NEXTera ENERGY
POINT BEACH

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U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Point Beach Nuclear Plant, Units 1 and 2
Dockets 50-266 and 50-301
Renewed License Nos. DPR-24 and DPR-27

License Renewal Commitment
Reactor Vessel Internals Program Submittal

References:
(1) NUREG-1839, Safety Evaluation Report Related to the License Renewal of the Point Beach Nuclear Plant, Units 1 and 2, dated December 2005 (ML053420134 and ML053420137)
(2) NextEra Energy Point Beach, LLC, letter to NRC, dated April 23, 2010, Revision to License Renewal Regulatory Commitment 29 Reactor Vessel Internals Program Implementation (ML101160025)
(3) NRC Safety Evaluation of EPRI Report, Materials Reliability Program Report 1016596 (MRP-227, Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," dated June 22, 2011 (TAC No. ME0680)

NUREG-1839 (Reference 1), Appendix A, Commitment 29, requires that an enhanced reactor vessel internals program be submitted for Commission review.

By letter dated April 23, 2010, (Reference 2), NextEra Energy Point Beach, LLC, (NextEra) submitted a revision to Regulatory Commitment 29 to modify the submittal date to be within 180 days following Commission approval of MRP-227.

The NRC issued the safety evaluation for MRP-227, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," on June 22, 2011 via Reference 3. The enclosure to this letter submits the NextEra enhanced reactor vessel internals program.

Summary of Regulatory Commitments

This letter fulfills the following Regulatory Commitment:

- Commitment 29 – "Implement an enhanced reactor vessel internals program." The due date of this commitment is "180 days following approval of the program by the NRC."

Very truly yours,

NextEra Energy Point Beach, LLC

FOR
Larry Meyer
Site Vice President

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Resident Inspector, Point Beach Nuclear Plant, USNRC
PSCW

ENCLOSURE

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

NP 7.7.30
REVISION 1
REACTOR VESSEL INTERNALS PROGRAM

55 pages follow

NP 7.7.30

REACTOR VESSEL INTERNALS PROGRAM

DOCUMENT TYPE:	Administrative
REVISION:	1
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TABLE OF CONTENTS

SECTION		TITLE	PAGE

1.0 PURPOSE

1.1 Scope

This document describes Point Beach Nuclear Plant (PBNP) requirements for implementation of a Reactor Vessel Internals Program for PBNP Units 1 and 2. The reactor vessel internals consist of two (2) basic assemblies: (1) an upper internals assembly which is removed during each refueling outage to access the reactor core; and (2) a lower internals assembly which can be removed following a complete core unload. The reactor vessel internals function to:

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

This document implements a commitment to the NRC to manage the effects of aging for systems, structures, and components (SSC) within the scope of License Renewal (LR) as described in NP 7.7.25, PBNP Renewed License Program. LR Regulatory commitments 4, 5, 6, 8, 29, and 39 require the implementation of a Reactor Vessel Internals Program. (B-1, B-2)

This document demonstrates that the Reactor Vessel Internals Program meets the requirements of NUREG-1801, Revision 1, "Generic Aging Lessons Learned (GALL) Report," Section XI.M16, "PWR Vessel Internals." This program was developed using EPRI MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," and NEI 03-08, "Guideline for the Management of Materials Issues." The Reactor Vessel Internals Program is a living document and will be revised periodically to reflect the latest plant configurations and industry experience.

1.2 Objective

The objectives of the Reactor Vessel Internals Program are to:

1.2.1 Demonstrate that the effects of aging on the Reactor Vessel Internals will be adequately managed for the period of extended operation in accordance with 10 CFR 54.

1.2.2 Summarize the role of existing PBNP Aging Management Programs (AMPs) in the Reactor Vessel Internals Program.

1.2.3 Participate in and implement the industry-defined (EPRI/MRP and PWROG) pressurized water reactor (PWR) reactor vessel internals requirements and guidance for managing aging of reactor internals.

1.2.4 Provide an inspection plan summary for the PBNP reactor vessel internals.

2.0 BACKGROUND

The management of aging degradation effects in reactor vessel internals is required for nuclear plants considering or entering license renewal, as specified in the NRC Standard Review Plan. 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. In 1997, the Westinghouse Owners Group (WOG) issued WCAP-14577, "License Renewal Evaluation: Aging Management for Reactor Internals," which was reissued as Revision 1-A in 2001 after receiving NRC Staff review and approval. Later, an effort was engaged by the EPRI MRP to address the PWR internals aging management issue for the 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 reporting. Based upon that framework and strategy, and on the accumulated data, three important precursor elements to an Aging Management Program were then developed.

- 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.
- PWR internals components were categorized, based on the screening criteria, into categories that ranged from:
  - Components for which the effects from the postulated aging mechanisms are insignificant,
  - Components that are moderately susceptible to the aging effects, and
  - Components that are significantly susceptible to the aging effects.
- Functionality assessment of components and assemblies of components based on representative plant designs using irradiated and aged material properties to determine the effects of the degradation mechanisms on functionality.

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. Items considered included component accessibility, operating experience, existing evaluations, and prior examination results.

The industry effort, as coordinated by the EPRI MRP, finalized initial Inspection and Evaluation Guidelines (I&E Guidelines) for reactor internals and submitted the document (MRP-227) to the NRC with a request for a formal Safety Evaluation Report (SER). The NRC issued an SER on June 22, 2011 (Reference 5.2.7). The changes proposed by the NRC in the Safety Evaluation as well those proposed by the NEI/EPRI in response to NRC Requests for Information (RAIs) are being incorporated into MRP-227-A.

A supporting document addressing inspection requirements has been issued. The industry guidance is contained within two separate EPRI MRP documents:

- MRP-227, "PWR Internals Inspection and Evaluation Guidelines," (hereafter referred to as "the I&E Guidelines" or simply "MRP-227") provides the industry background, listing of reactor internals components requiring inspection, type of NDE required for each component, timing for initial inspections, and 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).

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

The Pressurized Water Reactor Owners Group (PWROG) has also begun efforts to develop "generic acceptance criteria" for the MRP-227 inspections, where feasible, for some of the reactor internals components. Final reports are to be developed and be 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 where a generic approach is not practical.

The PBNP reactor vessel internals are integral with the reactor coolant system (RCS) of a Westinghouse two-loop nuclear steam supply system (NSSS). A typical illustration of which is provided in Attachment A, Figure 1.

As described in NUREG-1839, the PBNP reactor vessel internals are designed to support, align, and guide the core components and to support and guide in-core instrumentation. The reactor vessel internals consist of two basic assemblies: an upper internals assembly that is removed during each refueling operation to obtain access to the reactor core and a lower internals assembly that can be removed, if desired, following a complete core off-load.

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

The lower internals comprise the core barrel, thermal shield, core baffle assembly, lower core plate, intermediate diffuser plate, bottom support plate, and supporting structures. The upper internals package (upper core support structure) is a rigid member composed of the top support plate and deep beam sections, support columns, control rod guide tube assemblies, and the upper core plate. Upon upper internals assembly installation, the last three parts are physically located inside the core barrel.

PBNP was granted a license for extended operation by the NRC through the issuance of a SER in NUREG-1839. In the SER, the NRC concluded that the PBNP License Renewal Application (LRA) adequately identified the reactor vessel internals systems, structures, and components that are subject to an aging management review (AMR), as required by 10 CFR 54.21(a)(1).

The U.S. industry, as noted through the efforts of the MRP and PWROG, has further investigated the components and subcomponents that require aging management to support continued reliable function. As designated by the protocols of NEI 03-08, "Guidelines for the Management of Materials Issues," 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, and plant AMPs must therefore be completed by December 2011, or sooner, if required by plant-specific License Renewal commitments.

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.

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

3.0 RESPONSIBILITIES

The Nuclear Chief Operator Officer and Vice President, Nuclear Engineering Support are ultimately responsible for the successful implementation of the Reactor Vessel Internals Program.

The overall responsibility for the development, revision and implementation of the Reactor Vessel Internals Program resides with the PBNP Programs Engineering Department. Responsibilities of the various interfacing groups are described below.

3.1 Programs Engineering Department

3.1.1 Preparation, maintenance and ownership of the Reactor Vessel Internals Program which implement EPRI MRP requirements and guidance.

