Westinghouse Non-Proprietary Class 3

WCAP-17133-NP
Revision 0

November 2009

PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Palisades Nuclear Plant

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Karli Bowser*
Reactor Internals Design and Analysis

Cheryl Boggess*
Reactor Internals Design and Analysis

Joshua McKinley*
Materials Center of Excellence

Dr. Randy Lott*
Materials Center of Excellence

November 2009

Approved: Brian Gaia*, Manager
Reactor Internals Design and Analysis

*Electronically approved records are authenticated in the electronic document management system.

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

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LIST OF ACRONYMS

AMP	Aging Management Program
AMR	Aging Management Review
ARDM	age-related degradation mechanism
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
B&W	Babcock & Wilcox
BWR	boiling water reactor
CASS	cast austenitic stainless steel
CE	Combustion Engineering
CEOG	Combustion Engineering Owners Group
CFR	Code of Federal Regulations
CLB	current licensing basis
EFPY	effective full-power year
EPRI	Electric Power Research Institute
EVT	enhanced visual testing (a visual NDE method that includes EVT-1)
FMECA	failure modes, effects, and criticality analysis
GALL	Generic Aging Lessons Learned
I&E	Inspection and Evaluation
IASCC	irradiation-assisted stress corrosion cracking
ICI	in-core instrumentation
IGA	intergranular attack
INPO	Institute of Nuclear Power Operations
ISI	inservice inspection
ISR	irradiation-enhanced stress relaxation
LRA	License Renewal Application
MRP	Materials Reliability Program
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NES	Nuclear Engineering & Services
NOS	Nuclear Oversight Section
NRC	U.S. Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OBE	operating basis earthquake
ODSCC	outer diameter stress corrosion cracking
OE	Operating Experience
OER	Operating Experience Report
PH	precipitation-hardening
PWR	pressurized water reactor
PWROG	Pressurized Water Reactor Owners Group
QA	quality assurance
RCS	reactor coolant system
RO	refueling outage
RV	reactor vessel
RVI	reactor vessel internals

LIST OF ACRONYMS (cont.)

SCC	stress corrosion cracking
SER	Safety Evaluation Report
SRP	Standard Review Plan
SS	stainless steel
TLAA	time-limited aging analysis
UT	ultrasonic testing (a volumetric NDE method)
UGS	upper guide structure
VT	visual testing (a visual NDE method that includes VT-1 and VT-3)

ACKNOWLEDGEMENTS

The authors would like to thank the members of the Entergy Aging Management Program Team led by Keith Smith and our associates at Westinghouse for their efforts in supporting development of this WCAP.

1 PURPOSE

The purpose of this report is to document the Palisades Nuclear Plant (hereafter "Palisades") Reactor Vessel Internals (RVI) Aging Management Program (AMP). The purpose of the AMP is to manage the effects of aging on RVI through the license renewal period, which begins at the expiration of the original license in 2011. This document provides a description of the program as it relates to the management of aging effects identified in various regulatory and updated industry-generated documents in addition to the program described in the Palisades document LR-AMR-RVI [1] in support of license renewal program evaluations. This AMP is supported by existing Palisades documents and procedures and, as required by industry experience or directives in the future, will be updated or supported by additional documents to provide clear and concise direction for the effective management of aging degradation in reactor internals components. These actions provide assurance that operations at Palisades will continue to be conducted in accordance with the current licensing basis (CLB) for the RVI by fulfilling License Renewal commitments [2], following American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI Inservice Inspection (ISI) programs [3, 4], and meeting industry requirements [5]. Based on the programs sponsored by U.S. utilities through the Electric Power Research Institute (EPRI) managed Materials Reliability Program (MRP) and the Pressurized Water Reactor Owners Group (PWROG), this AMP fully captures the intent of the additional industry guidance for reactor internals augmented inspections.

The main objectives for the Palisades RVI AMP are to:

- Demonstrate that the effects of aging on the RVI will be adequately managed for the period of extended operation in accordance with 10 CFR 54 [6].
- Summarize the role of existing Palisades AMPs in the RVI AMP.
- Define and implement the industry-defined (EPRI/MRP and PWROG) pressurized water reactor (PWR) RVI requirements and guidance for managing aging of reactor internals.
- Provide an inspection plan summary for the Palisades reactor internals.

Palisades License Renewal Commitment 33 [2], "Reactor Vessel Internals Program," commits Palisades to:

> *"...participate in industry initiatives that will generate additional data on aging mechanisms relevant to reactor vessel internals (RVI), including void swelling, and develop appropriate inspection techniques to permit detection and characterization of features of interest. Recommendations for augmented inspections and techniques resulting from this effort will be incorporated into the Reactor Vessel Internals Program as applicable. The revised Reactor Vessel Internals Program will be submitted for NRC review and approval by March 24, 2009."* [2]

Entergy subsequently advised the NRC of a change to the Commitment 33 date for submittal of the revised Reactor Vessel Internals Program from March 24, 2009 to March 24, 2010 [7].

Augmented inspections, based on required program enhancements resulting from industry programs, will become part of the Reactor Vessel Internals Inspection Program [8]. Corrective actions for augmented inspections will be developed and will use repair and replacement procedures equivalent to those requirements in ASME B&PV Code, Section XI, or they will use processes determined to be equivalent to or more rigorous than currently defined procedures as determined independently by Entergy or in cooperation with the industry.

This AMP for the Palisades reactor internals demonstrates that the program adequately manages the effects of aging for reactor internals components and establishes the basis for providing reasonable assurance that the internals components will continue to perform their intended function through the Palisades license renewal period of extended operation. It supports the Palisades license renewal commitment to submit a Reactor Vessel Internals Program to the U.S. Nuclear Regulatory Commission (NRC) by March 24, 2010 [7]. It addresses the MRP-227 requirement for development of an AMP by December 2011 under the NEI-03-08 protocol for mandatory items.

The development and implementation of this program meets the license renewal amendment request commitment and industry mandatory directive for a reactor internals AMP.

2 BACKGROUND

The management of aging degradation effects in reactor internals is required for nuclear plants considering or entering license renewal, as specified in the NRC Standard Review Plan (SRP) [10]. The U.S. nuclear power industry has been actively engaged in recent years in a significant effort to support the industry goal of responding to these requirements. Various programs have been underway within the industry over the past decade to develop guidelines for managing the effects of aging within PWR reactor internals. Later, an effort was engaged by the EPRI MRP to address the PWR internals aging management issue for the following three currently operating U.S. reactor designs: Westinghouse, Combustion Engineering (CE), and Babcock & Wilcox (B&W).

The MRP first established a framework and strategy for the aging management of PWR internals components using proven and familiar methods for inspection, monitoring, surveillance, and communication. Based on that framework and strategy and on the accumulated industry research data, the following elements of an AMP were further developed [11, 12, 13]:

- Screening criteria were developed, considering chemical composition, neutron fluence exposure, temperature history, and representative stress levels, for determining the relative susceptibility of PWR internals components to each of eight postulated aging mechanisms (further discussed in Section 4 of this program).
- PWR internals components were categorized, based on the screening criteria, as follows:
  - Components for which the effects from the postulated aging mechanisms are insignificant
  - Components that are moderately susceptible to the aging effects
  - Components that are significantly susceptible to the aging effects
- Functionality assessments were performed based on representative plant designs of PWR internals components and assemblies of components, using irradiated and aged material properties, to determine the effects of the degradation mechanisms on component functionality.

Aging management strategies were developed combining the results of the functionality assessment with several contributing factors to determine the appropriate aging management methodology, baseline examination timing, and the need and timing of subsequent inspections. Factors considered included component accessibility, operating experience (OE), existing evaluations, and prior examination results.

The industry effort, as coordinated by the EPRI MRP, has produced finalized initial Inspection and Evaluation (I&E) Guidelines for reactor internals and document submittal to the NRC with a request for a formal Safety Evaluation Report (SER). A supporting document addressing inspection requirements was also completed. The industry guidance is contained within two separate EPRI MRP documents:

- MRP-227, "PWR Internals Inspection and Evaluation Guidelines" [5] (hereafter referred to as "the I&E Guidelines" or simply "MRP-227") provides industry background for the guidelines, lists of reactor internals components requiring inspection, and the timing for initial inspections of those components. For each component, the guidelines require a specific type of nondestructive examination (NDE) and give criteria for evaluating inspection results. MRP-227 provides a standardized approach to PWR internals aging management for each unique reactor design (Westinghouse, B&W, and CE). The document was submitted to the NRC for a formal evaluation and review.

- MRP-228 [14], "Inspection Standard for PWR Internals," provides guidance on the qualification/demonstration of the required NDE techniques and other criteria pertaining to the actual performance of the inspections.

The PWROG has also recently developed "Reactor Internals Acceptance Criteria Methodology and Data Requirements" for the MRP-227 inspections, where feasible [15]. Final reports are developed and available for industry use in support of planned license renewal inspection commitments. In some cases, individual plants will develop plant-specific acceptance criteria for some internals components if a generic approach is not practical.

As described in MRP-232 [16], the primary functions of the Palisades internals are to provide support, guidance, and protection to the reactor core, provide a passageway for the distribution of the reactor coolant flow to the reactor core, provide support, guidance, and protection for in-vessel/core instrumentation (ICI), and provide gamma and neutron shielding for the reactor vessel. The RVI consist of the following three main assemblies: an upper internals assembly (also known as an "upper guide structure" [UGS] at Palisades) that is removed during refueling as a single component to provide access to the fuel assemblies, a core support barrel, and a lower support structure. In addition, there are three other RVI assemblies in CE-designed plants: the control element assembly (CEA) shroud assembly, the core shroud assembly, and the in-core instrumentation support system. Since the design of Palisades is different from other CE-designed plants, some of the components have a different naming scheme. As mentioned, one of these is the "upper guide structure," which is similar to the "upper internals assembly" in other CE plants. Additionally, Palisades has a "control rod shroud assembly" instead of a "CEA shroud assembly" and an "in-core instrumentation guide system" rather than an "in-core instrumentation support system." These components have some differences in design, but they serve similar functions, and the MRP-227 augmented inspection requirements are still applicable. A brief summary of the design characteristics for these internals specific to Palisades is provided in the following subsection. Note that the component names at Palisades are used rather than the generic names for the other CE-designed plants. The general arrangement of the CE-designed PWR internals components as well as the specific configuration of Palisades is shown in Figure A-1.

Upper Guide Structure

The UGS is located above the reactor core within the core support barrel assembly. It is removed during refueling as a single component in order to provide access to the fuel assemblies. The UGS is an integral assembly that includes the upper fuel assembly alignment plate, the control rod shroud assemblies, the in-core instrumentation guide systems, and the hold-down assembly. At Palisades, the hold-down ring present in other CE designs has been replaced by a system of Belleville springs. The functions of the UGS are to provide alignment and support to the fuel assemblies, to prevent movement of the fuel assemblies in the case of a severe accident condition, and to protect the control rods from cross-flow effects in the upper plenum. The flange on the upper end of the UGS rests on the core support barrel.

Core Support Barrel

The core support barrel assembly consists of the core support barrel cylinders, nozzles, flange, alignment keys, and snubbers.

The core support barrel is a cylinder which contains the core and other internals components. Its function is to resist static loads from the fuel assemblies and other internals components and dynamic loads from normal operating hydraulic flow, seismic events, and loss-of-coolant accident (LOCA) events. The core support barrel supports the lower support structure, and the core support plate rests on top of the lower support structure. The core support barrel upper flange is a thick ring that suspends the core support barrel from a ledge on the reactor vessel.

Lower Support Structure

The lower support structure consists of the core support plate and the core support columns and beams. The core support plate positions and supports the reactor core and directs the reactor coolant flow into each fuel assembly. The weight of the core is transmitted to the core support barrel through the lower support structure via the core support columns. Fuel alignment holes in the core support plate engage lower fuel assembly alignment pins to provide guidance for and limit lateral movement of the individual fuel assemblies.

Core Shroud Assembly

The core shroud assembly is located within the core support barrel and directly below the UGS. Palisades has a bolted core shroud assembly where the core shroud plates are fastened to the former plates with structural bolts. The former plates, and thus the assembly, are attached to the core support barrel by structural bolts as well. The core shroud assembly provides a boundary between the reactor coolant bypass flow on the inside of the core support barrel and the reactor coolant flow through the fuel assemblies. It also limits the amount of coolant bypass flow and reduces the lateral motion of the fuel assemblies.

Control Rod Shroud Assemblies

The control rod shroud assemblies consist of the control rod shrouds, the control rod shroud bolts, and the extension shaft guide tubes. The control rod shrouds protect the cruciform control rods from cross-flow effects in the upper plenum. The lower end of the shrouds is bolted to the upper fuel assembly alignment plate. The extension shaft guides also protect the control rods from cross-flow effects in the upper plenum and provide lateral support and alignment of the control rod extension shafts during refueling operations. The extension shaft guides are permanently mounted to the inside of the reactor vessel closure head. The control rod drive mechanisms are positioned on the reactor vessel closure head and are coupled to the control rods by the control rod drive extension shafts. The control rod shroud assemblies were considered in the development of this AMP. The control rod drive mechanisms, extension shafts, and control rods were not included in the list of internals components.

In-Core Instrumentation Support System

The ICI support system consists of in-core instrumentation guide tubes and components that provide support to the in-core instrumentation. For plants with top-entry in-core instrumentation assemblies, such as Palisades, the in-core instrumentation is inserted through the reactor vessel head via a nozzle and into a guide tube. The guide tubes interface with the upper fuel alignment plate to align with holes in the fuel assemblies. ICI elements are inserted from the guide tubes directly into the fuel assemblies.

Palisades License Extension

Palisades was granted a license for extended operation by the NRC through the issuance of a SER in NUREG-1871 [2]. In the SER, the NRC concluded that the Palisades License Renewal Application (LRA) adequately identified the RVI system structures and components that are subject to an Aging Management Review (AMR), as required by 10 CFR 54.21(a)(1) [6]. A listing of the Palisades RVI components and subcomponents already reviewed by the NRC in the LRA, and that are subject to AMP requirements, is included in Table B-1.

In accordance with 10 CFR Part 54 [6], frequently referred to as the License Renewal Rule, Palisades has developed a procedure to direct the performance of AMRs of mechanical structures and components [17]. The U.S. industry, through the efforts of the MRP and PWROG, has further investigated the components and subcomponents that require aging management to support continued reliable function. MRP-227 contains a NEI 03-08 [18] "Mandatory" requirement that each plant will be required to use MRP-227 and MRP-228 to develop and implement an AMP for reactor internals no later than three years after the initial industry issuance of MRP-227. MRP-227 was issued in December 2008. Therefore, plant AMPs must be completed by December 2011 or sooner, as required by plant-specific license renewal commitments.

The information contained in this AMP fully complies with the requirements and guidance of the referenced documents. The AMP will manage aging effects of the RVI so that the intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

3 PROGRAM OWNER

The Reactor Vessel Internals Program [8] is part of the Palisades Nuclear Plant aging management programs. The successful implementation and comprehensive long-term management of the Palisades RVI AMP will require the integration of Entergy organizations, corporately and at Palisades, and interaction with multiple industry organizations including, but not limited to, the ASME, MRP, NRC, and PWROG. The responsibilities of the individual Entergy corporate and Palisades groups are provided in applicable plant procedures. The Engineering Programs department has overall responsibility for maintaining and implementing the RVI inspection program and the RVI AMP [8]. Entergy will maintain cognizance of industry activities related to PWR internals inspection and aging management and will address/implement industry guidance stemming from those activities, as appropriate under NEI 03-08 practices [5].

4 DESCRIPTION OF THE PALISADES REACTOR INTERNALS AGING MANAGEMENT PROGRAMS AND INDUSTRY PROGRAMS

The U.S. nuclear industry, through the combined efforts of utilities, vendors, and independent consultants, has defined a generic guideline to assist utilities in developing reactor internals plant-specific aging management programs based on inspection and evaluation. As noted in Section 3, the PWR Vessel Internals Program is a part of the Palisades Nuclear Plant aging management programs. The intent of this program is to ensure the long-term integrity and safe operation of the reactor internals components. Palisades has developed this AMP in conformance with the 10 Generic Aging Lessons Learned (GALL Report) [19] attributes and MRP-227.

This reactor internals AMP utilizes a combination of prevention, mitigation, and condition monitoring. Where applicable, credit is taken for existing programs such as water chemistry [12, 20] and inspections prescribed by the ASME Section XI Inservice Inspection Program [3, 4]. These existing programs combined with the LRA commitment for augmented inspections or evaluations as recommended by MRP-227 reflect on Entergy's proactive approach for aging management.

Aging degradation mechanisms that impact internals have been identified and documented in a Palisades AMR [1] prepared using the corporate procedural guidance document [17] in support of the License Renewal effort. The overall outcome of the reviews and the additional work performed by the industry, as summarized in MRP-227, are to provide appropriate augmented inspections for reactor internals components to provide early detection of the degradation mechanisms of concern. Therefore, this AMP is consistent with the existing Palisades AMR methodology and the additional industry work summarized in MRP-227. All sources are consistent and address concerns about component degradation resulting from the following eight material aging degradation mechanisms identified as affecting reactor internals:

- <u>Stress Corrosion Cracking</u>

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

- <u>Irradiation-Assisted Stress Corrosion Cracking</u>

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

- <u>Wear</u>

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

- Fatigue

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

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

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

- Thermal Aging Embrittlement

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

- Irradiation Embrittlement

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

- <u>Void Swelling and Irradiation Growth</u>

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

- <u>Thermal and Irradiation-Enhanced Stress Relaxation or Irradiation-Enhanced Creep</u>

  The loss of preload aging effect can be caused by the aging mechanisms of stress relaxation or creep. Thermal stress relaxation (or primary creep) is defined as the unloading of preloaded components due to long-term exposure to elevated temperatures, as seen in PWR internals. Stress relaxation occurs under conditions of constant strain where part of the elastic strain is replaced with plastic strain. Available data show that thermal stress relaxation appears to reach saturation in a short time (< 100 hours) at PWR internals temperatures.

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

The Palisades RVI AMP is focused on meeting the requirements of the 10 elements of an AMP as described in NUREG-1801, GALL Report Section XI.M16 for PWR Vessel Internals. In the Palisades RVI AMP, this is demonstrated through application of existing Palisades AMR methodology that credits inspections prescribed by the ASME Code Section XI Inservice Inspection Program, existing Palisades programs, and additional augmented inspections based on MRP-227 recommendations. A description of the applicable existing Palisades programs and compliance with the elements of the GALL Report is contained in the following subsections.

4.1 EXISTING PALISADES PROGRAMS

Palisades's overall strategy for managing aging in reactor internals components is supported by the following existing programs:

- ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program
- Reactor Vessel Internals Inspection Program
- Water Chemistry Program

These are established programs that support the aging management of RCS components in addition to the RVI components. Although affiliated with and supporting the RVI AMP, these programs will continue to be managed under the existing structure.

Brief descriptions of the programs are included in the following subsections.

4.1.1 ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program

The ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program [3, 4] is an existing program that facilitates inspections to identify and correct degradation in Class 1, 2, and 3 piping, components, their supports and integral attachments. The program includes periodic visual, surface and/or volumetric examinations and leakage tests of all Class 1, 2 and 3 pressure-retaining components, their supports and integral attachments, including welds, pump casings, valve bodies, pressure-retaining bolting, piping/component supports, and reactor head closure studs. These are identified in ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," or commitments requiring augmented inservice inspections, and are within the scope of license renewal. This program is in accordance with 10 CFR 50.55a.

The evaluation of this program against the 10 attributes in the GALL Report for Programs XI.M1, XI.M3, and XI.S3 in support of the Palisades LRA remains applicable.

