shall include descriptions of the bolting designs and damage mechanisms, a general description of the examination system, inspection parameters including procedure and personnel requirements, and a description of procedure experience. The damage mechanisms are required to be addressed in the Technical Justification for each primary inspection item on the basis of susceptibility. The primary components as described in Section 4 of the Inspection and Evaluation Guidelines (MRP-227) have at least one damage mechanism above the screening criteria and require additional aging management program elements to manage the effects. The expansion components are scheduled for examination only on the basis of the findings from examination of the primary examination. The examination requirements identified in MRP-227 for both primary and expansion components are summarized for information in Tables 5-1 and 5-2, and the damage mechanisms for the primary components identified in MRP-227 are described for information in Sections 5.2.1 and 5.2.2. Typical drawings for these components are shown for information only in Figures 5-2 through 5-18.

Table 5-1
Required Volumetric Examinations for CE Reactor Internals (Information)

Volumetric (UT) Examinations						
Primary Examination Item (See MRP-227, Table 4-2)	**Material (See MRP-227, Table 3-2)**	**Plant (See MRP-227, Table 3-2)**	**Figure**	**Damage Mechanism (See MRP-227, Table 3-2)**	**Expansion Components (See MRP-227, Table 4-5)**	**Requirements**
Core shroud assembly (bolted) Core shroud bolts	All: 316 SS	Bolted plant designs	n/a	Cracking (IASCC, fatigue)	Core support column bolts (Fig. 5-8) Barrel-shroud bolts	See MRP-227, Tables 4-2 and 4-5

Table 5-2
Required Visual Examinations for CE Reactor Internals (Information)

Visual (EVT-1 or VT-1) Examinations						
Primary Examination Item (See MRP-227, Table 4-2)	**Material (See MRP-227, Table 3-2)**	**Plant (See MRP-227, Table 3-2)**	**Figure**	**Damage Mechanism (See MRP-227, Table 3-2)**	**Expansion Component (See MRP-227, Table 4-5)**	**Requirements**
Core shroud assembly (welded) Core shroud plate-former plate weld	304 SS	Plants with core shrouds assembled in two vertical sections	5-2	Cracking (IASCC)	Remaining axial welds	See MRP-227, Tables 4-2 and 4-5
Core shroud assembly (welded) Shroud plates	304 SS	Plant designs with core shrouds assembled with full-height shroud plates.	n/a	Cracking (IASCC)	Remaining axial welds	
Core shroud assembly (welded) Entire assembly	Plates, ribs, rings: 304 SS Bolts: 316 SS Tie Rod: 348 SS Guide lugs, inserts: A286 SS	Plants with core shrouds assembled in two vertical sections	5-3	Distortion (void swelling)	None	
Core support barrel (CB) assembly Upper flange weld	304 SS	All	5-4	Upper flange and core barrel assembly welds: cracking (SCC) Core support column welds: cracking (SCC, IASCC, fatigue)	Other core barrel assembly welds Core support column welds (Fig. 5-8 through 5-10) Lower core barrel flange (Fig. 5-4)	

Table 5-2 (continued)
Required Visual Examinations for CE Reactor Internals (Information)

Visual (EVT-1 or VT-1) Examinations						
Primary Examination Item (See MRP-227, Table 3-2)	**Material (See MRP-227, Table 3-2)**	**Plant (See MRP-227, Table 3-2)**	**Figure**	**Damage Mechanism (See MRP-227, Table 3-2)**	**Expansion Component (See MRP-227, Table 4-5)**	**Requirements**
Core support barrel assembly Lower flange weld	304 SS	All	5-4	Cracking (fatigue)	None	See MRP-227 Tables 4-2 and 4-5
Lower support structure Core support plate	304/304L SS	All plants with a core support plate	5-10	Cracking (fatigue)	None	
Upper internals assembly Fuel alignment plate	304 SS	All plants with core shrouds assembled with full-height shroud plates	5-5	Cracking (fatigue)	None	
Lower support structure Deep beams	304 SS	All plants with core shrouds assembled with full-height shroud plates.	5-7	Cracking (fatigue)	None	

Table 5-2 (continued)
Required Visual Examinations for CE Reactor Internals (Information)

Visual (VT-3) Examinations						
Primary Examination Item (See MRP-227, Table 4-2)	**Material (See MRP-227, Table 3-2)**	**Plant (See MRP-227, Table 3-2)**	**Figure**	**Damage Mechanism (See MRP-227, Table 3-2)**	**Expansion Component (See MRP-227, Table 4-5)**	**Requirements**
Core shroud assembly (bolted) Entire assembly	Plates, ribs, rings: 304 SS Bolts: 316 SS Tie Rod: 348 SS Guide lugs, inserts: A286 SS	Bolted plant designs	5-12, 5-17, 5-18	Distortion (void swelling)	None	See MRP-227, Tables 4-2 and 4-5
Control element assembly Peripheral instrument tubes	304 SS	All	5-6	Cracking (SCC, fatigue)	Remaining instrument tubes within the CEA shroud assemblies	

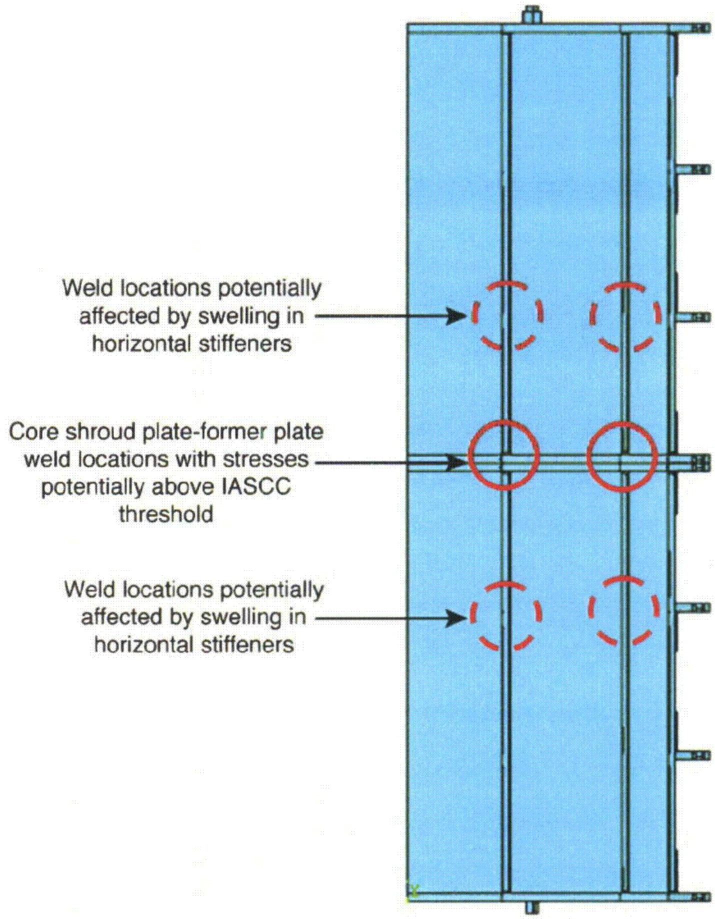


Figure 5-2
Potential Crack Locations for CE Welded Core Shroud Assembled in Stacked Sections

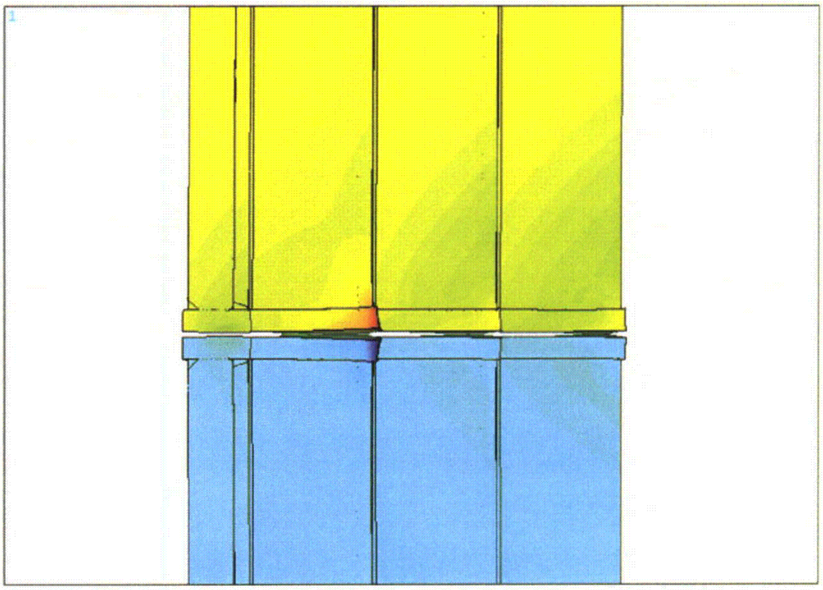


Figure 5-3
Illustration of Potential Separation Between Core Shroud Sections Caused by Swelling Induced Warping of Thick Flange Plates in CE Welded Core Shroud Assembled in Stacked Sections

