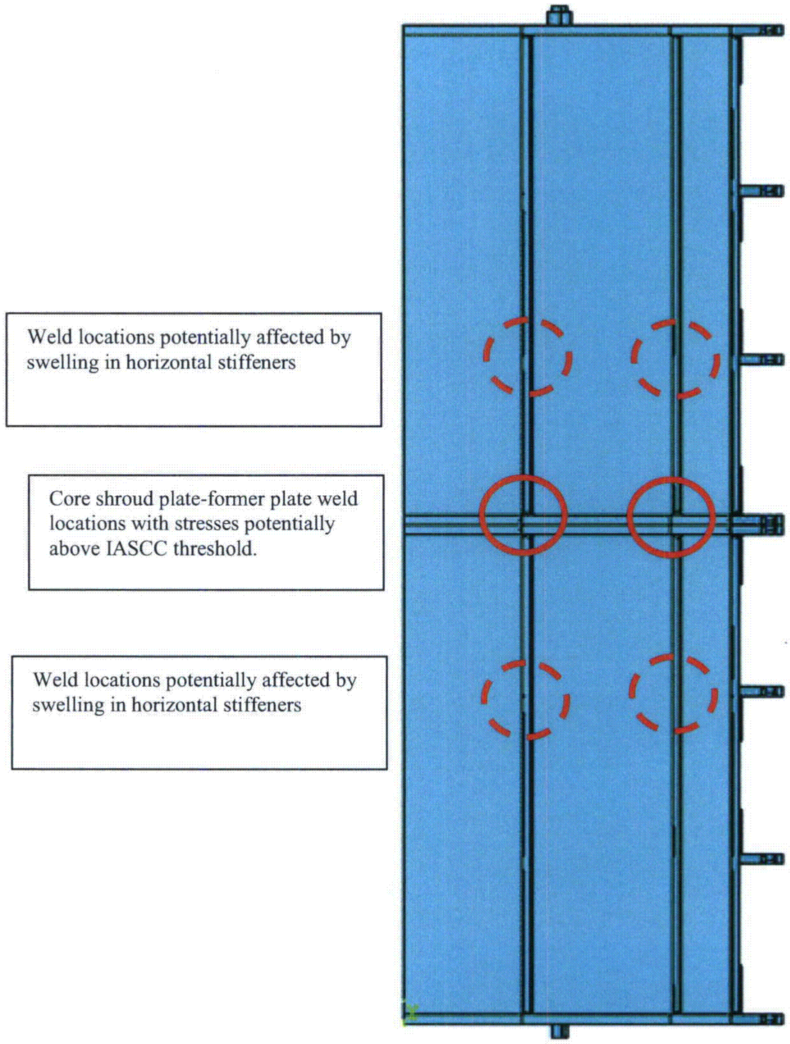


Figure 4-12
Potential crack locations for CE welded core shroud assembled in stacked sections

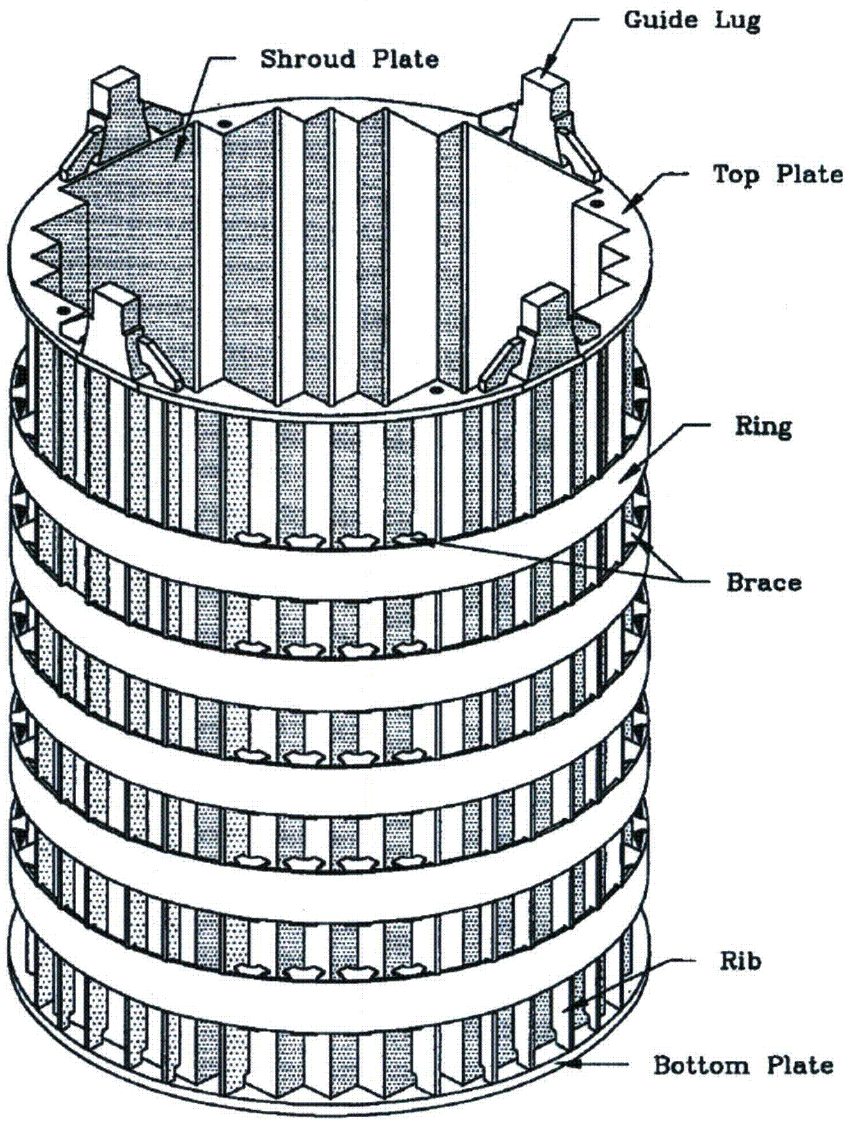


Figure 4-13
CE welded core shroud with full height panels

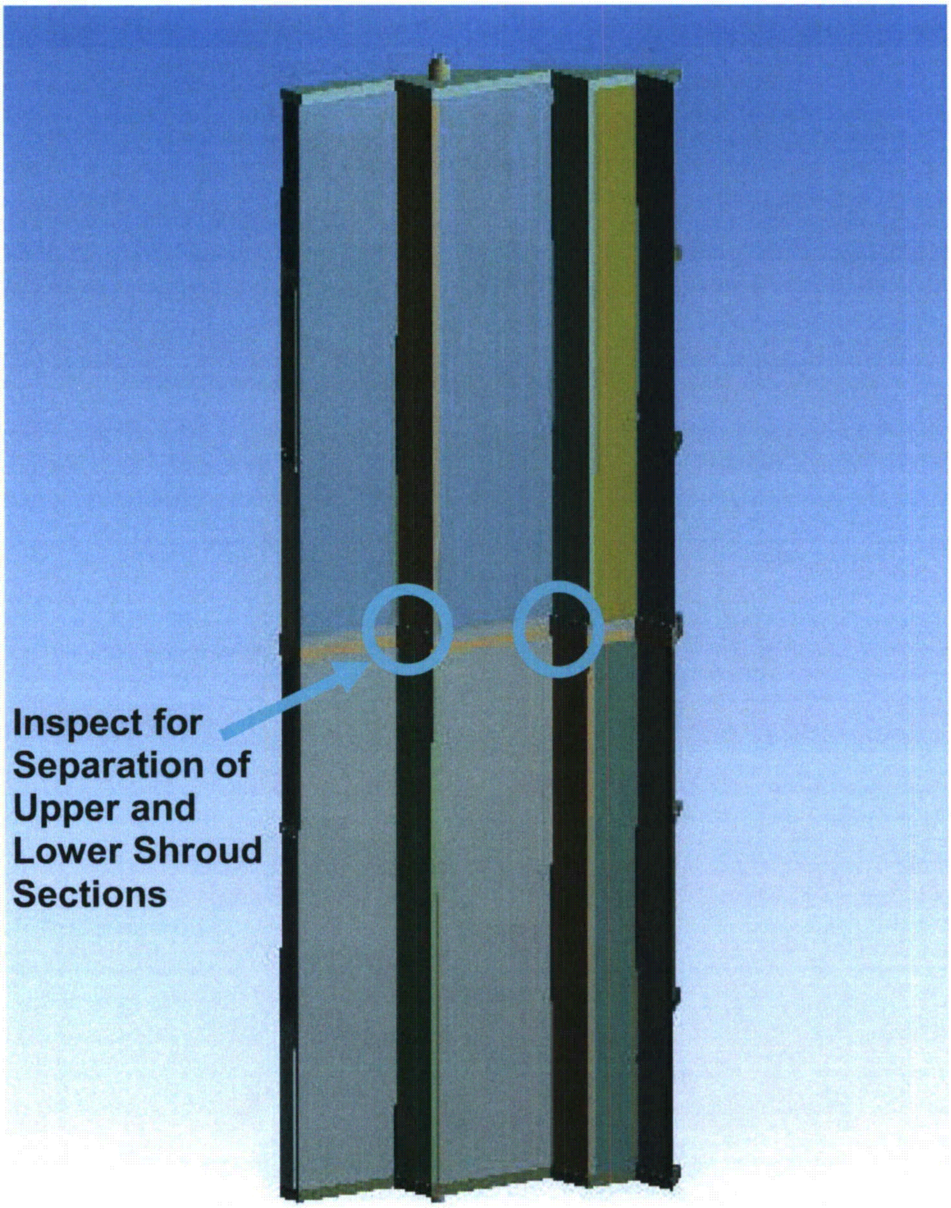


Figure 4-14
Locations of potential separation between core shroud sections caused by swelling induced warping of thick flange plates in CE welded core shroud assembled in stacked sections

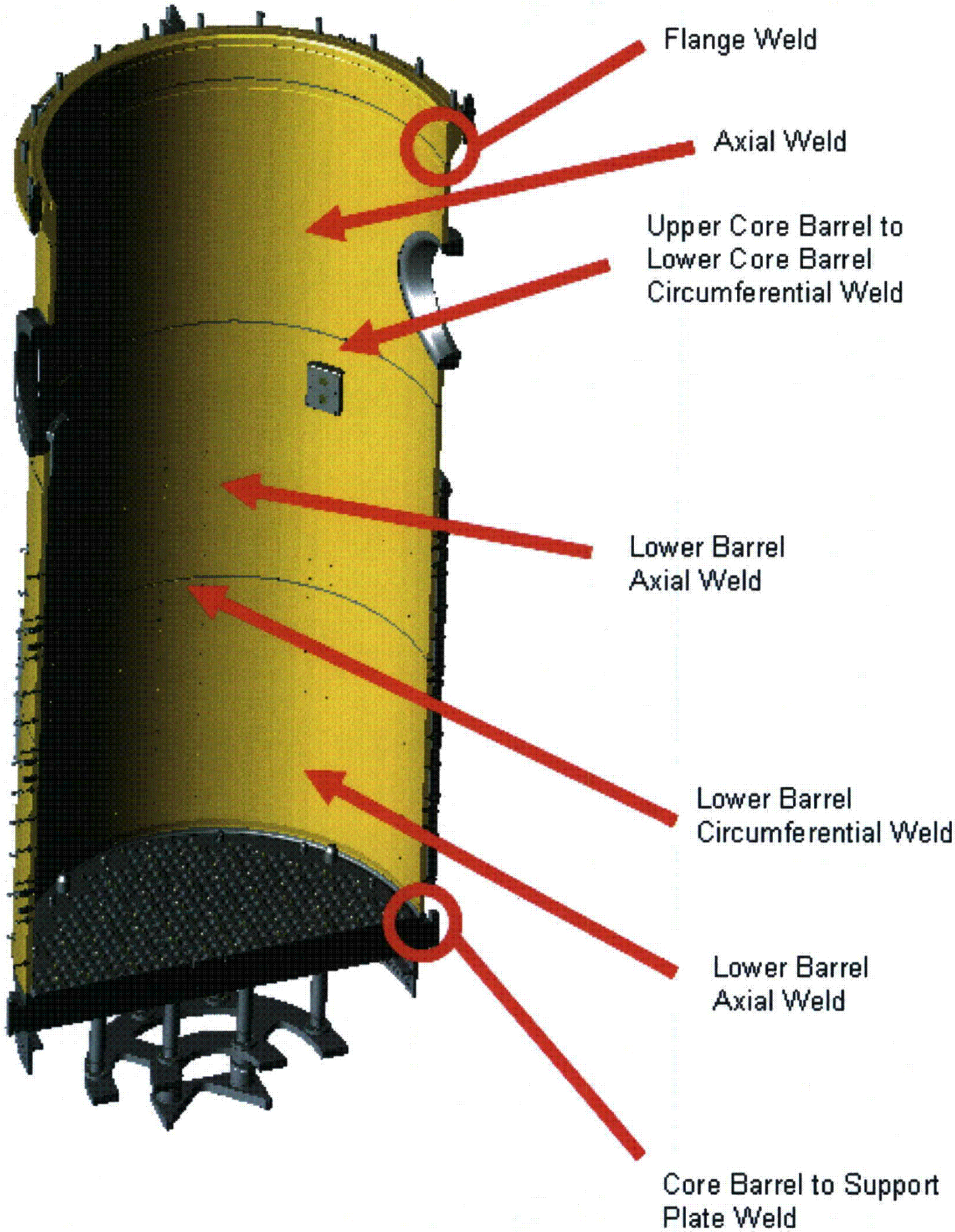
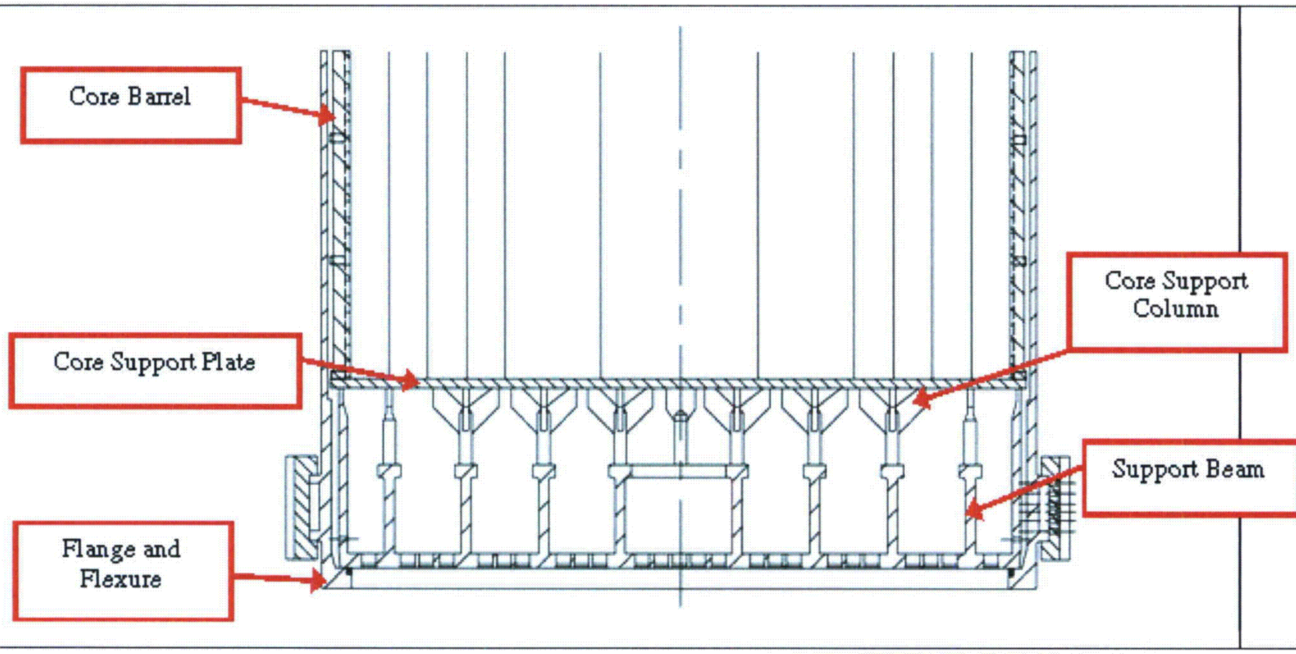
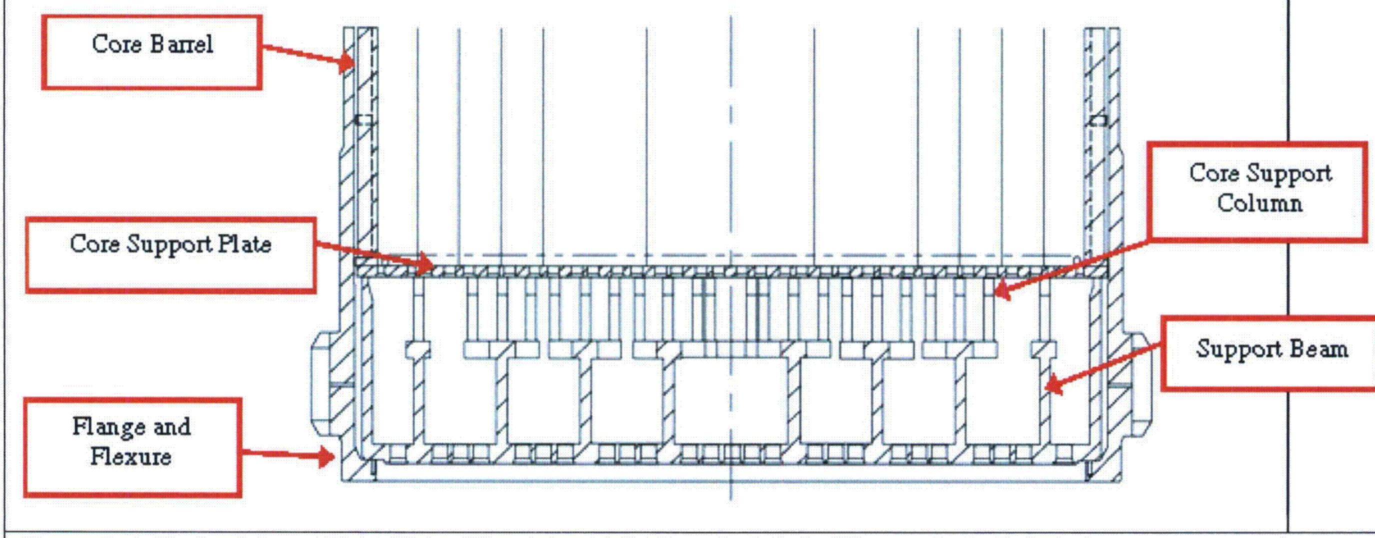