4.1.2 Reactor Vessel Internals Inspection Program

The Reactor Vessel Internals Inspection Program is an existing program that manages the aging effects for reactor vessel internals. The program provides for (a) Inservice Inspection (ISI) in accordance with ASME Section XI requirements, including examinations performed during the 10-year ISI examination; (b) Participation in industry initiatives to evaluate the significance of void swelling; (c) Monitoring and control of reactor coolant water chemistry in accordance with the EPRI guidelines in TR-105714 (see Water Chemistry Program) to mitigate SCC or IASCC; and (d) Participation in industry initiatives that will generate additional data on aging mechanisms relevant to RVI and develop appropriate inspection techniques to permit detection and characterization of features of interest.

The evaluation of this program against the 10 attributes in the GALL Report for Program XI.M16 in support of the Palisades LRA remains applicable.

4.1.3 Water Chemistry Program

The Water Chemistry Program is an existing program that is credited for managing aging effects by controlling the environment to which internal surfaces of systems and components are exposed. Such effects include the following:

- Loss of material due to general, pitting, and crevice corrosion
- Cracking due to SCC
- Steam generator tube degradation caused by denting, intergranular attack (IGA), and outer diameter stress corrosion cracking (ODSCC)

The aging effects are minimized by controlling the chemical species that cause the underlying mechanisms that produce them. The program provides assurance that an elevated level of contaminants and, where applicable, oxygen does not exist in the systems and components covered by the program, thus minimizing the occurrences of aging effects, and maintaining each component's ability to perform the intended functions. This is done according to EPRI PWR Primary Water Chemistry Guidelines [12].

The evaluation of this program against the 10 attributes in the GALL Report for Program XI.M2 in support of the Palisades LRA remains applicable.

4.2 SUPPORTING PALISADES PROGRAMS AND AGING MANAGEMENT SUPPORTIVE PLANT ENHANCEMENTS

4.2.1 Reactor Internals Aging Management Review Process

A comprehensive review of aging management of reactor internals was performed according to the requirements of the License Renewal Rule [6] as directed by Palisades corporate procedure LRPG 4, "Mechanical Aging Management Review" [17]. LR-AMR-RVI [1] documents the results of the AMR performed in support of Palisades license renewal for reactor internals. The Palisades LRA was approved by the NRC in NUREG-1871 [2]. RVI components specifically noted as requiring aging management, as identified in the NUREG, are summarized in Appendix B, Table B-1 of this AMP.

The referenced documents supported the LRA as follows:

1. Identified applicable aging effects requiring management
2. Associated AMPs to manage those aging effects
3. Identified enhancements or modifications to existing programs, new AMPs, or any other actions required to support the conclusions reached in the calculation

AMRs were performed for each Palisades system that contained long-lived, passive components requiring AMR, in accordance with the screening process of Palisades procedure LR-SS-RVI, "Scoping/Screening

Report of Reactor Vessel Internals" [21]. This review is not repeated here, but the results are fully incorporated in the Palisades RVI AMR.

4.3 INDUSTRY PROGRAMS

4.3.1 CE NPSD-1216, Aging Management for Reactor Internals

The Combustion Engineering Owner's Group (CEOG) topical report CE NPSD-1216 [11] contains a technical evaluation of aging degradation mechanisms and aging effects for CE RVI components. The CEOG report provided guidance for CEOG member plant owners to manage effects of aging on RVI during the period of extended operation, using approved aging management methodologies to develop plant-specific AMPs.

The AMR for the Palisades internals, documented in [1], was completed in a manner consistent with the approach of CE NPSD-1216 [11]. Both the Palisades-specific AMR document and the generic CE document were completed in accordance with 10 CFR 54.

4.3.2 MRP-227, Reactor Internals Inspection and Evaluation Guidelines

MRP-227, as discussed in Section 2, was developed by a team of industry experts including utility representatives, NSSS vendors, independent consultants, and international committee representatives who reviewed available data and industry experience on materials aging. The objective of the group was to develop a consistent, systematic approach for identifying and prioritizing inspection and evaluation requirements for reactor internals. The following subsections briefly describe the industry process.

4.3.2.1 MRP-227 RVI Component Categorizations

MRP-227 used a screening and ranking process to aid in the identification of required inspections for specific RVI components. MRP-227 credited existing component inspections, when they were deemed adequate, as a result of detailed expert panel assessments conducted in conjunction with the development of the industry document. Through the elements of the process, the reactor internals for all currently licensed and operating PWR designs in the U.S. were evaluated in the MRP program and appropriate inspection, evaluation, and implementation requirements for reactor internals were defined.

Based on the completed evaluations, the RVI components are categorized within MRP-227 as "Primary" components, "Expansion" components, "Existing Programs" components, or "No Additional Measures" components, described as follows:

- Primary

  Those PWR internals that are highly susceptible to the effects of at least one of the eight aging mechanisms were placed in the Primary group. The aging management requirements that are needed to ensure functionality of Primary components are described in the I&E guidelines. The Primary group also includes components that have shown a degree of tolerance to a specific aging degradation effect, but for which no highly susceptible component exists or for which no highly susceptible component is accessible.

- Expansion

  Those PWR internals that are highly or moderately susceptible to the effects of at least one of the eight aging mechanisms, but for which functionality assessment has shown a degree of tolerance to those effects, were placed in the Expansion group. The schedule for implementation of aging management requirements for Expansion components depends on the findings from the examinations of the Primary components at individual plants.

- Existing Programs

  Those PWR internals that are susceptible to the effects of at least one of the eight aging mechanisms and for which generic and plant-specific existing AMP elements are capable of managing those effects, were placed in the Existing Programs group.

- No Additional Measures Programs

  Those PWR internals for which the effects of all eight aging mechanisms are below the screening criteria were placed in the No Additional Measures group. Additional components were placed in the No Additional Measures group as a result of a failure mode, effects, and criticality analysis (FMECA) and the functionality assessment. No further action is required by these guidelines for managing the aging of the No Additional Measures components.

The categorization and analysis used in the development of MRP-227 are not intended to supersede any ASME B&PV Code Section XI requirements. Any components that are classified as core support structures, as defined in ASME B&PV Code Section XI IWB 2500, Category B-N-3, have requirements that remain in effect and may only be altered as allowed by 10 CFR 50.55a.

4.3.2.2 NEI 03-08 Guidance Within MRP-227

The industry program requirements of MRP-227 are classified in accordance with the requirements of the NEI 03-08 protocols. The MRP-227 guideline includes Mandatory, Needed, and Good Practice elements as follows:

- Mandatory

  There is one Mandatory element:

  > *Each commercial U.S. PWR unit shall develop and document a PWR reactor internals aging management program within 36 months following issuance of MRP-227, Rev. 0.*

  Palisades Applicability: MRP-227 was officially issued by the industry in December 2008. An AMP must therefore be developed by December 2011. Entergy is actively developing this AMP for Palisades to meet the commitment contained in MRP-227.

- Needed

There are three Needed elements:

1. *Each commercial U.S. PWR unit shall implement MRP-227, Tables 4-1 through 4-9 and Tables 5-1 through 5-3 for the applicable design within 24 months following issuance of MRP-227-A.*

   ...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.

   Palisades Applicability: MRP-227 augmented inspections will be incorporated in the Palisades ISI for the license renewal period. The applicable CE tables contained in MRP-227 are Table 4.2 (Primary), Table 4.5 (Expansion), and Table 4.8 (Existing) and are attached herein as Appendix C Tables C-1, C-2, and C-3, respectively.

   Since license renewal commitments of Palisades require the AMP submittal prior to the issuance of MRP-227-A, this AMP has been developed in accordance with MRP-227, Revision 0.

2. *Examinations specified in the MRP-227 guidelines shall be conducted in accordance with Inspection Standard MRP-228.*

   Palisades Applicability: Inspection standards will be in accordance with the requirements of MRP-228 [14]. These inspection standards will be used for augmented inspection at Palisades as applicable where required by MRP-227 directives.

3. *Examination results that do not meet the examination acceptance criteria defined in Section 5 of the MRP-227 guidelines shall be recorded and entered in the plant corrective action program and dispositioned.*

   Palisades Applicability: Palisades will comply with this requirement.

- Good Practice

There is one Good Practice element:

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

Palisades Applicability: As discussed in Section 4, Entergy will participate in future industry efforts and will adhere to industry directives for reporting, response, and follow-up.

4.3.2.3 MRP-227 AMP Development Guidance

It should be noted that MRP-227, Appendix A also includes a description of the attributes that make up an acceptable AMP. These attributes are similar to the previously discussed attributes of the GALL Report and are consistent with the Palisades AMR process. Evaluation of the Palisades RVI AMP against GALL Report attribute elements is provided in Section 5 of this program plan.

As part of License Renewal, Palisades agreed to participate in industry activities associated with the development of the standard Industry Guideline for Inspection and Evaluation of Reactor Internals. The industry efforts have defined the required inspections and examination techniques for those components critical to aging management of RVI. The results of the industry-recommended inspections, as published in MRP-227, serve as the basis for identifying any augmented inspections that are required to complete the Palisades RVI AMP.

The MRP-227 guideline has been submitted to the NRC with the ultimate goal of producing an SER and approval. Discussions between utilities and the NRC, however, have indicated that the utility program cannot be based solely on the MRP-227 work prior to the issuance of the SER. Therefore, the Palisades RVI AMP has been rooted in meeting the GALL Report attributes and the previous work of the CEOG in CE NPSD-1216. However, the Palisades RVI AMP also captures the results of additional evaluations, inspection recommendations, and forward-looking concepts of the MRP-227 work summarized in the tables of Appendix C.

4.3.2.4 MRP-227 Applicability to Palisades

The applicability of MRP-227 to Palisades requires compliance with the following MRP-227 assumptions:

- *Operation of 30 years or less with high-leakage core loading patterns (fresh fuel assemblies loaded in peripheral locations) followed by implementation of a low-leakage fuel management strategy for the remaining 30 years of operation.*

  Palisades Applicability: Palisades fuel management program changed from a high- to a low-leakage core loading pattern prior to 30 years of operation.

- *Base load operation, i.e., typically operates at fixed power levels and does not usually vary power on a calendar or load demand schedule.*

  Palisades Applicability: Palisades operates as a base load unit.

- *No design changes beyond those identified in general industry guidance or recommended by the original vendors.*

  Palisades Applicability: MRP-227 states that the recommendations are applicable to all U.S. PWR operating plants as of May 2007 for the three designs considered. Palisades has not made any modifications to reactor internals components since May 2007.

Based on the Palisades applicability, as stated, the MRP-227 work is representative for Palisades.

4.3.3 Ongoing Industry Programs

The U.S. industry, through both the EPRI/MRP and the PWROG, continues to sponsor activities related to RVI aging management, including planned development of a standard NRC submittal template, development of a plant-specific implementation program template for currently licensed U.S. PWR plants, and development of acceptance criteria and inspection disposition processes. Entergy will maintain cognizance of industry activities related to PWR internals inspection and aging management and will address/implement industry guidance stemming from those activities, as appropriate under NEI 03-08 practices.

4.4 SUMMARY

It should be noted that the Entergy, Palisades, MRP, and PWROG approaches to aging management are based on the GALL Report approach to aging management strategies. This approach includes a determination of which reactor internals passive components are most susceptible to the aging mechanisms of concern followed by determination of the proper inspection or mitigating program to provide reasonable assurance that the component will continue to perform its intended function through the period of extended operation. The GALL Report-based approach was used at Palisades for the initial basis of the LRA that resulted in the NRC SER in NUREG-1871 [2].

The approach used to develop Palisades AMPs is fully compliant with regulatory directives and approved documents. The additional evaluations and analysis completed by the MRP industry group have provided clarification on the level of inspection quality needed to determine the proper examination method and frequencies. The tables provided in MRP-227 and included as Appendix C provide the level of examination required for each of the components evaluated.

It is the Palisades position that use of the AMR produced by the LRA methodology, combined with any additional augmented inspections required by the MRP-227 industry tables provided in Appendix C, provides reasonable assurance that the reactor internals passive components will continue to perform their intended functions through the period of extended operation.

5 PALISADES REACTOR INTERNALS AGING MANAGEMENT PROGRAM ATTRIBUTES

The Palisades RVI AMP is credited for aging management of RVI components for the following eight aging degradation mechanisms and their associated effects:

- Stress corrosion cracking
- Irradiation-assisted stress corrosion cracking
- Wear
- Fatigue
- Thermal aging embrittlement
- Irradiation embrittlement
- Void swelling and irradiation growth
- Thermal and irradiation-enhanced stress relaxation or irradiation-enhanced creep

The attributes of the Palisades Reactor Internals AMP and compliance with NUREG-1801 (GALL Report), Section XI.M16, "PWR Vessel Internals" [19] are described in this section. The GALL Report identifies 10 attributes for successful component aging management. The framework for assessing the effectiveness of the projected program is established by the use of the 10 elements of the GALL Report.

Palisades fully utilized the GALL Report process contained in NUREG-1801 [19] in performing the AMR of the reactor internals in the license renewal process. Palisades has committed to: (1) participate in industry initiatives that will generate additional data on aging mechanisms relevant to reactor internals; (2) revise the Reactor Vessel Internals Program as applicable to incorporate industry recommendations for augmented inspections and techniques resulting from the industry initiatives; and (3) submit a revised Reactor Vessel Internals Program to the NRC for review and approval.

This AMP is consistent with the NUREG-1801 process and includes consideration of the augmented inspections identified in MRP-227. It fully meets the requirements of the Palisades commitments. Specific details of the Palisades Reactor Internals AMP are summarized in the following subsections.

5.1 NUREG-1801/SRP PROGRAM ELEMENT 1: SCOPE OF PROGRAM

GALL Report AMP Element Description

"The program is focused on managing the effects of crack initiation and growth due to stress corrosion cracking (SCC) or irradiation assisted stress corrosion cracking (IASCC), and loss of fracture toughness due to neutron irradiation embrittlement or void swelling. The program contains preventive measures to mitigate SCC or IASCC; ISI to monitor the effects of cracking on the intended function of the components; and repair and/or replacement as needed to maintain the ability to perform the intended function. Loss of fracture toughness is of consequence only if cracks exist. Cracking is expected to initiate at the surface and is detectable by augmented inspection. The program provides guidelines to assure safety function integrity of the subject safety related reactor pressure vessel internal components, both non-bolted and bolted components.

The program consists of the following elements: (a) identify the most susceptible or limiting items, (b) develop appropriate inspection techniques to permit detection and characterizing of the feature (cracks) of interest and demonstrate the effectiveness of the proposed technique, and (c) implement the inspection during the license renewal term. For example, appropriate inspection techniques may include enhancing visual VT-1 examinations for non-bolted components and demonstrated acceptable inspection methods for bolted components." [19]

Palisades Program Scope

The Palisades RVI consist of three basic assemblies: (1) an upper guide structure, (2) a core support barrel assembly, and (3) a lower support structure. Additional RVI details are provided in Section 2 of this WCAP and in applicable sections of the Palisades LRA [9].

The Palisades RVI subcomponents that required an AMR are indicated in Table 3.1.2-3 in the Palisades LRA [9]. The portion of this table associated with the internals is included as part of the tables in Appendix B. The tables in the LRA list the subcomponents of the RVI that required an AMR along with each subcomponent passive function(s) and reference(s) to the corresponding AMR table(s) in the Palisades LRA.

The Palisades Reactor Internals AMR was conducted and documented in the Palisades AMR [1]. The table summarizing the results of that review is also included in the tables of Appendix B. The tables identify those aging effects that require management for those components requiring AMR. A column in the tables lists the program/activity that is credited to address the component and aging effect during the period of extended operation. Palisades submitted the information to the NRC for review [9] and received approval in NUREG-1871 [2].

The results of the industry research provided by MRP-227, summarized in the tables of Appendix C, provide the basis for the required augmented inspections, inspection techniques to permit detection and characterizing of the aging effects (cracks, loss of material, loss of preload, etc.) of interest, prescribed frequency of inspection, and examination acceptance criteria. The Palisades RVI AMP scope is based on previously established GALL Report approaches through application of supporting methodologies to determine those components that require aging management. Likewise, the additional information provided in the industry document MRP-227 (results of Appendix C) is rooted in the GALL Report methodology and provides a basis for augmented inspections that were required to complete this Palisades RVI AMP by providing the inspection method, frequency of inspection, and examination acceptance criteria.

Conclusion

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the Palisades SER.

5.2 NUREG-1801/SRP PROGRAM ELEMENT 2: PREVENTIVE ACTIONS

GALL Report AMP Element Description

> *"The requirements of ASME Section XI, Subsection IWB, provide guidance on detection, but do not provide guidance on methods to mitigate cracking. Maintaining high water purity reduces susceptibility to cracking due to SCC. Reactor coolant water chemistry is monitored and maintained in accordance with the EPRI guidelines in TR-1014986. The program description and evaluation and technical basis of monitoring and maintaining reactor water chemistry are presented in Chapter XI.M2, Water Chemistry."* [19]

Palisades Preventive Action

The Palisades reactor internals AMP includes the following existing program that complies with the requirements of this element. A description and applicability to the Palisades reactor internals AMP is provided in the following subsection.

Primary Water Chemistry Program

To mitigate aging effects on component surfaces that are exposed to water as process fluid, chemistry programs are used to control water chemistry for impurities (e.g., dissolved oxygen, chloride, fluoride, and sulfate) that accelerate corrosion. This program relies on monitoring and control of water chemistry to keep peak levels of various contaminants below the system-specific limits. The Palisades PWR Primary Water Chemistry Program [20, 9] is based on the current, approved revisions of EPRI PWR Primary Water Chemistry Guidelines.

This program is consistent with the corresponding program described in the GALL Report [22].

The limits of known detrimental contaminants imposed by the chemistry monitoring program are consistent with the EPRI PWR Primary Water Chemistry Guidelines [12].

Conclusion

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the Palisades SER.

5.3 NUREG-1801/SRP PROGRAM ELEMENT 3: PARAMETERS MONITORED OR INSPECTED

GALL Report AMP Element Description

"The program monitors the effects of cracking on the intended function of the component by detection and sizing of cracks by augmentation of the inservice inspection requirements in accordance with the requirements of the ASME Code Section XI, Table IWB 2500-1." [19]

Palisades Parameters Monitored or Inspected

The Palisades AMP monitors, inspects, and/or tests for the effects of the eight aging degradation mechanisms on the intended function of the Palisades PWR internals components through inspection and condition monitoring activities in accordance with the augmented requirements defined under industry directives as contained in MRP-227 and in accordance with ASME Section XI.

For license renewal, the ASME Section XI Program consists of periodic volumetric, surface, and/or visual examination of components for assessment, signs of degradation, and corrective actions. This program is consistent with the corresponding program described in the GALL Report [22].

Appendices B and C of this AMP provide a detailed listing of the components and subcomponents and the parameters monitored, inspected, and/or tested.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the Palisades SER.

5.4 NUREG-1801/SRP PROGRAM ELEMENT 4: DETECTION OF AGING EFFECTS

GALL Report AMP Element Description

"The extent and schedule of the inspection and test techniques prescribed by the aging management program are designed to maintain structural integrity and ensure that aging effects will be discovered and repaired before the loss of intended function. Inspections can reveal crack initiation and growth. Vessel internal components that are inspected in accordance with the requirements of ASME Section XI, Subsection IWB examination category B-N-3 for all accessible surfaces of reactor core support structure that can be removed from the vessel. The ASME Section XI inspection specifies visual VT-3 examination to determine the general mechanical and structural condition of the component supports by (a) verifying parameters such as clearances, settings, and physical displacements, and (b) detecting discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion.

However, visual VT-3 examination is to be augmented to detect tight or fine cracks. Also, historically the VT-3 examination have not identified bolt cracking because cracking occurs at the juncture of the bolt head and shank, which is not accessible for visual inspection. Creviced and other inaccessible regions are difficult to inspect visually. This AMP recommends more stringent inspections such as enhanced visual VT-1 examinations or ultrasonic methods of volumetric inspection, for certain selected components and locations.