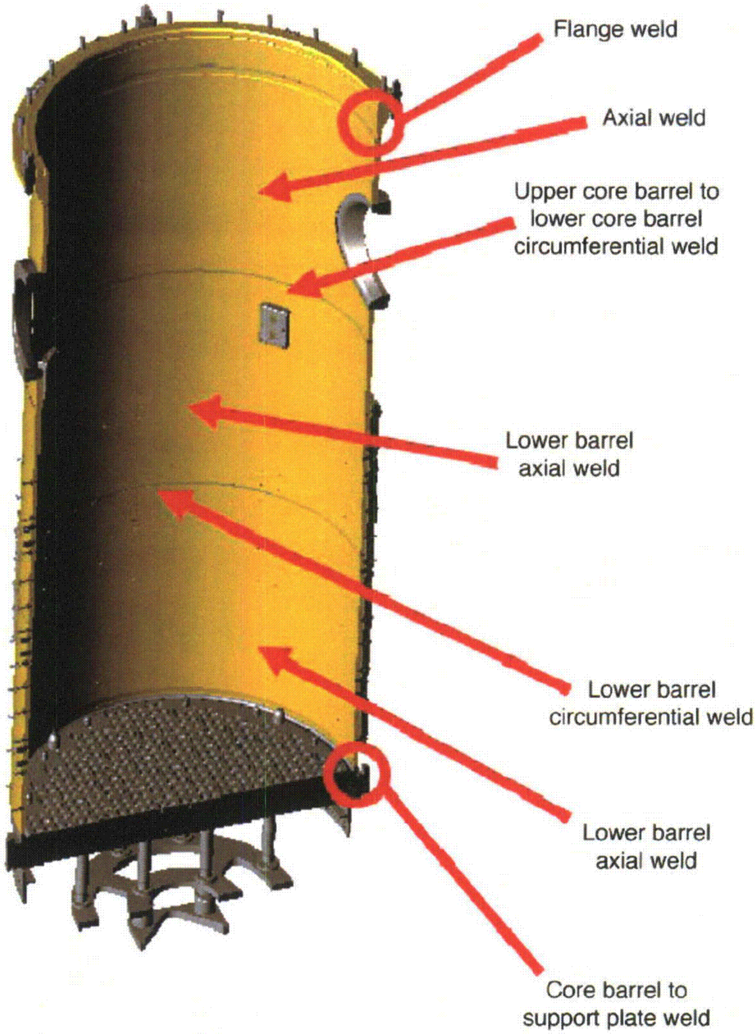


Figure 5-4
Typical CE Core Support Barrel Assembly

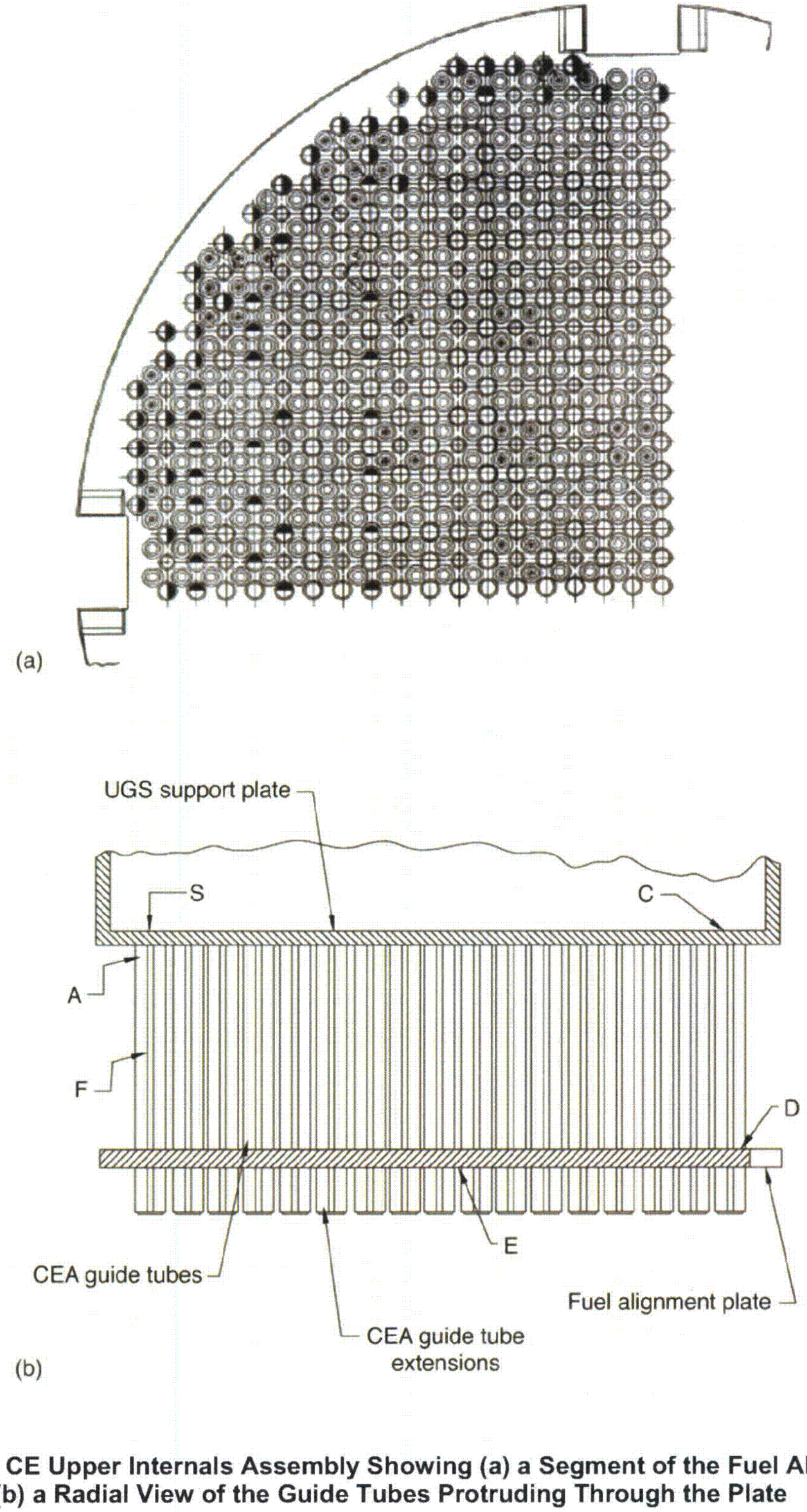


Figure 5-5
System 80 CE Upper Internals Assembly Showing (a) a Segment of the Fuel Alignment Plate and (b) a Radial View of the Guide Tubes Protruding Through the Plate

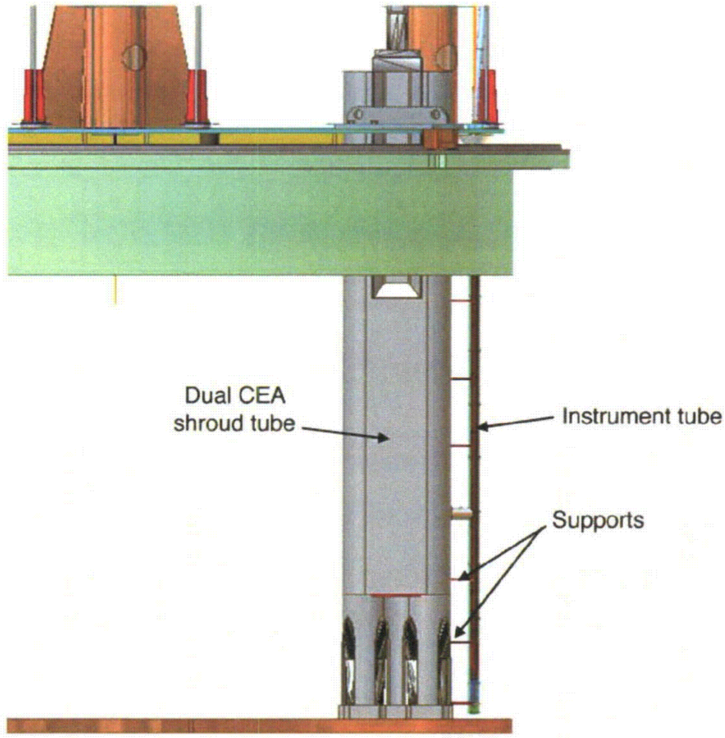
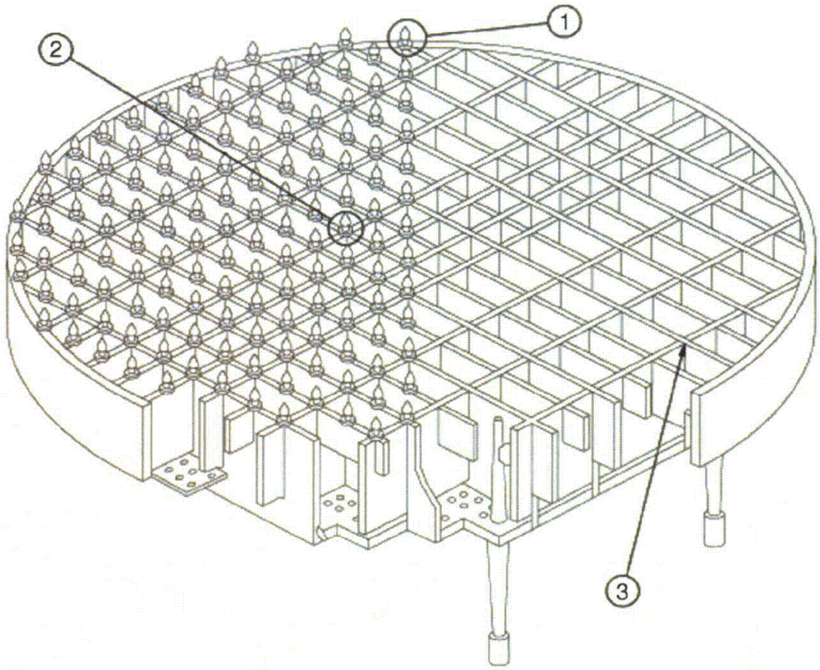


Figure 5-6
Individual Instrument Tube and Connected CEA Shroud Tube



Illustrates the deep beam grid structure (number 3), as well as the fuel alignment pins (numbers 1 and 2) on the left-side of the illustration.

Figure 5-7
Isometric View of the Lower Support Structure in the CE System 80 Units. Fuel Rests on Alignment Pins.

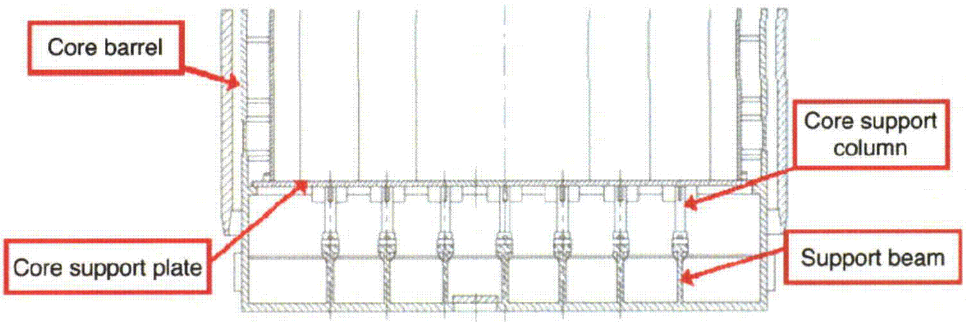
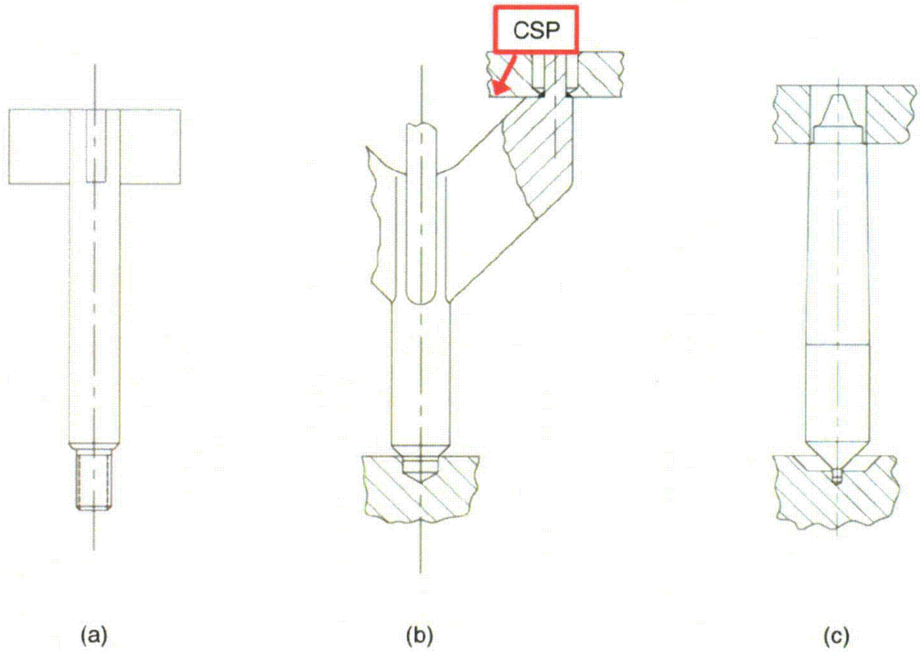
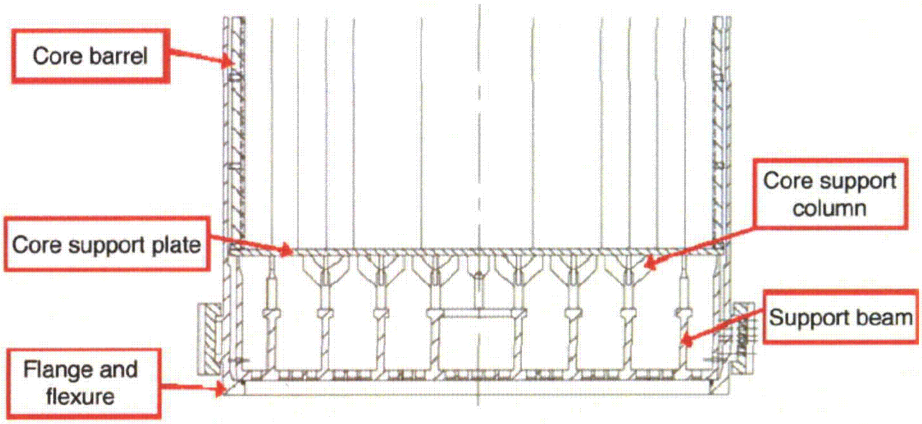