Figure 4-15
Typical CE core support barrel structure



"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

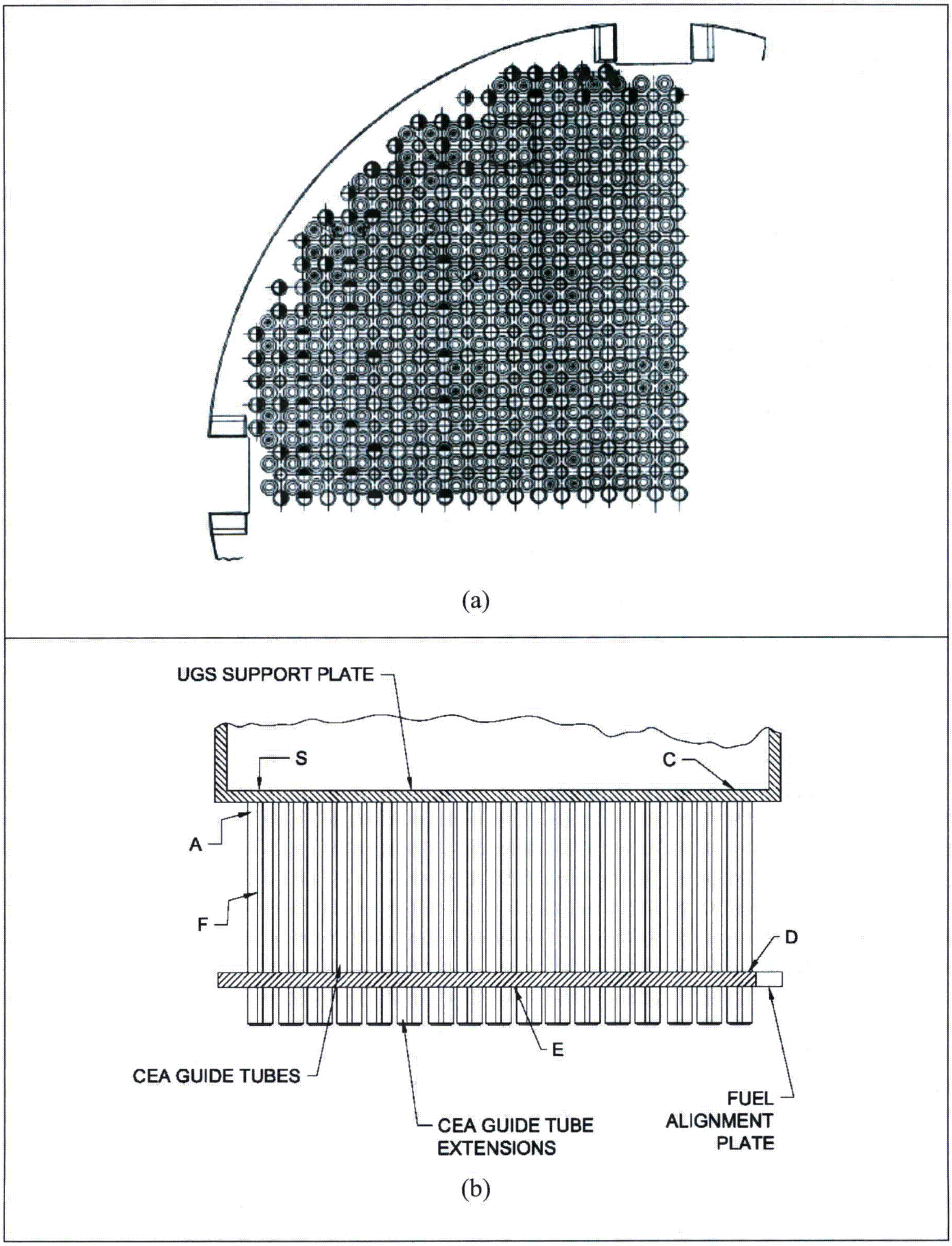


Figure 4-17
(a) Schematic illustration of a portion of the fuel alignment plate, and (b) Radial view schematic illustration of the guide tubes protruding through the plate in upper internals assembly of CE core shrouds with full-height shroud plates

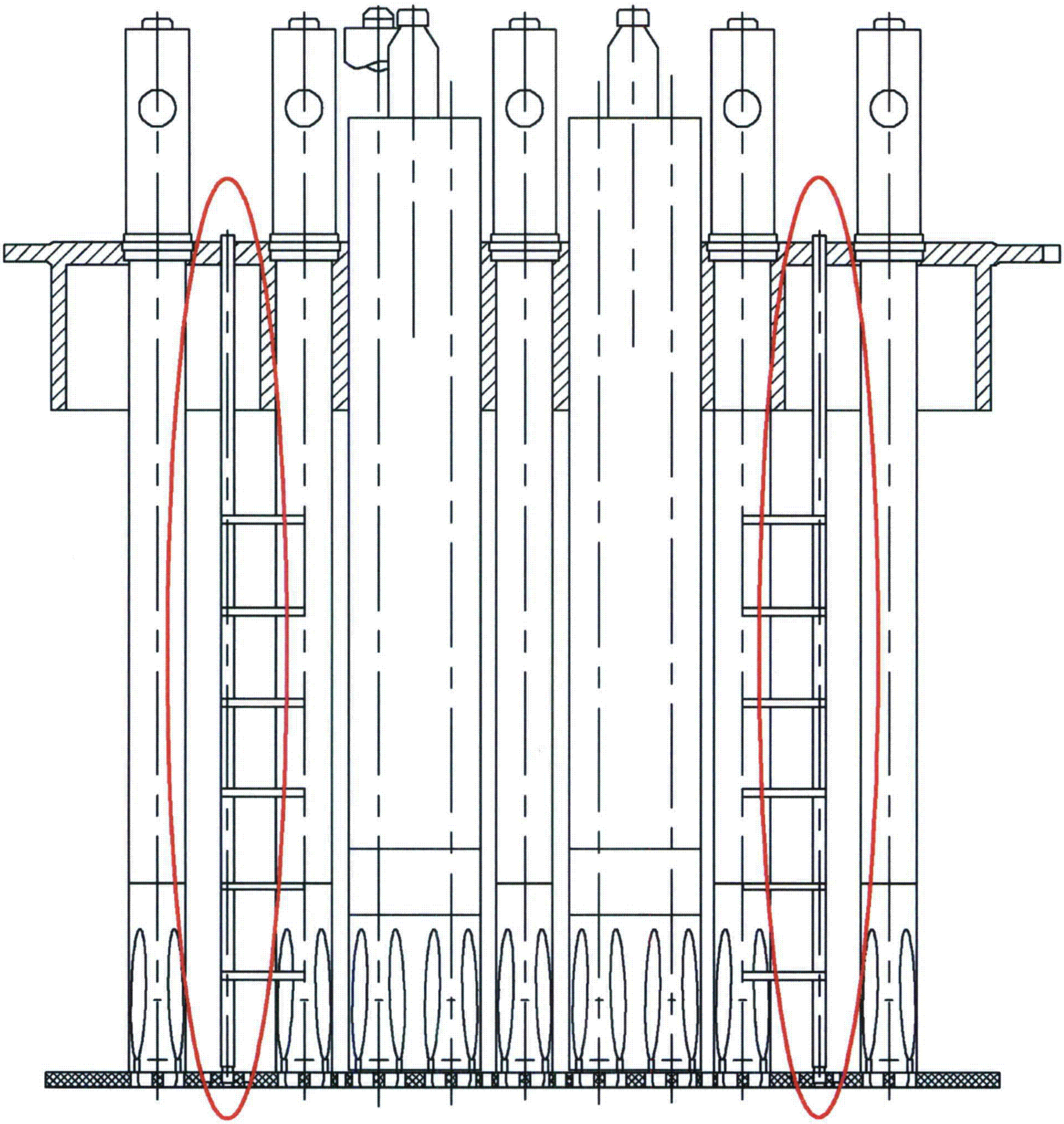
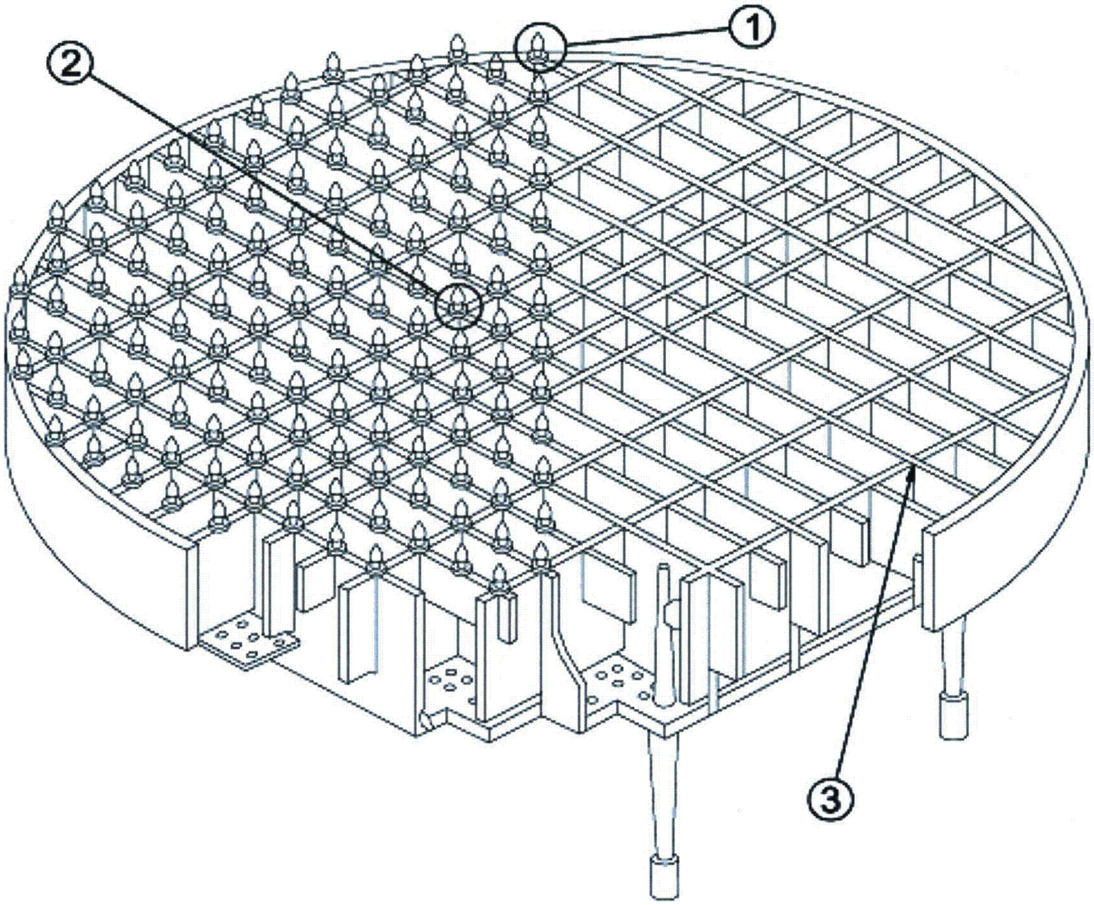


Figure 4-18
CE control element assembly (CEA) shroud instrument tubes (circled in red) are shown, along with the welded supports attaching them to the CEA shroud tube, in this schematic illustration



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

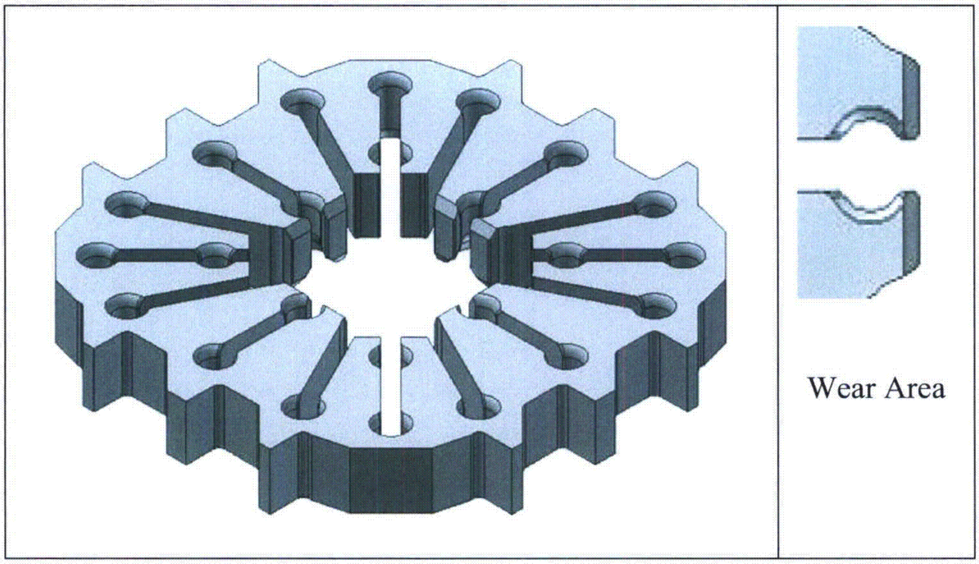


Figure 4-20
Typical Westinghouse control rod guide card (17x17 fuel assembly)