The inspection technique is capable of detecting the critical flaw size with adequate margin. The critical flaw size is determined based on the service loading condition and service-degraded material properties. For non-bolted components, augmented ISI may include enhancement of the visual VT-1 examination of Section XI IWA-2210. A description of such an enhanced visual VT-1 examination should include the ability to achieve a 0.0005-in. resolution, with the conditions (e.g., lighting and surface cleanliness) of the inservice examination bounded by those used to demonstrate the resolution of the inspection technique. For bolted components, augmented ISI is to include other demonstrated acceptable inspection methods to detect cracks between the bolt head and the shank. Alternatively, the applicant may perform a component-specific evaluation, including a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. If the loading is compressive or low enough (<5 ksi) to preclude fracture, then supplemental inspection of the component is not required. Failure to meet this criterion requires continued use of the augmented inspection methods." [19]

Palisades Detection of Aging Effects

Detection of indications that are required by the ASME Code Section XI ISI Program is well established and field-proven through the application of the Section XI ISI Program. Those augmented inspections that are taken from the MRP-227 recommendations will be applied through use of the MRP-228 Inspection Standard.

Inspection can be used to detect physical effects of degradation including cracking, fracture, wear, and distortion. The choice of an inspection technique depends on the nature and extent of the expected damage. The recommendations supporting aging management for the reactor internals, as contained in this report, are built around three basic inspection techniques: (1) visual, (2) ultrasonic, and (3) physical measurement. Three different visual techniques are VT-3, VT-1, and EVT-1. The assumptions and process used to select the appropriate inspection technique are described in the following subsections. Inspection standards developed by the industry for the application of these techniques for augmented reactor internals inspections are documented in MRP-228.

VT-1 Visual Examinations

The acceptance criteria for visual examinations conducted under categories B-N-2 (welded core support structures and interior attachments to reactor vessels) and B-N-3 (removable core support structures) are defined in IWB-3520 [23]. VT-1 visual examination is intended to identify crack-like surface flaws. Unacceptable conditions for a VT-1 examination are:

- Crack-like surface flaws on the welds joining the attachment to the vessel wall that exceed the allowable linear flaw standards of IWB-3510
- Structural degradation of attachment welds such that the original cross-sectional area is reduced by more than 10 percent

These requirements are defined to ensure the integrity of attachment welds on the ferritic pressure vessel. Although the IWB-3520 criteria do not directly apply to austenitic stainless steel internals, the clear intent is to ensure that the structure will meet minimum flaw tolerance fracture requirements. In the MRP-227 recommendations, VT-1 examinations have been identified for components requiring close visual examinations with some estimate of the scale of deformation or wear. In MRP-227, note that 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. Therefore, no additional VT-1 inspections over and above those required by ASME Section XI ISI have been specified.

EVT-1 Enhanced Visual Examination for the Detection of Surface Breaking Flaws

In the augmented inspections detailed in MRP-227 for reactor internals, the EVT-1 enhanced visual examination has been identified for inspection of components where surface-breaking flaws are a potential concern. Any visual inspection for cracking requires a reasonable expectation that the flaw length and crack mouth opening displacement meet the resolution requirements of the observation technique. The EVT-1 specification augments the VT-1 requirements to provide more rigorous inspection standards for SCC and has been demonstrated for similar inspections in boiling water reactor (BWR) internals. Enhanced visual examination (i.e., EVT-1) is also conducted in accordance with the requirements described for visual examination (i.e., VT-1) with additional requirements (such as camera scanning speed) currently being developed by the industry. Any recommendation for EVT-1 inspection will require additional analysis to establish flaw-tolerance criteria. The industry is currently developing a consensus approach for acceptance criteria methodologies to support plant-specific augmented examinations. Entergy has been an active participant in these initiatives and will follow the industry directive. These acceptance criteria methodologies may be determined either generically or on a plant-specific basis because both loads and component dimensions may vary from plant to plant within a typical PWR design.

VT-3 Examination for General Condition Monitoring

In the augmented inspections detailed in the MRP-227 for reactor internals, the VT-3 visual examination has been identified for inspection of components where general condition monitoring is required. The VT-3 examination is intended to identify individual components with significant levels of existing degradation. As the VT-3 examination is not intended to detect the early stages of component cracking or other incipient degradation effects, it should not be used when failure of an individual component could threaten either plant safety or operational stability. The VT-3 examination may be appropriate for inspecting highly redundant components, where a single failure does not compromise the function or integrity of the critical assembly.

The acceptance criteria for visual examinations conducted under categories B-N-2 (welded core support structures and interior attachments to reactor vessels) and B-N-3 (removable core support structures) are defined in IWB-3520. These criteria are designed to provide general guidelines. The unacceptable conditions for a VT-3 examination are:

- Structural distortion or displacement of parts to the extent that component function may be impaired
- Loose, missing, cracked, or fractured parts, bolting, or fasteners
- Foreign materials or accumulation of corrosion products that could interfere with control rod motion or could result in blockage of coolant flow through fuel
- Corrosion or erosion that reduces the nominal section thickness by more than 5 percent
- Wear of mating surfaces that may lead to loss of function
- Structural degradation of interior attachments such that the original cross-sectional area is reduced more than 5 percent

The VT-3 examination is intended for use in situations where the degradation is readily observable. It is meant to provide an indication of condition, and quantitative acceptance criteria are not generally required. In any particular recommendation for VT-3 visual examination, it should be possible to identify the specific conditions of concern. For instance, the unacceptable conditions for wear indicate wear that might lead to loss of function. Guidelines for wear in a critical-alignment component may be very different from the guidelines for wear in a large structural component.

Ultrasonic Testing

Volumetric examinations in the form of ultrasonic testing (UT) techniques can be used to identify and determine the length and depth of a crack in a component. Although access to the surface of the component is required to apply the ultrasonic signals, the flaw may exist in the bulk of the material. In this proposed strategy, UT inspections have been recommended exclusively for detection of flaws in bolts. For the bolt inspections, any bolt with a detected flaw should be assumed to have failed. The size of the flaw in the bolt is not critical because crack growth rates are generally high, and it is assumed that the observed flaw will result in failure prior to the next inspection opportunity. It has generally been observed through examination performance demonstrations that UT can reliably (90 percent or greater reliability) detect flaws that reduce the cross-sectional area of a bolt by 35 percent.

Failure of a single bolt does not compromise the function of the entire assembly. Bolting systems in the reactor internals are highly redundant. For any system of bolts, it is possible to demonstrate multiple minimum acceptable bolting patterns. The evaluation program must demonstrate that the remaining bolts meet the requirements for a minimum bolting pattern for continued operation. The evaluation procedures must also demonstrate that the pattern of remaining bolts contains sufficient margin such that continuation of the bolt failure rate will not result in failure of the system to meet the requirements for minimum acceptable bolting pattern before the next inspection.

Establishment of the minimum acceptable bolting pattern for any system of bolts requires analysis to demonstrate that the system will maintain reliability and integrity in continuing to perform the intended function of the component. This analysis is highly plant-specific, and it is recommended that, prior to UT inspection of bolts, a minimum acceptable bolting pattern be established to support continued operation.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the Palisades SER.

5.5 NUREG-1801/SRP PROGRAM ELEMENT 5: MONITORING AND TRENDING

GALL Report AMP Element Description

> *"Inspection schedules in accordance with IWB-2400, assessment of susceptible or limiting components or locations, and reliable examination methods provide timely detection of cracks. The scope of examination expansion and reinspection beyond the baseline inspection are required if flaws are detected."* [19]

Palisades Monitoring and Trending

Operating experience with PWR reactor internals has been generally proactive. The extremely low frequency of failure in reactor internals makes monitoring and trending based on operating experience somewhat impractical. The majority of the materials aging degradation models used to develop the MRP-227 Guidelines are based on test data from reactor internals components removed from service. The data are used to identify trends in materials degradation and forecast potential component degradation. The industry continues to share both material test data and operating experience through the auspices of the MRP and PWROG. Entergy has in the past and will continue to maintain cognizance of industry activities and shared information related to PWR internals inspection and aging management as demonstrated in their Corrective Action Program [24].

Inspections credited in Appendix B are based on utilizing the Palisades 10-year ISI program and the augmented inspections derived from the industry program contained in Appendix C. These inspections, where practical, are scheduled to be conducted in conjunction with typical 10-year ISI examinations.

Appendix C, Tables C-1, C-2, and C-3 identify the augmented primary and expansion inspection and monitoring recommendations, and the existing programs credited for inspection and aging management. As discussed in MRP-227, inspection of the "Primary" components provides reasonable assurance for demonstrating component current capacity to perform the intended functions.

Reporting requirements are included as part of the MRP-227 guidelines. Consistent reporting of inspection results across all PWR designs will enable the industry to monitor reactor internals degradation on an ongoing industry basis as the period of extended operation moves forward. Reporting of examination results will allow the industry to monitor and trend results and take appropriate preemptive action through update of the MRP guidelines.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the Palisades SER.

5.6 NUREG-1801/SRP PROGRAM ELEMENT 6: ACCEPTANCE CRITERIA

GALL Report AMP Element Description

> *"Any indication or relevant condition of degradation is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500."* [19]

Palisades Acceptance Criteria

Those recordable indications that are the result of inspections required by the existing Palisades ISI program scope are evaluated in accordance with the applicable requirements of the ASME Code through the existing Corrective Action Program [24].

Inspection acceptance and expansion criteria are provided in Appendix C, Table C-4. These criteria will be reviewed periodically as the industry continues to develop and refine the information and will be included in updates to Palisades procedures to enable the examiner to identify examination acceptance criteria considering state-of-the-art information and techniques.

Augmented inspections, as defined by the MRP-227 requirements, that result in recordable relevant conditions will be entered into the plant Corrective Action Program and addressed by appropriate actions that may include enhanced inspection, repair, replacement, mitigation actions, or analytical evaluations. The industry, through various cooperative efforts, is working to construct a consensus set of tools in line with accepted and proven methodologies to support this element, such as WCAP-17096 [15]. Additional analysis to establish Appendix C expansion component evaluation criteria is being performed through the efforts of the PWROG. Status is monitored through direct Entergy cognizance of industry (including PWROG) activities related to PWR internals inspection and aging management.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the Palisades SER.

5.7 NUREG-1801/SRP PROGRAM ELEMENT 7: CORRECTIVE ACTIONS

GALL Report AMP Element Description

> *"Repair and replacement procedures are equivalent to those requirements in ASME Section XI. Repair is in conformance with IWB-4000 and replacement occurs according to IWB-7000. As discussed in the regulatory documents integral with the GALL report, the staff considered the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions."* [19]

Palisades Corrective Action

Corrective actions are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B [6] and the Palisades quality assurance program, "Conduct of Nuclear Oversight" [25]. Controls are established to assure that conditions adverse to quality are identified and documented and that appropriate remedial action is taken. For significant conditions adverse to quality, necessary corrective action is promptly determined and recorded. Corrective action includes determining the cause and extent of the condition, and taking appropriate action to preclude similar problems in the future. The controls also assure that corrective action is implemented in a timely manner.

Corrective actions are implemented through the initiation of an Action Request in accordance with plant procedures [24]. Equipment deficiencies may be initially documented by a work order, but the corrective action process specifies that an Action Request also be initiated if required. This approach ensures that identified problems are corrected in a timely manner.

Conclusion

This element complies with the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the Palisades SER.

5.8 NUREG-1801/SRP PROGRAM ELEMENT 8: CONFIRMATION PROCESS

GALL Report AMP Element Description

> *"Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls."* [19]

Palisades Confirmation Process

Palisades has an established 10 CFR Part 50, Appendix B Program [6] that addresses the elements of corrective actions, confirmation process, and administrative controls. The Palisades Program includes non-safety-related structures, systems, and components. Quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the Palisades SER.

5.9 NUREG-1801/SRP PROGRAM ELEMENT 9: ADMINISTRATIVE CONTROLS

GALL Report AMP Element Description

See item 8 (Section 5.8).

Palisades Administrative Controls

See evaluation in Section 5.8.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the Palisades SER.

5.10 NUREG-1801/SRP PROGRAM ELEMENT 10: OPERATING EXPERIENCE

GALL Report AMP Element Description

> *"Because the ASME Code is a consensus document that has been widely used over a long period, it has been shown to be generally effective in managing aging effects in Class 1, 2, or 3 components and their integral attachments in light-water cooled power plants.*
>
> *In PWRs, stainless steel components have generally not been found to be affected by SCC because of low dissolved oxygen levels and control of primary water chemistry. However, the potential for SCC exists due to inadvertent introduction of contaminants into the primary coolant system from unacceptable levels of contaminants in the boric acid; introduction through the free surface of the spent fuel pool, which can be a natural collector of airborne contaminants (NRC IN 84-18); introduction of relatively high levels of oxygen during shutdown, or from aggressive chemistries that may develop in creviced regions. Cracking has occurred in SS baffle former bolts in a number of foreign plants (NRC IN 98-11) and has now been observed in plants in the United States."* [19]

Palisades Operating Experience

Extensive industry and Palisades operating experience has been reviewed during the development of the RVI AMP. The experience reviewed includes NRC Information Notices 84-18, "Stress Corrosion Cracking in PWR Systems" [26] and 98-11, "Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants" [27]. To date, no cracking has been discovered in bolting for CE-designed reactor vessel internals. In 1998, the CEOG performed an assessment of the cracking of the baffle former bolts reported in foreign PWRs, including the potential impact of the cracking on domestic CE plants. The CEOG report, NPSD-1098, "Evaluation of the Applicability of Baffle Bolt Cracking to Ft. Calhoun and Palisades Internals Bolts" [28], states that the most likely mechanism for the cracking of cold-worked 316 stainless steel baffle former bolts in foreign plants is IASCC [8].

Early plant operating experience related to hot functional testing and reactor internals is documented in plant historical records. Inspections performed as part of the 10-year ISI program have been conducted as designated by existing commitments and would be expected to discover overall general internals structure degradation. To date, very little degradation has been observed industry-wide.

Industry operating experience is routinely reviewed by Entergy engineers using Institute of Nuclear Power Operations (INPO) Operating Experience (OE), the Nuclear Network, and other information sources as directed under the applicable procedure [29], for the determination of additional actions and lessons learned. These insights, as applicable, can be incorporated in the plant systems quarterly health reports and further evaluated for incorporation in plant programs.

A review of industry and plant-specific experience with reactor vessel internals reveals that the U.S. industry, including Entergy and Palisades, has responded proactively to industry issues relative to reactor internals degradation. A key element of the MRP-227 Guideline is the reporting of age-related degradation of reactor vessel components. Entergy, through its participation in PWROG and EPRI-MRP activities, will continue to benefit from the reporting of inspection information and will share its own operating experience with the industry through those groups or INPO, as appropriate.

Conclusion

This element complies with or exceeds the corresponding aging management attribute in NUREG-1801, Section XI.M16 and Commitment 33 in the Palisades SER.

6 DEMONSTRATION

Palisades has demonstrated a long-term commitment to aging management of reactor internals. This AMP is based on an established history of programs to identify and monitor potential aging degradation in the reactor internals. Programs and activities undertaken in the course of fulfilling that commitment include:

- The examinations required by ASME Section XI for the Palisades RVI have been performed as required since plant operations commenced.
- As documented in Palisades operational procedures, Operating Experience Reports (OERs) are continuously reviewed by Palisades personnel for applicable issues that indicate a need for updated operating procedures or programs.
- Reviews of Nuclear Oversight Section (NOS) audit reports, NRC inspection reports, and INPO evaluations indicate no unacceptable issues related to reactor vessel internals inspections.
- The Primary Water Chemistry Program at Palisades has been effective in maintaining the levels of oxygen, halides, and sulfate sufficiently low to prevent SCC of the reactor vessel internals.
- Entergy has actively participated in past and ongoing EPRI and PWROG RVI activities. Entergy will continue to maintain cognizance of industry activities related to PWR internals inspection and aging management and will address/implement the industry guidance stemming from those activities as appropriate under NEI 03-08 practices.

This AMP fulfills the approved license renewal methodology requirement to identify the most susceptible components and to inspect those components using inspection techniques with the capability to detect the expected degradation mechanism indication or indications. Augmented inspections, derived from the information contained in the industry I&E Guidelines (MRP-227), have been utilized in this AMP to build on existing plant programs. This approach is expected to encourage detection of a degradation mechanism at its first appearance, which is consistent with the ASME Code approach to inspections. This approach provides reasonable assurance that the internals components will continue to perform their intended function through the period of extended operation.

Typically, MRP-227 augmented reactor internals examinations applicable to Palisades must be performed no later than two refueling outages from the beginning of the license renewal period. A schedule for performing reactor vessel internals inspections at Palisades is provided in Table 7-1. The augmented inspections needed for compliance with MRP-227 will be integrated into the Reactor Vessel Internals Inspection Program [8]. Integration of the required inspections will be tracked to completion, and according to the good practice element in MRP-227, Palisades will continue to participate in future industry efforts on reactor internals and will adhere to industry directives for reporting, response, and follow-up as experience is gained through inspection results. This feedback loop will enable updates based on actual inspection experience.

The augmented inspections described in this document, as summarized in Appendix C, combined with the ASME Code Section XI ISI program inspections, existing Palisades programs, and use of OERs, provide reasonable assurance that the reactor internals at Palisades will continue to perform their intended functions through the period of extended operation.

7 PROJECTED PROGRAM ENHANCEMENT AND IMPLEMENTATION SCHEDULE

The requirements of MRP-227 are based on an 18-month refueling cycle and consider both effective full-power years (EFPY) and cumulative operation. The information contained in Table 7-1 is based on this information and includes a description of the currently projected scope of inspections pertaining to the reactor internals AMP. Should a change occur in plant operational practices or should operating experience result in changes to the projections, appropriate updates will be made to affected plant documentation in accordance with approved procedures.

Table 7-1 Aging Management Program Enhancement and Inspection Implementation Summary

Refueling Outage	Project Month/Year	Estimated EFPY[1]	AMP-Related Scope	Inspection Method and Criteria	Comments
RO-21	Fall 2010	23.6	Not applicable	Not applicable	Original license expires in 2011.
RO-22	Spring 2012	24.9	ASME Code Section XI and MRP-227 augmented inspections for core shroud assembly, upper core support barrel flange weld, core support plate, fuel alignment plate, and instrument guide tubes	MRP-227 inspections in accordance with MRP-228 specifications and ASME Code Section XI	None
RO-23	Fall 2013	26.3	Not applicable	Not applicable	None
RO-24	Spring 2015	27.6	Not applicable	Not applicable	None
RO-25	Fall 2016	29.0	Not applicable	Not applicable	None
RO-26	Spring 2018	30.5	Not applicable	Not applicable	None
RO-27	Fall 2019	32.0	Not applicable	Not applicable	None
RO-28	Spring 2021	33.5	Not applicable	Not applicable	None

Table 7-1 (cont.) Aging Management Program Enhancement and Inspection Implementation Summary

Refueling Outage	Project Month/Year	Estimated EFPY	AMP-Related Scope	Inspection Method and Criteria	Comments
RO-29	Fall 2022	35.0	ASME Code Section XI and MRP-227 augmented inspections for core shroud assembly, upper core support barrel flange weld, core support plate, fuel alignment plate, instrument guide tubes, and core shroud bolts	MRP-227 inspections in accordance with MRP-228 specifications and ASME Code Section XI	None
RO-30	Spring 2024	36.5	Not applicable	Not applicable	None
RO-31	Fall 2025	38.0	Not applicable	Not applicable	None
RO-32	Spring 2027	39.5	Not applicable	Not applicable	None
RO-33	Fall 2028	41.0	Not applicable	Not applicable	None
RO-34	Spring 2030	42.5	Not applicable	Not applicable	None
RO-35	Fall 2031	44.0	Not applicable	Not applicable	License expires March 24, 2031

[1] EFPY values after RO-25 were estimated based on an assumption of 100% reliability.