3.1.2 Development of refueling outage examination plans.

3.1.3 Development of a recommended strategy for the management of Reactor Vessel Internals materials.

3.1.4 Ensure compliance with regulatory requirements.

3.1.5 Serve as the contact for outside technical communications (NEI, INPO, NRC, EPRI, ASME, PWR Owners Group, etc.).

3.1.6 Participate in industry owners groups and industry investigations of aging affects applicable to reactor vessel internals.

3.1.7 Monitor and participate in industry initiatives with regard to baffle/former and barrel/former bolt performance to support aging management for the Unit 1 bolting.

3.1.8 Incorporate applicable results of industry initiatives related to void swelling in the Reactor Vessel Internals Program.

3.1.9 Provide analysis and response to significant industry events.

3.1.10 Conduct periodic self-assessments of the Reactor Vessel Internals Program.

3.2 Design Engineering

3.2.1 Preparation of Design Change Packages (DCP) packages for repairs or modifications that would result in a configuration change to existing Reactor Vessel Internals components.

3.2.2 Disposition of Condition Reports associated with examination results.

4.0 PROCEDURE

4.1 Program Overview

This reactor internals program utilizes a combination of prevention, mitigation, and condition monitoring. Where applicable, credit is taken for existing programs such as water chemistry, inspections prescribed by the ASME Section XI Inservice Inspection Program, thimble tube inspections, and past and future mitigation projects such as split pin replacements, combined with augmented inspections or evaluations as recommended by MRP-227.

Aging degradation mechanisms that impact internals have been identified and documented in PBNP Aging Management Reviews in support of License Renewal. The overall outcome of the reviews and the additional work performed by the industry, as summarized in MRP-227, is to provide appropriate augmented inspections for reactor internals components to provide early detection of the degradation mechanisms of concern. Therefore, this Reactor Vessel Internals Program is consistent with the existing PBNP 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:

- Stress Corrosion Cracking

  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.

- Irradiation-Assisted Stress Corrosion Cracking

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

- Wear

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

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

- 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.

- Void Swelling and Irradiation Growth

  Void swelling is defined as a gradual increase in the volume of a component caused by formation of microscopic cavities in the material. These cavities result from the nucleation and growth of clusters of irradiation-produced vacancies. Helium produced by nuclear transmutations can have a significant impact on the nucleation and growth of cavities in the material. Void swelling may produce dimensional changes that exceed the tolerances on a component. Strain gradients produced by differential swelling in the system may produce significant stresses. Severe swelling (>5 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.

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

  The loss of preload aging effect can be caused by the aging mechanisms of stress relaxation or creep. Thermal stress relaxation (or primary creep) is defined as the unloading of preloaded components due to long-term exposure to elevated temperatures, 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 Reactor Vessel Internals Program is focused on meeting the requirements of the 10 elements of an aging management program as described in NUREG-1801, GALL Report, Revision 1, Section XI.M16 for PWR Vessel Internals. This is demonstrated through application of existing PBNP AMR methodology that credits inspections prescribed by the ASME Section XI Inservice Inspection Program, existing PBNP programs, and additional augmented inspections based on MRP-227 requirements and guidelines. A description of the applicable existing PBNP programs and compliance with the elements of the GALL is contained in the following subsections.

4.2 Existing PBNP Programs

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

- Water Chemistry Control Program
- ASME Section XI Subsections IWB, IWC, and IWD Inservice Inspection Program
- Thimble Tube Inspection 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, they will be managed under the existing programs.

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

4.2.1 Water Chemistry Control Program

The PBNP Water Chemistry Control Program is used 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 that accelerate corrosion and contaminants that may cause cracking due to SCC. This program relies on monitoring and control of water chemistry to keep peak levels of various contaminants below the system-specific limits. The PBNP Water Chemistry Control Program is based on the current revision of EPRI PWR Primary Water Chemistry Guidelines. Later revisions of the guidelines will be used when issued. The limits imposed by the PBNP program meet the intent of the industry standard for addressing water chemistry.

4.2.2 ASME Section XI Subsections IWB, IWC, and IWD Inservice Inspection Program

The PBNP ASME Section XI Subsections IWB, IWC, and IWD Inservice Inspection (ISI) Program is implemented to monitor for aging effects such as cracking, loss of preload due to stress relaxation or irradiation creep, loss of material, and reduction of fracture toughness due to thermal embrittlement. For PBNP, inspections conducted under the reactor internals program will be controlled as a combination of ASME Section XI ISI exams on core support structures and augmented exams performed under that ISI Program for the remaining reactor internals components addressed within MRP-227.

4.2.3 Thimble Tube Inspection Program

Thimble tubes are long, slender, stainless steel tubes that are seal welded at one end with flux thimble tube plugs, which pass through the vessel penetration, through the lower internals assembly, and finally extend to the top of the fuel assembly. The bottom-mounted instrumentation (BMI) column assemblies provide a path for the flux thimbles into the core from the bottom of the vessel and protect the flux thimbles during operation of the reactor. The thimble tube provides a path for the neutron flux detector into the core and is subject to reactor coolant pressure on the outside and containment pressure on the inside.

The PBNP thimble tube inspection program is an existing plant-specific program that satisfies NRC Bulletin 88-09 requirements that a tube wear inspection procedure be established and maintained for Westinghouse-supplied reactors that use bottom-mounted flux thimble tube instrumentation. The program follows Branch Technical Position RLSB-1, Aging Management Review – Generic, which is included in Appendix A of NUREG-1800. The program includes eddy current testing requirements for thimble tubes and criteria for determining sample size, inspection frequency, flaw evaluation, and corrective action in accordance with NRC Bulletin 88-09.

4.2.4 License Renewal Programs

The license renewal processes conducted at PBNP Units 1 and 2 created a number of programs to ensure that the integrity of structures and components is maintained throughout the periods of extended operation at both sites. Specific programs concerning Reactor Vessel Internals materials include:

- LR-AMP-015-RVINT, Reactor Vessel Internals Program Basis Document for License Renewal
- LR-AMP-001-WCHEM, Water Chemistry Control Program Basis Document for License Renewal
- LR-AMP-017-IWBCD, ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program Basis Document for License Renewal
- LR-AMP-006-TTI, Thimble Tube Inspection Program Basis Document for License Renewal

4.3 Joint Industry Issues Programs

Applicable issues programs and their current examination requirements include:

4.3.1 WCAP-14577, Aging Management for Reactor Internals

The Westinghouse Owners Group (WOG, now PWROG) topical report WCAP-14577 contains a technical evaluation of aging degradation mechanisms and aging effects for Westinghouse RVI components. The WOG sent the report to the NRC staff to demonstrate that WOG member plant owners that subscribed to the WCAP could adequately manage effects of aging on RVI during the period of extended operation, using approved aging management methodologies of the WCAP to develop plant-specific aging management programs.

The aging management review for the PBNP internals was completed in accordance with the requirements of WCAP-14577.

4.3.2 MRP-227, Reactor Internals Inspection and Evaluation Guidelines

MRP-227, as discussed in Section 2.0, 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.

a. 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 United States were evaluated in the MRP program; and appropriate inspection, evaluation, and implementation requirements for reactor internals were defined.

Based on the completed functionality assessments, all PBNP RVI components are categorized within MRP-227 as "Primary" components, "Expansion" components, "Existing Programs" components, or "No Additional Measures" components, as summarized below:

- Primary

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

- Expansion

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

- Existing Programs

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

- No Additional Measures

  Those PWR internals for which the effects of all eight aging mechanisms are below the screening criteria were placed in the No Additional Measures group. Additional components were placed in the No Additional Measures group as a result of 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.

b. 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 for management of aging of reactor internal components within 36 months following issuance of MRP-227, Rev. 0. (that is, no later than December 31, 2011).

PBNP Applicability: This document meets this mandatory element. This program was initially issued as AM 3-44 on September 30, 2009.

Needed

There are five (5) Needed elements:

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.

PBNP Applicability: The applicable Westinghouse tables contained in MRP-227 are Table 4.3 (Primary), Table 4.6 (Expansion), and Table 4.9 (Existing), and Table 5.3 (Examination Acceptance and Expansion Criteria). For PBNP, these tables are Attachments B, C, D, and E respectively.