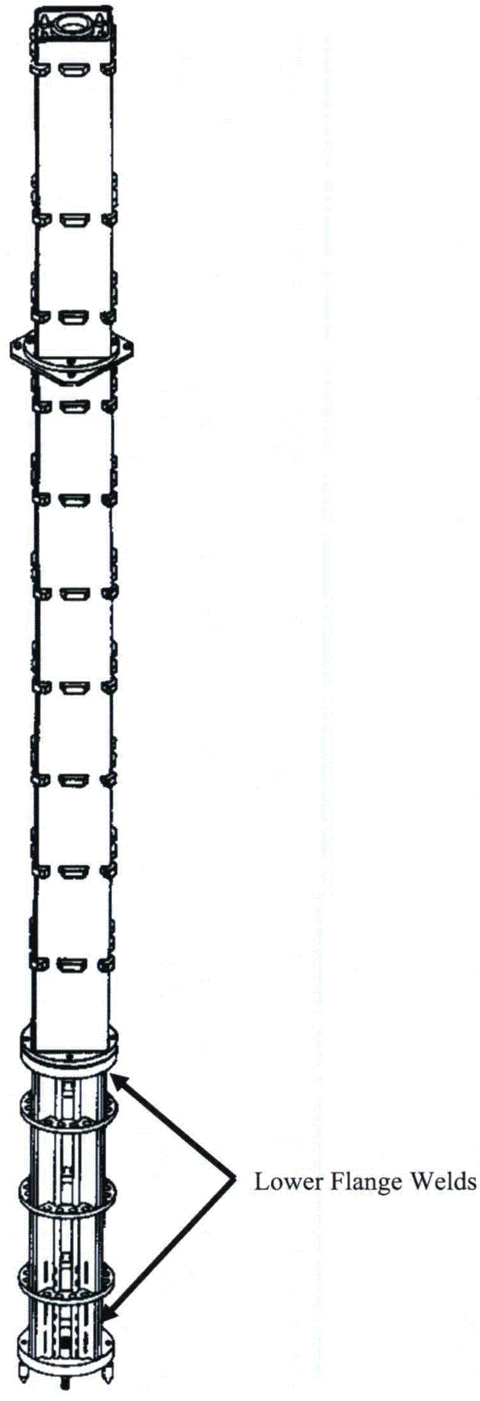


Figure 4-21
Typical Westinghouse control rod guide tube assembly

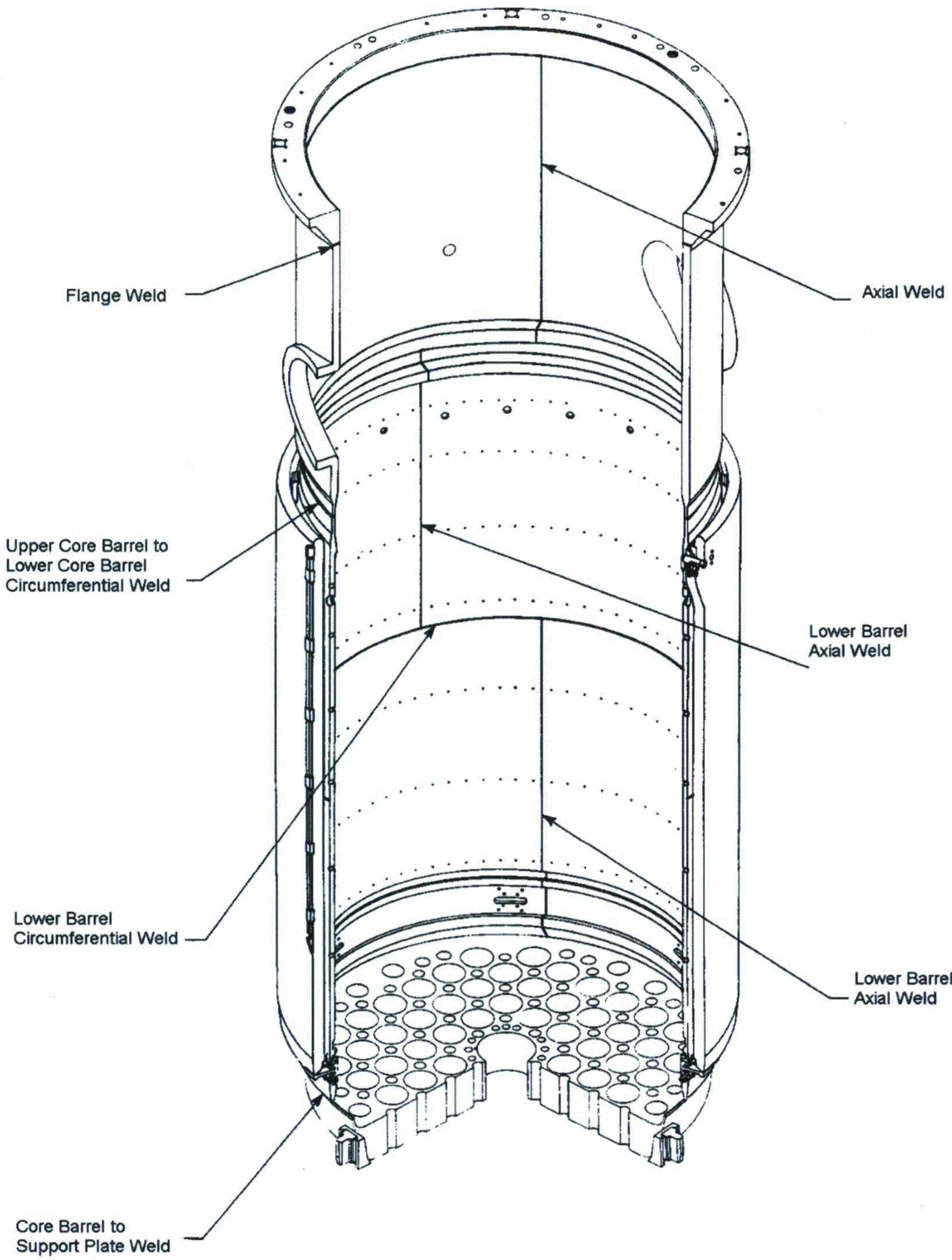


Figure 4-22
Major fabrication welds in typical Westinghouse core barrel

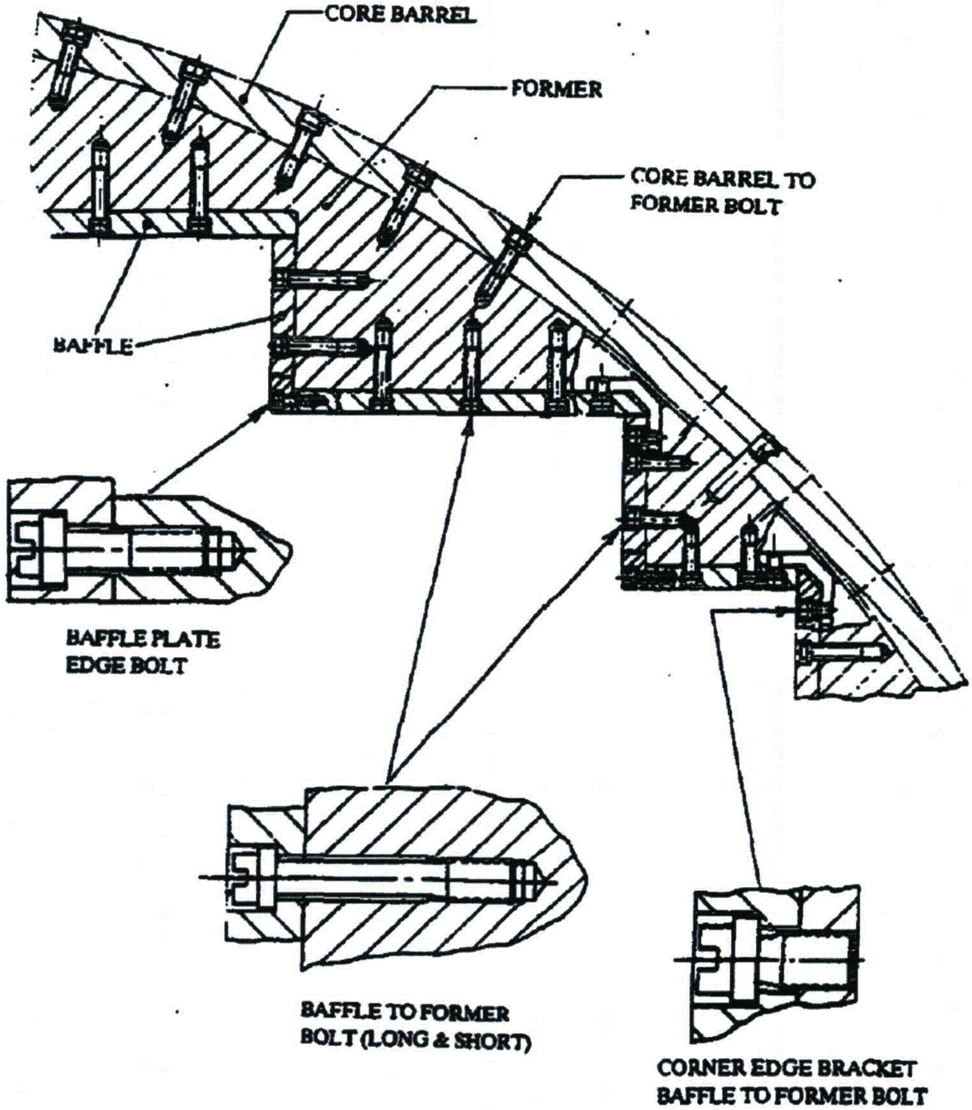


Figure 4-23
Bolt locations in typical Westinghouse baffle-former-barrel structure. In CE plants with bolted shrouds, the core shroud bolts are equivalent to baffle-former bolts and barrel-shroud bolts are equivalent to barrel-former bolts

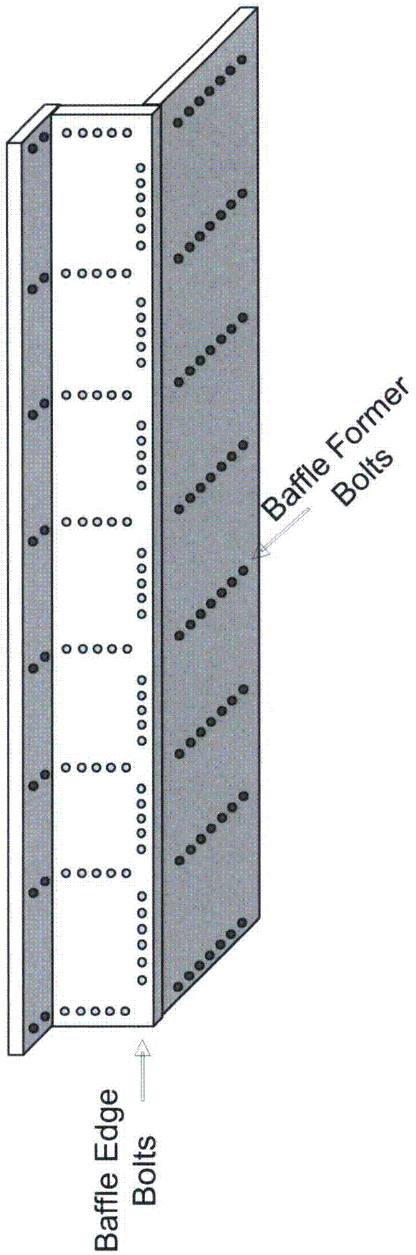


Figure 4-24
Baffle-edge bolt and baffle-former bolt locations at high fluence seams in bolted baffle-former assembly (note: equivalent baffle-former bolt locations in bolted CE shroud designs are core shroud bolts)

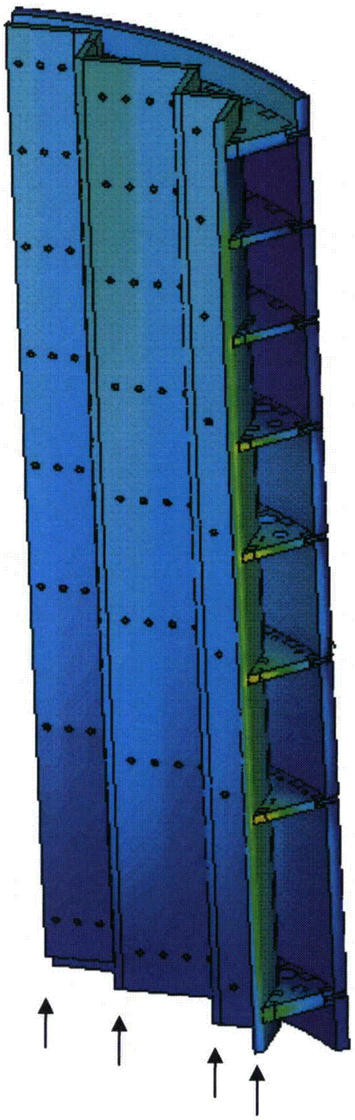


Figure 4-25
High fluence seam locations in Westinghouse baffle-former assembly (full axial length of each of the re-entrant baffle plate corners)

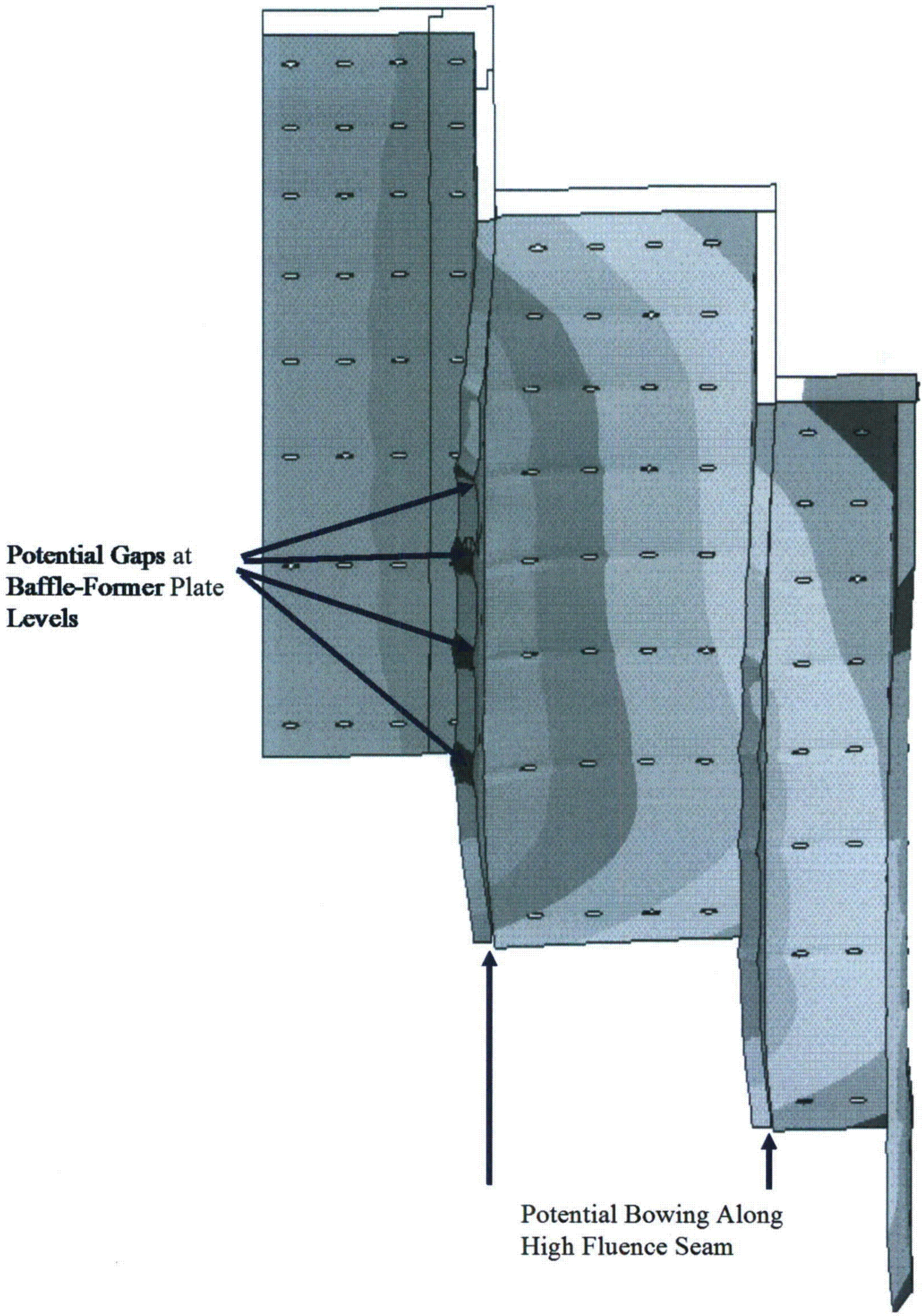


Figure 4-26
Exaggerated view of void swelling induced distortion in Westinghouse baffle-former assembly. This figure also applies to bolted CE shroud designs

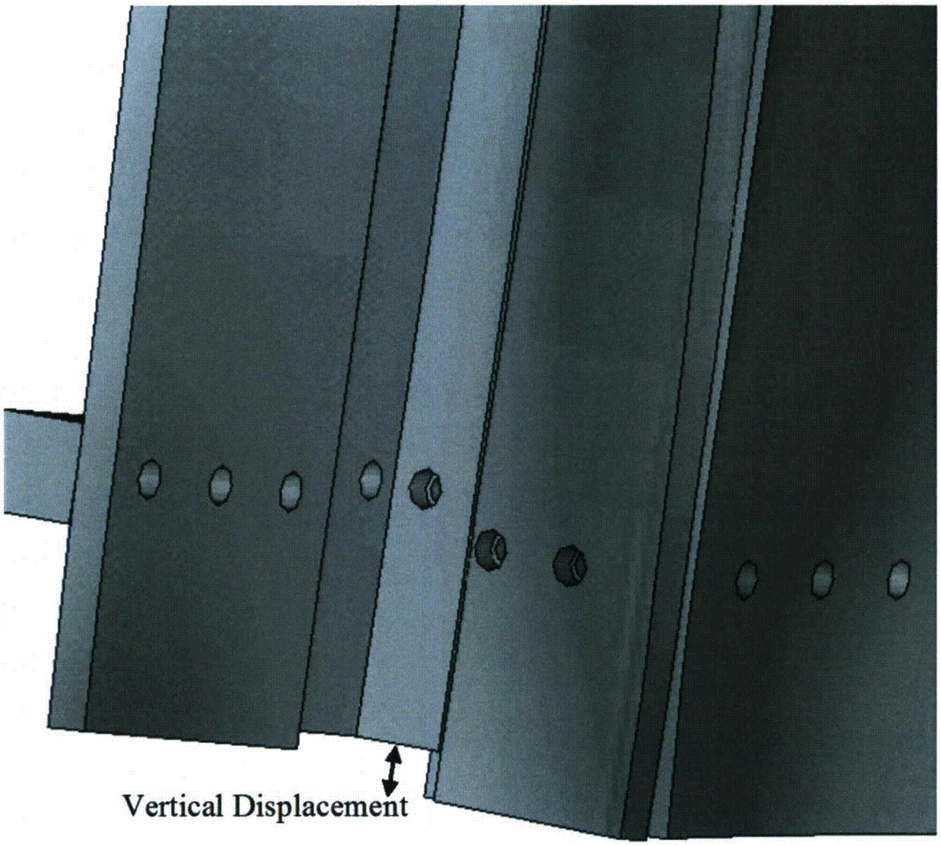


Figure 4-27
Vertical displacement of Westinghouse baffle plates caused by void swelling. This figure also applies to bolted CE shroud designs