8 IMPLEMENTING DOCUMENTS

As noted within this Palisades AMP document, the PWR Vessel Internals Program is a part of the Palisades Nuclear Plant aging management programs. The Palisades AMP also references the Reactor Vessel Internals Inspection Program, Water Chemistry Program and the ASME Code Section XI Inservice Inspection, Subsections IWB, IWC, IWD, and IWF Program. MRP-227 augmented examinations (Appendix C), recommended as a result of industry programs, will be included in the existing Reactor Vessel Internals Inspection Program [8].

Palisades documents associated with the existing Palisades programs and considered to be implementing documents of the PWR Vessel Internals Program are:

- Reactor Vessel Internals Inspection Program [8]
- Chemistry Program Implementation [20]
- Inservice Inspection Programs [3, 4]

The PWR Vessel Internals AMP relies on the Water Chemistry Program for maintaining high water purity to reduce susceptibility to cracking due to SCC. The Water Chemistry Program was evaluated [2, 22] and found to be consistent with the GALL Report with some exceptions related to augmented inspections expected to be defined through industry programs. Additional procedures may be updated or created as OE for augmented examinations is accumulated.

Based on this information, the updated AMP for Palisades RVI provides reasonable assurance that the aging effects will be managed such that the components within the scope of license renewal will continue to perform their intended functions consistent with the CLB for the period of extended operation.

9 REFERENCES

1. LR-AMR-RVI, Rev. 4, "Aging Management Review Reactor Vessel Internals Palisades Nuclear Plant License Renewal Project," October 20, 2005.
2. NUREG-1871, "Safety Evaluation Report Related to the License Renewal of Palisades Nuclear Plant," Docket 50-255, U.S. Nuclear Regulatory Commission, January 2007.
3. "Palisades Nuclear Plant Fourth 10-Year Interval Master Inservice Inspection Plan," Rev. 18.
4. EM-09-03, Rev. 16, "Inservice Inspection."
5. Materials Reliability Program: "PWR Internals Inspection and Evaluation Guidelines" (MRP-227-Rev. 0), Electric Power Research Institute, Palo Alto, CA, December 2008. 1016596.
6. Code of Federal Regulations, 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," U.S. Nuclear Regulatory Commission.
7. Palisades Letter, "Reactor Vessel Internals Program Submittal Commitment Date Change," Docket 50-255, U.S. Nuclear Regulatory Commission, March 23, 2009.
8. LR-AMPBD-23-VSLINTERNALS, Rev. 3, "Reactor Vessel Internals Inspection Program," November 16, 2006.
9. "Palisades Nuclear Plant Application for Renewed Operating License," March 2005.
10. NUREG-1800, U.S. NRC Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants (SRP-LR), U.S. Nuclear Regulatory Commission, July 2001.
11. CEOG Report CE NPSD-1216, Rev. 0, "Generic Aging Management Review Report for Reactor Vessel Internals," March 2001.
12. "Pressurized Water Reactor Primary Water Chemistry Guidelines," Volumes 1 and 2, Revision 6, Electric Power Research Institute, Palo Alto, CA: 2007, 1014986.
13. Materials Reliability Program: "Screening, Categorization and Ranking of Reactor Internals Components of Westinghouse and Combustion Engineering PWR Design (MRP-191)," Electric Power Research Institute, Palo Alto CA: 2006, 1013234.
14. Materials Reliability Program: "Inspection Standard for PWR Internals (MRP-228)," Electric Power Research Institute, Palo Alto, CA: 2009, 1016609.
15. Westinghouse Report WCAP-17096, Rev. 0, "Reactor Internals Acceptance Criteria Methodology and Data Requirements."
16. Materials Reliability Program: "Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals (MRP-232)," Electric Power Research Institute, Palo Alto, CA: 2008, 1016593.
17. LRPG 4, Rev. 3, "Mechanical Aging Management Review," Palisades Nuclear Plant License Renewal Project Guideline.
18. NEI 03-08, "Guidelines for the Management of Materials Issues," Nuclear Energy Institute, Washington, DC, March 2009 effective version.

19. NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," U.S. Nuclear Regulatory Commission, July 2001.

20. LR-AMPBD-26-CHEMISTRY, Rev. 3, "Water Chemistry Program," License Renewal Aging Management Program Basis Document.

21. LR-SS-RVI, Rev. 2, "Scoping/Screening Report of Reactor Vessel Internals."

22. Palisades Nuclear Plant FSAR; Sections: 1.9, 3.3.4, 4.3.3, 4.5.6 14.17.3, Table 1-3.

23. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2001 Edition, 2003 Addenda, American Society of Mechanical Engineers, New York, NY.

24. EN-LI-102, Rev. 13, "Corrective Action Process," December 2008.

25. EN-QV-100, Rev. 2, "Conduct of Nuclear Oversight."

26. U.S. Nuclear Regulatory Commission Information Notice 84-18, "Stress Corrosion Cracking in Pressurized Water Reactor Systems," March 7, 1984.

27. U.S. Nuclear Regulatory Commission Information Notice 98-11, "Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants," March 25, 1998.

28. CE NPSD-1098, Rev. 0, "Evaluation of the Applicability of Baffle Bolt Cracking to Ft. Calhoun and Palisades Internals Bolts," April 1998.

29. LRPG 12, Rev. 2, "LRP Operation Experience Database Development and Use Guideline," Palisades Nuclear Plant License Renewal Project Guideline.

APPENDIX A
ILLUSTRATIONS

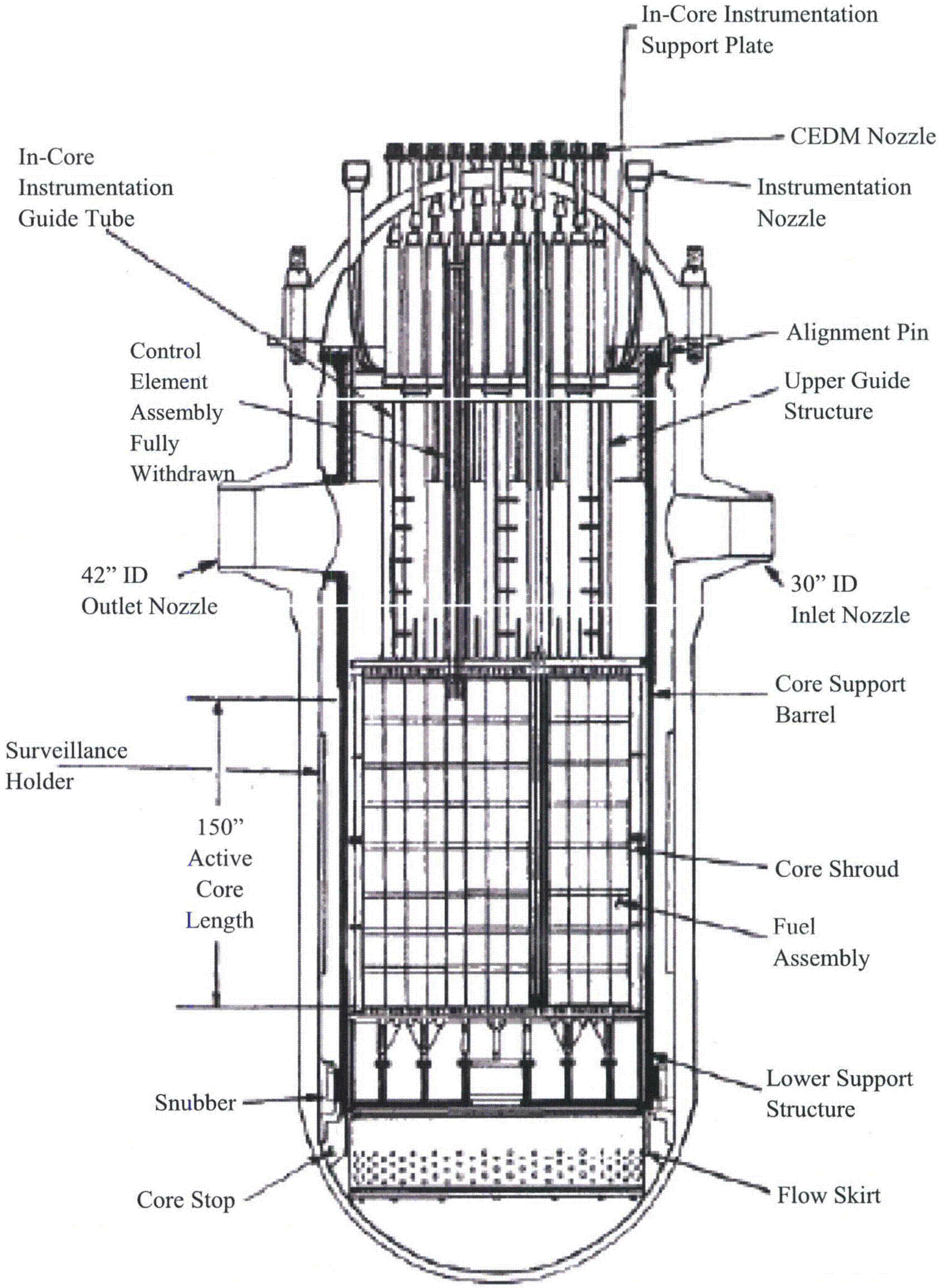


Figure A-1a Illustration of Typical CE Internals

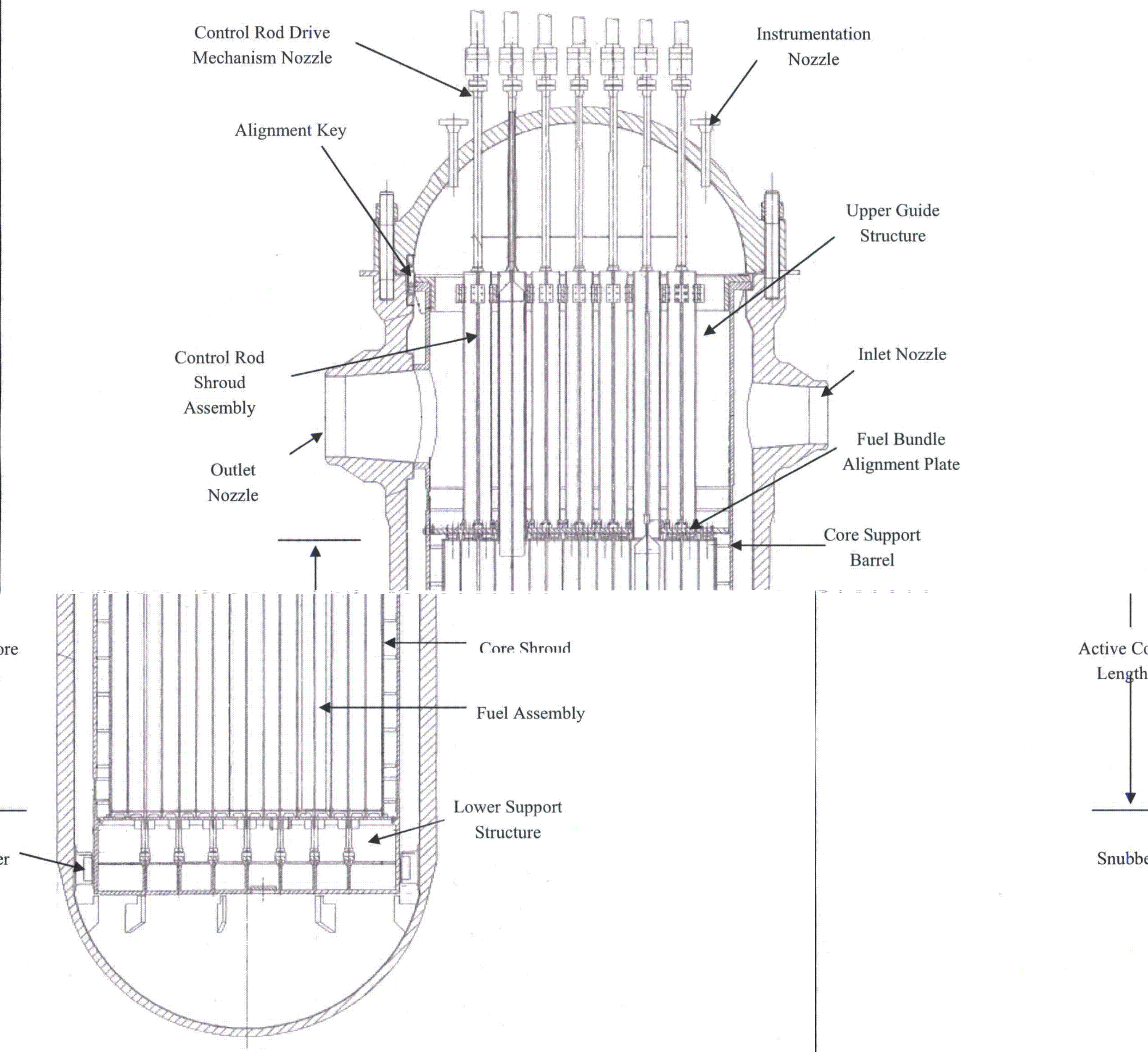


Figure A-1b Illustration of Palisades Internals

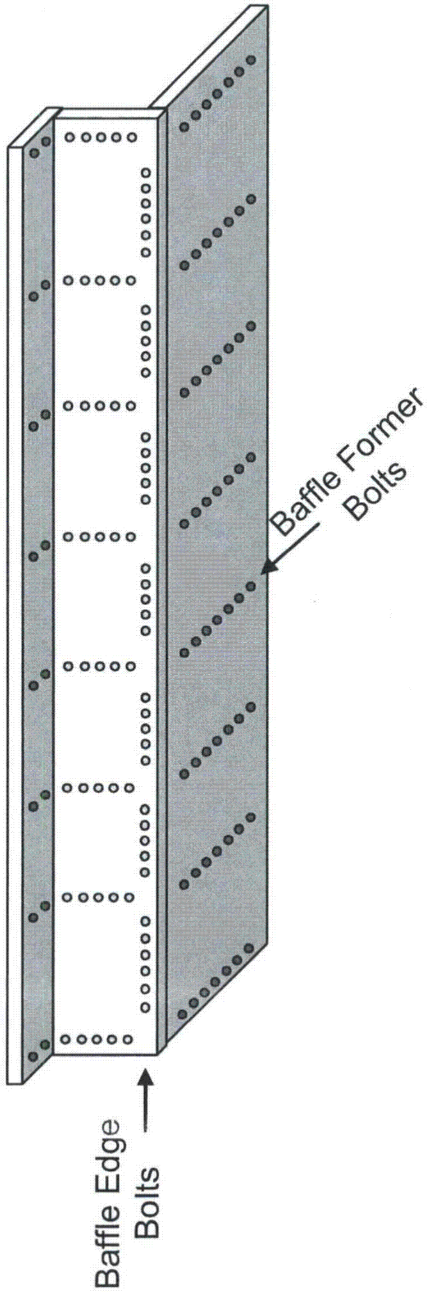


Figure A-2a Bolting Systems used in Typical Westinghouse Core Baffles
(Note: Palisades has a similar configuration in its core shroud where the baffle-former bolts are equivalent to core shroud bolts and there are no edge bolts.)

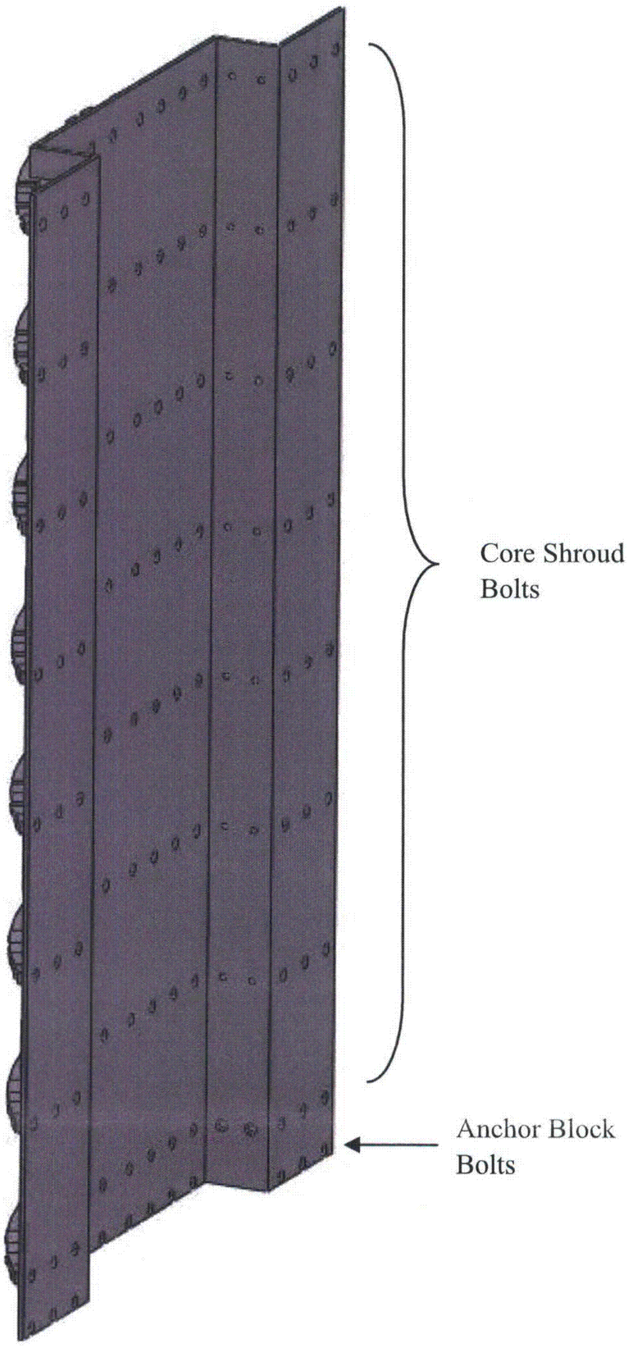


Figure A-2b Palisades Core Shroud Plate with Anchor Block Bolt Holes

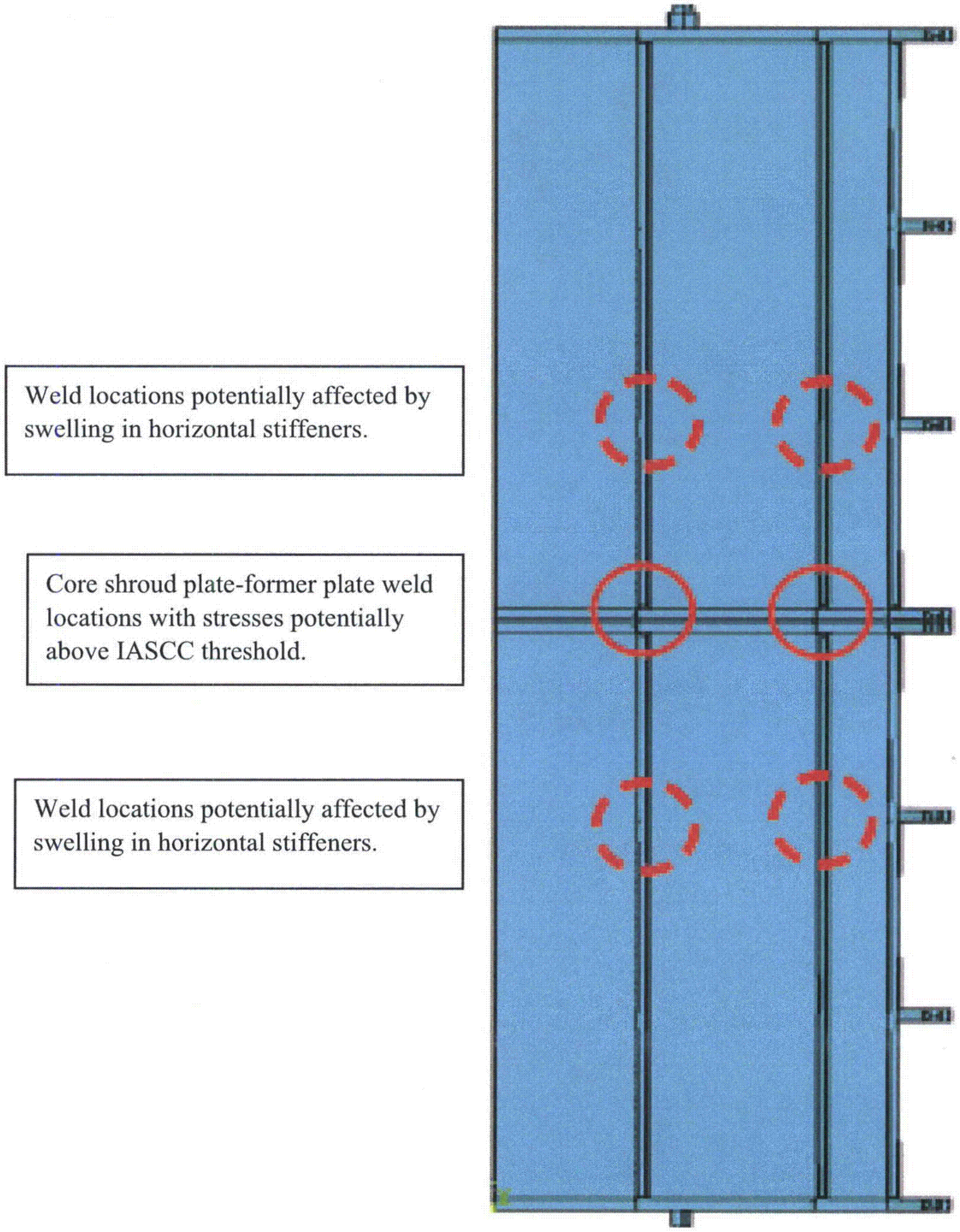


Figure A-3 Potential Crack Locations for CE Welded Core Shroud Assembled in Stacked Sections
(Not applicable for Palisades)