Figure 5-8
CE Lower Support Structure Assembly Design I: Integrated Core Barrel and Lower Support Structure Assembly with a Core Support Plate

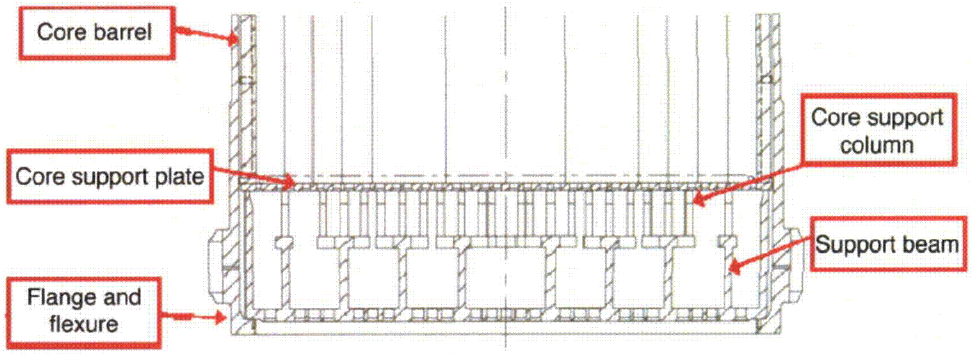


(a) Early support column design
(b) Winged support column design and plants with second generation core support assemblies
(c) Later support column design used in plants with second generation core support assemblies

Figure 5-9
CE Core Support Columns



Winged core support columns



Core support columns without winged design

Figure 5-10
CE Lower Support Structure Assembly Design II: Separate Core Barrel and Lower Support Structure Assembly with a Core Support Plate

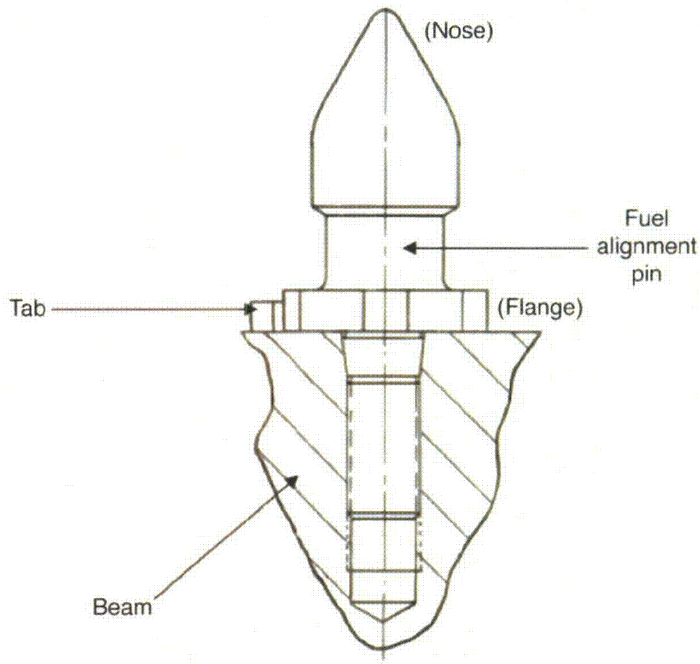


Figure 5-11
Cross-Section of the CE System 80 Fuel Alignment Pin (threaded and pre-loaded) in the Beam with Locking Tab

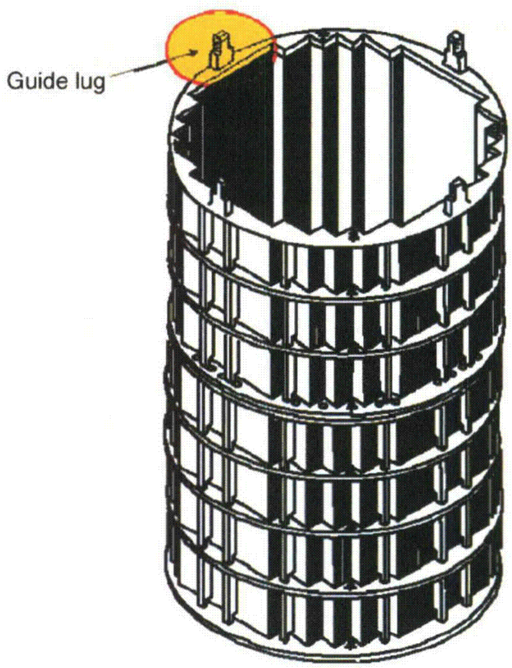


Figure 5-12
CE Core Shroud Assembly with One of the Guide Lug Inserts Highlighted

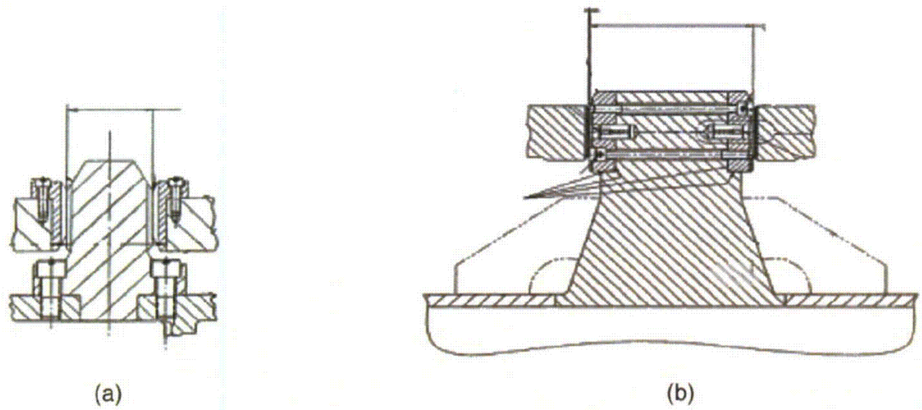


Figure 5-13
(a) Guide Lug and Insert; (b) Guide Lug and Insert for CE System 80

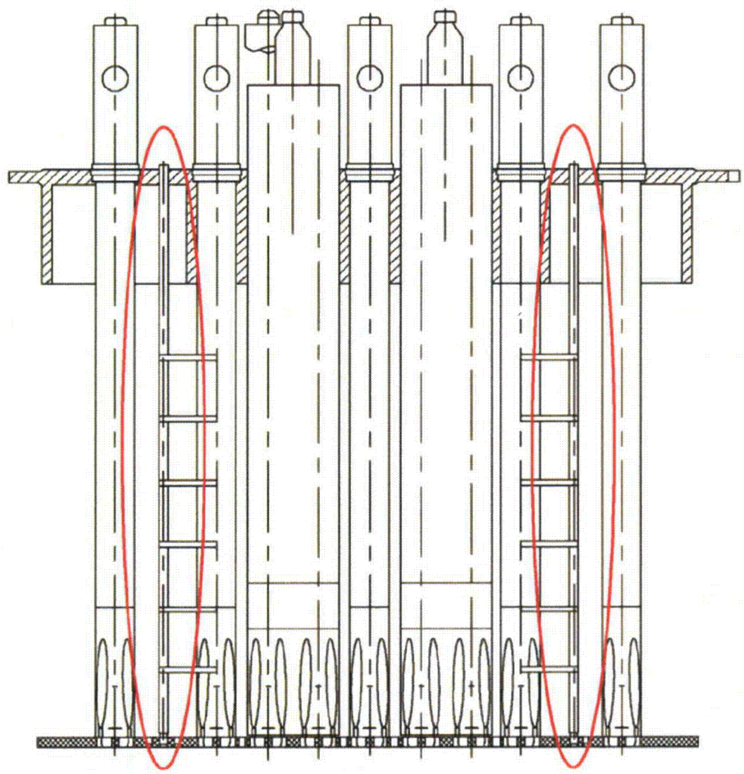


Figure 5-14
Schematic Illustration of CEA Shroud Instrument Tubes (circled in red) and the Welded Supports Attaching Them to the CEA Shroud Tube