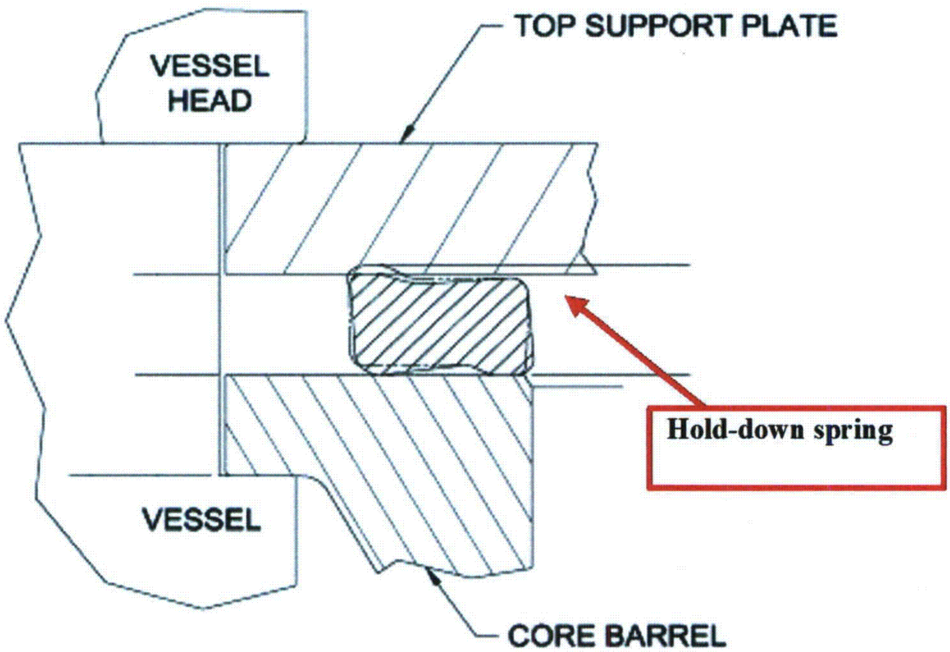


Figure 4-28
Schematic cross-sections of the Westinghouse hold-down springs

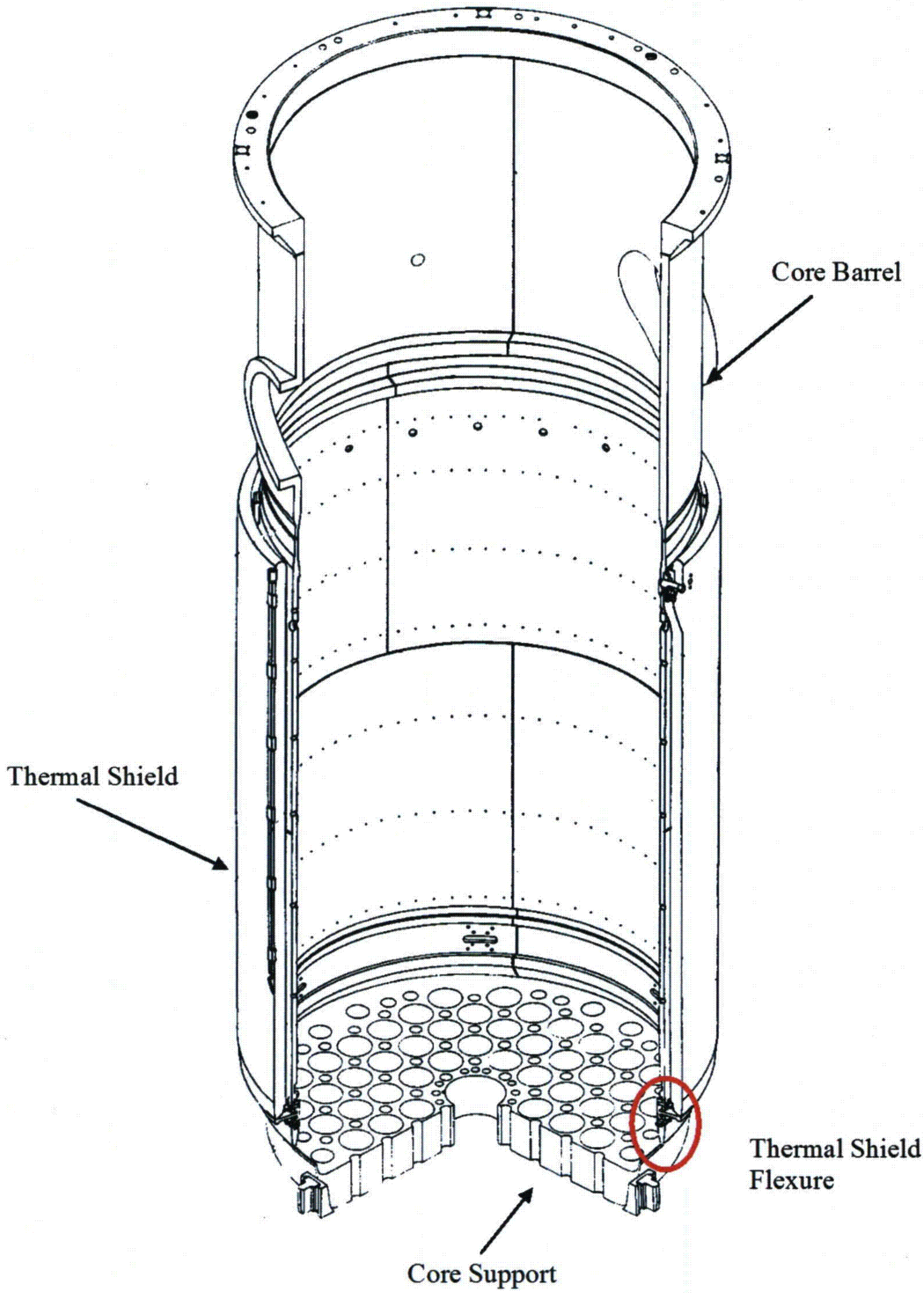


Figure 4-29
Location of Westinghouse thermal shield flexures

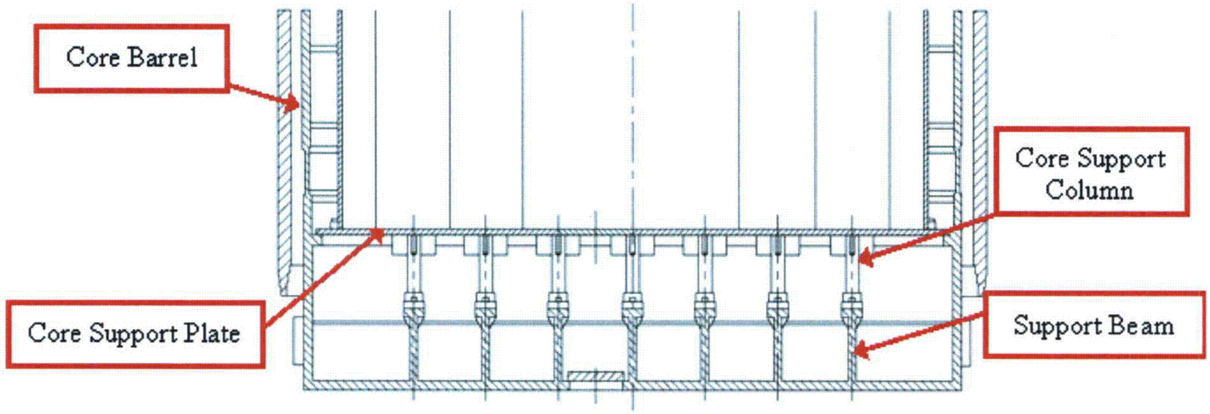
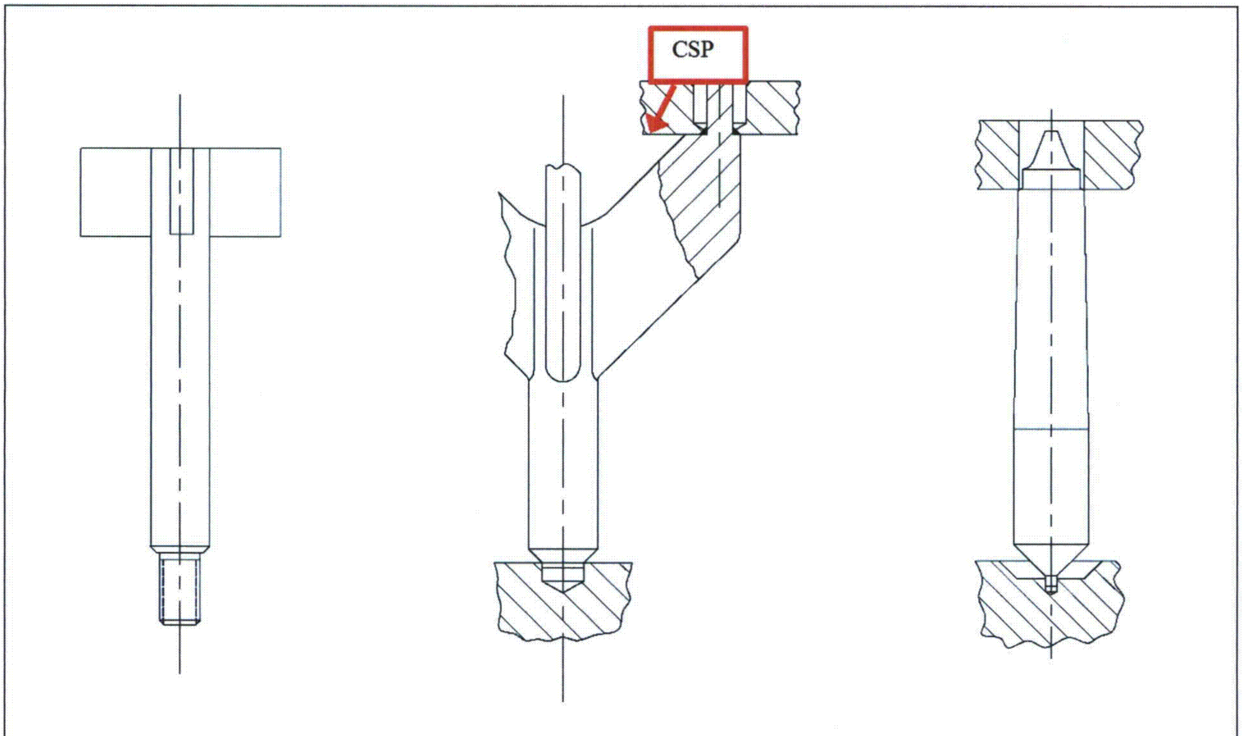


Figure 4-30
CE lower support structure assembly for plants with integrated core barrel and lower support structure with a core support plate (this design does not contain a lower core barrel flange)



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

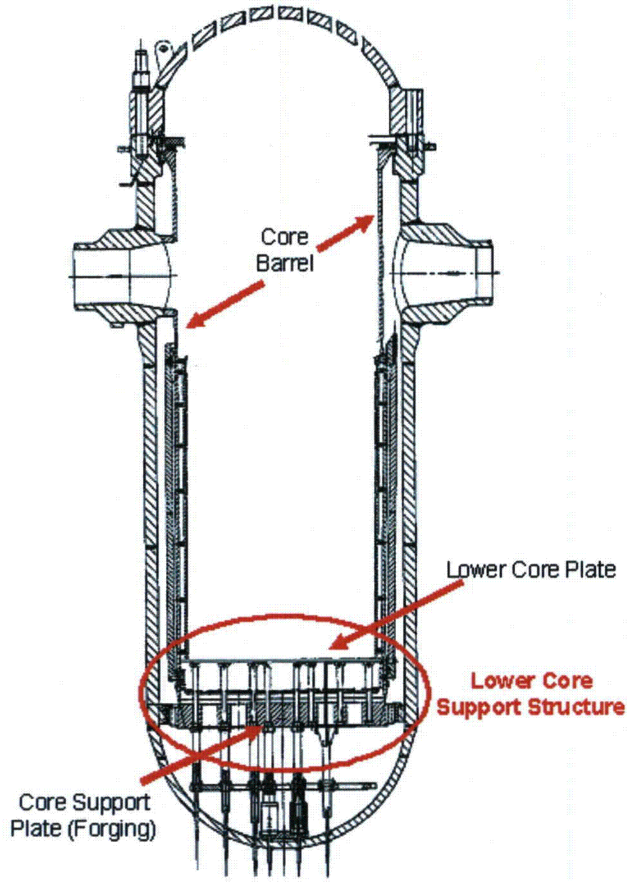


Figure 4-32
Schematic indicating location of Westinghouse lower core support structure. Additional details shown in Figure 4-33

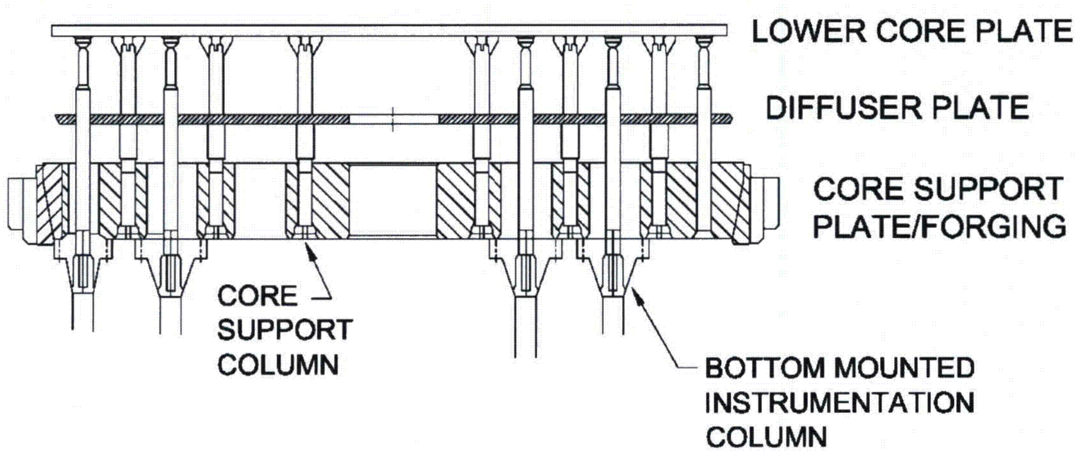


Figure 4-33
Westinghouse lower core support structure and bottom mounted instrumentation columns. Core support column bolts fasten the core support columns to the lower core plate

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

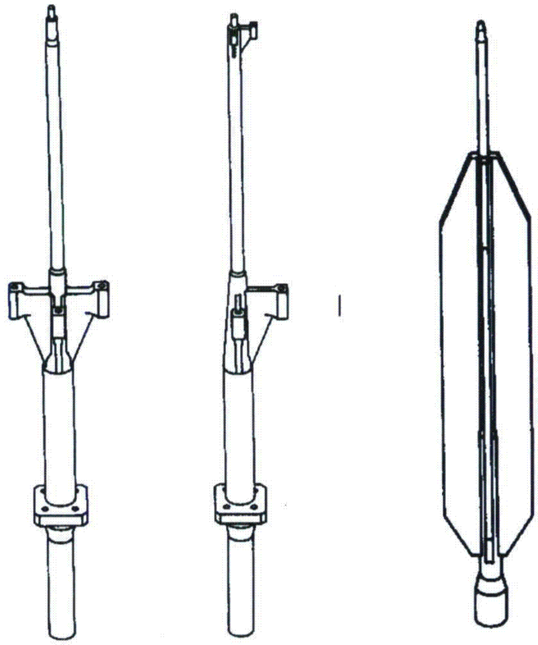


Figure 4-35
Examples of Westinghouse bottom mounted instrumentation column designs

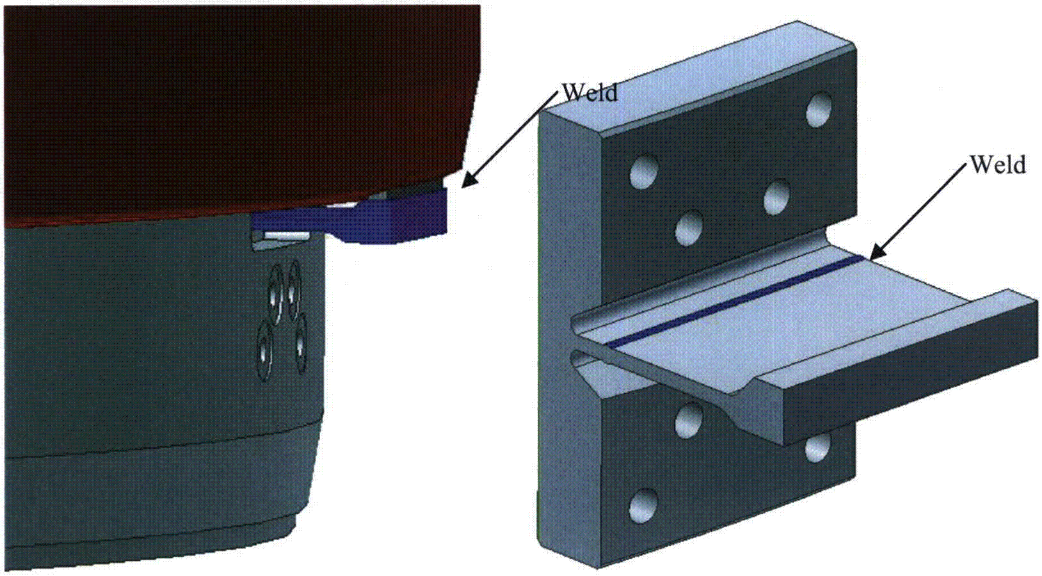


Figure 4-36
Typical Westinghouse thermal shield flexure

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)	Reference	Examination Method	Examination Coverage
Core Shroud Assembly Guide lugs Guide lug inserts and bolts	All plants	Loss of material (Wear) Aging Management (ISR)	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. Accessible surfaces at specified frequency.
Lower Support Structure Fuel alignment pins	All plants with core shrouds assembled with full-height shroud plates	Cracking (SCC, IASCC, Fatigue) Aging Management (IE and ISR)	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) Aging Management (IE and ISR)	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)	Reference	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) Aging Management (IE)	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 to Table 4-9:

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 and in Reference 26. 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 vent valve top retaining ring CSS vent valve bottom retaining ring	All plants	Visual (VT-3) examination. The specific relevant condition is evidence of damaged or fractured retaining ring material, and missing items.	None	N/A	N/A
Control Rod Guide Tube Assembly CRGT spacer castings	All plants	The specific relevant condition for the VT-3 of the CRGT spacer castings is evidence of fractured spacers or missing screws.	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.	UTS bolts and LTS bolts or studs/nuts and their locking devices SSHT studs/nuts or bolts and their locking devices (CR-3 and DB only) Lower grid shock pad bolts and their locking devices (TMI-1 only)	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 bolts and 100% of the accessible LTS bolts or studs/nuts, *Additionally for TMI-1* 100% of the accessible lower grid shock pad bolts, *Additionally for CR-3 and DB* 100% of the accessible SSHT studs/nuts or 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 bolt and 100% of the accessible LTS bolt or stud/nut 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 or stud/nut 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 VT-3 of the expansion 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 bolts and LTS bolts or studs/nuts and their locking devices SSHT studs/nuts or bolts and their locking devices (CR-3 and DB only) Lower grid shock pad bolts and their locking devices (TMI-1 only)	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 bolts and 100% of the accessible LTS bolts or studs/nuts *Additionally for TMI-1* 100% of the accessible lower grid shock pad bolts, *Additionally for CR-3 and DB* 100% of the accessible SSHT studs/nuts or 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 bolts and 100% of the accessible LTS bolt or stud/nut 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 stud/nut or 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 VT-3 of the expansion 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 baffle 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, including locking welds.	Locking devices, including locking welds, for the external baffle-to-baffle bolts and core 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, including locking welds, 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 grid 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 grid fuel assembly support pads by the completion of the next refueling outage.	The specific relevant condition for the VT-3 of the expansion dowel locking weld 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
Flow Distributor Assembly Flow distributor (FD) bolts and their locking devices	All plants	1) Volumetric (UT) examination of the FD bolts. The examination acceptance criteria for the UT of the FD bolts shall be established as part of the examination technical justification. 2) Visual (VT-3) examination of the FD bolt locking devices. The specific relevant condition for the VT-3 of the FD bolt locking devices is evidence of broken or missing bolt locking devices.	UTS bolts and LTS bolts or studs/nuts and their locking devices SSHT studs/nuts or bolts and their locking devices (CR-3 and DB only) Lower grid shock pad bolts and their locking devices (TMI-1 only)	1) Confirmed unacceptable indications exceeding 10% of the FD 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 bolts and 100% of the accessible LTS bolts or studs/nuts *Additionally for TMI-1* 100% of the accessible lower grid shock pad bolts, *Additionally for CR-3 and DB* 100% of the accessible SSHT studs/nuts or bolts. 2) Confirmed evidence of relevant conditions exceeding 10% of the FD 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 bolts and 100% of the accessible LTS bolt or stud/nut 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 stud/nut or 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 VT-3 of the expansion 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
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.	Lower fuel grid assembly support pad items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screw, and their locking welds	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.	The specific relevant conditions for the VT-3 of the lower grid fuel assembly support pad items (pads, pad-to-rib section welds, Alloy X-750 dowels, cap screws, and their locking welds) are separated or missing welds, missing support pads, dowels, cap screws and locking welds, or misalignment of the support pads.