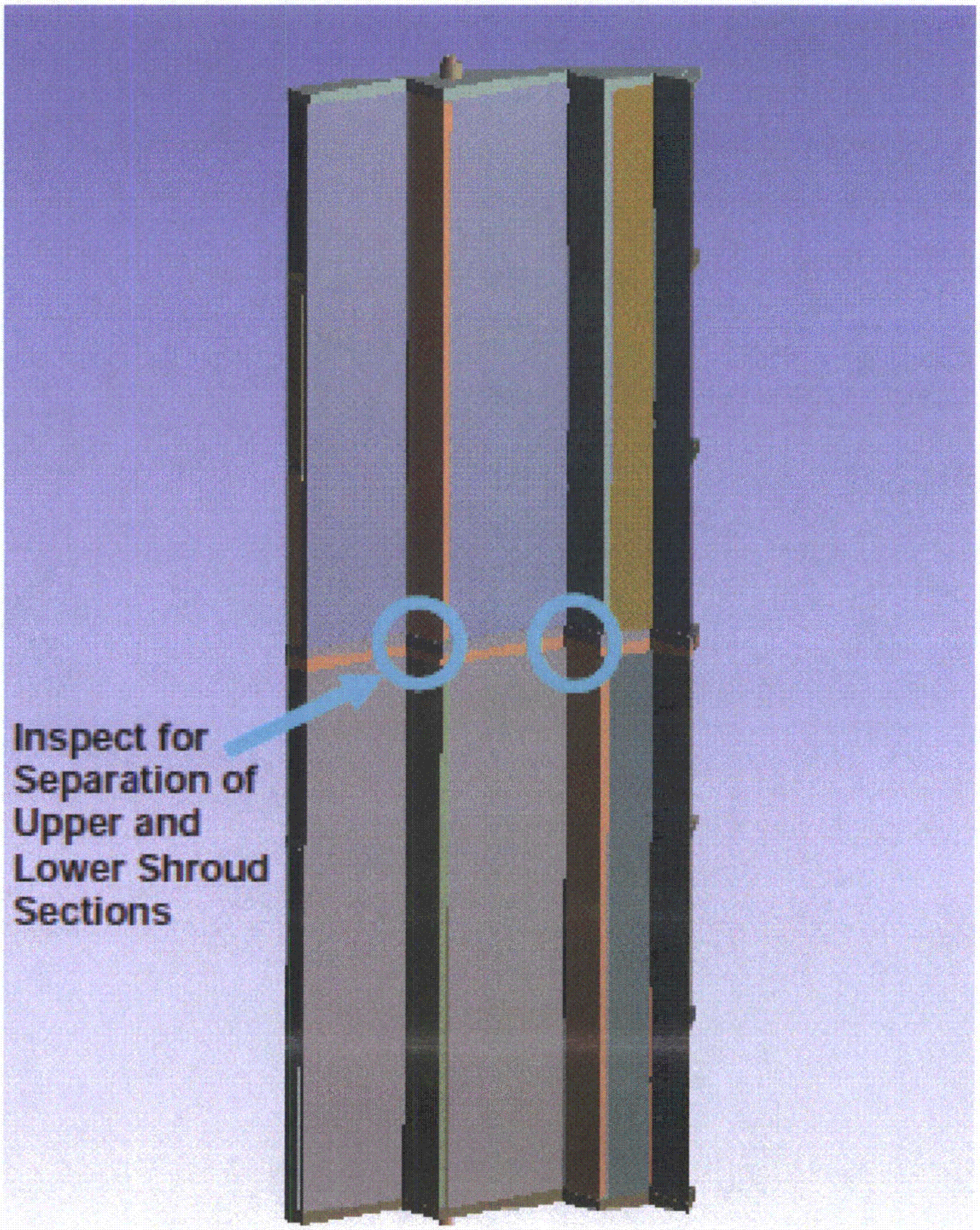


Figure A-4 Typical Flange Plates in CE Welded Core Shroud
(Not applicable for Palisades)

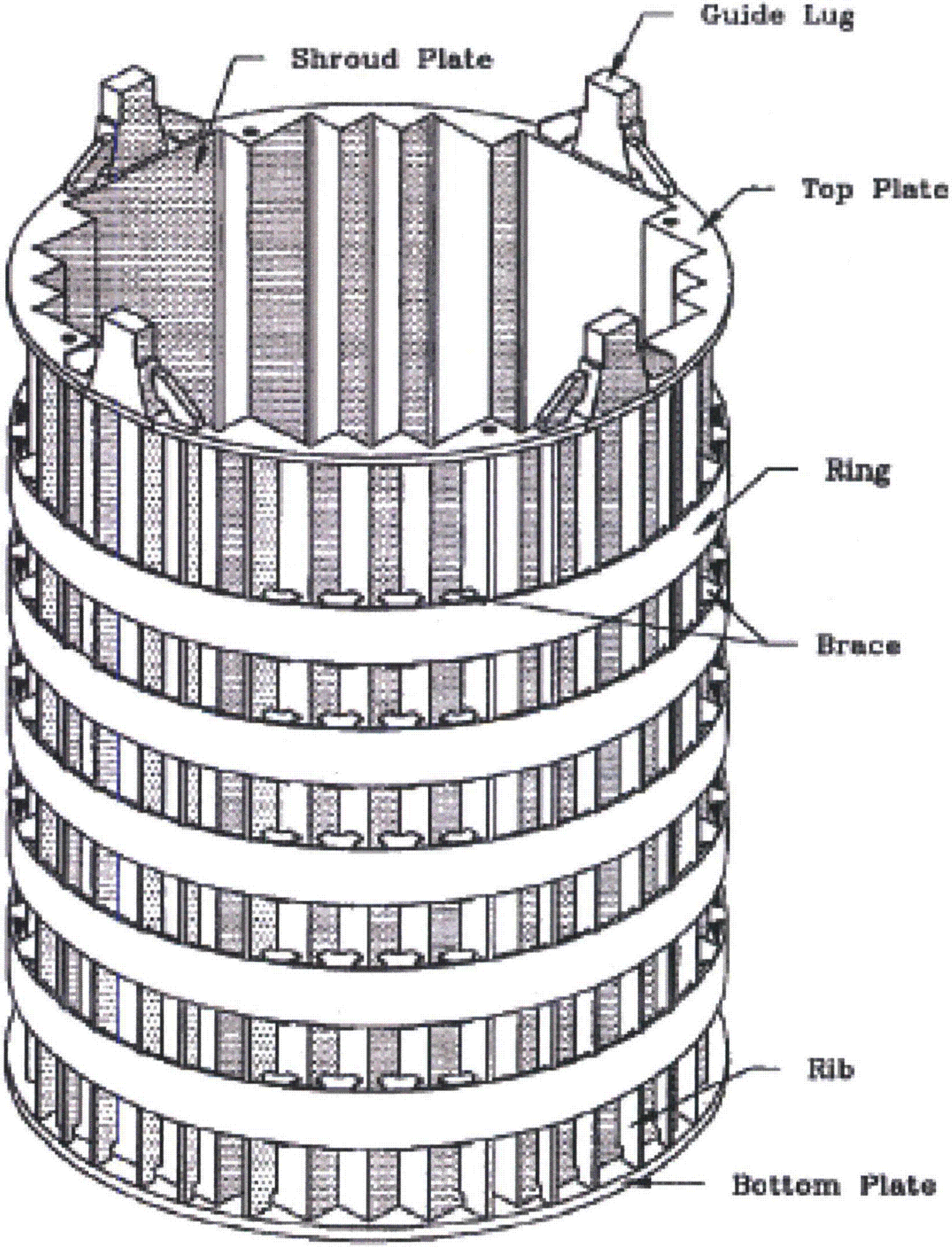


Figure A-5a Typical CE Welded Core Shroud with Full-Height Panels
(Not applicable for Palisades)

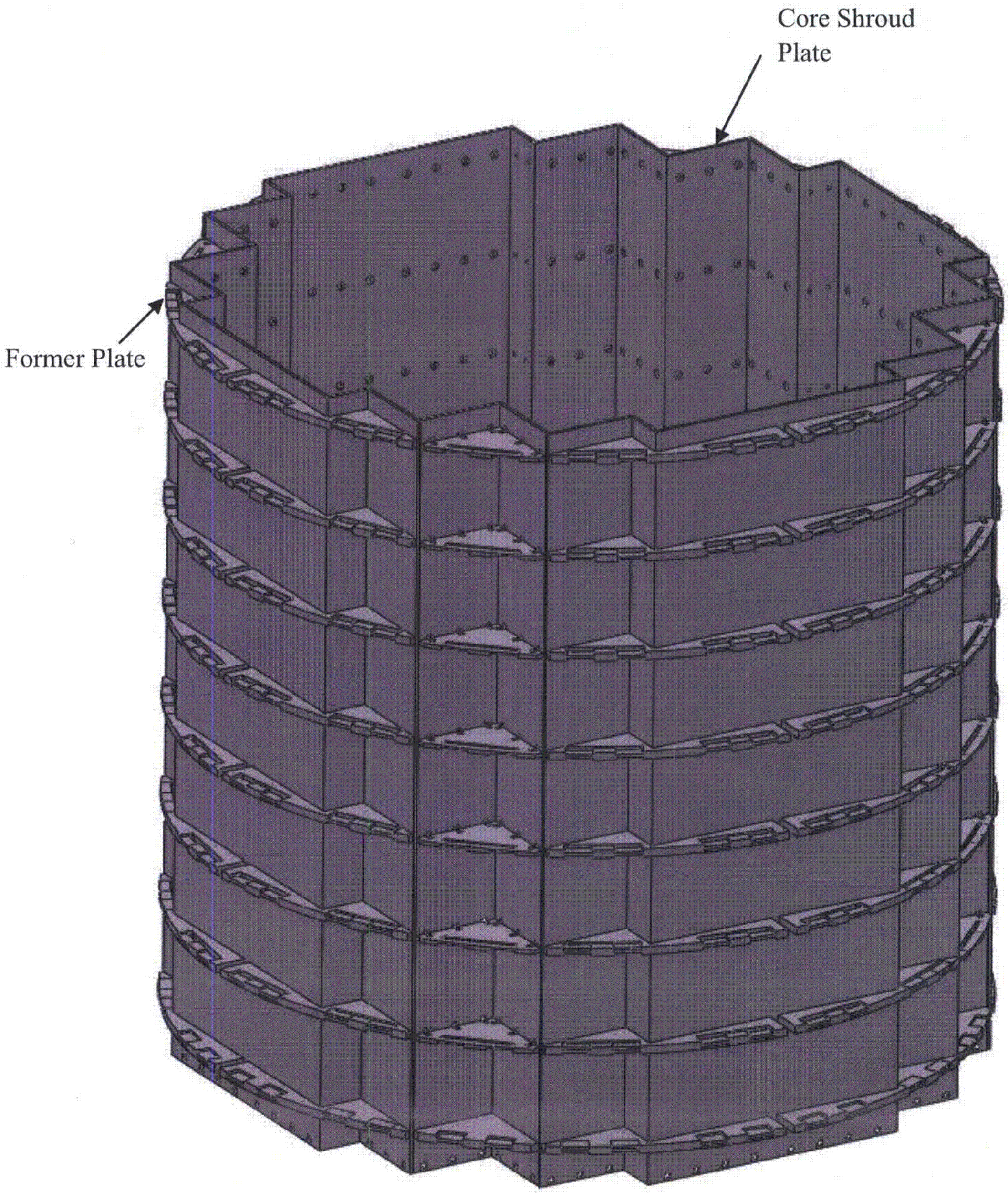


Figure A-5b Palisades Bolted Core Shroud Assembly

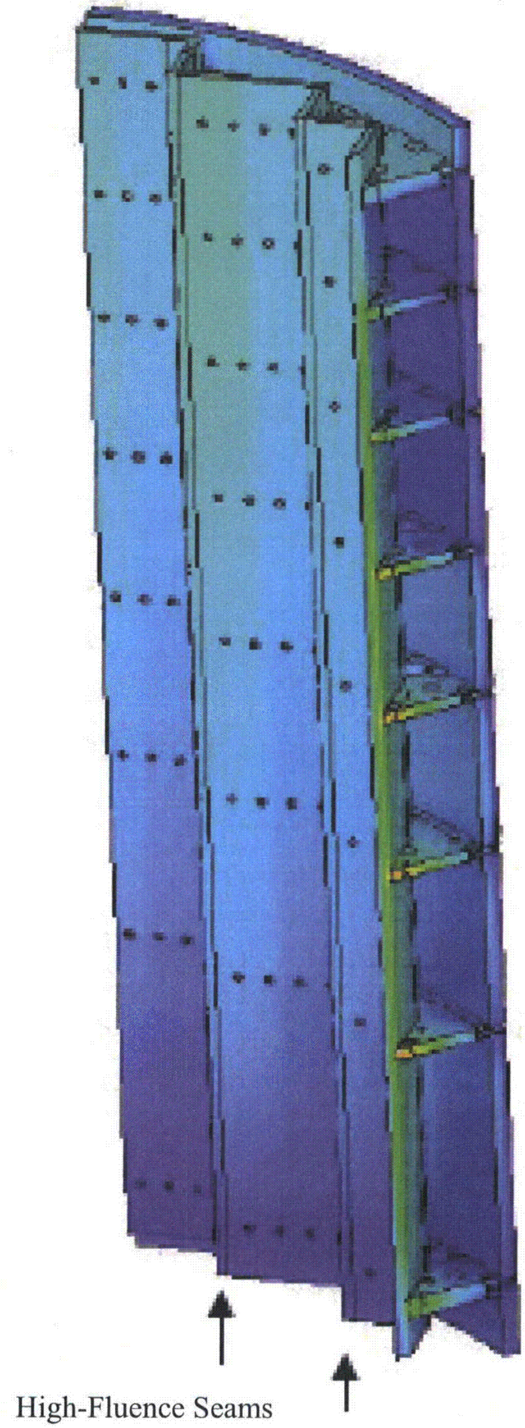


Figure A-6 High-Fluence Seam Locations in Westinghouse Baffle-Former Assembly
(Note: This also applies to bolted CE shroud designs.)

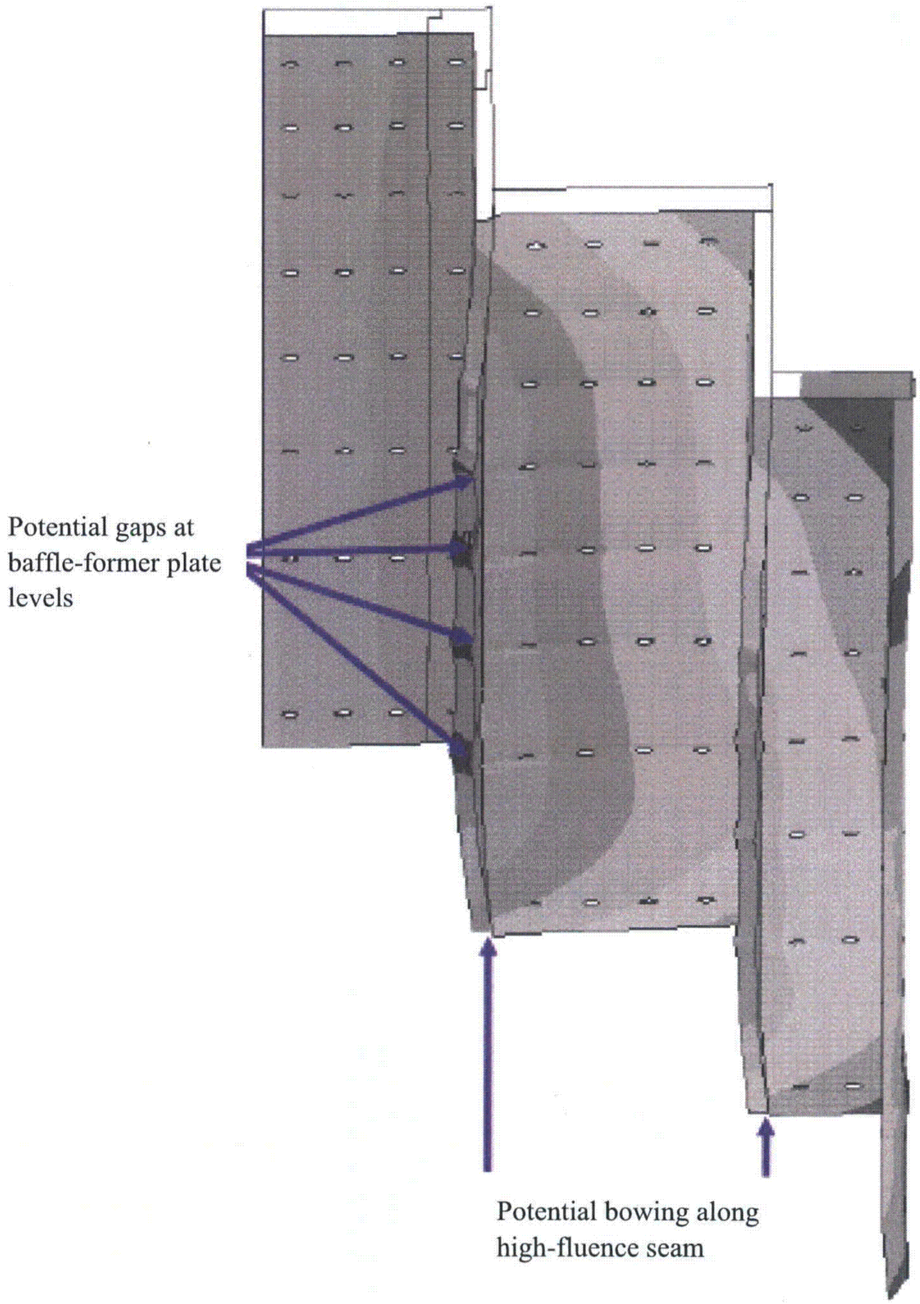


Figure A-7 Exaggerated View of Void Swelling Induced Distortion in Westinghouse Baffle-Former Assembly
(Note: This also applies to bolted CE shroud designs.)