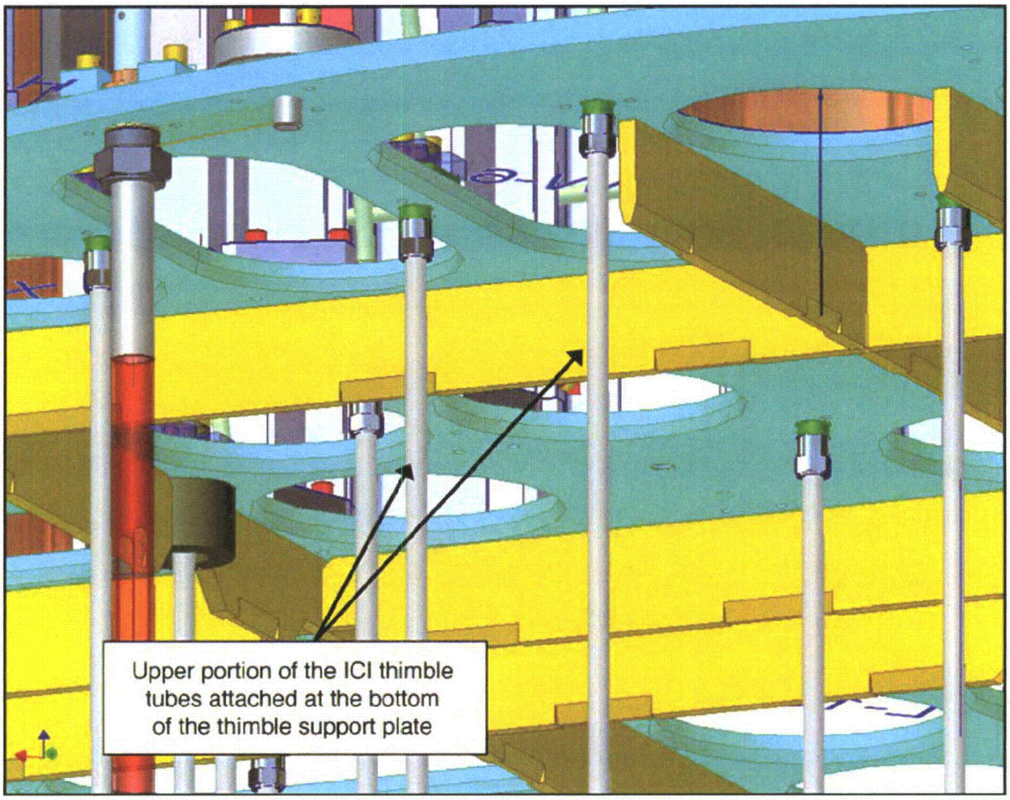
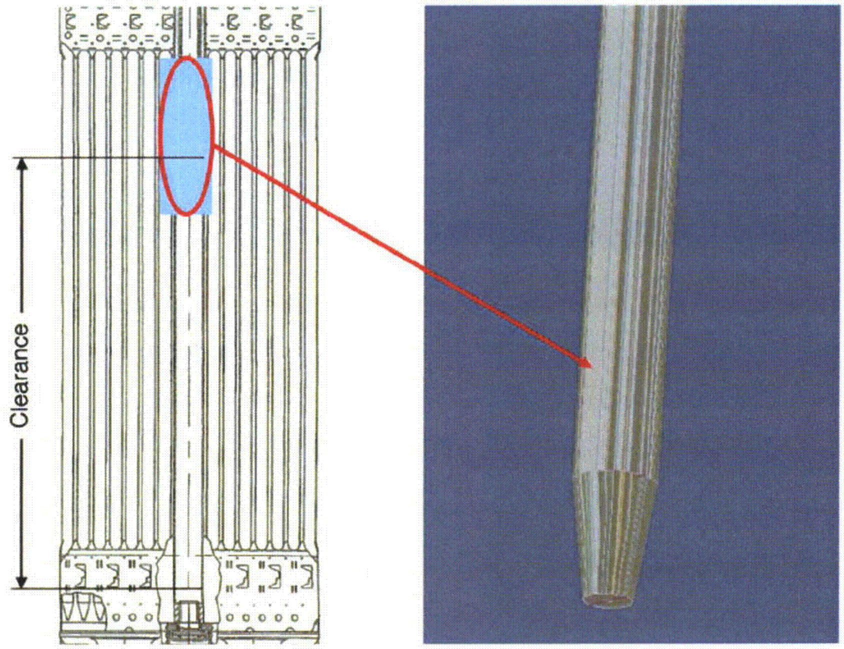


Figure 5-15
Bottom View of the CE Replacement ICI Support Plate and Thimble Tube Assembly



(a) (b)

Figure 5-16
(a) Cross-Section View of a CE ICI Thimble Tube Showing Clearance in the Bottom of a Fuel Assembly; (b) Magnified View of the Bullet-Nose End of the Tube

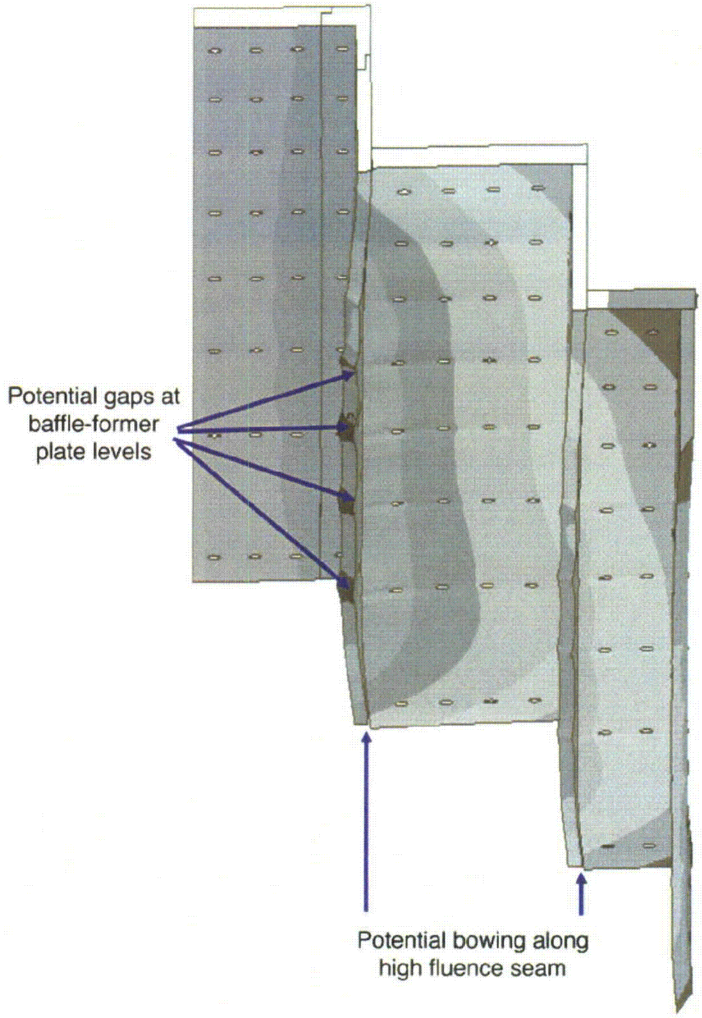


Figure 5-17
Exaggerated View of Void Swelling Induced Distortion in a Bolted CE Shroud Assembly. (This figure for a Westinghouse baffle-former assembly with the same damage mechanism is also representative of the CE bolted shrouds.)

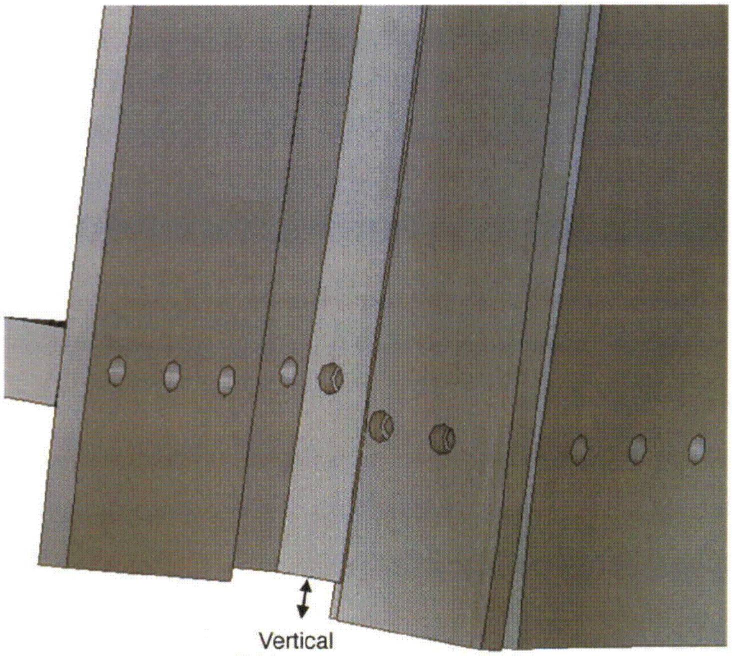


Figure 5-18
Vertical Displacement in Bolted CE Shroud Assembly Caused by Void Swelling. (This figure showing Westinghouse baffle plates with the same damage mechanism is also representative of the CE bolted shrouds.)

5.2.1 Inspection and Damage Mechanisms for Bolting

For core shroud assemblies designed with bolting, the bolting requires examination using the ultrasonic method. The potential damage mechanisms are SCC and fatigue cracking. No other bolting is identified for examination as a primary component. In addition to fatigue, IASCC has been identified as a potential damage mechanism for the barrel-shroud bolts and core support column bolts examined as expansion components. The MRP-227 references for required bolting examinations and damage mechanisms are summarized for information in Table 5-1.

5.2.2 Inspection and Damage Mechanisms for Components

Reactor internal components, other than bolting, are subject to EVT-1, VT-1, or VT-3 examinations. IASCC is the potential damage mechanism for welded core shroud plates and shroud plate–to–former plate welds, which are subject to EVT-1 examination. EVT-1 examination is also required for the core support barrel upper flange weld, which is subject to potential SCC. Fatigue cracking is the potential damage mechanism for the core support barrel lower flange weld, the core support plate, the deep beams of the lower support structure, and the fuel alignment plate—all of which are subject to EVT-1 examination. VT-1 is performed on some designs for detection of separation between upper and lower core shroud segments due to distortion. VT-3 examination is required for the instrument tubes, which are subject to both SCC and fatigue, and for the inside surfaces of the core shroud assembly where distortion due to void swelling may occur. The MRP-227 references for the required EVT-1, VT-1, and VT-3 examinations and damage mechanisms for both primary and expansion components are provided for information in Table 5-1.

5.3 Mockups of CE Internal Components and Bolting

Mockups may be used for developing and modifying procedures, determining examination system parameters, training and qualifying personnel, and supporting the Technical Justifications. Mockups shall be representative of the components or bolting to be examined, and discontinuities in mockups shall be sufficient in number, size, and orientation to demonstrate and evaluate the procedure and system variables relevant to the specific examination. The specific configuration and number of component mockups and bolting, including bolting sizes and types, shall be described in the Technical Justification.

Vendor and utility mockups can be used when available. Mockups of the core barrel assembly, including flange welds, may be prepared by the MRP in the absence of vendor and utility mockups to simulate potential cracking caused by SCC and the fatigue damage mechanisms identified for these components. The mockups may be full size with crack locations and orientations as would be expected for those damage mechanisms.

5.4 Existing Programs

Existing programs for inspection of reactor internals, such as are required for compliance with ASME Section XI, Subsection IWB for examination categories B-N-1 and B-N-3, are discussed in the I&E Guideline (MRP-227) but are not in the scope of this standard. The inspection methodologies and techniques as well as the Technical Justifications may be used to enhance and support the existing programs.

6 WESTINGHOUSE REACTOR INTERNALS INSPECTION

6.1 Summary Reactor Internals Description

The internals of Westinghouse PWRs consist of two major assemblies: the upper internals assembly, which is removed during each refueling operation to obtain access to the reactor core, and a lower internals assembly that may be removed after a full core offload. 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. There are a number of significant variations in the internals design of Westinghouse reactors operating in the United States, including the baffle-barrel region flow design (downflow, upflow, and converted upflow) and upper support plate configuration. The general arrangement of Westinghouse PWR internal components is shown in Figure 6-1.