Notes to Table 5-1:

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

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.	Lower core support beams Upper core barrel cylinder (including welds) Upper core barrel flange	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 core support beams, upper core barrel cylinder and upper core barrel flange be performed by the completion of the next refueling outage.	The specific relevant condition is a detectable crack-like surface indication.
Core Support Barrel Assembly Lower cylinder girth welds	All plants	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Lower cylinder axial welds	Confirmation that a surface-breaking indication >2 inches in length has been detected and sized in the lower cylinder girth weld shall require an EVT-1 examination of all accessible lower cylinder axial welds by the completion of the next refueling outage.	The specific relevant condition for the expansion lower cylinder axial welds is a detectable crack-like surface indication.
Lower Support Structure Core support column welds	All plants	Visual (VT-3) examination. The specific relevant condition is missing or separated welds.	None	None	

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 to Table 5-2:
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), upper core plate and lower support forging or casting	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, upper core plate and lower support forging/castings 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, upper core plate and lower support forging/castings, 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. Core barrel outlet nozzle 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 be expanded to include the core barrel outlet nozzle welds by the completion of the next refueling outage. b. If extensive cracking in the core barrel outlet nozzle 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 for the expansion core barrel outlet nozzle weld and lower support column body examination is a detectable crack-like surface indication.
Core Barrel Assembly Lower core barrel flange weld (Note 2)	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	None	None	None

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 cylinder girth welds	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Upper core barrel cylinder axial welds	The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the upper core barrel cylinder axial welds by the completion of the next refueling outage.	The specific relevant condition for the expansion upper core barrel cylinder axial weld examination is a detectable crack-like surface indication.
Core Barrel Assembly Lower core barrel cylinder girth welds	All plants	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Lower core barrel cylinder axial welds	The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the lower core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the lower core barrel cylinder axial welds by the completion of the next refueling outage.	The specific relevant condition for the expansion lower core barrel cylinder axial weld examination 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 to Table 5-3:

1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s).
2. The lower core barrel flange weld may alternatively be designated as the core barrel-to-support plate weld in some Westinghouse plant designs.

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 or pins 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 guidance provided in this section is general; Reference 26 should be consulted for more detailed guidance.

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 3×10^{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 of crack growth is 216 KJ/m^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 of crack growth is 75.8 KJ/m^2. If the tangent to the crack growth resistance curve at 1.5 mm 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 100 MPa√m. 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 of crack growth is only 64 KJ/m^2. Extrapolating the tangent of the crack growth resistance curve back to zero crack growth and converting gives K_{IC} = 92 MPa√m. 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 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 for exposure levels between 5 and 15 dpa, and with a limiting fracture toughness K_{IC} = 38 MPa√m for exposure levels greater than 15 dpa.

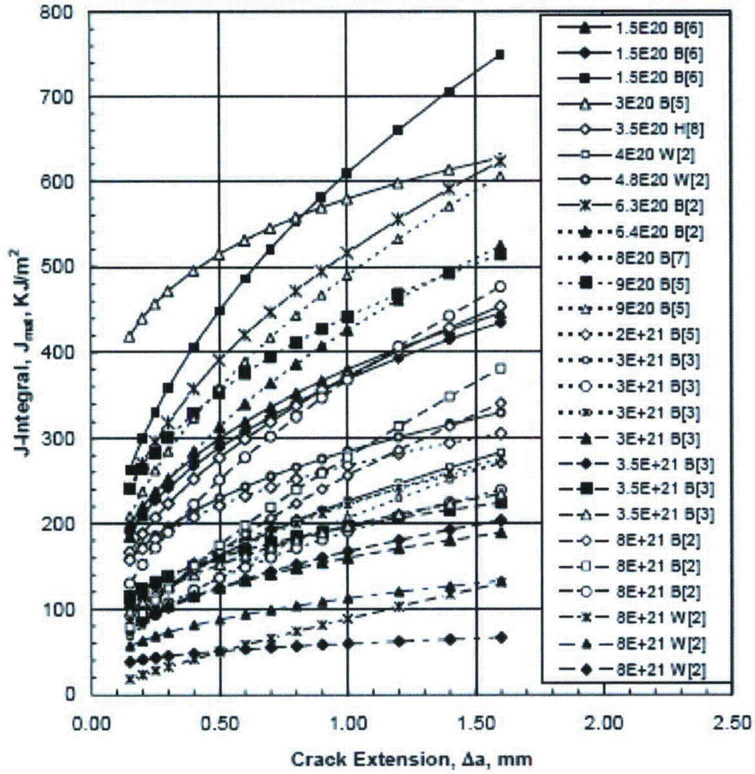


Figure 6-1
Experimental $J_{material}$ versus crack extension curves for stainless steel materials at various fluence levels [19]

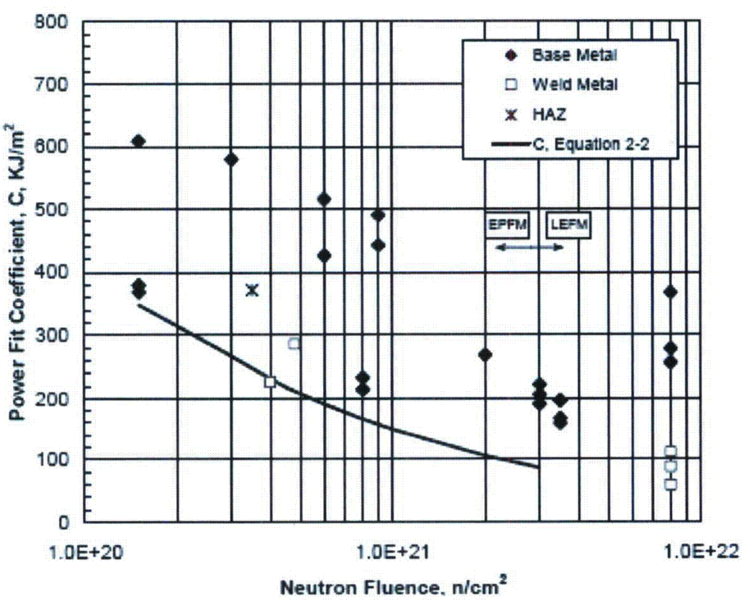


Figure 6-2
J-R curve power law parameter C as a function of neutron fluence for stainless steel, applicable for fluence less than $3x10^{21}$ n/cm^2 [19]

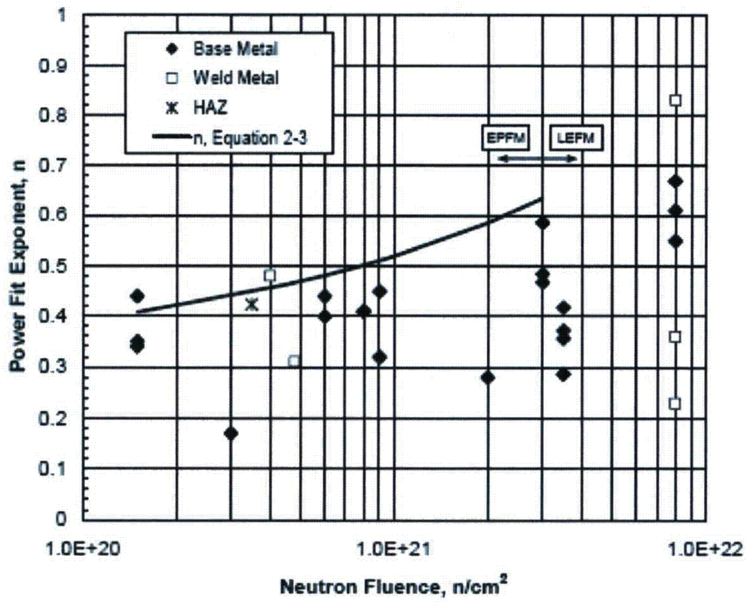


Figure 6-3
J-R curve power law parameter n as a function of neutron fluence for stainless steel, applicable for fluence less than $3x10^{21}$ n/cm^2 [19]

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.

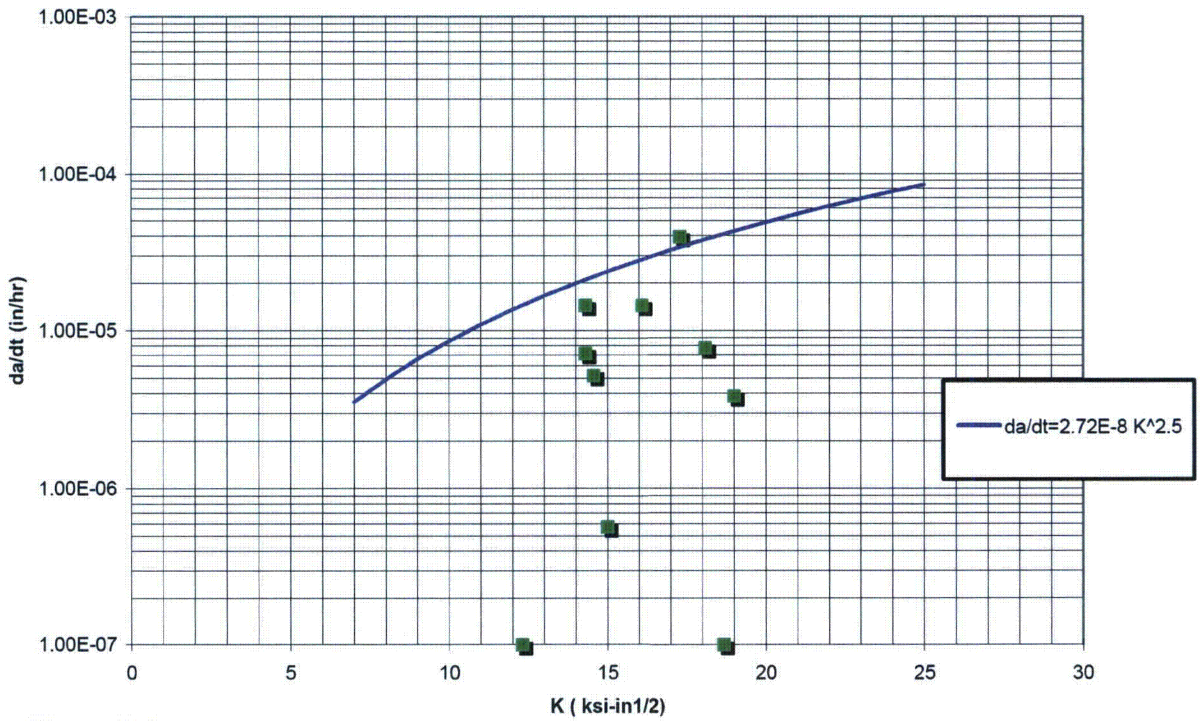


Figure 6-4
Proposed BWR hydrogen water chemistry crack growth curves for stainless steel irradiated between $5x10^{20}$ to $3x10^{21}$ n/cm^2 [24]

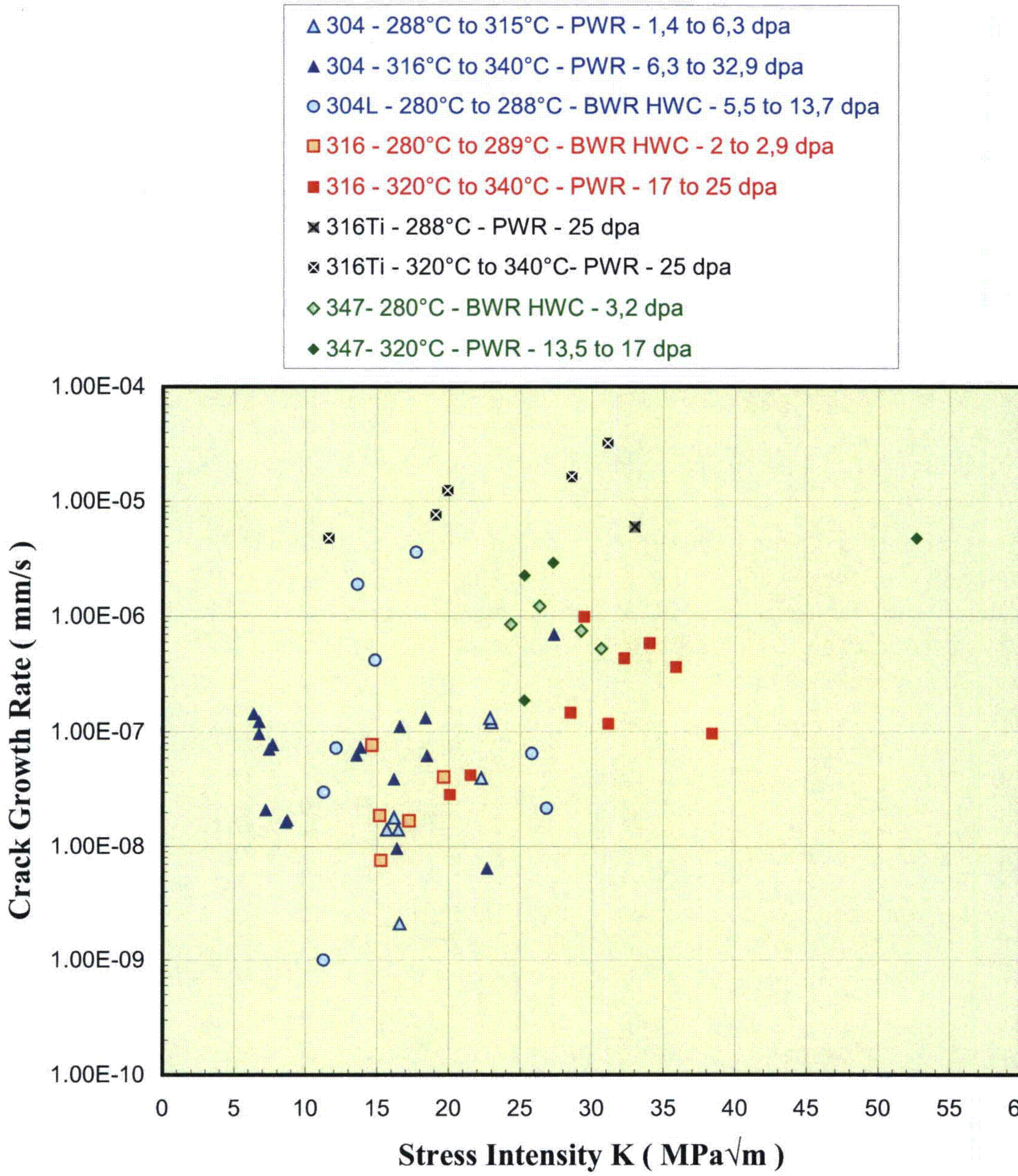


Figure 6-5
Effect of stress intensity on IASCC crack growth rate [25]

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]. Appendix B 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 by 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 7.7 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 Appendix B 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 program for management of aging of reactor internal components within thirty-six months following issuance of MRP-227-Rev. 0 (that is, no later than December 31, 2011).*

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 (that is, no later than December 31, 2013). Implementation means performance of inspections of applicable components within the time frame specified in the guidance provided in the applicable tables. MRP-227-A is the current version that has 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 (MRP-227 Rev. 0).

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.