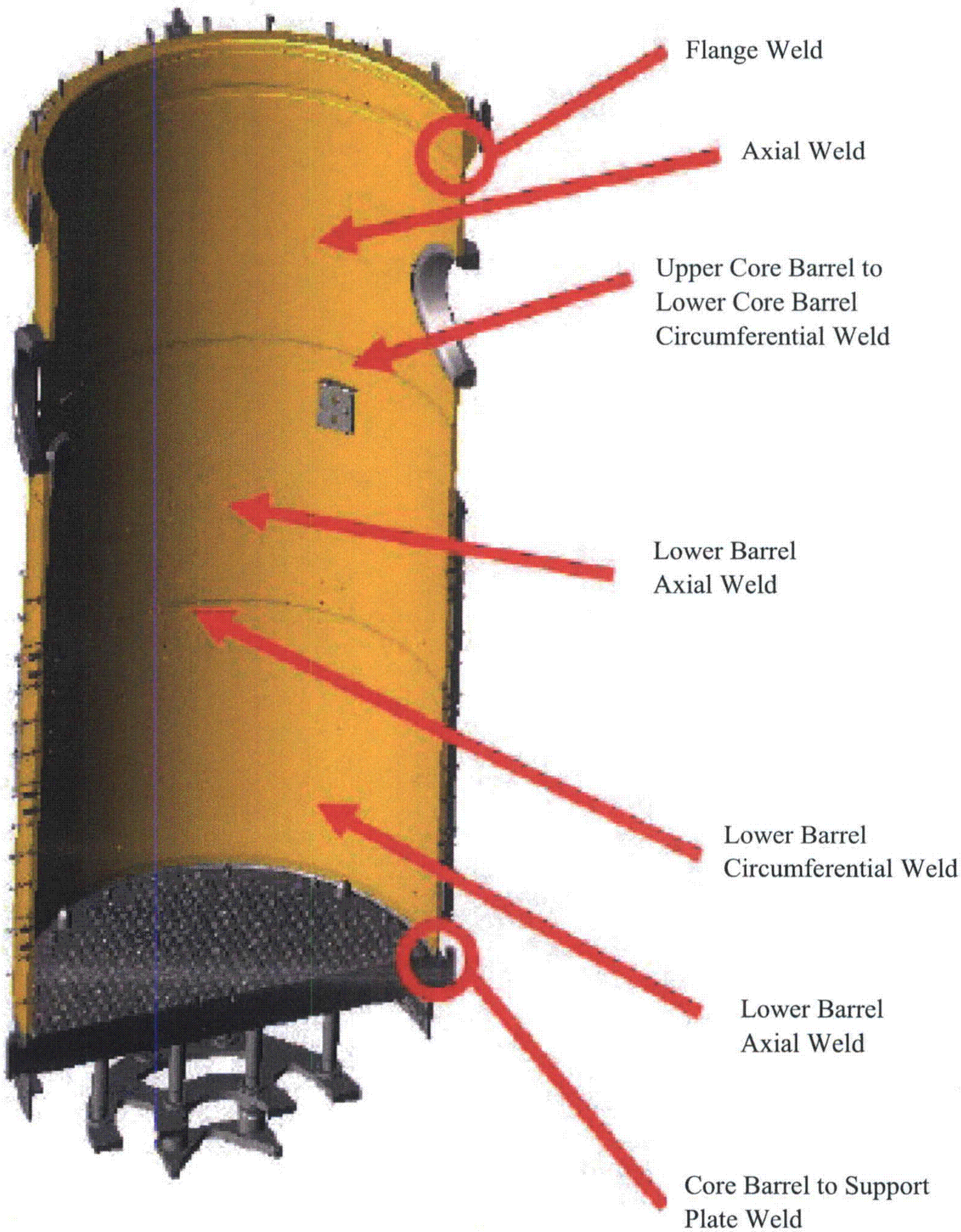


Figure A-8a Typical CE Core Support Barrel Structure

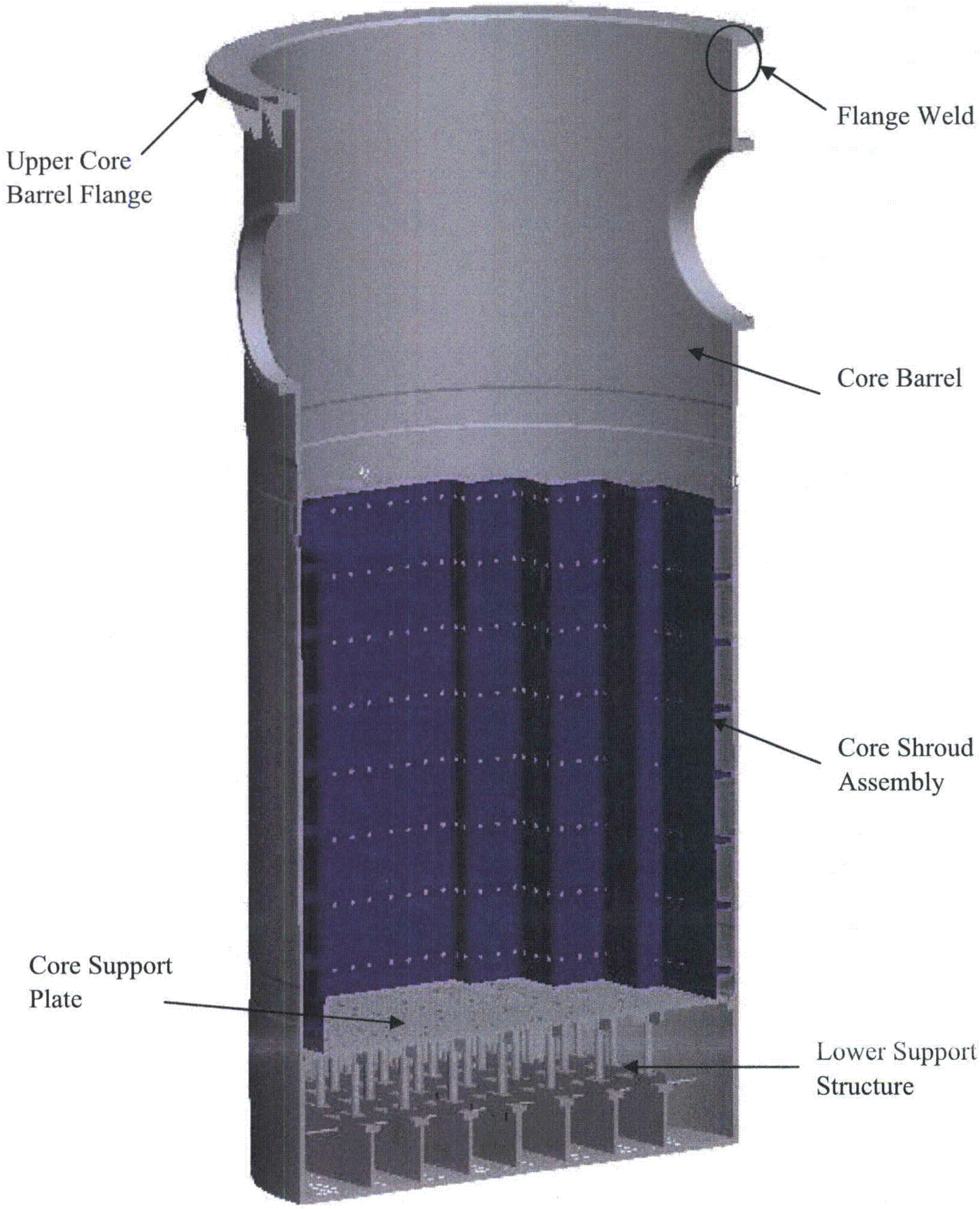


Figure A-8b Palisades Core Support Barrel, Core Shroud Assembly, and Lower Support Structure

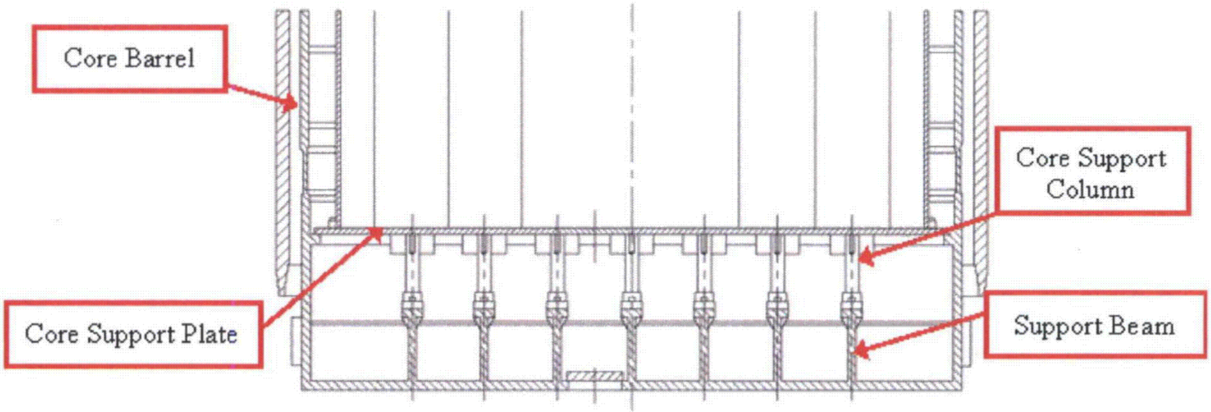


Figure A-9a Schematic View of Palisades Lower Support Structure Assembly

(Note: Palisades does not have a thermal shield as shown in the figure.)

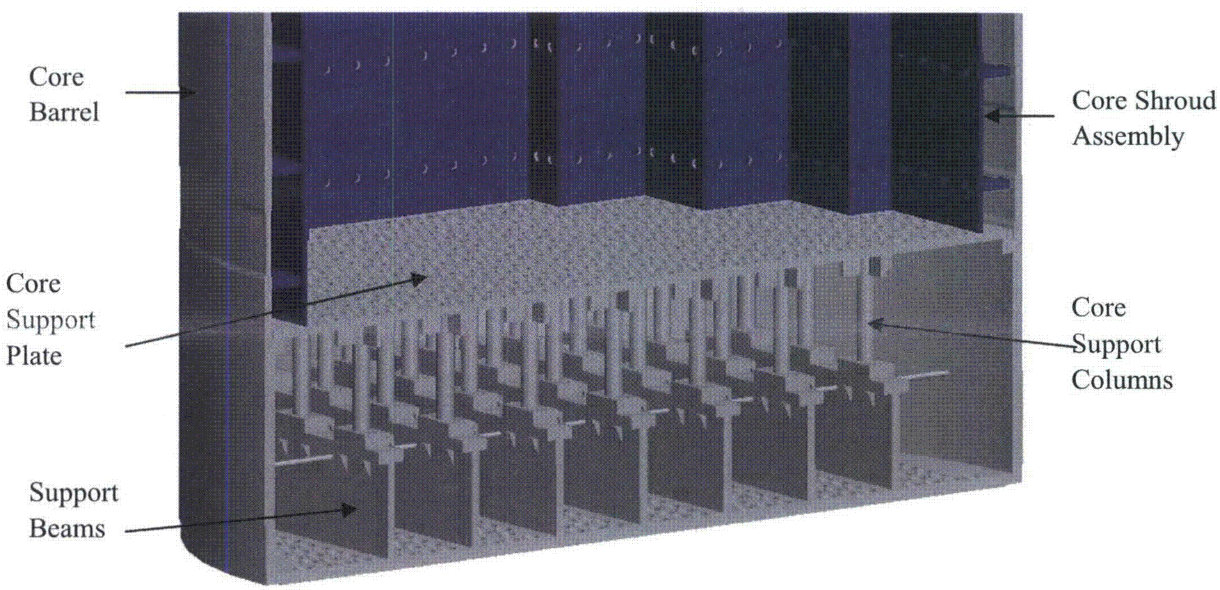


Figure A-9b Palisades Lower Support Structure Assembly

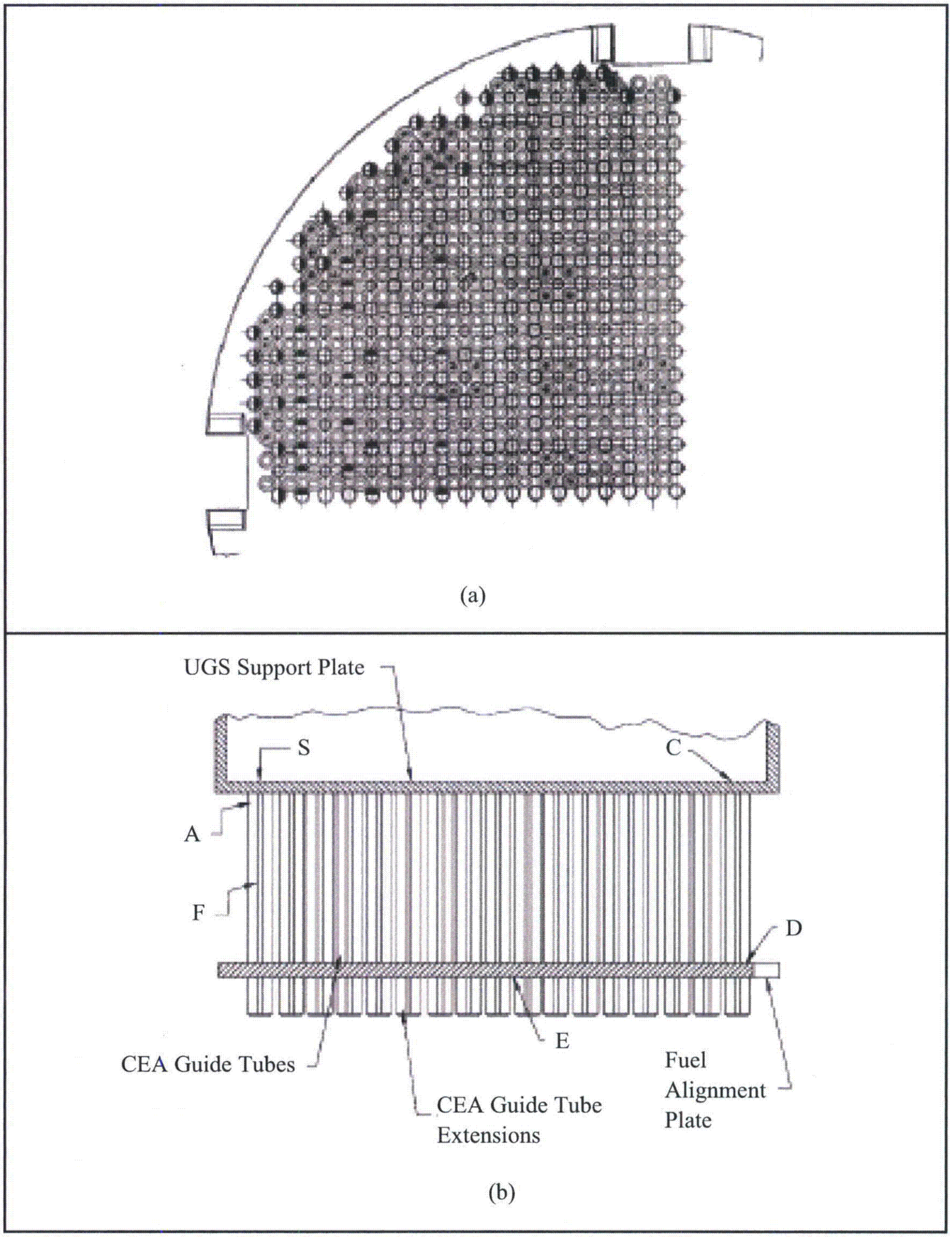


Figure A-10a (a) CE Schematic Illustration of a Portion of the Fuel Alignment Plate, and (b) CE Radial-View Schematic Illustration of the Guide Tubes
(Not applicable for Palisades)

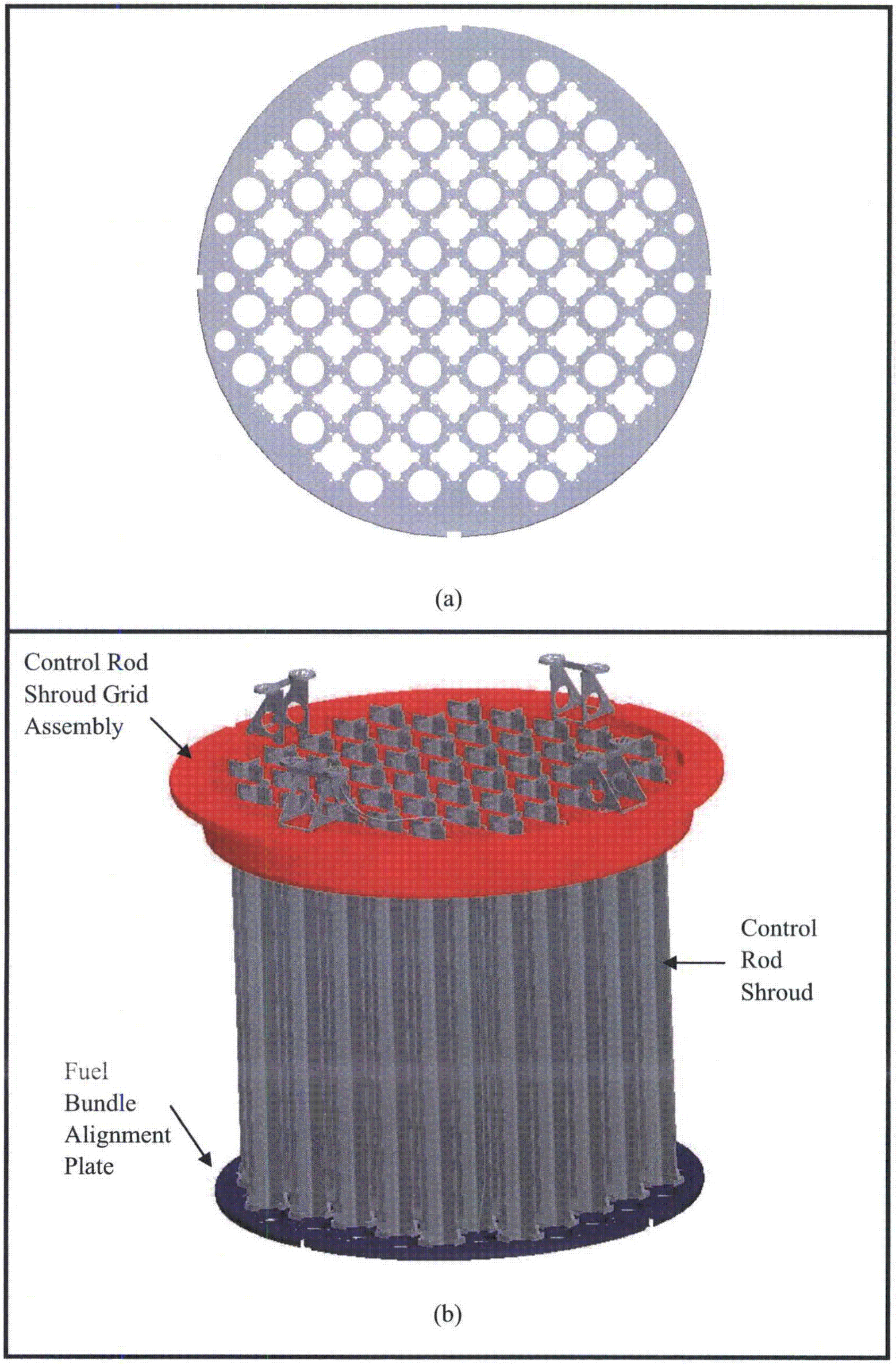


Figure A-10b Palisades (a) Fuel Bundle Alignment Plate and (b) Upper Guide Structure