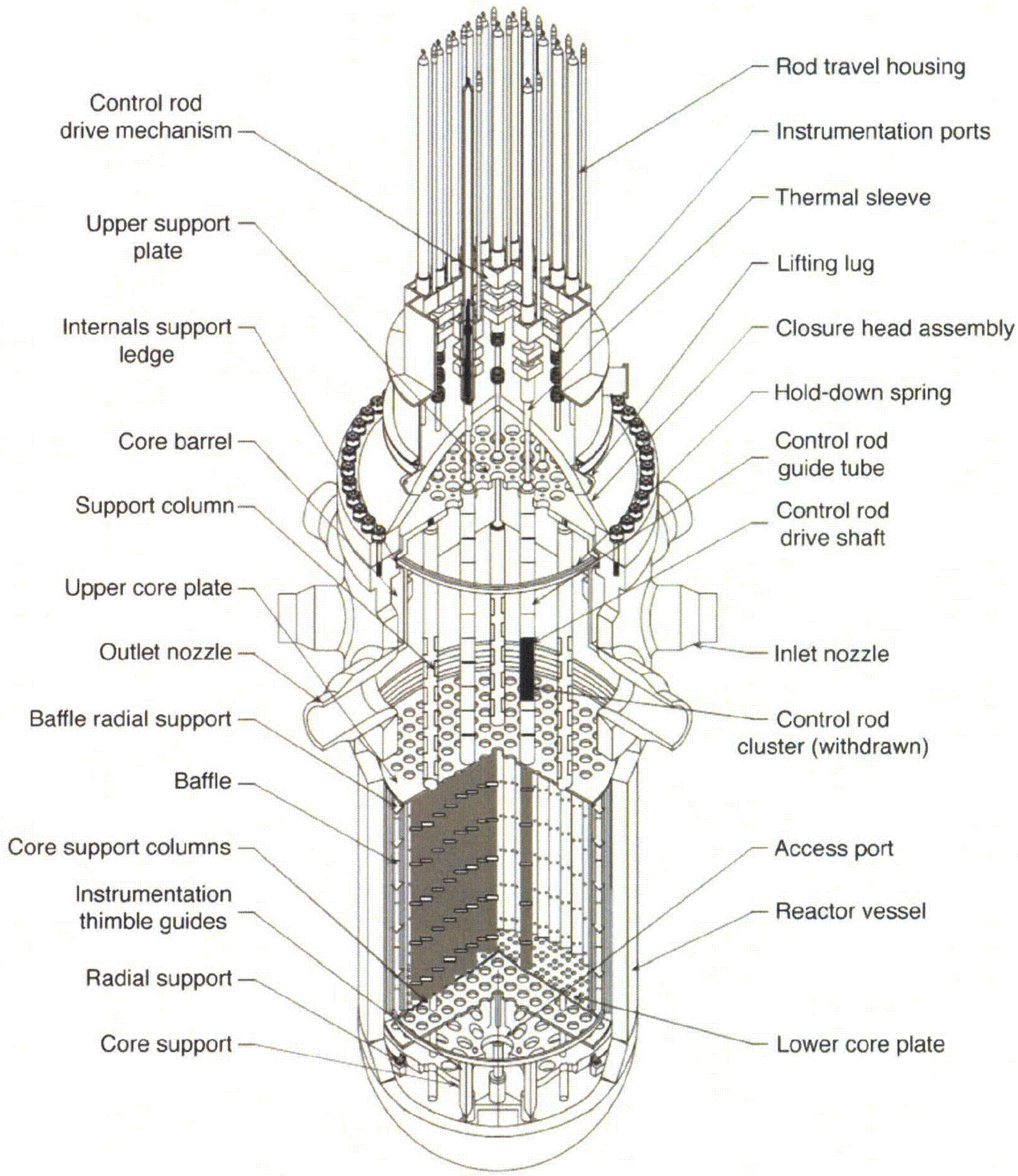


Figure 6-1
Typical Arrangement of Westinghouse Reactor Internals

The major subassemblies of the upper internals assembly are the upper core plate (UCP) and fuel alignment pins, the upper support column assemblies, the control rod guide tube assemblies and flow downcomers, the upper plenum, and the upper support plate assembly. The upper support plate (USP) acts as the divider between the upper plenum and the reactor vessel head and provides a base for the rest of the upper internals components. The upper support columns and the guide tubes are attached to the USP. The UCP alignment pins locate the UCP laterally with respect to the lower internals assembly and are the interfacing components between the UCP and the core barrel. The upper instrumentation columns are also attached, by bolting, to the USP.

There are three different USP designs: deep beam, top hat, and an inverted top hat. The USP, the upper support columns, and the UCP are core support structures.

Both the lower internals and upper internals assemblies position and support the reactor core. The lower internals are aligned with the upper internals primarily by the UCP alignment pins and secondarily by the head/vessel alignment pins. The lower core plate (LCP) positions and supports the core. It is located above the lower core support inside the core barrel near the bottom of the lower support assembly, as shown in Figure 6-1. The lower support is a forging in most plants, but some four-loop plants use a cast lower support. The core sits directly on the LCP, which is supported by the lower support columns (see Figures 6-2 and 6-3).

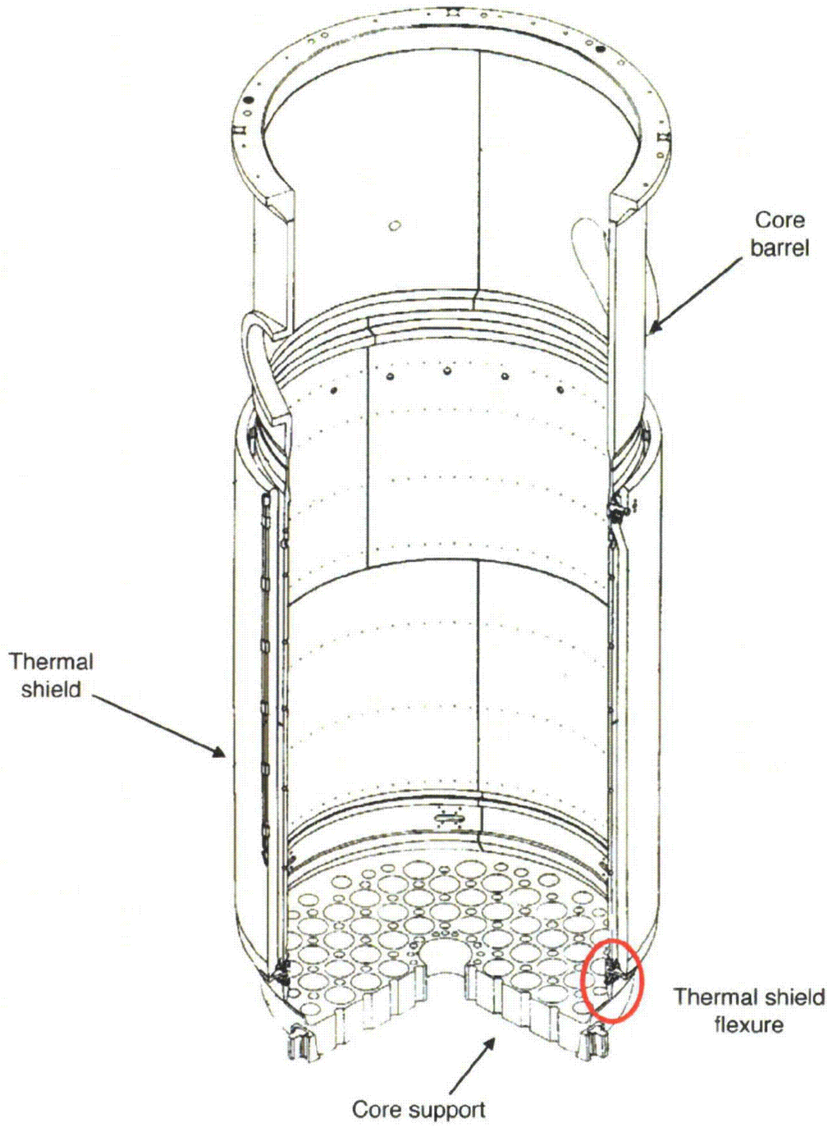


Figure 6-2
Westinghouse Thermal Shield Flexures

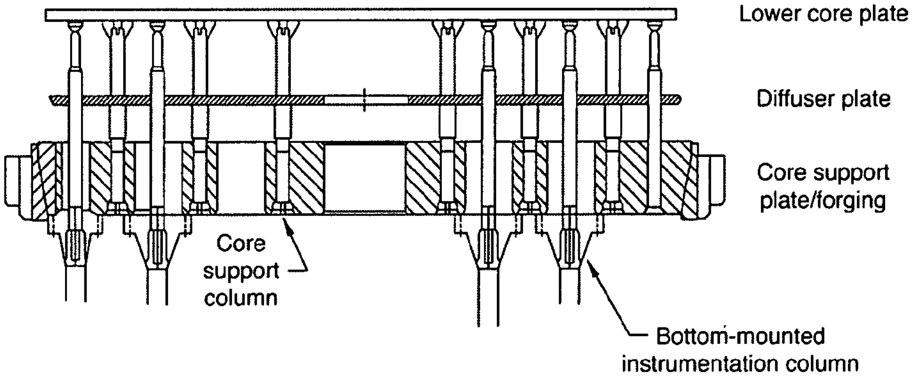


Figure 6-3
Westinghouse Lower Core Support Structure Detail Showing Bottom-Mounted Instrumentation Columns

The core barrel is part of the lower internals assembly. A number of components are attached to the core barrel or its 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 core support, and the LCP. The radial support keys restrain transverse motion of the core barrel while allowing radial and axial thermal movement from expansion. The baffle/former assembly forms the interface between the core and the core barrel. The vertical baffle plates are attached to the horizontal former plates by bolts. The individual baffle plates are attached to each other by edge bolts at selected corners, and some plants have bolted corner brackets behind the baffle plates. 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 all core support structures.

The bottom-mounted instrumentation (BMI) columns, shown in Figures 6-3 and 6-4, provide a path for the flux thimbles into the core from the bottom of the vessel. The flux thimble itself is a long, slender stainless steel sealed tube that passes through the vessel penetration, through the lower internals assembly, and finally extends to the top of the fuel assembly.

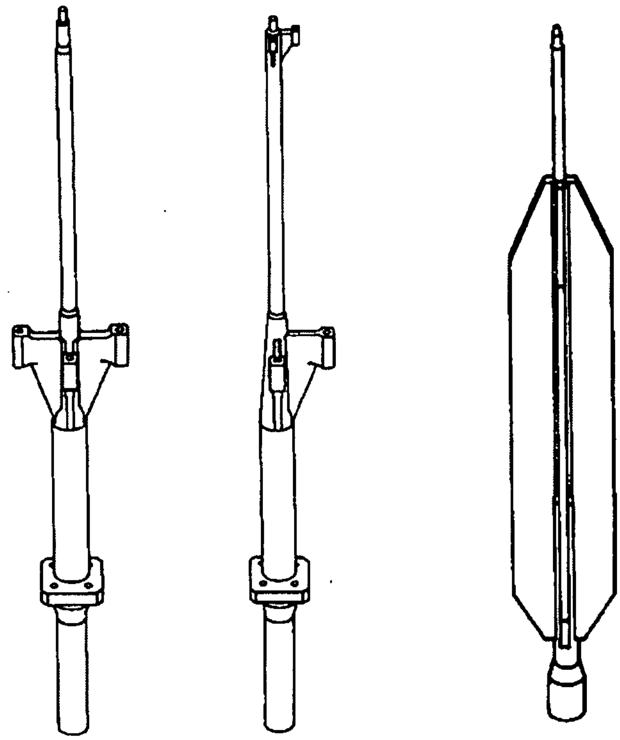


Figure 6-4
Examples of Westinghouse BMI Column Designs

6.2 Technical Justifications

The qualifying Technical Justifications are required for UT and examinations of Westinghouse internal components and bolting, respectively. As described in Section 2, the Technical Justification shall include descriptions of the bolting designs and damage mechanisms, a general description of the examination system, inspection parameters including procedure and personnel requirements, and a description of procedure experience. The damage mechanisms are required to be addressed in the Technical Justification for each primary inspection item on the basis of susceptibility. The primary components as described in Section 4 of the Inspection and Evaluation Guidelines (MRP-227) have at least one damage mechanism above the screening criteria and require additional aging management program elements to manage the effects. The expansion components are scheduled for examination only on the basis of the findings from examination of the primary examination. The examination requirements identified in MRP-227 for both primary and expansion components are summarized for information in Tables 6-1 and 6-2. The damage mechanisms identified in MRP-227 for the primary components are described for information in Sections 6.2.1 and 6.2.2. Typical drawings for these components are shown for information only in Figures 6-5 through 6-18.