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

PBNP Applicability: Inspection standards developed under MRP-228 will be used for augmented inspection at PBNP as applicable where required by MRP-227 directives.

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.

PBNP Applicability: PBNP will comply with this requirement.

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

PBNP Applicability: PBNP will participate in future industry efforts and will adhere to industry directives for reporting, response, and follow-up.

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

PBNP Applicability: PBNP will prepare engineering evaluations used to disposition reactor vessel internal results in accordance with a NRC-approved evaluation methodology.

Good Practice

There is no Good Practice element

c. MRP-227 Applicability to PBNP

The applicability of MRP-227 to PBNP 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.

  PBNP Applicability: A low leakage fuel management strategy was implemented starting with fuel cycle 8 for Unit 1 and fuel cycle 6 for Unit 2. The core loading pattern was changed significantly 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.

  PBNP Applicability: Early in operating life, PBNP followed load as needed depending upon Wisconsin Electric Power Company system requirements. Load following operations diminished in the late 1970's as system requirements changed. Since this time, PBNP has typically operated at a fixed power level. The resulting load follow cycles are well within the reactor vessel internal's design basis transient analysis.

The existence of the load following history will not affect the component categorizations nor recommended inspections since PBNP has been operated within its design and licensing basis.

In addition, other conservatisms exist in that PBNP only operated with a high leakage core for 7 cycles on Unit 1 and 5 cycles on Unit 2, which is significantly less than the 30 years assumed in MRP-227.

Thus, MRP-227 is applicable to PBNP.

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

  PBNP Applicability: PBNP has made modifications to the reactor internals. These modifications were all performed with the involvement of Westinghouse, the reactor vessel internals designer. MRP-227 states that the recommendations are applicable to all U.S. PWR operating plants as of May 2007 for the three designs considered. PBNP has not made any modifications to reactor internals components since May 2007 other than replacement of split pins for Unit 1 with an upgraded material in 2008. The split pin replacement was performed by Westinghouse. The modification will have no impact on the applicability of MRP-227 and is an example of the PBNP proactive approach to managing aging reactor internals.

  Based on the above, MRP-227 is applicable for PBNP.

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. PBNP 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.

4.3.4 Point Beach Reactor Internals Aging Management Program Attributes

The key elements of the previously discussed aging management activities, which are used in the Reactor Vessel Internals Program, are described below. The results of an evaluation of each key element against NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Sections XI.M16, "PWR Vessel Internals," is also provided below.

4.3.5 GALL Element 1: Scope of Program

The PBNP reactor vessel internals consist of two basic assemblies: (1) an upper internals assembly that is removed during each refueling operation to obtain access to the reactor core, and (2) a lower internals assembly that can be removed, if desired, following a complete core unload.

The results of the industry research provided by MRP-227, summarized in the tables of Attachments B, C and D 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 PBNP Reactor Vessel Internals Program scope is based on previously established and approved GALL Report approaches through application of the WCAP-14577 methodologies to determine those components that require aging management. Likewise, the additional information provided in the industry MRP-227 is rooted in the GALL methodology and provides a basis for augmented inspections that were required to complete this PBNP Reactor Vessel Internals Program by providing the inspection method, frequency of inspection, and examination acceptance criteria.

This program also credits the One-Time Inspection Program for the management of stress relaxation of the lower internals hold-down spring.

This element is consistent with the corresponding NUREG-1801 aging management program elements.

4.3.6 GALL Element 2: Preventative Actions

The Water Chemistry Control Program is credited for monitoring and control of reactor coolant water chemistry to prevent or mitigate the effects of SCC and IASCC. The PBNP Water Chemistry Control Program is based on the EPRI PWR Water Chemistry Guidelines.

The PBNP Unit 1 and 2 split pins were replaced with a 316 stainless steel material in 2008 and 2005, respectively. A portion of the baffle former bolts in the PBNP Unit 2 internals have been replaced with a more crack resistant material. There are no preventive actions to mitigate thermal aging, neutron irradiation embrittlement or void swelling.

This element is consistent with the corresponding NUREG-1801 aging management program elements.

4.3.7 GALL Element 3: Parameters Monitored or Inspected

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

The ASME Section XI Program consists of periodic volumetric, surface, and/or visual examination of components for assessment, signs of degradation, and corrective actions. Attachments B, C, and D provide a detailed listing of the components and subcomponents and the parameters monitored, inspected, and/or tested.

This element is consistent with the corresponding NUREG-1801 aging management program elements.

4.3.8 GALL Element 4: Detection of Aging Effects

Detection of indications that are required by the ASME 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 include 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

Visual (VT-1) examination is defined in the ASME Code Section XI as an examination "conducted to detect discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, corrosion, or erosion." For MRP-227, VT-1 has only been selected to detect distortion as evidenced by small gaps between the upper-to-lower mating surfaces of CE welded core shrouds assembled in two vertical sections.

The examination acceptance criterion is thus the absence of the relevant condition of gaps that would be indicative of distortion from void swelling. Since PBNP is a Westinghouse design, VT-1 examinations are not used.

EVT-1 Enhanced Visual Examination

Enhanced visual (EVT-1) examination has the same requirements as the ASME Code Section XI visual (VT-1) examination, with additional requirements given in MRP-228. These enhancements are intended to improve the detection and characterization of discontinuities taking into account the remote visual aspect of reactor internals examinations. As a result, EVT-1 examinations are capable of detecting small surface breaking cracks and surface crack length sizing when used in conjunction with sizing aids (e.g. landmarks, ruler, and tape measure). EVT-1 examination has been selected to be the appropriate NDE method for detection of cracking in plates or their welded joints. Thus the relevant condition applied for EVT-1 examination is the same as found for cracking in ASME Code Section XI which is crack-like surface breaking indications.

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

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 (such as baffle-edge bolts), 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;
- 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; and
- 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. Therefore, any recommendation for UT inspection of bolts assumes that the plant owner will work with the designer to establish minimum acceptable bolting patterns prior to the inspection to support continued operation. For Westinghouse-designed plants, minimum acceptable bolting patterns for baffle-former and barrel-former bolts are available through the PWROG. PBNP has been a full participant in the development of the PWROG documents and has full access and use.

This element is consistent with the corresponding NUREG-1801 aging management program elements.

4.3.9 GALL Element 5: Monitoring and Trending

Operating experience with PWR reactor internals has been generally proactive. Flux thimble wear and control rod guide tube split pin cracking issues were identified by the industry and continue to be actively managed. 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 is 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. PBNP has in the past and will continue to maintain cognizance of industry activities and shared information related to PWR internals inspection and aging management.

Inspections credited in Attachment D are based on utilizing the PBNP ASME Section XI Inservice Inspection Program and the augmented inspections derived from the industry program contained in Attachments B and C. These inspections, where practical, will be scheduled to be conducted in conjunction with typical 10-year interval ISI examinations.

Attachments B, C and D 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.

This element is consistent with the corresponding NUREG-1801 aging management program elements.

4.3.10 GALL Element 6: Acceptance Criteria

For examinations conducted in accordance with ASME Section XI, indications or relevant conditions of degradation detected will be evaluated in accordance with IWB-3100, which refers to acceptance standards contained in IWB-3400 and IWB-3500.

Inspection acceptance and expansion criteria are provided in Attachment E. These criteria will be reviewed periodically as the industry continues to develop and refine the information and will be included in updates to PBNP 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. An example of an analytical evaluation is using a minimum bolting WCAP approach such as those commonly used to support continued component or assembly functionality. Additional analysis to establish acceptable bolting pattern evaluation criteria for the baffle-former bolt assembly, as contained in various industry documents, is also considered in determining the acceptance of inspection results to support continued component or assembly functionality. 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. Additional analysis to establish Attachment E expansion component evaluation criteria is being performed through the efforts of the PWROG. Status is monitored through direct PBNP cognizance of industry (including PWROG) activities related to PWR internals inspection and aging management.

This element is consistent with the corresponding NUREG-1801 aging management program elements.

4.3.11 GALL Element 7: Corrective Actions

Corrective actions are implemented in accordance with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," and FPL-1, Quality Assurance Topical Report. The PBNP corrective action program includes non-safety related structures, systems, and components."

This element is consistent with the corresponding NUREG-1801 aging management program elements.