Consistent with the requirements of NEI 03-08, if the guidance contained in Table 4-1 through 4-9 and/or Tables 5-1 through 5-3 cannot, need not, or will not be implemented as written, a technical justification must be prepared that clearly states what requirement cannot, need not, or will not be met and why; what alternative action is being taken to satisfy the objective or intent of the guidance; and, why the alternative action is acceptable. Examples of alternatives that may be justifiable are: elevation of an Expansion component to Primary; substitution of an equivalent or more rigorous examination than is required by the tables; or destructive testing in lieu of nondestructive examination, such as the case where one or more of the primary components is being replaced. Since the Expansion components are also "needed" requirements, the technical justification for not fully implementing a Primary component examination or not implementing it in a manner consistent with its intent, would be expected to include disposition of the associated Expansion components.

When submittal of a deviation from work products or elements is required, the justification shall be reviewed and approved in accordance with the applicable plant procedures with the additional responsibility for deviation from a 'Needed' element that an internal independent review is performed and that concurrence is obtained from the responsible utility executive. Further, as stipulated in the Implementation Protocol (Appendix B) of NEI 03-08, a utility is required to notify the Issue Program (e.g., the MRP) and the NRC.

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

Needed: Each commercial U.S. PWR unit shall 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 within the scope of MRP-227 are examined.

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.

7.7 Evaluation Requirement

Needed: If an engineering evaluation is used to disposition an examination result that does not meet the examination acceptance criteria in Section 5, this engineering evaluation shall be conducted in accordance with a NRC-approved evaluation methodology.

8 REFERENCES

1. *Guidelines for the Management of Materials Issues*, NEI 03-08, Nuclear Energy Institute, Washington, DC, Latest Edition.
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. *Materials Reliability Program: Inspection Standard for Reactor Internals (MRP-228)*. EPRI, Palo Alto, CA: 2009. 1016609.
4. 10 CFR 50.55a – Codes and Standards, Title 10 (Energy), Part 50 (Domestic Licensing of Production and Utilization Facilities) of the Code of Federal Regulations, U.S. Nuclear Regulatory Commission, Washington, DC, 2005.
5. *Materials Reliability Program: Framework and Strategies for Managing Aging Effects in Reactor Internals (MRP-134)*. EPRI, Palo Alto, CA: 2005. 1008203.
6. *Materials Reliability Program: Characterization of Decommissioned PWR Vessel Internals Material Samples – Material Certification, Fluence, and Temperature (MRP-128)*. EPRI, Palo Alto, CA: 2003. 1008202.
7. *Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175)*. EPRI, Palo Alto, CA: 2005. 1012081.
8. *Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals (MRP-189-Rev. 1)*. EPRI, Palo Alto, CA: 2009. 1018292.
9. *Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals (MRP-190)*. EPRI, Palo Alto, CA: 2006. 1013233.
10. *Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs (MRP-191)*. EPRI, Palo Alto, CA: 2006. 1013234.
11. *Materials Reliability Program: Functionality Analysis for B&W Representative PWR Internals (MRP-229)*. EPRI, Palo Alto, CA: 2008. 1016598.
12. *Materials Reliability Program: Functionality Analysis for Westinghouse and Combustion Engineering Representative PWR Internals (MRP-230)*. EPRI, Palo Alto, CA: 2008. 1016597.
13. *Materials Reliability Program: Aging Management Strategies for B&W PWR Internals (MRP-231)*. EPRI, Palo Alto, CA: 2008. 1016592.
14. *Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals (MRP-232)*. EPRI, Palo Alto, CA: 2008. 1016593.

15. Letter to Reactor Internals Focus Group from MRP, Subject: *Minutes of the Expert Panel Meetings on Expansion Criteria for Reactor Internals I&E Guidelines*, MRP 2008-036 (via email), June 12, 2008.

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

17. *Nondestructive Evaluation: Evaluation of Remote Visual Examination Methods*. EPRI, Palo Alto, CA: 2006. 1013537.

18. Topical Report BAW-2248A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," March 2000.

19. *BWRVIP-100-A: BWR Vessel and Internals Project, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds.* EPRI, Palo Alto, CA: 2006. 1013396.

20. *ASME Boiler & Pressure Vessel Code, Section XI, Division 1, Nonmandatory Appendices, Appendix C, "Evaluation of Flaws in Austenitic Piping*," American Society of Mechanical Engineers, New York, NY, 2004 Edition, July 1, 2004.

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.

22. *Materials Reliability Program: Fracture Toughness Evaluation of Highly Irradiated PWR Stainless Steel Internal Components (MRP-210)*. EPRI, Palo Alto, CA: 2007. 1016106.

23. *BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel Reactor Pressure Vessel Internals (BWRVIP-14)*. EPRI TR-105873, EPRI, Palo Alto, CA: 1996.

24. *BWRVIP-99: BWR Vessels and Internals Project, Crack Growth Rates in Irradiated Stainless Steels in BWR Internal Components*. EPRI TR-1003018, EPRI, Palo Alto, CA: 2001.

25. *Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data – State of Knowledge (MRP-211)*. EPRI, Palo Alto, CA: 2007. 1015013.

26. WCAP-17096-NP, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" Revision 2 (December 2009), or latest NRC-approved revision.

27. U. S. Nuclear Regulatory Commission letter "Revision 1 to the Final Safety Evaluation of EPRI Report, Material Reliability Program Report 1016596 (MRP-227), Revision 0, *Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines*, (TAC NO. ME0680)," dated December 16, 2011.

A
REACTOR INTERNALS OPERATIONAL EXPERIENCE

Note that in Revision 0 to MRP-227, Appendix A provided guidance for development of an Aging Management Program (AMP) for PWR internals components. This guidance has been deleted from MPR-227. Guidance for AMP preparation may be found in AMP XI.M16A of NUREG-1801, Revision 2 (or subsequent revisions).

Commercial PWR vessel internals in the United States have experienced safe, relatively trouble-free operation. There have been no instances to date in which PWRs in the U.S. have posed a threat to public safety as a result of PWR internals material aging degradation. While relatively few incidents of PWR vessel internals aging degradation have been reported in operating U.S. commercial PWR plants, a summary of the current operating experience is useful for licensees developing aging management programs. This summary is organized first by the aging effect and subsequently by the age-related degradation mechanism leading to that effect. This compilation neither replaces efforts by licensees to review and document their plant-specific operating experience that may impact plant programs, nor does it preclude licensee participation in industry initiatives that perform these functions.

<u>Cracking</u>

IGSCC — Multiple PWR internals bolt failures of the lower thermal shield bolts were discovered during the 1981 and 1982 in-service inspections performed at three B&W-design PWRs. The thermal shield bolt locking clips at these three plants were visually observed to be missing or loose. Subsequent examinations during 1982, 1983, and 1984 revealed bolt failures at four additional units. These failures included upper core barrel, lower core barrel, upper thermal shield, and surveillance specimen holder tube bolts. All of the affected fasteners were fabricated from Alloy A-286 ASTM A 453, Grade 660, Condition A or B material. The results of an extensive evaluation program revealed the failure mechanism was predominantly due to an environmentally-assisted IGSCC mechanism. However, for some bolts, there was evidence that fatigue was also a contributor, likely in the form of corrosion fatigue.

In general, the primary mechanism causing cracking and failure of the Alloy A-286 PWR internals bolts was IGSCC. All the failures occurred in the bolt head-to-shank fillet. Information Notice (IN) 90-68 provides information about IGSCC cracking in Alloy A-286 bolts used to hold the turning vanes to reactor coolant pumps at a foreign plant. The IN 90-68 document includes a general discussion of the problems experienced with cracking of Alloy A-286 bolting materials, including the problems identified with respect to B&W PWR internals bolting.

In 2005, cracking of replacement core barrel-to-former plate bolts fabricated from cold-worked Type 316Ti stainless steel was observed in a German PWR by visual inspection. These bolts had replaced the original Alloy X-750 core barrel bolts in the late 1980s, which had exhibited failure due to PWSCC (described below). Subsequent UT inspection and failure analysis confirmed that the cracking was confined to the bolt head initiating from the bolt fillet transition, but the bolt threads and shank were free from cracking. The failure mechanism of the cold-worked Type 316Ti stainless steel replacement core barrel bolts has been identified as IGSCC. To date, all known failures of core barrel bolts have been limited to the original Alloy X-750 and the replacement cold-worked Type 316Ti stainless steel in German PWRs.

PWSCC — Alloy X-750 has experienced numerous worldwide failures in the Westinghouse-designed PWR internals involving the control rod guide tube support pins (a.k.a., split pins). As noted in IN 82-29, these failures first appeared in Japan in the late 1970s. Split pin failures prompted investigations and modifications to manufacturing practices. The original heat treatment condition AH[1] of the age-hardenable material has shown the most susceptibility to PWSCC cracking. By the early 1980s, nearly all of the original design split pins had been replaced with the improved HTH heat treatment Condition.

In 1987, failures of Alloy X-750 HTH Condition control rod guide tube support pins in French PWRs occurred at much shorter times and lower stresses than expected. Foucault, et al., showed that these early failures were due to the surface condition of the pins. Any heat treatment after machining degrades the performance of Alloy X-750. The greatest resistance to IGSCC was found when machining or polishing was performed after heat treatment, which removes an oxide layer from the surface of the material. Additional refinements have since been made to the manufacturing practices used to produce a newer version of Alloy X-750 HTH split pins.

After an extensive worldwide industry program to develop a material heat treatment for Alloy X-750 that would have maximum resistance to SCC, Westinghouse and utility customers conducted a campaign during the 1980s to replace guide tube support pins. Ultimately, Westinghouse developed a cold-worked Type 316 stainless steel support pin as a replacement and a number of utilities have performed replacements with this design. A few utilities have opted to perform ultrasonic inspections rather than initiate wholesale replacements, while still other utilities have preferred to take no action at this time.

Alloy X-750, in a condition similar to AH, was used for the baffle-to-former plate bolts in the German Biblis-type reactors. After about four years of service, several bolts were found either cracked or severed. The cracking occurred in the bolt head-to-shank fillet area and was attributed to IGSCC (a.k.a., PWSCC in nickel-base materials). The bolt stress levels were reportedly at the yield strength of the material.

Failures have been attributed to three factors:

1. Heat treatment condition
2. High peak stresses
3. Surface damage due to fabrication processes

Failures of Alloy X-750 clevis insert bolts were reported by one Westinghouse-designed plant in 2010. The lower clevis structure works with the radial keyways on the core barrel to provide rotational alignment for the lower internals. The Alloy X-750 bolting was used to fasten the Alloy 600 clevis inserts to the RV lugs. Although the failed clevis insert bolts were not removed for metallurgical examination, it can be surmised that the most likely cause of failure was

[1] Hot rolled, "equalized" at 1625°F (885°C) followed by 20 hours at 1300°F (704°C).

PWSCC. The clevis insert bolting had been heat treated in a condition similar to the AH treatment that has proven to be susceptible to PWSCC in the guide tube support pins. The relatively long time to failure in the clevis insert bolting may be attributed to the lower service temperature.

IASCC — A considerable amount of PWR internals IASCC has been observed in European PWRs since the 1980s, with emphasis on cracking of baffle-former bolting. Ultrasonic (UT) testing of baffle-former bolts in six French PWRs discovered failure rates ranging from 1.2% to 11% of the 960 total bolts. For this reason, the U.S. PWR owners and operators began a program to inspect the baffle-former bolting in order to determine whether similar problems might be expected in U.S. plants. One benefit of this program was the experience gained with the UT examination techniques used in the inspections. In addition, the industry began laboratory testing projects in order to gather the materials data necessary to support future inspections and evaluations.

As part of the U. S baffle-former bolt program, UT inspections were performed at two units with cold-worked Type 316 stainless steel bolting (1998/1999). In one unit, 1086 of 1088 bolts were inspected with no indications. Two bolts could not be inspected due to accessibility and were replaced. In the second unit, all 1088 bolts were inspected, again with no indications. A proactive minimum bolt pattern replacement was performed at these plants (276 bolts for Unit 1 and 203 bolts for Unit 2). Bolts removed from these plants were subject to follow-on mechanical testing and hot cell examination. This follow-on testing confirmed the NDE results.

The program also included inspection of two plants with solution-annealed Type 347 stainless steel baffle-former bolting. In one plant, all 728 baffle-former bolts were inspected by UT in 1998, with 55 bolts (7.5%) having indications that exceeded the UT acceptance criteria. At another unit, 639 out of the 728 solution-annealed Type 347 stainless steel baffle-former bolts were examined in 1999, with 59 bolts (9.2%) having indications failing to meet the UT acceptance criteria. At the first unit, on-site underwater mechanical testing of the removed baffle-former bolts indicated that the actual number of defective bolts was lower than suggested by the UT inspection. However, these known European or domestic baffle-former bolt IASCC indications are not necessarily applicable to all PWR designs. To date, the incidents have been generally associated with cold-worked Type 316 stainless steel or solution-annealed Type 347 stainless steel.

Bolts fabricated from solution-annealed Type 304 stainless steel appear to be less susceptible. An inspection was performed at one B&W-designed unit in 2005 on all 864 baffle-former bolts and UT indications were not observed.

In 2010, one Westinghouse plant reported finding several broken Type 347 stainless steel baffle-former bolt heads and Type 304 stainless steel locking bars on the lower core plate during a normal refueling outage. Subsequent investigation identified a region containing approximately 40 broken or severely damaged bolts. The damage was limited to the upper half of a single baffle plate.

Baffle-former bolt inspections have been conducted under the guidance of MRP-227 at three different Westinghouse-design U.S. domestic plants. The original bolting material at all three plants is Type 347 stainless steel; though, one of the units did replace a subset of the bolts in 1999 with Type 316 stainless steel.

The first of these inspections was a full UT examination of the baffle-former bolts conducted in 2010. The UT inspection detected one likely flaw out of 1088 bolts. Additionally, visual inspection of the baffle-former assembly detected two baffle-former bolts with missing lock bar welds. Each lock bar should have two welds to hold it in place, and these two bolts only had one weld. The missing welds were dispositioned as fabrication errors. The missing welds and flawed bolt were left in service.

The second inspection was a full UT examination of the baffle-former bolts conducted in 2011. The UT inspection detected two likely flaws out of 1088 bolts. Visual examination of the baffle-former assembly did not detect any reportable indications. The flawed bolts were left in service.

The one unit which replaced a subset of bolts in 1999 conducted a UT examination on a subset of bolts. All of the 1999 replacement bolts were inspected and approximately 100 of the original bolts were inspected. Additionally, a small number of bolts were removed and replaced. When possible, these bolts were inspected by UT after removal. Of the bolts inspected, only one defective bolt was detected. This bolt was left in service.

Flow-induced Vibration — In the earlier PWRs, a number of incidents occurred indicating that thermal shields and their support system could be vulnerable to the high flow forces in the vessel-core barrel downcomer. Westinghouse, CE, and B&W responded to these experiences in different ways. The Westinghouse approach was to add vibration-resistance to the shields and to embark on a program to develop advanced thermal shield designs for future plants. For CE plants, thermal shields were removed from operation for all but one facility, which has maintained integrity through positioning pin replacement, tightening, and inspection. The B&W approach was to modify and repair the thermal shields for improved resistance to vibration.

The dominant degradation mechanisms in thermal shields are high-cycle fatigue and SCC resulting from flow-induced vibration, with mechanical wear as a potential consequence. These degradation events appeared predominantly in the earliest thermal shield designs. Typically, the degraded components were fasteners or thermal shield support structures, not the thermal shield itself.

Two CE plants reported cases where failures in the thermal shield resulted in damage to the core barrel. The thermal shields were removed from both plants and the damage to the core barrels was mitigated.

Three early Westinghouse plants identified thermal shield degradation. The thermal shield degradation in these three plants was repaired; however, they are no longer operating and no operating plant has the same thermal shield design. Two additional Westinghouse plants have reported isolated failures of core barrel bolting that may be linked to flow-induced vibration.

Loss of Material

Wear — Wear of the in-core instrumentation thimble tubes was observed in the top part of the Zircaloy-4 thimble tubes at three CE-designed units. These tubes experienced through-wall tube degradation as a result of flow-induced vibration in the vicinity of the fuel alignment plate. This particular wear phenomenon was addressed by making modifications to the fuel alignment plate to alter the flow conditions in the vicinity of the entry point of the thimble tubes into the plate. Wear as a result of flow-induced vibration has not been observed in these components after implementing the modifications to the fuel alignment plate. Accordingly, these components are not considered susceptible to this type of wear in the future.

Problems were noted involving the original locking devices for the B&W-design vent valve jackscrews in the late 1970s and early 1980s. The jackscrew locking mechanism was vibrating and wearing through the locking cup. A new locking mechanism was designed and supplied to most B&W units. At least four of the eight vent valves were modified with the redesigned locking devices. The four vent valves next to the two outlet nozzles were replaced. In the late 1970s and early 1980s, problems were also noted involving the original jackscrew guide bushing, which was found to be improperly secured on some valves. Procedures were developed to install the modified locking device on the jackscrew and to secure the bushing when necessary.

Wear of the Westinghouse control rod guide tube assembly guide cards has been reported at several domestic and international plants. The wear enlarges the guide card holes that guide the control rods through the assembly and maintain the alignment of the rods. A program is currently in progress through the PWROG to establish guidelines for managing this wear.

The wear surfaces on the radial keyways and clevis inserts are routinely examined as part of the PWR internals ASME B&PV Code Section XI inservice inspection programs. While reports of scratches, superficial wear, or both are common in these inspections, one European plant has reported significant wear scars at these surfaces. Efforts to establish quantitative acceptance criteria are ongoing.