Table 6-1
Required Volumetric Examinations for Westinghouse Reactor Internals (Information)

Volumetric (UT) Examinations						
Primary Examination Item (See MRP-227, Table 4-3)	**Material (See MRP-227, Table 3-3)**	**Plant (See MRP-227, Table 3-3)**	**Figure**	**Damage Mechanism (See MRP-227, Table 3-3)**	**Expansion Components (See MRP-227, Table 4-6)**	**Requirements**
Baffle-former assembly Baffle-former bolts	Baffle-former and core barrel bolts: 316 SS, 347 SS Lower support column bolts: 304 SS	All	6-9, 6-10	Cracking (IASCC, fatigue)	Barrel-former bolts (Fig. 6-14) Lower support column bolts (Fig. 6-11)	UT examination method for both primary and expansion components. Baffle-former bolts are accessible from the core side with possible interference by head and locking device design. Access to barrel-former bolts is limited by thermal shields or neutron pads.
Note: As expansion components, UT may be required on baffle-edge bolts and baffle-to-former-bolts, based on the findings of the primary VT-3 examination of the baffle-edge bolts (see explanation in Table 6-2).						

Table 6-2
Required Visual Examinations for Westinghouse Reactor Internals (Information)

Primary Examination Item (See MRP-227, Table 4-3)	Material (See MRP-227, Table 3-3)	Plant (See MRP-227, Table 3-3)	Figure	Damage Mechanism (See MRP-227, Table 3-3)	Expansion Component (See MRP-227, Table 4-6)	Requirements
Visual (EVT-1) Examinations						
Control rod guide tube assembly Lower flanges	Lower flange and lower support column: CF-8 BMI column bodies: 304 SS	All	6-7	Lower flange: cracking (SCC, fatigue) BMI column bodies: cracking (fatigue) Lower support column bodies: cracking (IASCC)	BMI column bodies, lower support column bodies (cast)	See MRP-227, Tables 4-3 and 4-6
Core barrel assembly Upper core barrel flange weld	304 SS	All	6-8	Core barrel: cracking (SCC) Lower support column bodies: cracking (IASCC)	Other core barrel welds, lower support column bodies (non-cast)	
Direct Measurement (EVT-1, VT-1 or 3 Resolution Check Not Required))						
Alignment and interfacing components Internals hold-down spring	304 SS	All plants with 304 stainless hold-down springs	6-13, 6-18	Distortion/relaxation (loss of load)	None	See MRP-227, Tables 4-3 and 4-6

Table 6-2 (continued)
Required Visual Examinations for Westinghouse Reactor Internals (Information)

Visual (VT-3) Examinations						
Primary Examination Item (See MRP-227, Table 4-3)	**Material (See MRP-227, Table 3-3)**	**Plant (See MRP-227, Table 3-3)**	**Figure**	**Damage Mechanism (See MRP-227, Table 3-3)**	**Expansion Component (See MRP-227, Table 4-6)**	**Requirements**
Control rod guide tube assembly Guide plates (cards) C tubes Sheaths	304 SS	All	6-6	Wear	None	
Baffle-former assembly Baffle-edge bolts	316 SS 347 SS	All	6-9	Cracking (IASCC)	Volumetric (UT) of baffle-edge bolts and/or baffle-to-former bolts if primary VT-3 reveals evidence of extensive failures	
Baffle-former assembly Entire assembly	304 SS	All	6-9, 6-10, 6-11, 6-12	Distortion (void swelling)	None	
Thermal shield flexures	304 SS	All	6-2	Cracking (fatigue) and wear	None	See MRP-227, Tables 4-4 and 4-6

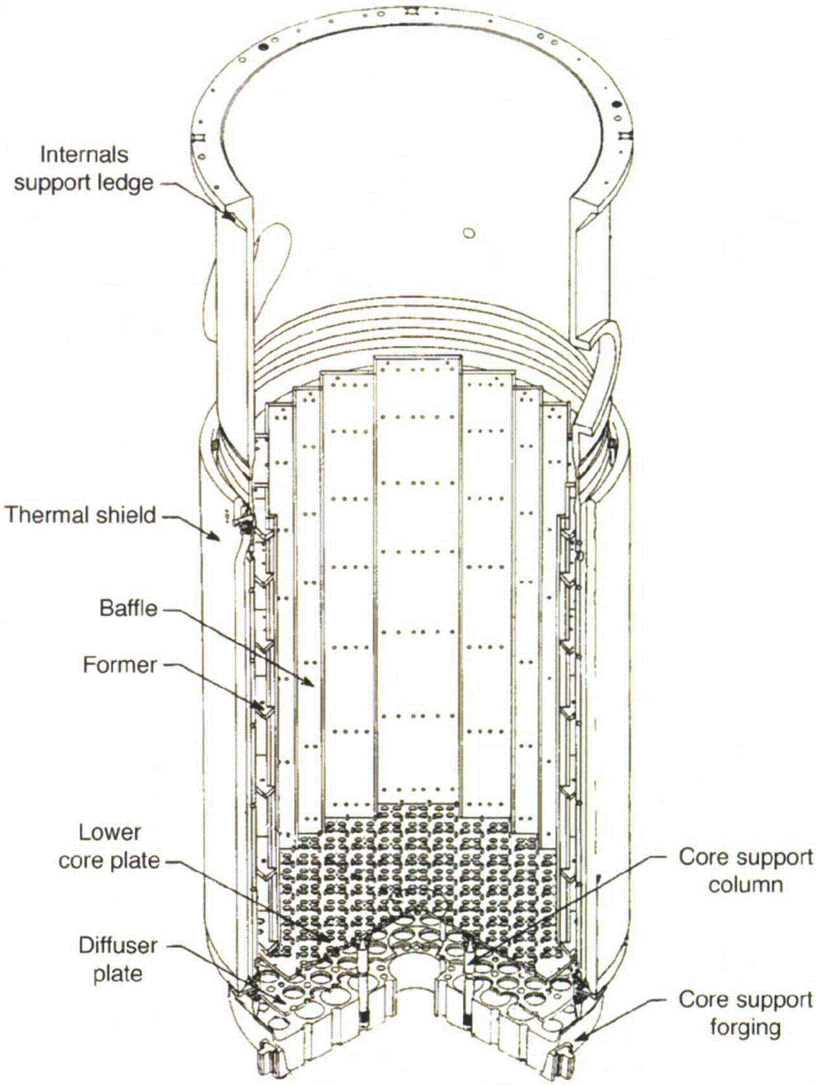


Figure 6-5
Westinghouse Core Baffle-Barrel Structure

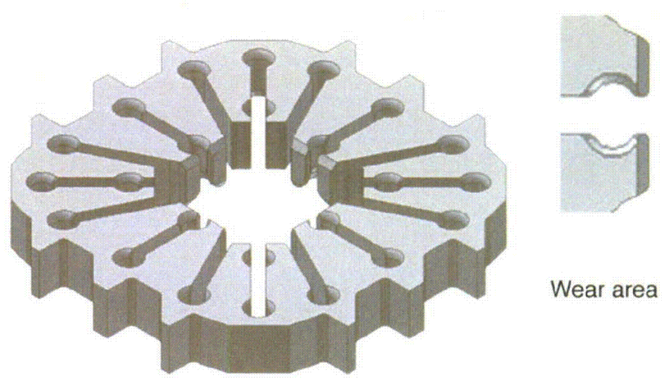


Figure 6-6
Typical Westinghouse Control Rod Guide Card (17x17 Fuel Assembly)

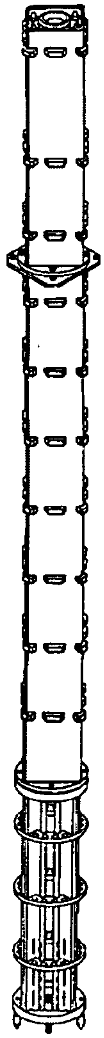


Figure 6-7
Typical Westinghouse Control Rod Guide Tube Assembly

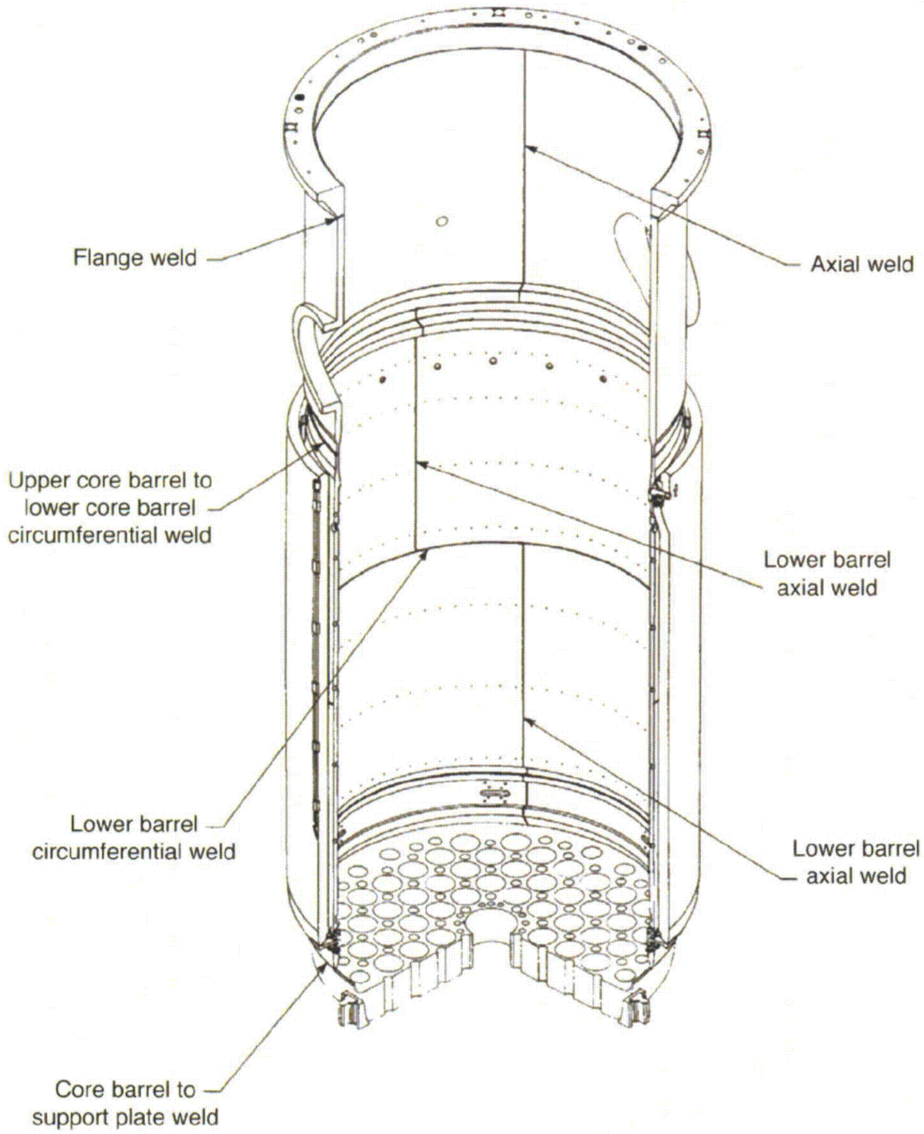


Figure 6-8
Typical Westinghouse Core Barrel Showing Weld Locations

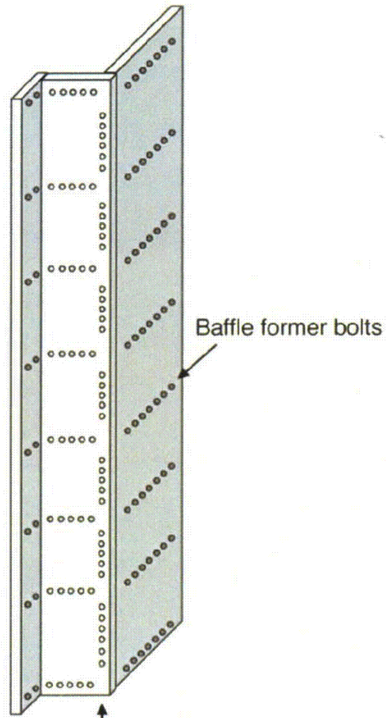


Figure 6-9
Westinghouse Baffle Edge Bolt and Baffle-Former Bolt Locations at High Fluence Seams in the Baffle-Former Assembly