4.3.12 GALL Element 8: Confirmation Process

The confirmation process is part of the corrective action program, which is implemented in accordance with the requirements of 10 CFR 50, Appendix B and FPL-1, Quality Assurance Topical Report. The PBNP corrective action program includes non-safety related structures, systems, and components."

This element is consistent with the corresponding NUREG-1801 aging management program elements.

4.3.13 GALL Element 9: Administrative Controls

The Reactor Vessel Internals Program is implemented through the documents discussed in Section 5.2. These implementing documents are subject to administrative controls, including a formal review and approval process, in accordance with the requirements of 10 CFR 50, Appendix B and FPL-1, Quality Assurance Topical Report.

This element is consistent with the corresponding NUREG-1801 aging management program elements.

4.3.14 GALL Element 10: Operating Experience

Extensive industry and PBNP operating experience has been reviewed during the development of the Reactor Vessel Internals Aging Management Program. The experience reviewed includes NRC Information Notices 84-18, "Stress Corrosion Cracking in PWR Systems" and 98-11, "Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants." Most of the industry operating experience reviewed has involved cracking of austenitic stainless steel baffle-former bolts, or SCC of high-strength internals bolting. SCC of control rod guide tube split pins has also been reported.

A review of plant-specific operating experience with reactor vessel internals reveals that PBNP has responded to industry operating experience regarding reactor vessel internals degradation. Examples that demonstrate PBNP's response to industry operating experience with RVI are described below:

PBNP Control Rod Guide Tubes Split Pins

The control rod guide tube split pins were replaced at PBNP Units 1 and 2. The new pins are a material upgrade from Inconel X-750 to 316SS in support of managing aging in the component.

Fuel Damage Due to Baffle Gap Water Jetting -Conversion to Upflow Design

Both Point Beach units have experienced fuel rod damage due to baffle gap jetting. Both units were placed in service with a "downflow" design for coolant flow in the former region between the baffle plates and the core barrel. This design minimizes core bypass flow, but exhibits a fairly large pressure differential across the baffle plates near the top of the core. The baffle plates at the two units are of both a buttjointed bolted configuration and a customized joint configuration.

One fuel pin failure at PBNP Unit 1 in 1975 was mentioned in NRC IE Circular 80-17, "Fuel Pin Damage due to Water Jet from Baffle Plate Corner." This same document describes the peening techniques that were used to reduce the gaps between baffle segments. The fuel rod damage was limited to only one of eight fuel assemblies in core locations adjacent to identified susceptible joints.

Baffle plate/joint peening at PBNP was done in accordance with:

- Baffle Peening (Unit 1), Westinghouse Procedure "MRS 2.3.1 Gen-1," Revision 1
- Baffle Peening (Unit 2), Westinghouse Procedure "MRS 2.3.1 Gen-1," Revision 2

While peening the susceptible locations appeared to be an effective solution to the baffle jetting problem, subsequent fuel damage was experienced after several years.

At that time, a more effective solution was implemented by converting the baffle/barrel flow design to the "upflow" configuration. This modification effectively eliminated the pressure differential across the baffle joints which caused the jetting problem. The upflow modifications were performed in the 1986-1987 time frame in accordance with:

- PBNP Modification 86-058 (Unit 1) – RV lower internals upflow conversion.
- PBNP Modification 86-059 (Unit 2) – RV lower internals upflow conversion.

Installation of Flexureless Inserts

The original internals configuration at both units incorporated removable inserts at the top of the control rod guide tubes to minimize bypass flow from the core outlet region to the upper head region. These inserts were attached to the upper guide tube housing plate by four flexures, which were manufactured from Inconel X-750. These flexures, over time, exhibited cracking due to primary water stress corrosion cracking, and a modified "flexureless" design was developed as a replacement.

Replacement of the original inserts at Point Beach has been performed in accordance with:

- PBNP Modification 84-235 (Unit 1) – Remove flexures and install flexureless inserts.
- PBNP Modification 84-236 (Unit 2) – Remove flexures and install flexureless inserts.

Examination and Replacement of Baffle to Former Bolts at PBNP Unit 2

In 1997, the Westinghouse Owners Group (WOG) formed a task group to investigate the issue of reactor vessel internals bolting integrity. A small group of domestic utilities that operate Westinghouse two-loop plants collectively developed a proactive plan for inspection and replacement of bolts that join the baffle to former plates in the reactor lower internals. An augmented examination via UT was conducted on the baffle-former bolts of PBNP Unit 2. The UT examination identified a number of bolts with indications indicative of crack like flaws. A number of bolts sufficient to guarantee the structural margins of the baffle-former joints were replaced, including all bolts with UT indications. The replacement bolts are fabricated from a more IASCC-resistant material.

PBNP Unit 2 Hold-Down Spring Evaluation

The preload from the PBNP Unit 2 core barrel hold-down spring was evaluated to justify extending the use of the current core barrel hold-down spring for the period of extended operation. During the October 2006 outage, Westinghouse measured the height of the core barrel hold-down spring for PBNP Unit 2.

The material for the PBNP Unit 2 hold-down spring is quenched and tempered modified Type 403 martensitic stainless steel per Westinghouse Specification PDS-10725-HA,

Based on the new and as-found core barrel hold-down spring height measurements, the lower internals hold down capacity of PBNP Unit 2 was shown to be adequate for an increased design life of 60 years.

PWROG Control Rod Guide Cards Inspection Program at PBNP Unit 1

The PWROG is currently conducting upper internals control rod guide tube card wear measurements on a sample of guide tubes from selected representative pilot plants to approximate the remaining life of the guide tube guide cards. This is a proactive effort by the U.S. industry to establish criteria for inspection and gather data to support aging management of the component. PBNP proactively inspected guide cards in Unit 1 during the 2008 outage as one of the representative pilot inspection plants. Inspection outcomes are being evaluated to ensure compliance with MRP-227 specifications.

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

A key element of the MRP-227 Guideline is the reporting of age-related degradation of reactor vessel components. PBNP, 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.

PBNP will continue to participate in industry groups studying RVI materials degradation issues, such as the EPRI MRP RI-ITG and Westinghouse Owner's Group (WOG), so as to gain the full benefit of the data that will be generated. The WOG also sponsors research related to RVI degradation issues that are specific to Westinghouse-built PWRs. As new information and technology becomes available, the Reactor Vessel Internals Program will be modified to incorporate enhanced inspections of appropriate components as necessary. As such, PBNP committed to submit the Reactor Vessel Internals Program to the NRC for review and approval one year prior to entering the period of extended operation.

A review of NRC Inspection Reports, QA Audit/Surveillance Reports, and Self-Assessments since 1999 revealed no issues or findings that could impact the effectiveness of the Reactor Vessel Internals Program. As additional operating experience is obtained, lessons learned may be used to adjust this program.

This element is consistent with the corresponding NUREG-1801 aging management program elements.

4.3.15 Summary

The Reactor Vessel Internals Program complies with or exceeds the corresponding aging management attribute in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Section XI.M16, "PWR Vessel Internals."

5.0 REFERENCES

5.1 Source Documents

5.1.1 WCAP-15960, "Aging Management Review Report for the Point Beach Reactor Vessel Internals"

5.1.2 LR-AMR-0127-RC, License Renewal Aging Management Review Report, Reactor Vessel Internals

5.1.3 LR-AMP-015-RVINT, Reactor Vessel Internals Program Basis Document for License Renewal

5.1.4 Westinghouse Letter WEP-02-8, "Point Beach Units 1 and 2 Reactor Internals CMTR Summary," dated September 12, 2002

5.2 Reference Documents

5.2.1 NEI 03-08, "Guideline for the Management of Materials Issues"

5.2.2 EPRI MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"

5.2.3 ER-AA-105, "Reactor Coolant System Materials Degradation Management Program (RCS MDMP)"

5.2.4 NP 7.7.25, PBNP Renewed License Program

5.2.5 NDE-756, Remote Visual Examination of Reactor Pressure Vessel Interior and Components

5.2.6 NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Revision 1, September, 2005

5.2.7 NRC Final Safety Evaluation of EPRI Report, Materials Reliability Program Report 106596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" (TAC No. ME0680), dated June 22, 2011