In all currently operating Westinghouse and B&W plants, the incore flux detectors are directed through the RV bottom head via thimble tubes or guideways. For the bottom-mounted instrumentation design, the thimble tubes are retractable, and the insertion and retraction of these tubes are directed by long-radius guides below the bottom head and by internals guides between the bottom head and fuel assemblies. There is significant variation among plants with regard to thimble tube diameters (outer and inner), thimble tube-to guide path clearance, length of thimble tube exposed to coolant, and flow conditions.

The primary historical concerns with flux thimble degradation in Westinghouse-designed plants have been obstruction of the flux detector pathways, wear due to flow-induced vibration of the thimble tube, flow-induced vibration fatigue damage to thimble tube guideways, and damage to in-core instrumentation flange seating surfaces at refueling. The obstruction problem can often be mitigated by appropriate cleaning procedures at refueling. All Westinghouse plants are required by NRC Bulletin 88-09 to have an inspection program to periodically confirm incore neutron monitoring system thimble tube integrity. Reductions in wall thickness due to wear are normally monitored with an eddy-current inspection. Many plants have chosen to replace the flux thimbles with improved designs. These programs have been successful in managing thimble tube degradation.

A visual inspection in 1973 at one CE-designed plant revealed worn areas in the RV flange and head resulting from inadequate hold-down spring design and subsequent PWR internals vibration. Prior to shutdown, higher than normal ex-core neutron detector readings had suggested the possibility of excessive internals vibration. Wear was found on the mating surfaces, alignment keys and slots, snubbers, and outlet nozzle faces. The worn surfaces were repaired and a new design using Belleville spring assemblies greatly increased hold-down capacity and mitigated the issue.

In the Westinghouse-design and in two CE-designed units, PWR internals hold-down rings (or springs) were fabricated with Type 304 stainless steel. The subsequent CE-designed units switched to a modified Type 403 stainless steel hold-down ring, which shows less reduction in preload over the lifetime of the component. At least one international Westinghouse-designed plant has replaced their Type 304 stainless steel hold-down rings. Those that have not are managing potential degradation through physical measurement.

Change in Dimension

Irradiation-Induced Growth — Although irradiation-induced growth of zirconium alloys in CE-designed plants was not explicitly identified in MRP-175 as an age-related degradation mechanism to be evaluated as part of the screening process, irradiation-induced growth in the axial direction of the in-core instrumentation thimble tubes has reduced the clearance between the thimble nose and the bottom of the fuel assembly. Some plants had observed that the thimble tube support plate was raised above its normal support position when the upper internals structure was set in place after fuel reload. This indicated that for some of the thimbles the gap tolerance between the thimble tube and the bottom end fitting of the fuel assembly had been reduced until the tube contacted the bottom fitting of the fuel assembly and was being loaded in compression. Ten plants affected by this issue have taken actions. Six of these plants have already replaced the thimble tube assemblies with modified designs that are shorter in length to accommodate the expected irradiation-induced growth. Two additional plants have replacement designs in fabrication and have made preparations to install the replacement thimbles in an upcoming outage. The remaining two plants have not yet begun preparations for a full replacement of the thimble tubes, but one of these two has instead taken the intermediate step of raising the thimble support plate to accommodate additional axial growth. These plants are planning to execute a thimble assembly replacement program during a future refueling outage that is not currently encumbered with other large-scale replacements of major components. All affected plants will likely have replaced their thimble tubes prior to license extension.

Miscellaneous

B&W-design Vent Valves — Vent valve jackscrew locking cup damage has also been observed at some units, which was due to an interaction with the plenum assembly during insertion and removal activities. Vent valves are replaceable items and as noted above, have been replaced as necessary.

Mechanism Unidentified to Date — Visual examinations at one B&W-designed unit in 2005 indicated that three or four internal baffle-to-baffle bolts were found protruding. The bolt heads extended beyond the baffle plate surface. This was an indication that the locking devices, and potentially the bolts as well, had failed. As noted above, a UT inspection of 100% of the baffle-former bolts was performed, with no detected indications of broken bolts. No UT inspection was performed on the internal baffle-to-baffle bolts, and the potentially failed baffle-to-baffle bolts have yet to be removed to confirm failure and, if failed, the mechanism. As a result of the observations, AREVA performed a unit-specific evaluation to assess operational and safety functions for continued operation. That evaluation included thermal hydraulic evaluation, structural evaluation, fuel evaluation, and loose parts evaluation.

RVI Component Replacements — Replacement of upper internals in Westinghouse and CE designs have been made.

Beginning in 2004, replacement of the complete internals (upper and lower internals) at three Japanese PWRs has also been performed. It has been stated that these replacements have been performed for the following reasons:

1. To keep and improve operational reliability, safety, and a high load factor for the nuclear power units
2. To maintain the plant against aging degradation of the PWR internals
3. To mitigate degradation risks that would rise with increasing operational time in the future

B REQUESTS FOR ADDITIONAL INFORMATION AND MRP RESPONSES

May 28, 2009

Mr. Christian B. Larsen
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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1006596, "MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227-REV. 0)" (TAC NO. ME0680)

Dear Mr. Larsen:

By letter dated January 12, 2009, Electric Power Research Institute (EPRI) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review Topical Report (TR) 1006596, "Materials Reliability Program (MRP): Pressurized Water Reactor Internals Inspection and Evaluation Guidelines MRP-227-Rev. 0)." Upon review of the information provided, the NRC staff has determined that additional information (RAI) is needed to complete the review. During a conference call on May 21, 2009, the NRC staff requested that additional EPRI reports referenced in TR MRP-227 be provided expeditiously to the NRC staff. The NRC staff is providing its request for these additional EPRI reports in the enclosed RAI. Please note that a second set of RAIs will be issued separately to capture the technical questions related to the NRC staff's review of TR MRP-227. During a conference call on May 21, 2009, Ann Deema, Senior Project Manager, and I agreed that the NRC staff will receive your response to the enclosed RAI questions by June 12, 2009.

If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-3610.

Sincerely,

/RA/

Tanya M. Mensah, Senior Project Manager
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project Nos. 669 and 689

Enclosure: Request for Additional Information

cc w/encl: See next page

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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1006596, "MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227-REV. 0)" (TAC NO. ME0680)

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Project Nos. 669 and 689

Enclosure: Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT 1006596, "MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES

(MRP-227-REV. 0)

ELECTRIC POWER RESEARCH INSTITUTE

PROJECT NO. 669

During a conference call on May 21, 2009, the NRC staff requested that the following Electric Power Research Institute (EPRI) documents be submitted expeditiously to the NRC to support the staff's review of Topical Report (TR) Materials Reliability Program (MRP)-227.

(1) MRP-189, Revision 1, "Materials Reliability Program: Screening, Categorization, and Ranking of B& W-Designed PWR Internals."

(2) MRP-190, "Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals."

(3) MRP-191, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs."

(4) MRP-210, "Materials Reliability Program: Fracture Toughness Evaluation of Highly Irradiated PWR Stainless Steel Internal Components."

(5) MRP-229, "Materials Reliability Program: Functionality Analysis for B&W Representative PWR Internals."

(6) MRP-230, "Materials Reliability Program: Functionality Analysis for Westinghouse and Combustion Engineering Representative PWR Internals."

(7) MRP-231, "Materials Reliability Program: Aging Management Strategies for B&W PWR Internals."

(8) MRP-232, "Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals."

Based upon this conference call, the NRC staff understood that the EPRI would provide both proprietary and non-proprietary versions of these documents, with the exception of MRP-229 and MRP-230. The NRC staff and EPRI agreed during the conference call that these two documents were currently not needed by the NRC staff. However, if the NRC staff determines at a later date that they need to review MRP-229 and MRP-230 to support the review of MRP-227, then the EPRI will submit proprietary and non-proprietary versions of these documents expeditiously to the NRC Document Control Desk.

ENCLOSURE

ELECTRIC POWER
RESEARCH INSTITUTE

MRP **Materials Reliability Program** MRP 2009-043

(via email)

June 10, 2009

Tanya M. Mensah
U.S. Nuclear Regulatory Commission
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Rockville, Maryland 20852-2738

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1006596,'MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227-REV. 0)' (TAC NO. ME0680)

Reference:

1. Letter Tanya Mensah (NRC) to Christian B. Larsen (EPRI), same subject, dated May 28, 2009
2. Letter and Affidavit, Christian B. Larsen (EPRI) to NRC Document Control Desk, Request for Withholding Commercial Documents, dated June 10, 2009

Dear Ms. Mensah:

In response to your May 28 letter (Reference 1) requesting that EPRI provide copies of reports to support NRC review of EPRI Report 1006596, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)" we are forwarding eight copies of the following four documents:

1) *Materials Reliability Program:* Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items (MRP-189-Rev. 1). EPRI, Palo Alto, CA: 2009. 1018292;

2) *Materials Reliability Program:* Fracture Toughness Evaluation of Highly Irradiated PWR Stainless Steel Internal Components (MRP-210). EPRI, Palo Alto, CA: 2007. 1016106;

3) *Materials Reliability Program:* Aging Management Strategies for B&W PWR Internals (MRP-231). EPRI, Palo Alto, CA: 2008. 1016592;

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

These documents have been forwarded to the Document Control Desk by Reference 2 (copy attached) requesting that this copyrighted information be withheld from public disclosure. Two of the requested documents, MRP-190, "Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals" and MRP-191, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustions Engineering PWR Designs" have been made publicly available and are no longer controlled documents. They may be downloaded, at no cost, from the EPRI web site (www.EPRI.com).

As discussed during the conference call on May 21, 2009, AREVA has identified a minor error in the functionality analysis for the B&W plants that is reflected in one of the documents forwarded with this transmittal, MRP-231. The error does not affect the aging management strategies contained in MRP-231 nor the related recommendations in MRP-227; however, for completeness and accuracy the error is being corrected and will result in a revision to MRP-231. At such time as MRP-231, Rev. 1 is available, it will be provided to the NRC under a separate cover letter.

If you have any questions, please contact Christine King at 650-855-2605, or Anne Demma at 650-855-2026.

Best Regards,

Dennis P. Weakland

Dennis Weakland
FirstEnergy Nuclear Operating Co.
Chairman, Materials Reliability Program

Cc: Mike Melton (NEI)
Don Dyksterhouse (INPO)
Chris Larsen (EPRI)
Christine King (EPRI)
Anne Demma (EPRI)

NEI Project Nos. 669 and 689

August 24, 2009

Mr. Christian B. Larsen
Nuclear Vice President & Chief Officer
Electric Power Research Institute
3420 Hillview Avenue
Palo Alto, CA 94304-1338

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1006596, "MATERIALS RELIABILITY PROGRAM (MRP): PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227 – REV. 0) (TAC NO. ME0680)

Dear Mr. Larsen:

By letter dated January 12, 2009, Electric Power Research Institute submitted for U.S. Nuclear Regulatory Commission (NRC) staff review Topical Report (TR) 1006596, "Materials Reliability Program (MRP): Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." Upon review of the information provided, the NRC staff has determined that additional information (RAI) is needed to capture the initial set of technical questions related to the NRC staff's review of TR MRP-227 to support completion of the review. The NRC staff will however issue another set of RAIs based on its review of the MRP-227 report and its supporting reports. By e-mail dated August 22, 2009, Ms. Christine King, Program Manager, MRP, and I agreed that the NRC staff will receive your response to the enclosed RAI questions within 60 days of issuance of this letter. If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-3610.

Sincerely,

/RA/

Tanya M. Mensah, Senior Project Manager
Special Projects Branch
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Office of Nuclear Reactor Regulation

Project Nos. 669 and 689

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August 24, 2009

Mr. Christian B. Larsen
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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1006596, "MATERIALS RELIABILITY PROGRAM (MRP): PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227 – REV. 0) (TAC NO. ME0680)

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Project Nos. 669 and 689

Enclosure:
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ADAMS ACCESSION NO.: ML092250603 **NRR-106**

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REQUEST FOR ADDITIONAL INFORMATION (RAI)

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT 1006596, "MATERIALS RELIABILITY PROGRAM (MRP): PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227 – REV. 0)

ELECTRIC POWER RESEARCH INSTITUTE

PROJECT NO. 669

In a letter dated January 12, 2009, the Electric Power Research Institute (EPRI) submitted a Topical Report (TR) MRP-227, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," which addresses the development of an aging management program (AMP) for PWR reactor vessel internal (RVI) components. On July 2, 2009, EPRI provided additional reports that support the technical bases used for developing the AMP, and these reports were submitted to the NRC staff for information only. The NRC staff has reviewed TR MRP-227and developed an initial set of RAIs. Based on further review of the supporting reports, the NRC staff may issue additional RAIs at a later date.

RAI-1 Many components are placed on a standard 10-year inservice inspection interval coincident with typical American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) inspection requirements. It's not clear, however, whether this 10-year interval is technically acceptable for PWR RVI components. No justification in light of the specific degradation mechanisms being managed has been provided. Other inspection intervals and requirements are based on a certain number of operating cycles. The acceptability of these intervals has also not been established. Please provide a technical justification of the intervals chosen relative to the mechanisms being managed in TR MRP-227.

RAI-2 In Tables 4-1 through 4-6 and Tables 4-8 and 4-9 of TR MRP-227, the MRP intends to implement visual testing (VT-3) examinations to identify cracking in some PWR RVI components. Historically, enhanced visual testing (EVT-1) or ultrasonic testing (UT) methods are used to effectively identify cracks. Explain why the use of a VT-3 inspection method should be considered acceptable for identifying cracking in some PWR RVI components.

RAI-3 Eddy current testing (ET) is identified in TR MRP-227 as an inspection method to be used to identify cracking in some PWR RVI components. Clarify whether the acceptance criterion for ET inspections will be based on a "pass – no pass" acceptance criterion (i.e., any ET signals indicating a relevant ET indication would fail the acceptance criterion).

RAI-4 The accessibility of the primary inspection RVI components is not typically addressed. It is therefore not clear how much inspection coverage is necessary to ensure timely detection of aging effects in the primary inspection RVI components. Discuss whether guidance should be provided in TR MRP-227 regarding minimum inspection volumes/areas which must be achieved to take credit for having effectively inspected a particular RVI component.

ENCLOSURE

RAI-5 During the extended period of operation, some PWR RVI components are subject to high levels of neutron radiation which may lead to irradiation embrittlement and a loss of fracture toughness and the potential for irradiation-assisted stress corrosion cracking. In combination, these effects may lead to the potential for component failure under some design basis loading conditions. Explain how licensees will be expected to account for potential reduction in fracture toughness when evaluating cracks that are detected during the required inspections, in particular when establishing the frequency of subsequent inspections after cracking is identified.

RAI-6 Loose parts could be generated due to deterioration of some PWR RVI components during the extended period of operation. Provide information which addresses how the following consequences of loose parts generation were considered in development of the inspection program given in TR MRP-227.

(a) potential for fuel bundle flow blockage and consequential fuel damage,

(b) potential for interference with control rod operation, and

(c) potential for impact damage on reactor internals.

RAI-7 Alloy 600 PWR RVI components and their associated welds manufactured from Alloys 82 and 182 are susceptible to primary water stress corrosion cracking (PWSCC) when exposed to PWR reactor coolant water. In Table 3-1 of TR MRP-227, the following Babcock and Wilcox (B&W) Alloy X-750 PWR RVI components were welded with Alloy 82 material and yet they were classified under "N" category which excludes inspections for these PWR RVI components:

(1) dowel-to-core barrel cylinder welds, (2) dowel-to-upper grid rib section bottom flange welds, (3) dowel locking welds, (4) dowel-to-guide block welds, and (5) dowel-to-distributor flange welds.

Even though stress levels in these components may not exceed the threshold levels, the NRC staff considers it to be likely that PWSCC can potentially occur due to the introduction of cold work during fabrication. In light of this observation, provide an explanation for excluding inspection requirements for these B&W PWR RVI components.

RAI-8 When exposed to a light-water reactor temperatures of approximately 500 ºF or higher, the 17-4 precipitation hardened (PH) martensitic stainless steel (MSS) that has previously been subjected to aging (heat treatment) at about 1100 ºF can experience thermal embrittlement and an increase in hardness and a reduction in Charpy V-notch impact test toughness. Operating experience from Oconee Nuclear Station (Information Notice (IN) 2007-02, ADAMS Accession Number ML070100459) shows that thermally embrittled 17-4 PH MSS is susceptible to failure when exposed to unexpected loading conditions. In IN 2007-02, the NRC staff recommended that licensees prevent the deleterious effects of thermal embrittlement in the 17-4 PH MSS components by identifying aging degradation (i.e., cracks), implementing early corrective actions, and monitoring and trending age-related degradation. Therefore, the NRC staff requests that the TR MRP-227 report should include thermal embrittlement as an aging effect for any 17-4 PH MSS RVI components.

RAI-9 With respect to the management of cast austenitic stainless steel (CASS) aging and embrittlement TR MRP-227 does not appear to address the program's compliance with the requirements specified in the relevant Generic Aging Lessons Learned (GALL) Report AMPs. Provide a discussion of how TR MRP-227 adequately addresses the requirements specified in GALL AMP, XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)," and GALL AMP XI.M13, "Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS)," for CASS materials used in PWR RVI components. Alternatively, if the management CASS PWR RVI component aging is not treated within the scope of TR MRP-227, provide a proposed modification of the report which documents how licensees are expected to manage this mechanism outside of the TR MRP-227 program.