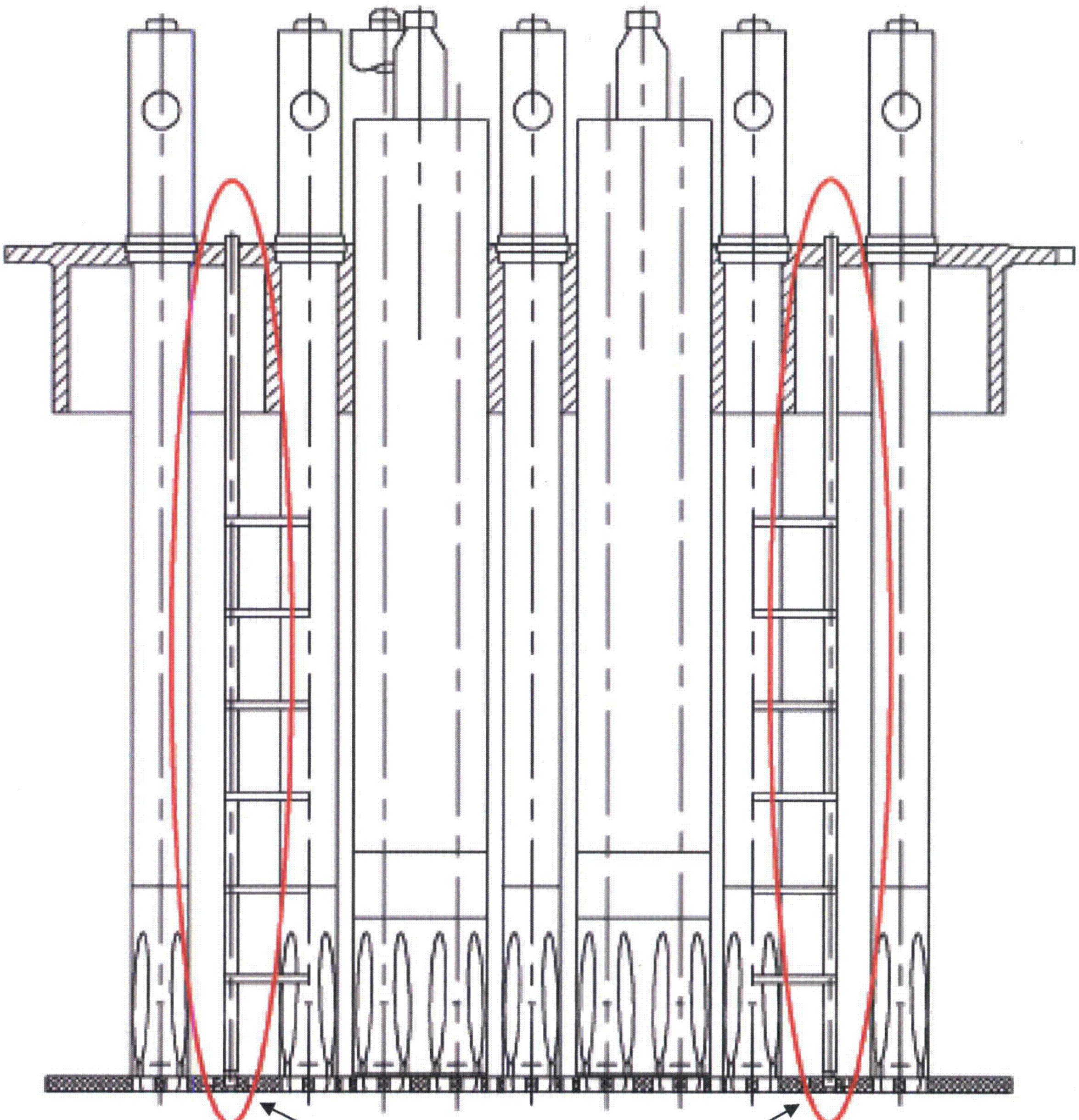


Figure A-11a CE Schematic Illustration of the Control Element Assembly (CEA)
(Not applicable for Palisades)

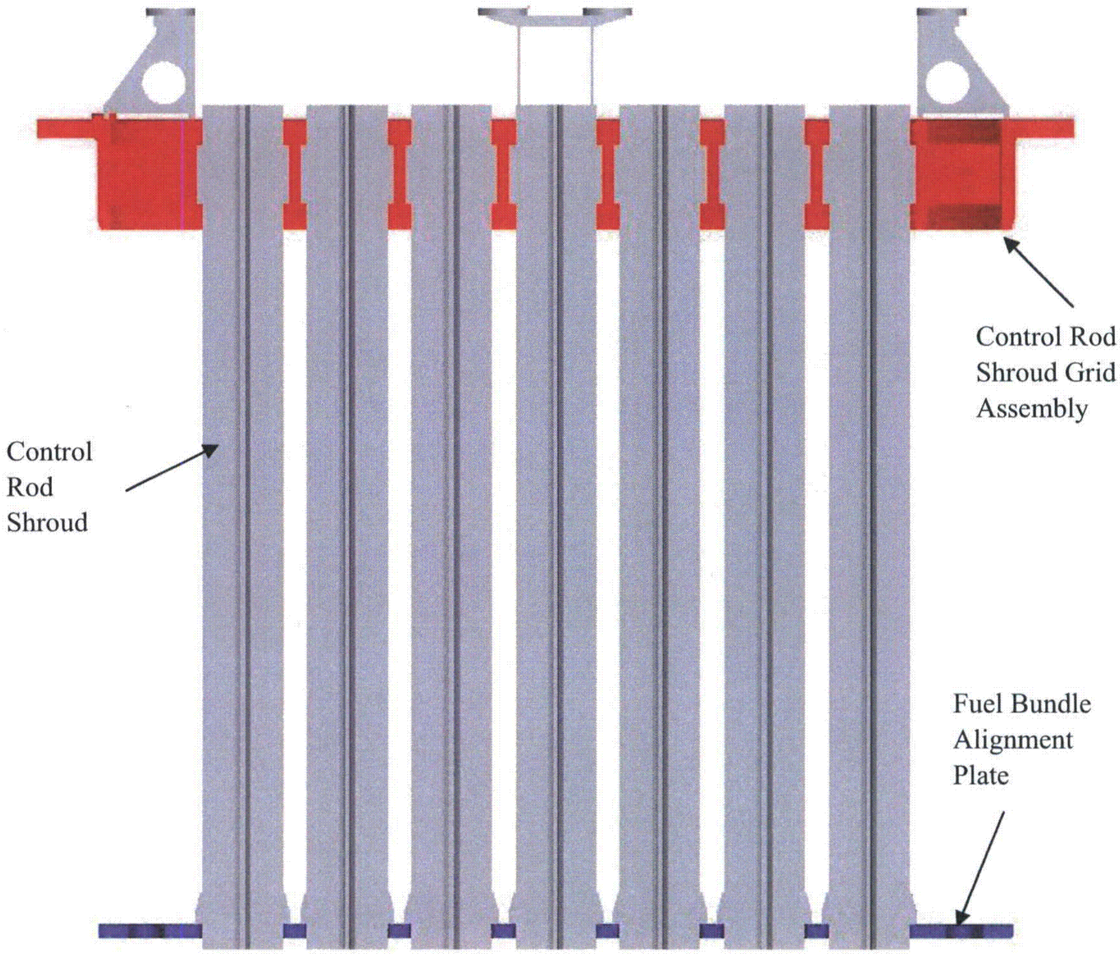
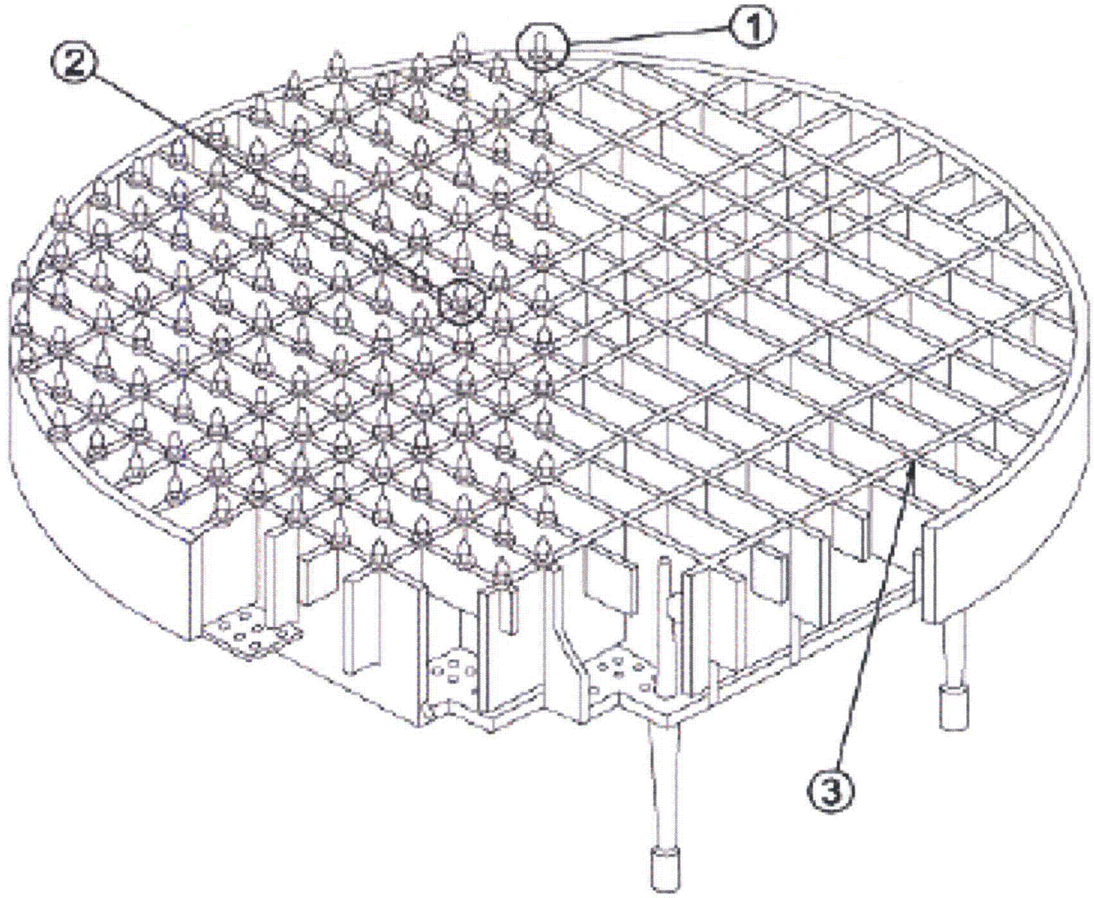


Figure A-11b Illustration of Palisades Control Rod Shroud Assembly



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

Figure A-12 Isometric View of the Lower Support Structure in the CE Core Shrouds with Full-Height Shroud Plates Units

(Not applicable for Palisades)

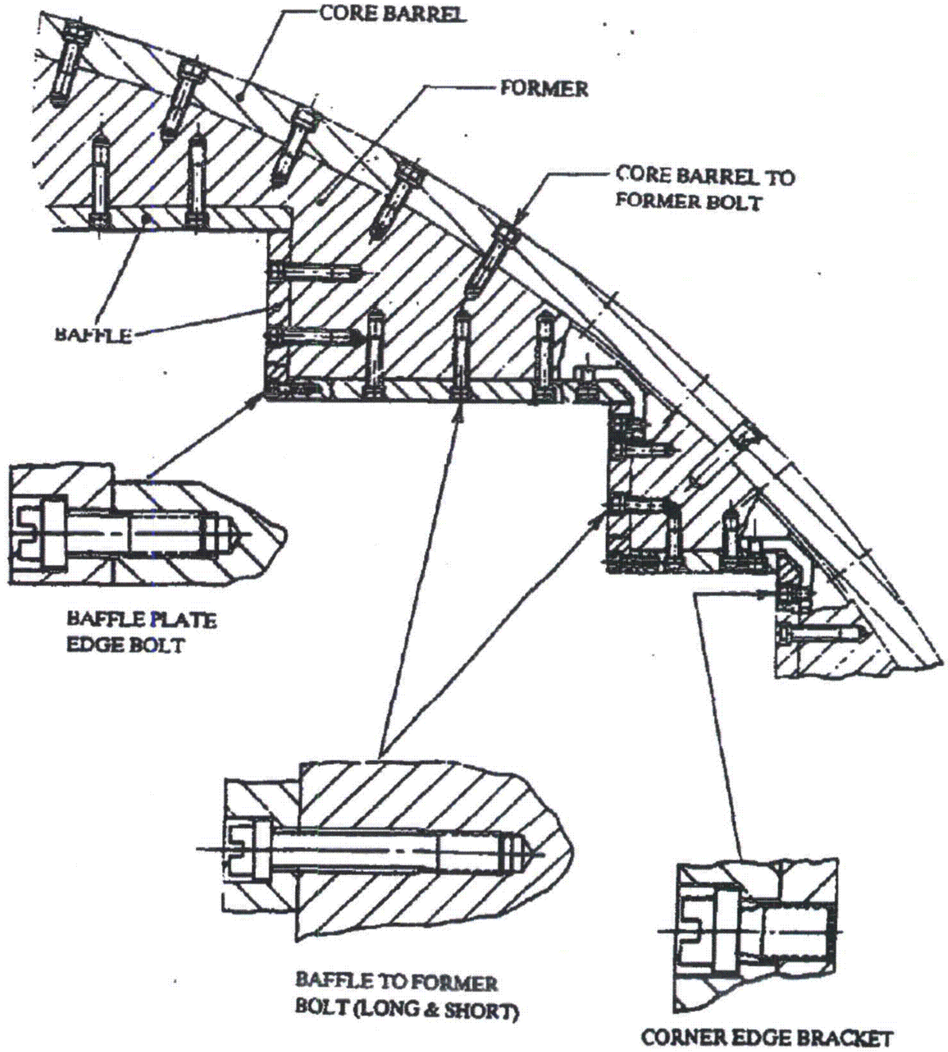
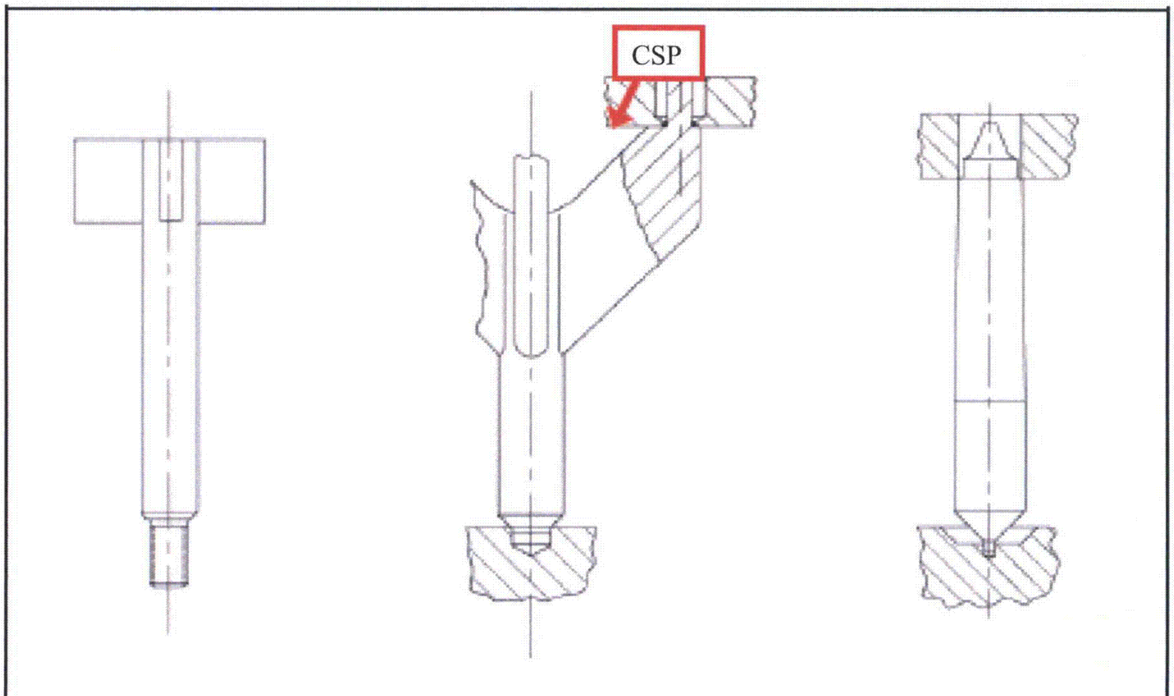


Figure A-13 Bolting in a Typical Westinghouse Baffle-Former Structure
(Note: In CE plants with bolted shrouds, the core shroud bolts are equivalent to baffle-former bolts and barrel-shroud bolts are equivalent to barrel-former bolts.)



a)	b)	c)

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

Figure A-14 CE Core Support Columns

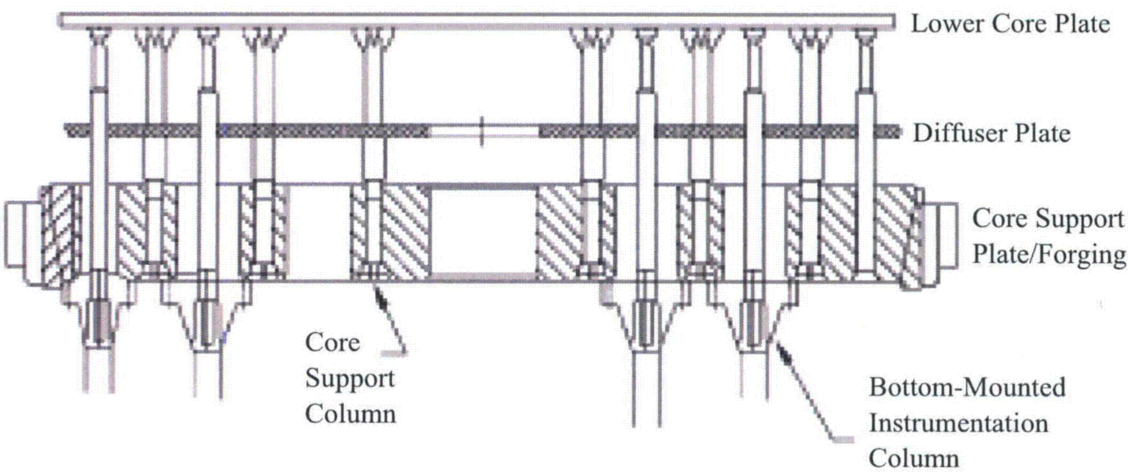


Figure A-15 Westinghouse Lower Core Support Structure – Cross-Section
(Not applicable for Palisades)

APPENDIX B
PALISADES NUCLEAR PLANT LICENSE RENEWAL AGING MANAGEMENT REVIEW SUMMARY TABLES

The content and numerical identifiers in Table B-1 of this Appendix are extracted from Table 3.1.2-3 Reactor Coolant System – Reactor Vessel Internals – Summary of Aging Management Evaluation in the Palisades LRA [9].