5.3 Records

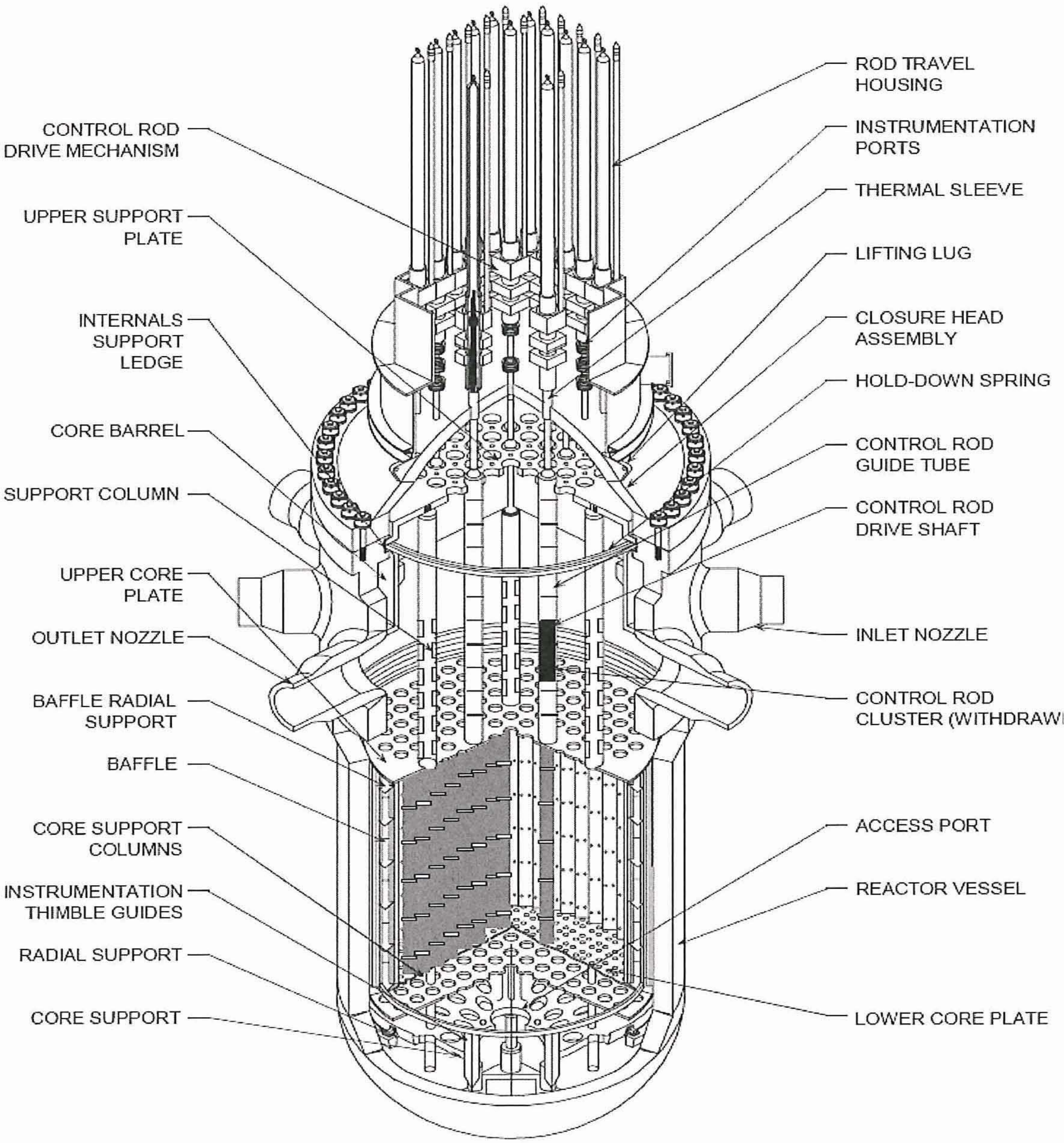
None.

6.0 BASES

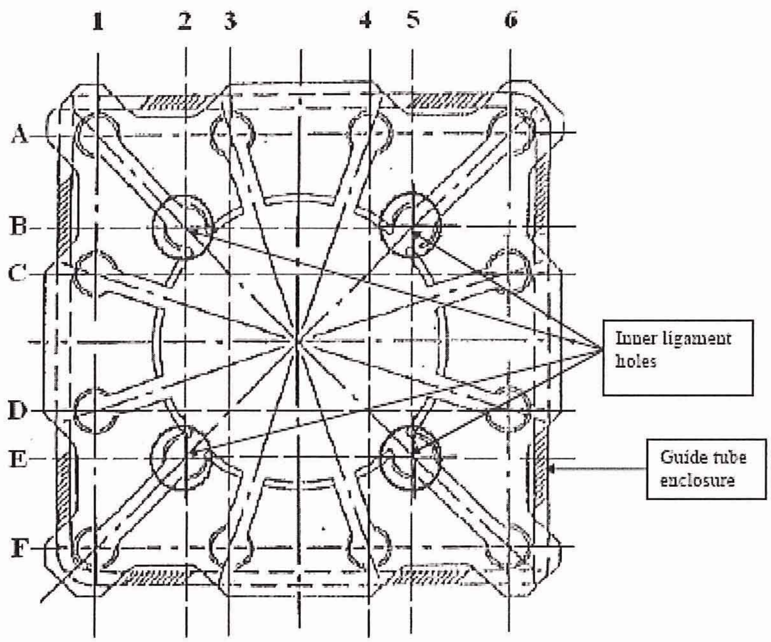
B-1 LR-AMP-015-RVINT, Reactor Vessel Internals Program Basis Document for License Renewal

B-2 NUREG-1839, "US NRC Safety Evaluation Report Related to the License Renewal of the Point Beach Nuclear Plant, Unit 1 and 2"

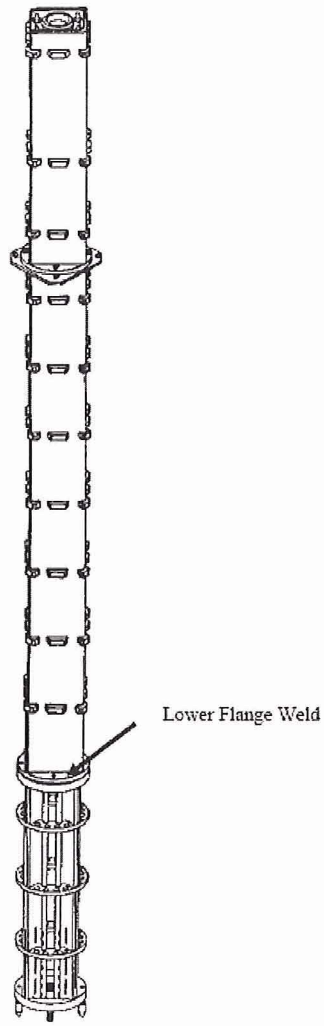
ATTACHMENT A
FIGURE 1
Overview Of Typical Westinghouse Internals