RAI-10 According to Section A.1.4 in MRP-175, "Materials Reliability Program: PWR Internal Aging Degradation Mechanism Screening Threshold Values," susceptibility to SCC in nickel-based Alloy X-750 PWR RVI components depends on the type of heat treatment that is performed on the alloy. High temperature heat treatment processes that are used on Alloy X-750 components offer better resistance to SCC than the other age hardened heat treatment processes. Licensee determination of the heat treatment applied to their Alloy X-750 PWR RVI components would appear to be a critical parameter in ensuring the licensee's AMP will adequately manage the potential effects of aging. Discuss whether this determination should be included as a license renewal application action item.

RAI-11 Following on to RAI-10, additional aspects of the TR MRP-227 methodology may need to be addressed by license renewal applicant action items for applications currently under review or those that have yet to be submitted to the NRC. The NRC staff requests the MRP's assistance in identifying potential action items which are: (1) necessary to provide plant-specific information to complete the AMP; (2) necessary to confirm applicant compliance with important assumptions underlying the MRP-227 methodology; or (3) other considerations.

RAI-12 Provide the loading combinations that were used in determining the peak stress values for any given PWR RVI component. The NRC staff believes that plants that have been implementing power uprates will have to assess whether the peak stress values for any given PWR RVI component are affected by power uprate conditions to determine if their plant is bounded by the assumptions underlying TR MRP-227.

RAI-13 Certain degradation mechanisms (e.g., void swelling in B&W PWR RVI components) are not inspected for in a particular reactor type. Why does the program not require the most susceptible location for each mechanism in each reactor-type (i.e., B&W, Combustion Engineering, or Westinghouse) be inspected as a primary component to insure that each degradation mechanism is not occurring within the reactor?

RAI-14 Discuss how the PWR RVI components in each reactor design considered to be the most susceptible to (or most likely to first demonstrate the effects of) a particular degradation mechanism did, or did not, get binned in the primary inspection component group for that design.

RAI-15 The failure modes, effects, and criticality analysis (FMECA) uses a probabilistic approach with regard to structural stability of any given RVI component and includes the development of a "failure probability factor". What methodology was used to establish the failure probability factor of any RVI component?

RAI-16 Clarify the conditions under which design basis event (DBE) effects on component performance were considered. How does this approach provide reasonable assurance that the margins against failure are adequately maintained during the license renewal period?

RAI-17 Component failure due to the same degradation mechanism is not considered to be a common cause failure because of the expectation that damage initiation and growth occurs at different times. However, certain DBEs could potentially lead to a plant condition (damage state) that would not occur unless multiple components were degraded. Discuss how the potential for multi-component failure due to a DBE was considered as part of the development of the MRP-227 program.

RAI-18 Clarify how plant-specific differences were considered within the FMECA. Discuss whether any additional plant-specific analyses are required, either as a supplement to TR MRP-227 or as indentified plant-specific action items, in order to assure that FMECA analysis supporting the TR MRP-227 program is applicable to a given facility.

RAI-19 Discuss how a licensee will demonstrate adherence to the reference core loading pattern on a unit-specific basis. Address plant-to-plant variability in neutron flux at various peripheral core locations. Confirm, based on significant operating experience, that "low-leakage" core designs, when normalized by power density, have peripheral neutron fluxes that are consistently within the estimates for the generically studied plants.

RAI-20 Provide a technical basis to justify the examination acceptance criteria, the sufficiency and relevancy of the links between primary and expansion group components (why were those particular links chosen), and the expansion criteria. Discuss also the technical basis that applied to place certain components in the primary category while others were placed in the expansion category.

RAI-21 Many of the acceptance criteria provided in TR MRP-227 are vague such as finding "detectable crack-like surface indications," or "damaged or fractured material," or "readily detectable cracking." It's not clear that these criteria will be uniformly interpreted or implemented from plant to plant. Discuss the need to develop more detailed acceptance criteria on a plant-specific basis and how will the sufficiency of these criteria be established.

RAI-22 The screening criteria groups materials into susceptibility levels for each degradation mechanism: highly susceptible, moderately susceptible, susceptible, and "below the screening criteria." Discuss the criteria used to distinguish among the different levels of susceptibility.

RAI-23 Discuss whether an evaluation was performed for any specific high consequence of failure PWR RVI components such that their inspection might be warranted even in the absence of a currently identifiable mechanism. Are there any PWR RVI components that should be monitored through in-service inspection to protect against unforeseen failure due to the emergence of a potential future degradation mechanism?

RAI-24 Relevant US and international operating experience with respect to RVI components is not summarized. It is important to indicate what prior RVI component inspections have identified, in particular with respect to justifying the adequacy of existing programs and as part of the basis for the examination requirements (e.g., type, periodicity, importance) identified in MRP-227.

<u>RAI-25</u> The cumulative usage factor values for several B&W components need to be confirmed during a comprehensive search of all existing stress and fatigue calculations for the PWR internals. Discuss how such items are intended to translate into plant-specific action items.

<u>RAI-26</u> The implications of void swelling are indicated as “dimensional change and distortion…” and it is also noted that “severe void swelling may result in cracking under stress.” However, it is not indicated that void swelling can lead to reduced fracture toughness in materials even though it is noted in Section 3.2.7 of TR MRP-227 that “severe swelling (>5%) has been correlated with extremely low fracture toughness values.” It is not clear how much void swelling is needed before distortion is detectable via VT-3 examination in susceptible PWR RVI components and whether this threshold for dectectability will also address the concern over potential loss of fracture toughness due to void swelling. Provide a discussion of this topic.

ELECTRIC POWER RESEARCH INSTITUTE

MRP Materials Reliability Program ______________________________ MRP 2010-004

February 1, 2010

Ms. Tanya M. Mensah
Senior Project Manager
Special Projects Branch, Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Subject: EPRI MRP Responses to: REQUEST FOR ADDITIONAL INFORMATION RE: ELECTRIC POWER RESEARCH INSTITUTE TOPICAL REPORT 1016596,'MATERIALS RELIABILITY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS INSPECTION AND EVALUATION GUIDELINES (MRP-227-REV. 0)' (TAC NO. ME0680), August 24, 2009 (please note the corrected EPRI Product Number – 1016596 – from that in the actual RAI request letter – 1006596)

Reference:

1. Letter Tanya Mensah (NRC) to Christian B. Larsen (EPRI), Request for Additional Information, dated August 24, 2009

Dear Ms. Mensah:

In response to your August 24 letter (Reference 1) we are forwarding ten copies of the subject response, eight copies for the staff and two for the Document Control Desk.

If you have any questions, please contact Christine King at 650-855-2605, or Anne Demma at 650-855-2026.

Best Regards,

Terry McAlister

Terry McAlister
South Carolina Electric & Gas Co.
Chairman, Materials Reliability Program

cc: NRC Document Control Desk (with Attachment – 2 copies)
Victoria Anderson, NEI
William Greeson, INPO
Christine King, EPRI
Anne Demma, EPRI

Final Responses to the 2nd set of RAIs on MRP-227-Rev. 0
02/01/2010

Titles of MRP Reports Referenced in MRP-227-Rev. 0 or Referred to in RAI Responses

MRP #	Title	EPRI #
MRP-128	*Materials Reliability Program: Characterization of Decommissioned PWR Vessel Internals Material Samples – Material Certification, Fluence, and Temperature,* 2003	1008202
MRP-134	*Materials Reliability Program: Framework and Strategies for Managing Aging Effects in Reactor Internals,* 2005	1008203
MRP-135 - Rev. 1	*Materials Reliability Program: Development of Material Constitutive Model for Irradiated Austenitic Stainless Steel,* 2009	1018291
MRP-156	*Materials Reliability Program: Pressurized Water Reactor Issue Management Table, PWR-IMT, Consequence of Failure,* 2005	1012110
MRP-157	*Materials Reliability Program: Updated B&W Design Information for the Issue Management Tables,* 2005	1012132
MRP-175	*Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values,* 2005	1012081
MRP-189 - Rev. 1	*Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals,* 2009	1018292
MRP-190	*Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals,* 2006	1013233
MRP-191	*Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs,* 2006	1013234
MRP-210	*Materials Reliability Program: Fracture Toughness Evaluation of Highly Irradiated PWR Stainless Steel Internal Components,* 2007	1016106
MRP-211	*Materials Reliability Program: PWR Internals Age-Related Material Properties, Degradation Mechanisms, Models, and Basis Data – State of Knowledge,* 2007	1015013
MRP-228	*Materials Reliability Program: Inspection Standard for Reactor Internals,* 2009	1016609

MRP-229 - Rev. 1	*Materials Reliability Program: Functionality Analysis for B&W-Designed Representative PWR Internals,* 2009	1019090
MRP-230 - Rev. 1	*Materials Reliability Program: Functionality Analysis for Westinghouse & CE-Designed Representative PWR Internals,* 2009	1019091
MRP-231	*Materials Reliability Program: Aging Management Strategies for B&W-Designed PWR Internals,* 2008	1016592
MRP-232	*Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals,* 2008	1016593

RAI-1 Many components are placed on a standard 10-year inservice inspection interval coincident with typical American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) inspection requirements. It's not clear, however, whether this 10-year interval is technically acceptable for PWR RVI components. No justification in light of the specific degradation mechanisms being managed has been provided. Other inspection intervals and requirements are based on a certain number of operating cycles. The acceptability of these intervals has also not been established. Please provide a technical justification of the intervals chosen relative to the mechanisms being managed in TR MRP-227.

Response:
The supporting documentation for MRP-227, such as the aging management strategy reports MRP-231 and MRP-232, evaluated each Primary, Expansion, or Existing Programs component and the associated degradation effects to determine the timing of initial and subsequent examinations or other inspections. The evaluation considered available operating experience for those components and laboratory data for the component materials, as well as information derived from the functionality analyses documented in MRP-229 and MRP-230. Based upon these evaluations, unless information gained from the first round of augmented inspections demonstrates otherwise, the schedule for initial examinations and the 10-year periodicity of subsequent examinations provide confidence that the aging effects for those components will be managed adequately and the functionality of the components will continue to be maintained.

However, the intention is that MRP-227, through its proactive approach to managing aging effects in PWR internals components, is and will continue to be a living document. The results of the first and subsequent rounds of augmented examinations will be reported to the MRP (see Section 7 of MRP-227) and may lead to changes in the periodicity of subsequent examinations. For example, if more aggressive degradation effects are identified through these augmented examinations, or through other operating experience, the findings would be evaluated to determine what, if any, changes are needed in the MRP-227 requirements.

Two sets of investigations performed during the development of MRP-227 provide strong support for the adequacy of the chosen inspection intervals. First, as documented in MRP-229 and MRP-230, the time history analyses of the individual and combined degradation mechanisms modeled in the functionality analyses showed a gradual, rather than a sudden, progression of potential degradation, with the majority of those aging effects accumulating during the conservatively assumed 30-years of high leakage core loading. The functionality analyses account for the combined effects of aging degradation mechanisms by including the irradiation/temperature-induced effects on material response, e.g. mechanical property changes, void swelling, creep/relaxation, etc., in the time history analysis. Second, the operating history review discussed in MRP-231 and MRP-232 confirms the finding that the required 10-year ASME Code Examination Category B-N-3 inspections and other, voluntary U.S. and international industry baffle-to-former bolt examinations show that 10-year examination intervals are appropriately conservative.

RAI-2 In Tables 4-1 through 4-6 and Tables 4-8 and 4-9 of TR MRP-227, the MRP intends to implement visual testing (VT-3) examinations to identify cracking in some PWR RVI components. Historically, enhanced visual testing (EVT-1) or ultrasonic testing (UT) methods are used to effectively identify cracks. Explain why the use of a VT-3 inspection method should be considered acceptable for identifying cracking in some PWR RVI components.

Response: The VT-3 examination was specified to identify general degradation in the aged components. In none of the cases where VT-3 is specified is the examination objective the detection of the onset of cracking with accompanying tight crack opening displacements (CODs). In the cases where cracking associated with IASCC/SCC is the anticipated mechanism or one of the mechanisms postulated and VT-3 is specified, the objective of the examination is detection of:

- Broken or missing bolt locking devices and welds
- Protruding bolts
- Broken or missing pieces (e.g., supports, spider arms, dowels)

In these cases, VT-3 as currently defined in Section XI of the ASME Code (0.106" character height resolution) is quite capable of this level of detection. This practice was deemed appropriate for the existing examination requirements listed in MRP-227 Tables 4-8 and 4-9.

As used in MRP-227, VT-3 is consistent with Section XI rules for pipe supports looking for:

- Missing bolts
- Gross degradation
- Misalignment
- General structural condition

Table RAI-2-1 provides a summary of the Primary and Expansion components from the B&W design where VT-3 is specified for cracking. In each case, the general condition of concern has been identified. The VT-3 inspections have been for cases where extensive cracking can occur without threatening the structural integrity of the internals. The use of VT-3 examinations to monitor cracking in the B&W internals can be summarized in four categories:

- Bolt locking devices (stress corrosion cracking, SCC), specified as separated or missing locking devices or welds
- Core Support Shield components (irradiation or thermal embrittlement, IE/TE), specified as detection of surface irregularities such as damaged, fractured, or missing material
- Baffle plates (irradiation embrittlement, IE), specified as "readily detectable" cracking
- IMI spiders/spider arms (TE/IE), specified as fractured or missing spider arms or separation of spider arms from the lower grid rib section at the weld

Table RAI-2-2 provides a summary of the of the Primary and Expansion components from the CE design where VT-3 is specified for cracking. There are only two CE components in this category:

- Instrument guide tubes attached to the upper CEA structure and the welds in the lower support structure. The inspection for this component specifies the concern as cracking that results in missing supports or separation at the welded joint between the tubes and the supports
- Core support column welds. The inspection for this component specifies the concern as damaged or fractured material.

Table RAI-2-3 provides a summary of the of the Primary and Expansion components from the Westinghouse design where VT-3 is specified for cracking. There are only three Westinghouse components in this category:

- Baffle-Edge Bolts. Although these bolts have a purpose, they do not carry primary loads required for structural integrity. The inspection is required to detect the following conditions:
  - Lost or broken locking devices
  - Failed or missing bolts
  - Protrusion of bolt heads.
- Thermal Shield Flexures. The inspection is required to detect excessive wear, fracture, or complete separation.
- Bottom Mounted Instrumentation (BMI) columns. The inspection is required to detect completely fractured column bodies

VT-3 examinations for cracking were not specified in cases where the data would potentially be used in a fracture mechanics analysis to demonstrate the structural integrity of the vessel internals. These more sensitive crack detection capabilities of the UT and EVT-1 examinations have been specified for other components where it was determined that early detection and protection against fracture was critical. However even in these cases the example calculations performed in MRP-210, demonstrate that the internals structures are extremely flaw-tolerant. See for example, the critical flaw sizes for the postulated stresses for a through-wall edge crack in a flat plate (Figure 3-18 and Table 3-3 in MRP-210). Thus, a VT-3 examination is capable of identifying subcritical crack growth well before a crack would become critical. For justification, see for example the final paragraph of Section 3.2.3.1 of MRP-231.

The MRP-227 inspection recommendations for VT-3 to monitor for the effects of cracking are appropriate because they are limited to cases where the intent of the examination is to monitor the general condition of the component. These recommendations are consistent with the approach used in the ASME Section XI examinations, which require VT-3 inspections for accessible core support structures. The practice is also consistent with the approach used in BWR applications. Further discussion of the inspection strategies for these components may be found in MRP-231 and MRP-232.

Table RAI-2-1 Primary and Expansion Components from B&W Designed Plants with Cracking Identified as an Effect and VT-3 Examinations Specified

Item	Effect (Mechanism)	Examination Method/Frequency	Examination Coverage
Primary **Core Support Shield Assembly** CSS cast outlet nozzles	Cracking (TE), including the detection of surface irregularities, such as damaged or fractured material	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
Primary **Core Support Shield Assembly** CSS vent valve discs (Note 1)			100% of accessible surfaces. (See BAW-2248A, page 4.3 and Table 4-1.) See Figures 4-10 and 4-11
Primary **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)	Cracking (TE), including the detection of surface irregularities, such as damaged, fractured material, or missing items	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
Primary **Core Support Shield Assembly** Upper core barrel (UCB) bolts and their locking devices	Cracking (SCC)	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
Primary **Core Barrel Assembly** Lower core barrel (LCB) bolts and their locking devices	Cracking (SCC)	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
Primary **Core Barrel Assembly** Baffle plates	Cracking (IE), including the detection of readily detectable cracking in the baffle 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

Item	Effect (Mechanism)	Examination Method/Frequency	Examination Coverage
Primary **Core Barrel Assembly** Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	Cracking (IASCC, IE, Overload), including the detection of missing, non-functional, or removed locking devices or welds	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
Primary **Lower Grid Assembly** Alloy X-750 dowel-to-guide block welds	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	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
Primary **Incore Monitoring Instrumentation (IMI) Guide Tube Assembly** IMI guide tube spiders IMI guide tube spider-to-lower grid rib section welds	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	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 Figures 4-3 and 4-6
Expansion **Upper Grid Assembly** Alloy X-750 dowel-to-upper fuel assembly support pad welds	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	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)
Expansion **Control Rod Guide Tube Assembly** CRGT spacer castings	Cracking (TE), including the detection of fractured spacers or missing screws.	Visual (VT-3) examination	100% of accessible surfaces at the 4 screw locations (at every 90°) (limited accessibility) See Figure 4-5

Item	Effect (Mechanism)	Examination Method/Frequency	Examination Coverage
Expansion **Lower Grid Assembly** <u>Lower fuel assembly support pad items</u>: 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)	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	Visual (VT-3) examination	100% of accessible pads, dowels, and cap screws, and associated welds See Figure 4-6
Expansion **Lower Grid Assembly** Alloy X-750 dowel-to-lower fuel assembly support pad welds	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	Visual (VT-3) examination	100% of accessible dowels welds See Figure 4-6