Table B-1 LRA Aging Management Evaluation Summary Table 3.1.2-3, Palisades LRA

Component/Commodity	Aging Effect Requiring Management	Aging Management Programs	Comments
Control Rod Shroud Assembly Control Rod Shroud	Changes in Dimensions Cracking	Reactor Vessels Internals Inspection Program Water Chemistry Program	
Control Rod Shroud Assembly Shroud Top Support	Cracking	Reactor Vessels Internals Inspection Program Water Chemistry Program	
Control Rod Shroud Assembly Shroud Support Lug	Cracking	Reactor Vessels Internals Inspection Program Water Chemistry Program	
Control Rod Shroud Assembly Control Rod Support Lug	Changes in Dimensions	Reactor Vessels Internals Inspection Program	
Control Rod Shroud Assembly Fuel Guide Pin	Changes in Dimensions Cracking Loss of Preload	Reactor Vessels Internals Inspection Program Water Chemistry Program ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	

Table B-1 LRA Aging Management Evaluation Summary Table 3.1.2-3, Palisades LRA (cont.)

Component/Commodity	Aging Effect/Mechanism	Aging Management Program	Comments
Control Rod Shroud Assembly Fuel Guide Pin Nuts	Changes in Dimensions Cracking Loss of Preload	Reactor Vessels Internals Inspection Program Water Chemistry Program ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	
Control Rod Shroud Assembly Fuel Plate Cap Screw	Loss of Preload	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	
Core Shroud Assembly Anchor Block	Changes in Dimensions Cracking Reduction in Fracture Toughness	Reactor Vessels Internals Inspection Program Water Chemistry Program	
Core Shroud Assembly Centering Plate	Changes in Dimensions Cracking Reduction in Fracture Toughness	Reactor Vessels Internals Inspection Program Water Chemistry Program	
Core Shroud Assembly Core Shroud Plate	Changes in Dimensions Cracking Reduction in Fracture Toughness	Reactor Vessels Internals Inspection Program Water Chemistry Program	
Core Shroud Assembly Anchor Screw & Pin	Cracking Changes in Dimensions Reduction in Fracture Toughness Loss of Preload	Reactor Vessels Internals Inspection Program Water Chemistry Program ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	

Table B-1 LRA Aging Management Evaluation Summary Table 3.1.2-3, Palisades LRA (cont.)

Component/Commodity	Aging Effect/Mechanism	Aging Management Program	Comments
Core Shroud Assembly Centering Screw & Pin	Cracking Changes in Dimensions Reduction in Fracture Toughness Loss of Preload	Reactor Vessels Internals Inspection Program Water Chemistry Program ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	
Core Shroud Assembly Positioning Screw	Cracking Changes in Dimensions Reduction in Fracture Toughness Loss of Preload	Reactor Vessels Internals Inspection Program Water Chemistry Program ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	
Core Shroud Assembly Shroud Bolt & Pin	Cracking Changes in Dimensions Reduction in Fracture Toughness Loss of Preload	Reactor Vessels Internals Inspection Program Water Chemistry Program ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	
Core Support Barrel Assembly Core Support Barrel	Cracking	Reactor Vessels Internals Inspection Program Water Chemistry Program	
Core Support Barrel Assembly Core Support Barrel Integral Upper Flange	Cracking Changes in Dimensions	Reactor Vessels Internals Inspection Program Water Chemistry Program	

Table B-1 LRA Aging Management Evaluation Summary Table 3.1.2-3, Palisades LRA (cont.)

Component/Commodity	Aging Effect/Mechanism	Aging Management Program	Comments
Incore Instrument Guide Tubes Guide Tube Plug Screw	Loss of Preload Changes in Dimensions Cracking Loss of Material Reduction in Fracture Toughness	Reactor Vessels Internals Inspection Program Water Chemistry Program ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	
Incore Instrument Guide Tubes Instrument Guide Tube	Changes in Dimensions Cracking Loss of Material Reduction in Fracture Toughness	Reactor Vessels Internals Inspection Program Water Chemistry Program ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	
Incore Instrument Guide Tubes Guide Tube Bracket	Changes in Dimensions Cracking Loss of Material Reduction in Fracture Toughness	Reactor Vessels Internals Inspection Program Water Chemistry Program ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	
Incore Instrument Guide Tubes Guide Tube Plugs	Changes in Dimensions Cracking Loss of Material Reduction in Fracture Toughness	Reactor Vessels Internals Inspection Program Water Chemistry Program ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	

Table B-1 LRA Aging Management Evaluation Summary Table 3.1.2-3, Palisades LRA (cont.)

Component/Commodity	Aging Effect/Mechanism	Aging Management Program	Comments
Incore Instrument Guide Tubes Guide Tube Support	Changes in Dimensions Cracking Loss of Material Reduction in Fracture Toughness	Reactor Vessels Internals Inspection Program Water Chemistry Program ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	
Lower Internal Assembly Core Support Barrel Cap Screws	Changes in Dimensions Cracking Reduction in Fracture Toughness	Reactor Vessels Internals Inspection Program Water Chemistry Program	
Lower Internal Assembly Core Support Barrel Snubber Lug	Changes in Dimensions Cracking Loss of Material Reduction in Fracture Toughness	Reactor Vessels Internals Inspection Program Water Chemistry Program ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	
Lower Internal Assembly Core Support Column	Changes in Dimensions Cracking	Reactor Vessels Internals Inspection Program Water Chemistry Program	
Lower Internal Assembly Core Support Column Support Beams and Tie Rods	Changes in Dimensions Cracking Reduction in Fracture Toughness	Reactor Vessels Internals Inspection Program Water Chemistry Program	
Lower Internal Assembly Core Support Plate	Changes in Dimensions Cracking Reduction in Fracture Toughness	Reactor Vessels Internals Inspection Program Water Chemistry Program	

Table B-1 LRA Aging Management Evaluation Summary Table 3.1.2-3, Palisades LRA (cont.)

Component/Commodity	Aging Effect/Mechanism	Aging Management Program	Comments
Upper Guide Structure Instrument Sleeve	Changes in Dimensions Reduction in Fracture Toughness Cracking Loss of Material	Reactor Vessels Internals Inspection Program ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	
Upper Guide Structure Spacer Shim	Cracking Loss of Material	Reactor Vessels Internals Inspection Program ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	
Upper Internal Assembly Brace Grid Beam	Changes in Dimensions Cracking	Reactor Vessels Internals Inspection Program Water Chemistry Program	
Upper Internal Assembly Cross Brace Screw	Changes in Dimensions Cracking	Reactor Vessels Internals Inspection Program Water Chemistry Program	
Upper Internal Assembly Shroud Grid Ring	Changes in Dimensions Cracking	Reactor Vessels Internals Inspection Program Water Chemistry Program	
Upper Internal Assembly Fuel Alignment Plate	Changes in Dimensions Cracking Loss of Material	Reactor Vessels Internals Inspection Program Water Chemistry Program ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	

Table B-1 LRA Aging Management Evaluation Summary Table 3.1.2-3, Palisades LRA (cont.)

Component/Commodity	Aging Effect/Mechanism	Aging Management Program	Comments
Upper Internal Assembly Fuel Plate Align Lug	Changes in Dimensions Cracking Loss of Material	Reactor Vessels Internals Inspection Program Water Chemistry Program ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	
Upper Internal Assembly Fuel Plate Cap Screw	Changes in Dimensions Cracking Loss of Material	Reactor Vessels Internals Inspection Program Water Chemistry Program ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	
Upper Internal Assembly Fuel Plate Guide Pin	Changes in Dimensions Cracking Loss of Material	Reactor Vessels Internals Inspection Program Water Chemistry Program ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	
Upper Internal Assembly Holddown Ring Plunger	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	
Upper Internal Assembly Holddown Ring Strap	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	
Upper Internal Assembly Holddown Ring	Loss of Material	ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program	

APPENDIX C
MRP-227 AUGMENTED INSPECTIONS

Table C-1 MRP-227 Primary Inspection and Monitoring Recommendations for CE-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Core Shroud Assembly (Bolted) Core shroud bolts	Bolted plant designs	Cracking (IASCC, Fatigue)	Core support column bolts, barrel-shroud bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination after 10 to 15 additional EFPY to confirm stability of bolting pattern. Re-examination for high-leakage core designs requires continuing inspections on a ten-year interval.	100% of accessible bolts, or as support by plant-specific justification. Heads are accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. See Figure A-2.
Core Shroud Assembly (Welded) Core shroud plate-former plate weld (Not applicable to Palisades)	Plant designs with core shrouds assembled in two vertical sections	Cracking (IASCC)	Remaining axial welds	Enhanced visual (EVT-1) examination no later than two refueling outages from the beginning of the license renewal period and subsequent examination on a 10-year interval.	Axial and horizontal weld seams at the core shroud re-entrant corners as visible from the core side of the shroud, within 6 inches of central flange and horizontal stiffeners. See Figures A-3 and A-4.
Core Shroud Assembly (Welded) Shroud plates (Not applicable to Palisades)	Plant designs with core shrouds assembled with full-height shroud plates	Cracking (IASCC)	Remaining axial welds, ribs and rings	Enhanced visual (EVT-1) examination no later than two refueling outages from the beginning of the license renewal period and subsequent examination on a 10-year interval.	Axial weld seams at the core shroud re-entrant corners, at the core mid-plane (± 3 feet in height) as visible from the core side of the shroud. See Figure A-5.

Table C-1 (cont.) MRP-227 Primary Inspection and Monitoring Recommendations for CE-Designed Internals

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Core Shroud Assembly (Bolted) Assembly	Bolted plant designs	Distortion (Void Swelling) including: Abnormal interaction with fuel assemblies Gaps along high-fluence shroud plate joints Vertical displacement of shroud plates near high-fluence joint	None	Visual (VT-3) examination no later than two refueling outages from the beginning of the license renewal period. Subsequent examinations on a 10-year interval.	Core-side surfaces as indicated. See Figures A-6 and A-7.
Core Shroud Assembly (Welded) Assembly (Not applicable to Palisades)	Plant designs with core shrouds assembled in two vertical sections	Distortion (Void Swelling), as evidenced by separation between the upper and lower core shroud segments	None	Visual (VT-1) examination no later than two refueling outages from the beginning of the license renewal period. Subsequent examinations on a 10-year interval.	If a gap exists, make three to five measurements of gap opening from the core side at the core shroud re-entrant corners. Then, evaluate the swelling on a plant-specific basis to determine frequency and method for additional examinations. See Figures A-3 and A-4.
Core Support Barrel Assembly Upper (core support barrel) flange weld	All plants	Cracking (SCC)	Remaining core barrel assembly welds, core support column welds	Enhanced visual (EVT-1) examination no later than two refueling outages from the beginning of the license renewal period. Subsequent examinations on a 10-year interval.	100% of the accessible surfaces of the upper flange weld. See Figure A-8.

Table C-1 MRP-227 Primary Inspection and Monitoring Recommendations for CE-Designed Internals (cont.)

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Core Support Barrel Assembly Lower flange weld (Not applicable to Palisades)	All plants	Cracking (Fatigue)	None	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), enhanced visual (EVT-1) examination, no later than two refueling outages from the beginning of the license renewal period. Subsequent examination on a 10-year interval.	Examination coverage to be defined by plant-specific fatigue analysis. See Figure A-8.
Lower Support Structure Core support plate	All plants with a core support plate	Cracking (Fatigue)	None	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), enhanced visual (EVT-1) examination, no later than two refueling outages from the beginning of the license renewal period. Subsequent examination on a 10-year interval.	Examination coverage to be defined by plant-specific fatigue analysis. See Figure A-9.
Upper Internals Assembly Fuel alignment plate (Not applicable to Palisades)	All plants with core shrouds assembled with full-height shroud plates	Cracking (Fatigue)	None	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), enhanced visual (EVT-1) examination, no later than two refueling outages from the beginning of the license renewal period. Subsequent examination on a 10-year interval.	Examination coverage to be defined by plant-specific fatigue analysis. See Figure A-10.

Table C-1 MRP-227 Primary Inspection and Monitoring Recommendations for CE-Designed Internals (cont.)

Item	Applicability	Effect (Mechanism)	Expansion Link	Examination Method/Frequency	Examination Coverage
Control Element Assembly Instrument guide tubes	All plants with instrument guide tubes in the CEA shroud assembly	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports	Remaining instrument guide tubes within the CEA shroud assemblies	Visual (VT-3) examination, no later than two refueling outages from the beginning of the license renewal period. Subsequent examination on a 10-year interval. Plant-specific component integrity assessments may be required if degradation is detected and remedial action is needed.	100% of tubes in peripheral CEA shroud assemblies (i.e., those adjacent to the perimeter of the fuel alignment plate). See Figure A-11.
Lower Support Structure Deep beams (Not applicable to Palisades)	All plants with core shrouds assembled with full-height shroud plates	Cracking (Fatigue) that results in a detectable surface-breaking indication in the welds or beams	None	Enhanced visual (EVT-1) examination, no later than two refueling outages from the beginning of the license renewal period. Subsequent examination on a 10-year interval, if adequacy of remaining fatigue life cannot be demonstrated.	Examine beam-to-beam welds, in the axial elevation from the beam top surface to 4 inches below. See Figure A-12.

Table C-2 MRP-227 Expansion Inspection and Monitoring Recommendations for CE-Designed Internals					
Item	**Applicability**	**Effect (Mechanism)**	**Primary Link**	**Examination Method**	**Examination Coverage**
Core Shroud Assembly (Bolted) Barrel-shroud bolts	Bolted plant designs	Cracking (IASCC, Fatigue)	Core shroud bolts	Volumetric (UT) examination, with initial and subsequent examination frequencies dependent on the results of core shroud bolt examinations.	100% (or as supported by plant-specific justification) of barrel-shroud and guide lug insert bolts with neutron fluence exposures >3 displacements per atom (dpa). See Westinghouse design Figure A-13.
Core Support Barrel Assembly Lower core barrel flange (Not applicable to Palisades)	All plants	Cracking (SCC, Fatigue)	Upper (core support barrel) flange weld	Enhanced visual (EVT-1) examination, with initial and subsequent examinations dependent on the results of the upper (core support barrel) flange weld examinations.	100% of accessible welds and adjacent base metal. See Figure A-8.
Core Support Barrel Assembly Remaining core barrel assembly welds	All plants	Cracking (SCC)	Upper (core support barrel) flange weld	Enhanced visual (EVT-1) examination, with initial and subsequent examinations dependent on the results of core barrel assembly upper flange weld examinations.	100% of one side of the accessible weld and adjacent base metal surfaces for the weld with the highest calculated operating stress. See Figure A-8.
Lower Support Structure Core support column welds (Not applicable to Palisades)	All plants except those with core shrouds assembled with full-height shroud plates	Cracking (SCC, IASCC, Fatigue) including damaged or fractured material	Upper (core support barrel) flange weld	Visual (VT-3) examination, with initial and subsequent examinations based on plant evaluation of SCC susceptibility and demonstration of remaining fatigue life.	Examination coverage determined by plant-specific analysis. See Figures A-9 and A-14.

Table C-2 (cont.) MRP-227 Expansion Inspection and Monitoring Recommendations for CE-Designed Internals

Item	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Core Shroud Assembly (Bolted) Core support column bolts	Bolted plant designs	Cracking (IASCC, Fatigue)	Core shroud bolts	Ultrasonic (UT) examination, with initial and subsequent examination frequencies dependent on the results of core shroud bolt examinations.	100% (or as supported by plant-specific analysis) of core support column bolts with neutron fluence exposures >3 dpa. See Figures A-9 and A-15.
Core Shroud Assembly (Welded) Remaining axial welds (Not applicable to Palisades)	Plant designs with core shrouds assembled in two vertical sections	Cracking (IASCC)	Core shroud plate-former plate weld	Enhanced visual (EVT-1) examination, with initial and subsequent examination frequencies dependent on the results of the core shroud weld examinations.	Axial weld seams other than the core shroud re-entrant corner welds at the core mid-plane. See Figure A-4.
Core Shroud Assembly (Welded) Remaining axial welds Ribs and rings (Not applicable to Palisades)	Plant designs with core shrouds assembled with full-height shroud plates	Cracking (IASCC)	Shroud plates of welded core shroud assemblies	Enhanced visual (EVT-1) examination, with initial and subsequent examination frequencies dependent on the results of the core shroud weld examinations.	Axial weld seams other than core shroud re-entrant corner welds at the core mid-plane, plus ribs and rings. See Figure A-5.
Control Element Assembly Remaining instrument guide tubes	All plants with instrument guide tubes in the CEA shroud assembly	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports.	Peripheral instrument guide tubes within the CEA shroud assemblies	Visual (VT-3) examination, with initial and subsequent examinations dependent on the results of the instrument guide tubes examinations.	100% of tubes in CEA shroud assemblies. See Figure A-11.

Table C-3 MRP-227 Existing Inspection and Aging Management Programs Credited in Recommendations for CE-Designed Internals					
Item	**Applicability**	**Effect (Mechanism)**	**Primary Link**	**Examination Method**	**Examination Coverage**
Core Shroud Assembly Guide lugs Guide lug inserts and bolts	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination, general condition examination for detection of excessive or asymmetrical wear.	First 10-year ISI after 40 years of operation, and at each subsequent inspection interval.
Lower Support Structure Fuel alignment pins (Not applicable to Palisades)	All plants with core shrouds assembled with full-height shroud plates	Cracking (SCC, IASCC, Fatigue)	ASME Code Section XI	Visual (VT-3) examination to detect severed fuel alignment pins, missing locking tabs, or excessive wear on the fuel alignment pin nose or flange.	Accessible surfaces at specified frequency.
Lower Support Structure Fuel alignment pins	All plants with core shroud assembled in two vertical sections	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	Accessible surfaces at specified frequency.
Core Barrel Assembly Upper flange	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	Area of the upper flange potentially susceptible to wear.

Table C-4 MRP-227 Acceptance Criteria and Expansion Criteria Recommendations for CE-Designed Internals					
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 (Not applicable to Palisades)	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 C-4 MRP-227 Acceptance Criteria and Expansion Criteria Recommendations for CE-Designed Internals (cont.)

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

Table C-4 (cont.) MRP-227 Acceptance Criteria and Expansion Criteria Recommendations for CE-Designed Internals					
Item	**Applicability**	**Examination Acceptance Criteria (Note 1)**	**Expansion Link(s)**	**Expansion Criteria**	**Additional Examination Acceptance Criteria**
Core Shroud Assembly (Welded) Shroud plates (Not applicable to Palisades)	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 detectable crack-like surface indication.
Core Shroud Assembly (Welded) Assembly (Not applicable to Palisades)	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 C-4 (cont.) MRP-227 Acceptance Criteria and Expansion Criteria Recommendations for CE-Designed Internals

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Support Barrel Assembly Upper (core support barrel) flange weld	All plants	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Remaining core barrel assembly welds beginning with: a. lower flange weld, followed by: (Go directly to item b, due to Palisades design considerations) b. remaining accessible core barrel assembly welds, and c. core support column welds (cast) (Not applicable to Palisades)	a. Confirmation that a surface-breaking indication >2 inches in length has been detected and sized in the upper flange weld shall require that an EVT-1 examination of the lower flange weld be performed by the completion of the next refueling outage. b. Confirmation that a surface-breaking indication >2 inches in length has been detected and sized in the lower flange weld shall require an EVT-1 examination of all remaining accessible core barrel assembly welds by the completion of the next refueling outage. c. Confirmation of cracking in any of the remaining accessible core barrel assembly welds shall require a VT-3 examination of cast core support column welds, taking into account the general compressive loading of these columns and the potential for thermal aging embrittlement of the castings.	a and b. The specific relevant condition is a detectable crack-like surface indication. c. The specific relevant condition is damaged or fractured material of the cast core support column welds.

Table C-4 MRP-227 Acceptance Criteria and Expansion Criteria Recommendations for CE-Designed Internals (cont.)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Support Barrel Assembly Lower flange weld (Not applicable to Palisades)	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 (Not applicable to Palisades)	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 C-4 (cont.) MRP-227 Acceptance Criteria and Expansion Criteria Recommendations for CE-Designed Internals

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 (Not applicable to Palisades)	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

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