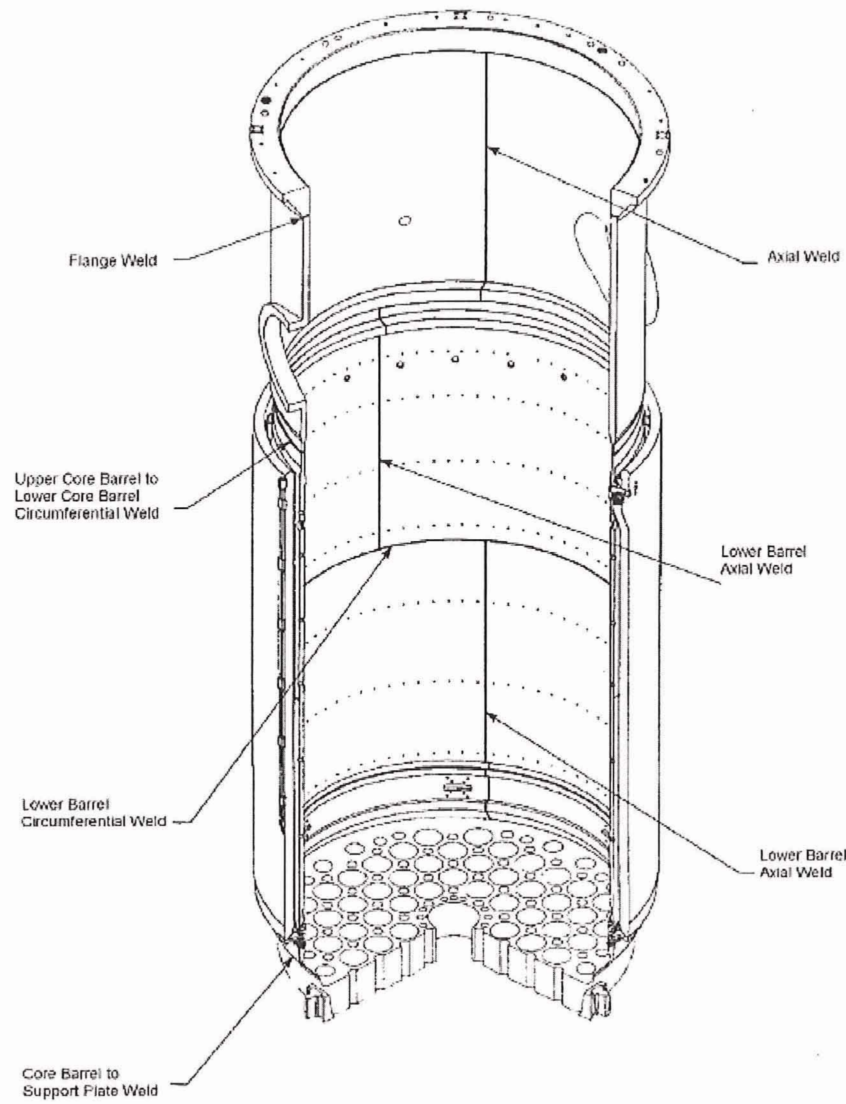
ATTACHMENT A
FIGURE 2
Westinghouse Control Rod Guide Card (PBNP 14 X 14 Fuel Assembly)



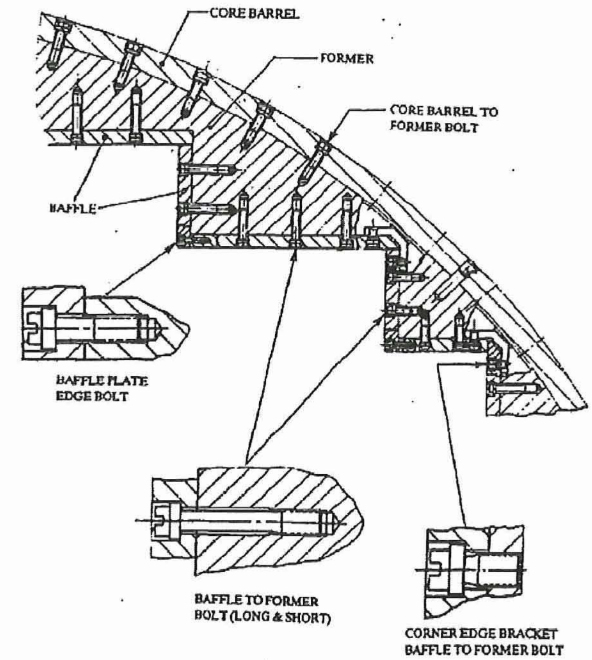
ATTACHMENT A
FIGURE 3
Typical Westinghouse Control Rod Guide Tube Assembly



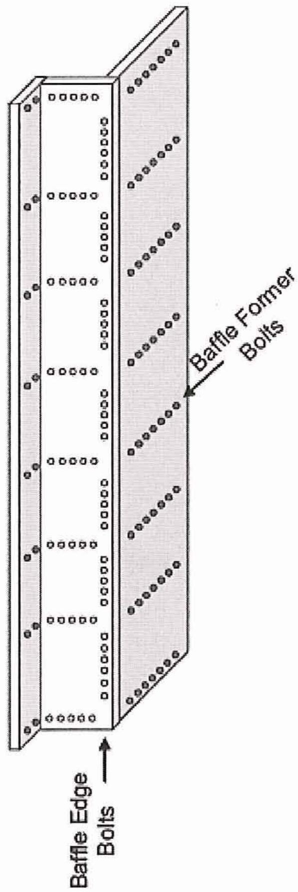
ATTACHMENT A
FIGURE 4
Major Fabrication Welds In Typical Westinghouse Core Barrel



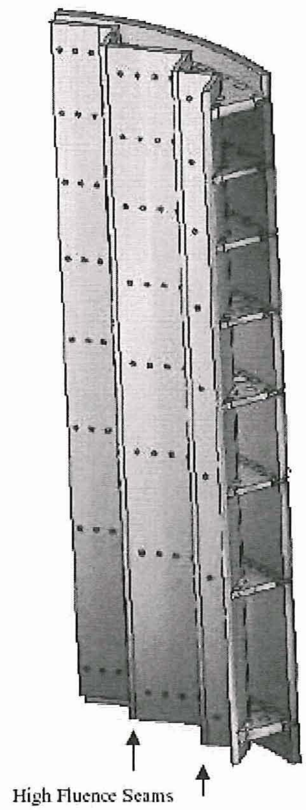
ATTACHMENT A
FIGURE 5
Bolt Locations In Typical Westinghouse Baffle-Former-Barrel Structure



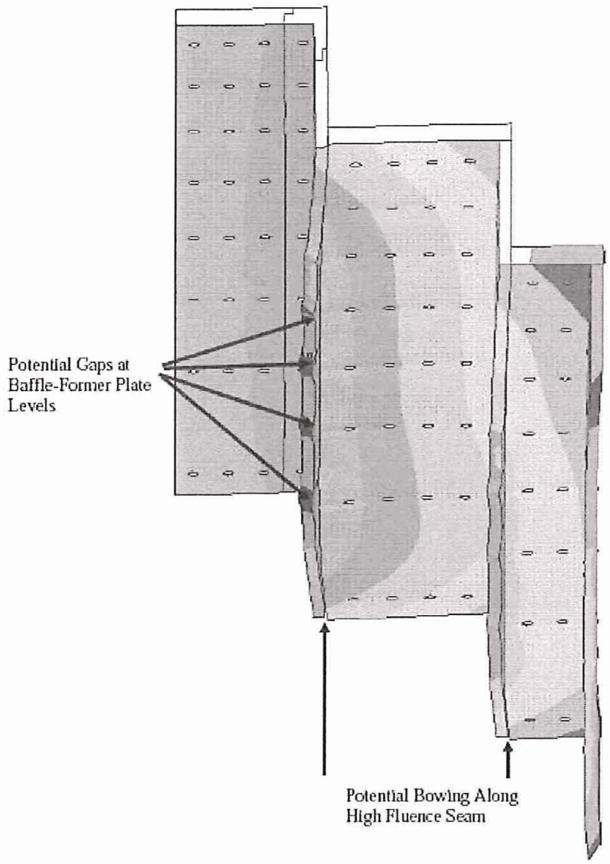
ATTACHMENT A
FIGURE 6
Baffle-Edge Bolt And Baffle-Former Bolt Locations At High Fluence Seams In Bolted Baffle-Former Assembly



ATTACHMENT A
FIGURE 7
High Fluence Seam Locations In Westinghouse Baffle-Former Assembly



ATTACHMENT A
FIGURE 8
Exaggerated View Of Void Swelling Induced Distortion In Westinghouse Baffle-Former Assembly



ATTACHMENT A
FIGURE 9
Vertical Displacement Of Westinghosue Baffle Plates Caused By Void Swelling