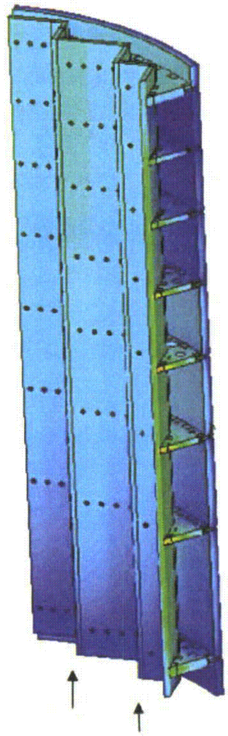


Figure 6-10
Westinghouse High Fluence Seams for Baffle-Former Bolt Locations

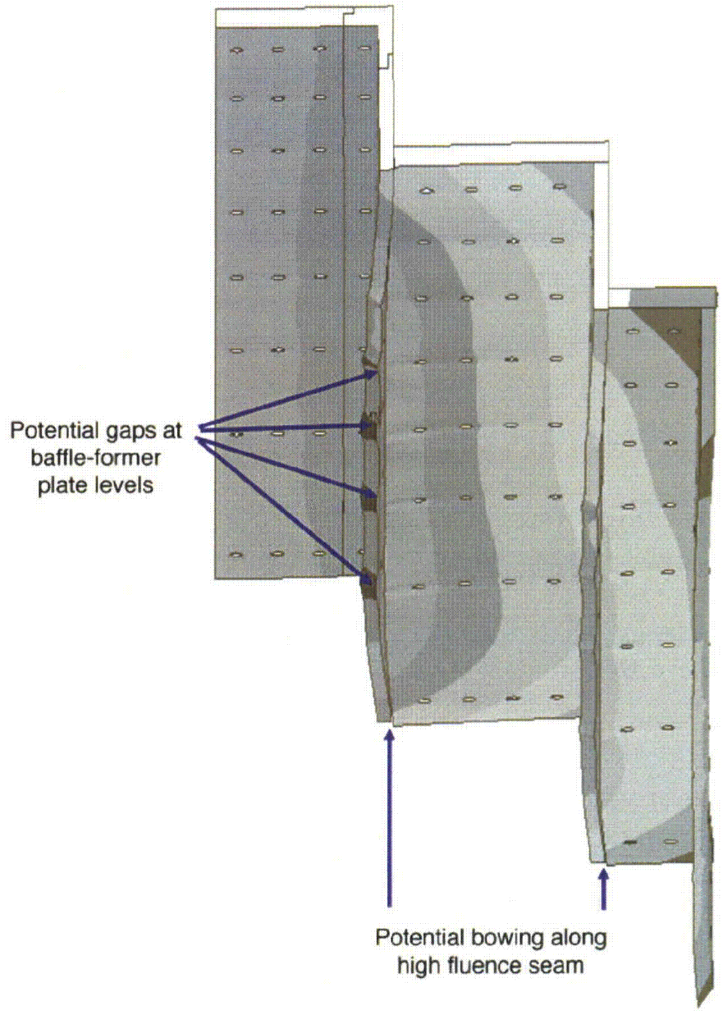


Figure 6-11
Exaggerated View of Void Swelling Induced Distortion in Westinghouse Baffle-Former Assembly

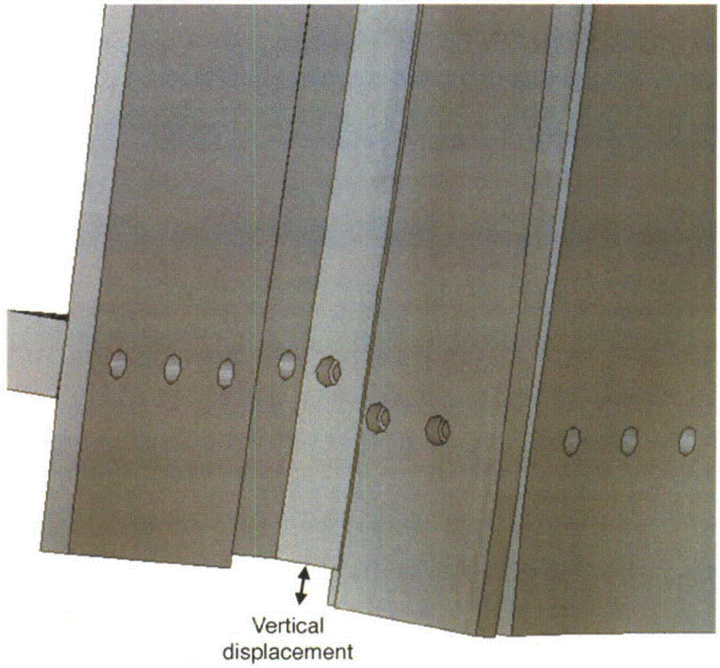


Figure 6-12
Vertical Displacement of Westinghouse Baffle Plates Caused by Void Swelling

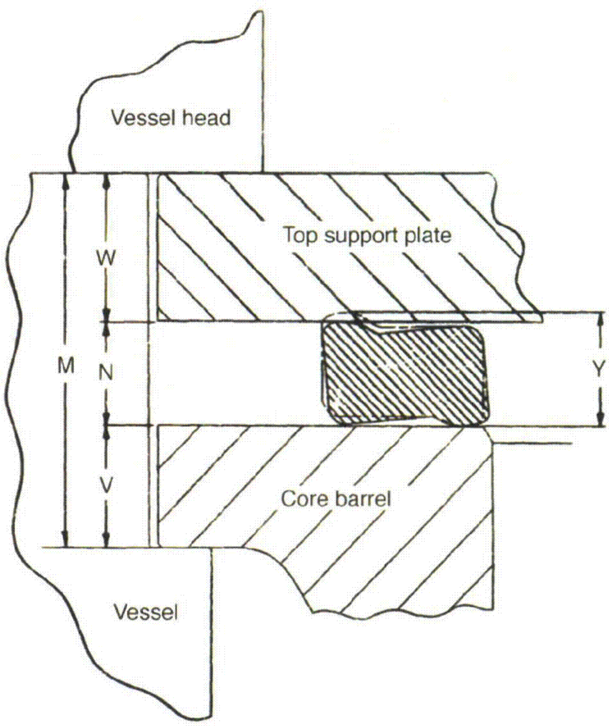


Figure 6-13
Cross-Section of Internals Hold Down Spring Compressed Between Westinghouse Core Barrel and Top Support Plate

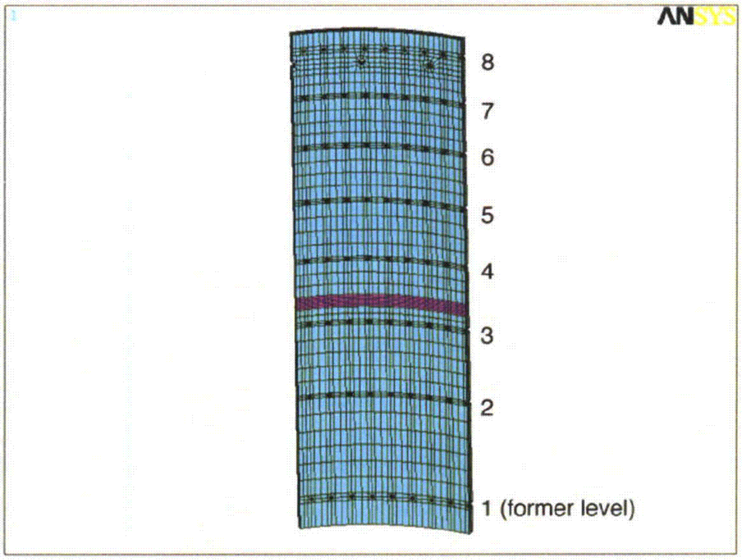


Figure 6-14
Westinghouse Barrel-Former Bolt Locations and Location of Circumferential Weld in Section of Core Barrel (shown in purple)