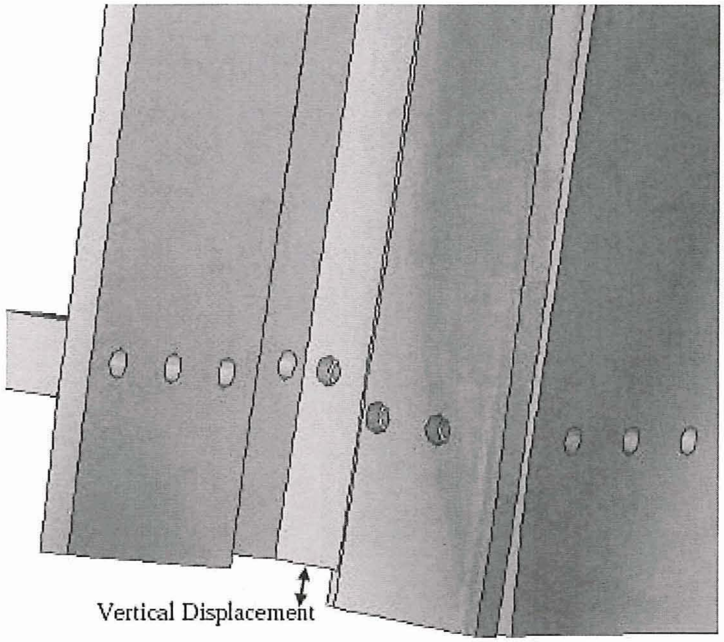
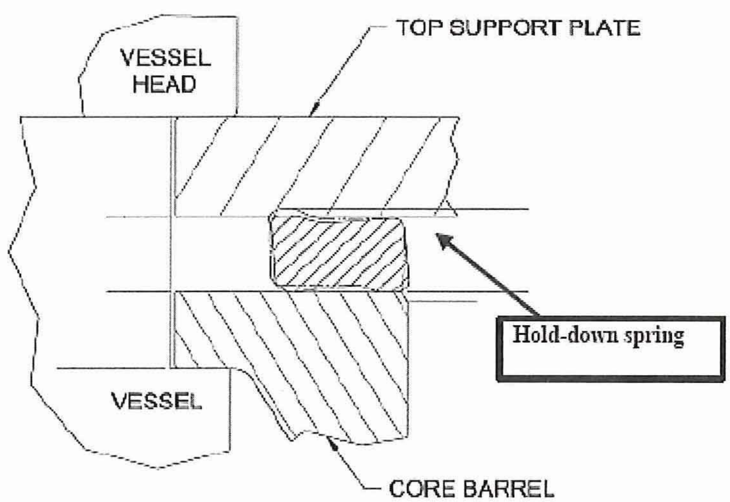
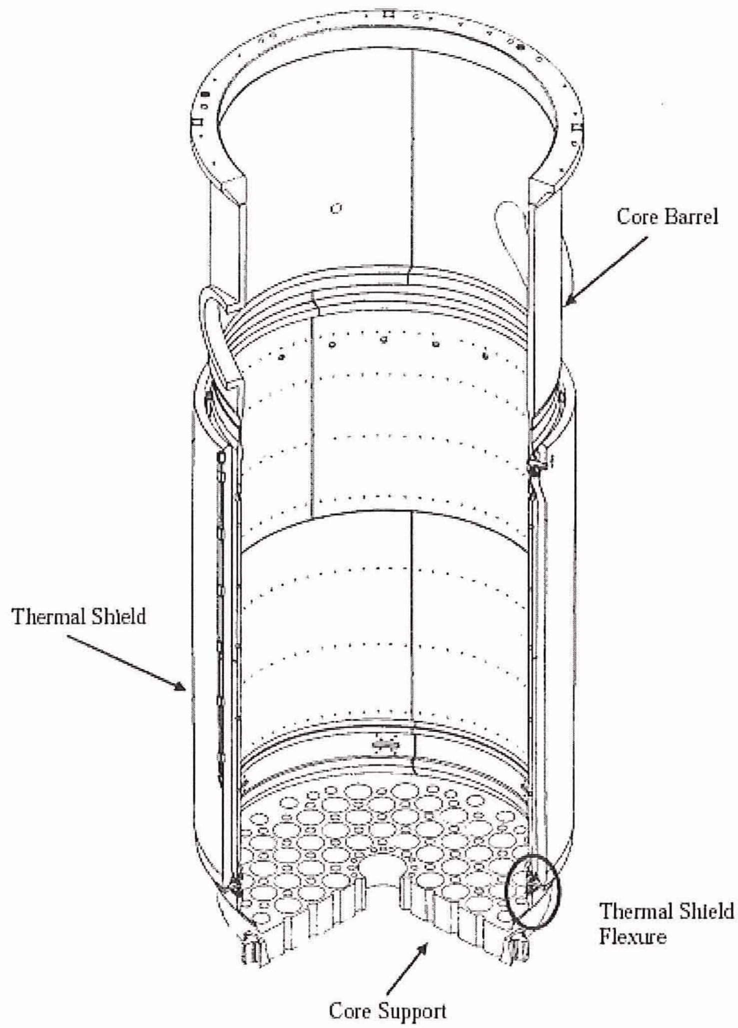


FIGURE 10
Schematic Cross-Sections Of The Westinghouse Hold-Down Springs



ATTACHMENT A
FIGURE 11
Location Of Westinghouse Thermal Shield Flexures



ATTACHMENT A

FIGURE 12

Schematic Indicating Location Of Westinghouse Lower Core Support Structure. Additional Details Shown In Figure 13

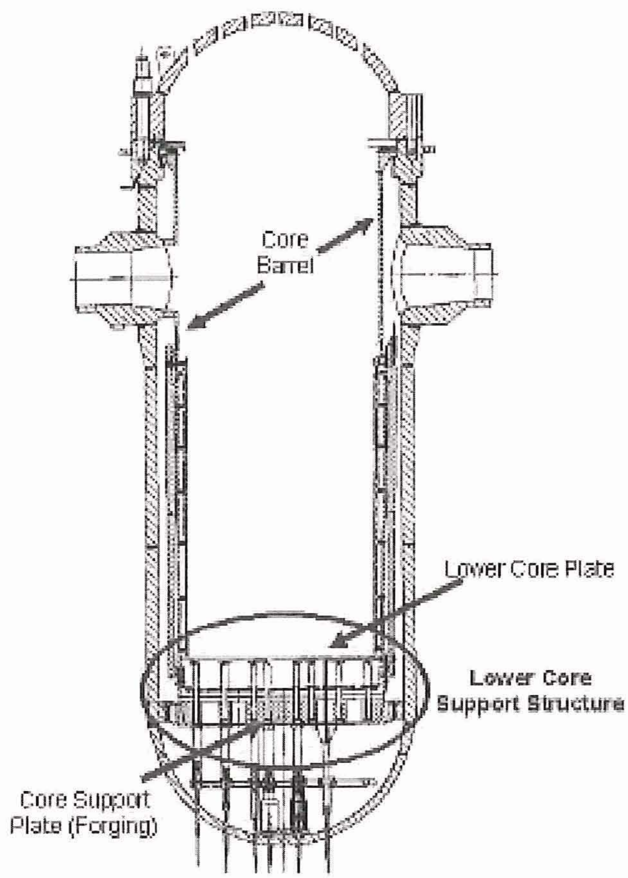
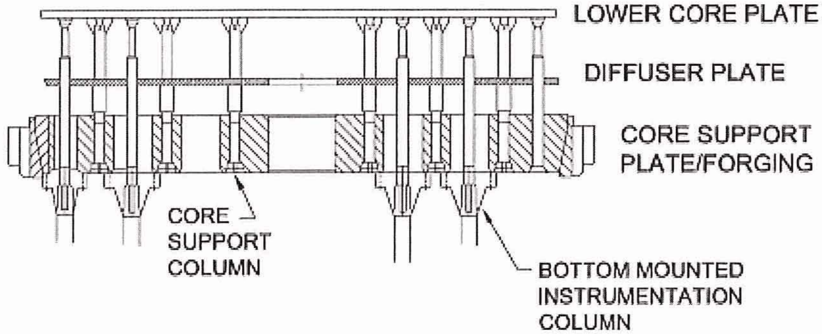


FIGURE 13

Westinghouse Lower Core Support Structure And Bottom Mounted Instrumentation Columns. Core Support Column Bolts Fasten The Core Support Columns To The Lower Core Plate.