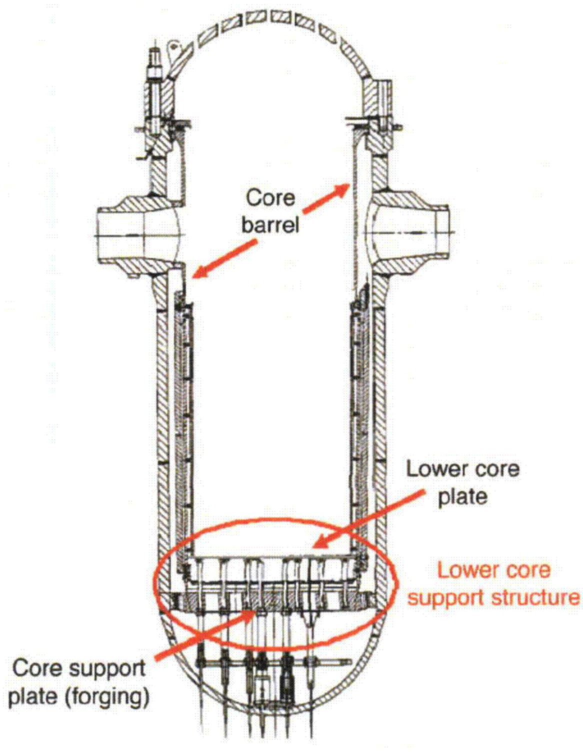


Figure 6-15
Location of Westinghouse Lower Core Support Structure

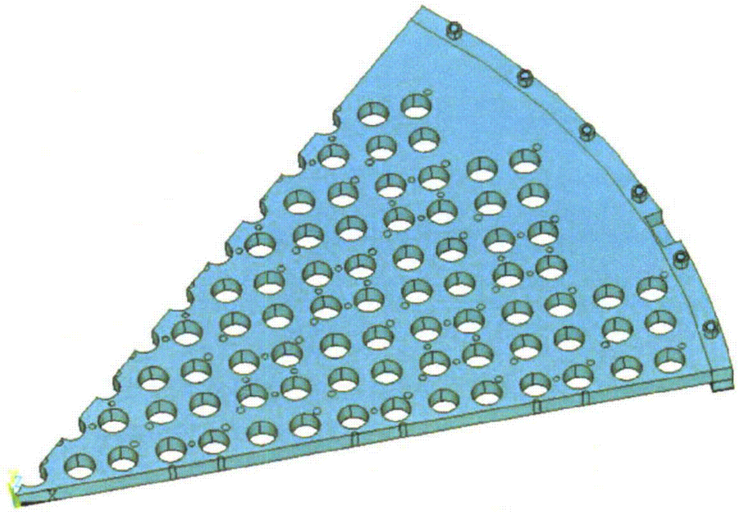
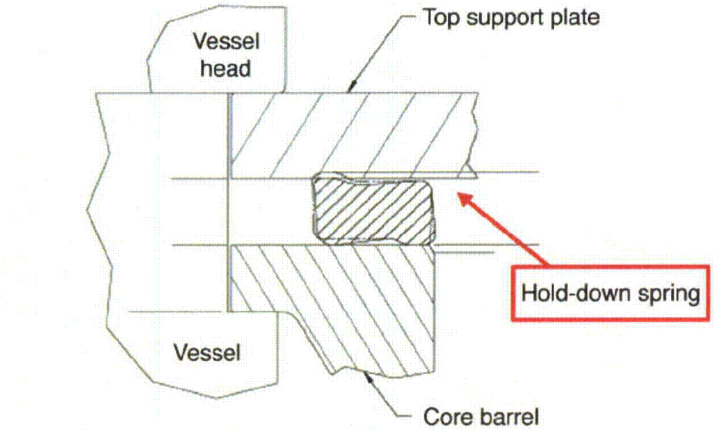
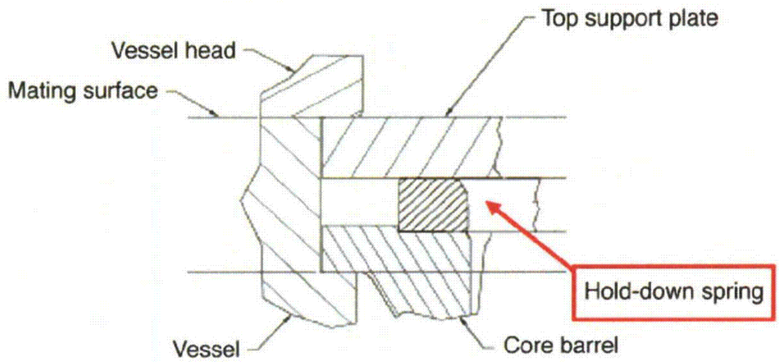


Figure 6-16
Example of Westinghouse Bolted Lower Core Plate: One-Eighth Segment from a Three-Loop Plant

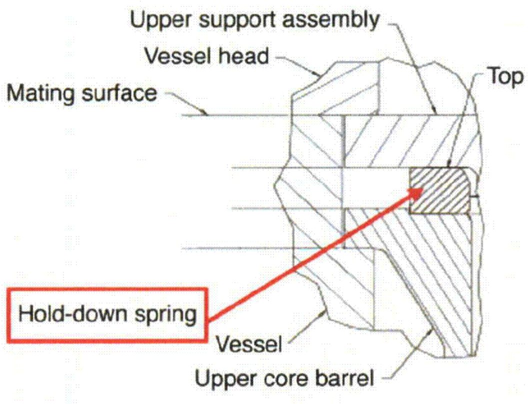
Figure 6-17
Typical Westinghouse Core Support Column



(a) Two-loop plant



(b) Three-loop plant



(c) Four-loop plant

Figure 6-18
Variations on Hold-Down Spring Configurations

6.2.1 Inspection and Damage Mechanisms for Bolting

Baffle-to-former bolting requires examination using the ultrasonic method. The potential damage mechanisms are IASCC and fatigue cracking. No other bolting is identified for examination as a primary component. IASCC and fatigue have also been identified as potential damage mechanisms for the barrel-to-former bolts and lower support column bolts, which are examined as expansion components. The MRP-227 references for required bolting examinations and damage mechanisms are summarized for information in Table 6-1.

6.2.2 Inspection and Damage Mechanisms for Components

Reactor internal components other than bolting are subject to EVT-1, VT-1, or VT-3 examinations; for the internals hold-down spring, direct measurement is required. EVT-1 is required for the lower flanges of the control rod guide tube assemblies and for the upper core barrel flange weld. Both are subject to potential damage from SCC; the lower flanges of the control rod guide tube assemblies are also subject to potential damage from fatigue. VT-3 examination is required to identify potential wear in the guide plates, C-tubes, and sheaths of the control rod guide tube assemblies. The thermal shield flexures require VT-3 examination for potential damage due to fatigue and wear. The baffle-to-former assembly is subject to VT-3 examination for the effects of potential IASCC and the effects of potential distortion due to void swelling. Both IASCC and fatigue are potential damage mechanisms for the baffle-edge bolts. The direct measurement of the internals hold-down spring identifies distortion and loss of spring stiffness. The MRP-227 references for required EVT-1, VT-1, and VT-3 examinations and damage mechanisms for both primary and expansion components are summarized for information in Table 6-2.

6.3 Mockups of Westinghouse Internal Components and Bolting

Mockups may be used for developing and modifying procedures, determining examination system parameters, training and qualifying personnel, and supporting the Technical Justifications. Mockups shall be representative of the components or bolting to be examined, and discontinuities in mockups shall be sufficient in number, size, and orientation to demonstrate and evaluate the procedure and system variables relevant to the specific examination. The specific configuration and number of component mockups and bolting, including bolting sizes and types, shall be described in the Technical Justification.

Vendor and utility mockups can be used when available. Mockups of the core barrel assembly including the upper core barrel flange weld may be prepared by the MRP in the absence of vendor and utility mockups to simulate SCC, the potential damage mechanism identified for these components. The mockups may be full size with crack locations and orientations as would be expected for SCC.

6.4 Existing Programs

Existing programs for inspection of reactor internals, such as are required for compliance with ASME Section XI, Subsection IWB for examination categories B-N-1 and B-N-3, are discussed in the I&E Guideline (MRP-227) but are not in the scope of this standard. The inspection methodologies and techniques and the Technical Justifications may be used to enhance and support the existing programs.

7 IMPLEMENTATION REQUIREMENTS

This section summarizes the implementation requirements for this standard. As stated previously, the requirements of this standard do not replace any portion of the plant's current ASME B&PV Code Section XI Inservice Inspection program or other plant-specific licensing requirements related to inservice inspection.

7.1 NEI 03-08 Implementation Protocol

The NEI 03-08 [6] definitions for *mandatory*, *needed*, and *good practice* were used as guidelines in identifying the appropriate implementation categories for portions of this document. Several sections of this report have been identified as *needed* and *good practice*, as described in Sections 7.2 through 7.6.

A failure to meet a *needed* or a *mandatory* requirement is a deviation from the guidelines and shall be processed in accordance with NEI 03-08 and the MRP administrative guidelines, MRP-130 [7].

7.2 Examination Procedures

Needed: Procedures, associated equipment, and personnel using those procedures shall be qualified in accordance with the requirements of Section 2.

Needed: Technical Justifications are required for each examination procedure in accordance with Section 2.1, except procedures for visual examination.

Needed: The generic requirements for visual examination (EVT-1, VT-1, and VT-3) described in Section 2.3 shall be met, including those addressing the personnel training and experience requirements for individuals performing those examinations.

7.3 Classification of Indications and Reporting of Examination Results

Needed: Indications detected by visual and ultrasonic examinations shall be classified and measured in accordance with the requirements of Sections 2.3 and 2.4. Reports of examinations shall comply with the applicable requirements of Sections 2.3 and 2.4.

Evaluation for acceptance of indications is in accordance with the requirement of the Inspection and Evaluation Guidelines, MRP-227 [1].

7.4 Flaw Length Measurement by Visual Examination

Good Practice: When measuring flaw indications by visual examinations, the guidance given in Section 3 for direct and comparative measurement techniques and determining uncertainty should be considered.

7.5 Inspection Planning

Good Practice: Planning for inspections should consider the optimum sequence and combination for inspections, such as combining visual examinations with ultrasonic examination of bolting in the same area of the vessel in order to minimize radiation dose to the examination personnel and reduce inspection task durations (see Section 2.3).

7.6 Inspection of Reactor Internals

Information: Sections 4, 5, and 6 have been provided for the user's information on inspection of reactor internals for Babcock & Wilcox, Combustion Engineering, and Westinghouse designs.

8 REFERENCES

1. *Materials Reliability Program: PWR Internals Inspection and Evaluation Guidelines (MRP-227)*. EPRI, Palo Alto, CA: 2008. 1016596.
2. ASME Boiler & Pressure Vessel Code, Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components," American Society of Mechanical Engineers, New York, NY: 2001 Edition, Plus 2003 Addenda, or later.
3. ASME Boiler & Pressure Vessel Code, Section V, Nondestructive Examination, American Society of Mechanical Engineers, New York, NY, 2004 Edition, July 1, 2004.
4. NUREG/CR-6860, An Assessment of Visual Testing, Pacific Northwest National Laboratory for the U.S. Nuclear Regulatory Commission, November 2004.
5. *TR-105696-R10 (BWRVIP-03) Revision 10: BWR Vessel and Internals Project.* EPRI, Palo Alto, CA: 2007. 1014993.
6. Appendix D, "Materials Guidelines: Implementation Protocol," *Guidelines for the Management of Materials Issues*, NEI 03-08, Nuclear Energy Institute, Washington, D.C., Latest Edition.
7. Pressurized Water Reactor Materials Reliability Program Administrative Procedures (MRP-130, Revision 1, October 2008, issued on November 6, 2008 via MRP letter MRP 2008-063).

A
SUBMITTED TECHNICAL JUSTIFICATIONS

Technical Justifications have been completed by two inspection vendors for examination of baffle-former bolts. These Technical Justifications were previously prepared by the vendors and, along with the associated ultrasonic examination procedures, were reviewed by EPRI MRP staff and found to be acceptable. They included more than 10 bolts in their blind demonstration test sets and detected more than 80% of the flaws with a cross-sectional area greater than 24% with a false call rate of less than 20%.

One vendor has successfully demonstrated its UT procedures on upper core barrel bolt mockups and another on upper and lower core barrel bolt mockups that were fabricated by the MRP. These were used to successfully demonstrate the UT procedures employed by both vendors. The results of these demonstrations are documented in the following:

- *WesDyne Core Barrel Bolt Demonstration.* EPRI, Palo Alto CA: 2008. IR-2008-320.
- Upper Core Barrel Bolting (UCBB) Examination Demonstration (AREVA), Letter to Dave Whittaker, Duke Energy Corp. dated October 3, 2007.

The vendors shall reference the applicable report in their Technical Justification to support field experience justifying their procedure.