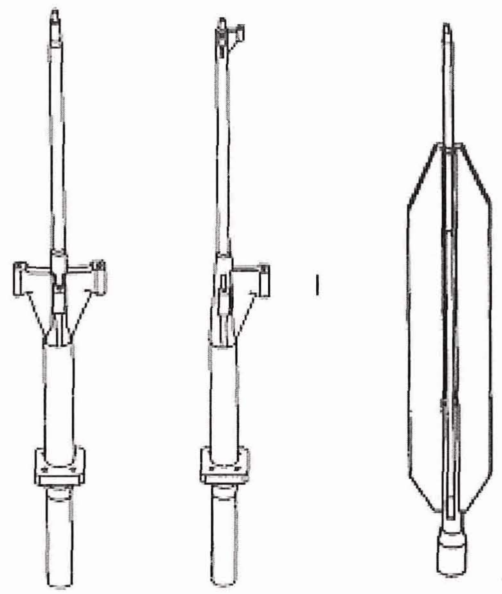
ATTACHMENT A

FIGURE 14

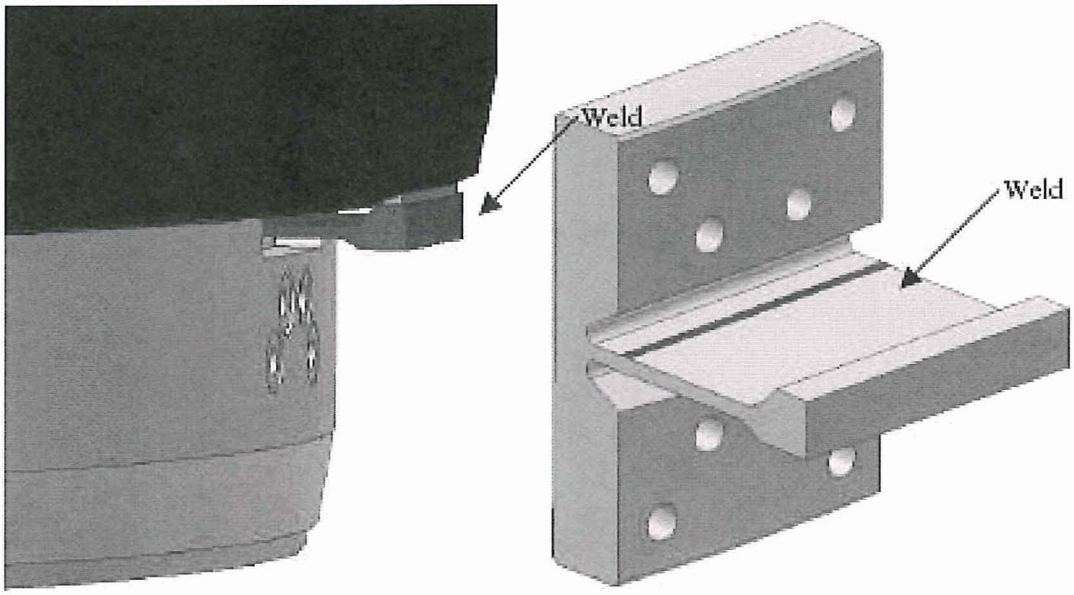
Typical Westinghouse Core Support Column. Core Support Column Bolts Fasten The Top Of The Support Column To The Lower Core Plate

FIGURE 15

Examples Of Westinghouse Bottom Mounted Instrumentation Column Designs



ATTACHMENT A
FIGURE 16
Typical Westinghouse Thermal Shield Flexure



ATTACHMENT B
WESTINGHOUSE PLANTS PRIMARY COMPONENTS

Item	Schedule	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Control Rod Guide Tube Assembly Guide plates (cards)	Unit 1 – 2008 Unit 2 – 2015	Loss of Material (Wear)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period, and no earlier than two refueling outages prior to the start of the license renewal period. Subsequent examinations are required on a ten-year interval.	20% examination of the number of CRGT assemblies, with all guide cards within each selected CRGT assembly examined. See Figure 2
Control Rod Guide Tube Assembly Lower flange welds and adjacent base metal	Unit 1 – 2013 Unit 2 - 2015	Cracking (SCC, Fatigue) Aging Management (IE and TE)	Bottom-mounted instrumentation (BMI) column bodies, Lower support column bodies (cast) Upper core plate, Lower support forging/casting	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds and adjacent base metal no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal on the individual periphery CRGT assemblies. (Note 5) See Figure 3
Core Barrel Assembly Upper core barrel flange weld	Unit 1 – 2013 Unit 2 - 2015	Cracking (SCC)	Lower support column bodies (non cast) Core barrel outlet nozzle welds	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal. (Note 3) See Figure 4
Core Barrel Assembly Upper and lower core barrel cylinder girth weld	Unit 1 – 2013 Unit 2 - 2015	Cracking (SCC, IASCC, Fatigue)	Upper and lower core barrel cylinder axial welds	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal. (Note 3) See Figure 4
Core Barrel Assembly Lower core barrel flange weld (Note 4)	Unit 1 – 2013 Unit 2 - 2015	Cracking (SCC, Fatigue)	None	Periodic enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal. (Note 3) See Figure 4

ATTACHMENT B
WESTINGHOUSE PLANTS PRIMARY COMPONENTS

Item	Schedule	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Baffle-Former Assembly Baffle-edge bolts	Unit 1 – 2013 Unit 2 - 2015	Cracking (IASCC, Fatigue) that results in • Lost or broken locking devices • Failed or missing bolts • Protrusion of bolt heads Aging Management (IE and ISR) (Note 6)	None	Visual (VT-3) examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Bolts and locking devices on high fluence seams. 100% of components accessible from core side. (Note 2) See Figure 5
Baffle-Former Assembly Baffle-former bolts	Unit 1 – 2013 Unit 2 - 2015	Cracking (IASCC, Fatigue) Aging Management (IE and ISR) (Note 6)	Lower support column bolts, Barrel-former bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examinations on a ten-year interval.	100% of accessible bolts (Note 2). Heads accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. See Figures 5 and 6
Baffle-Former Assembly Assembly (Includes: Baffle plates, baffle edge bolts and indirect effects of void swelling in former plates).	Unit 1 – 2013 Unit 2 – 2015	Distortion (Void Swelling), or Cracking (IASCC) that results in: • Abnormal interaction with fuel assemblies • Gaps along high fluence baffle joint • Vertical displacement of baffle plates near high fluence joint • Broken or damaged edge bolt locking systems along high fluence baffle joint.	None	Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Core side surface as indicated. See Figures 6, 7, 8 and 9

ATTACHMENT B
WESTINGHOUSE PLANTS PRIMARY COMPONENTS

Item	Schedule	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Alignment and Interfacing Components Internals hold down spring (304 SS)	N/A - Point Beach hold down springs are 403 SS.	Distortion (Loss of Load)	None	N/A	N/A See Figure 10
Thermal Shield Assembly Thermal shield flexures	Unit 1 – 2013 Unit 2 – 2015	Cracking (Fatigue) or Loss of Material (Wear) that results in thermal shield flexures excessive wear, fracture, or complete separation	None	Visual (VT-3) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of thermal shield flexures. See Figures 11 and 16

Notes:

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

ATTACHMENT C
WESTINGHOUSE PLANTS EXPANSION COMPONENTS

Item	Schedule	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Upper Internals Assembly Upper core plate	Based on results of Primary Link Components	Cracking (Fatigue, Wear)	CRGT lower flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2).
Lower Internals Assembly Lower support forging or castings	Based on results of Primary Link Components	Cracking Aging Management (TE in Casting)	CRGT lower flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2).
Core Barrel Assembly Barrel-former bolts	Based on results of Primary Link Components	Cracking (IASCC, Fatigue) Aging Management (IE, Void Swelling and ISR)	Baffle-former bolts	Volumetric (UT) examination Re-inspection every 10 years following initial inspection..	100% of accessible bolts. Accessibility may be limited by presence of thermal shields of neutron pads. (Note 2) See Figure 5
Lower Support Assembly Lower support column bolts	Based on results of Primary Link Components	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	Baffle-former bolts	Volumetric (UT) examination Re-inspection every 10 years following initial inspection..	100% of accessible bolts or as supported by plant-specific justification. (Note 2) See Figures 12 and 13
Core Barrel Assembly Core barrel outlet nozzle welds	Based on results of Primary Link Components	Cracking (SCC, Fatigue) Aging Management (IE of lower sections)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal. (Note 2) See Figure 4

ATTACHMENT C
WESTINGHOUSE PLANTS EXPANSION COMPONENTS

Item	Schedule	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Barrel Assembly Upper and lower core barrel cylinder axial welds	Based on results of Primary Link Components	Cracking (SCC, IASCC) Aging Management (IE)	Upper and lower core barrel cylinder girth welds	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of one side of the accessible surfaces of the selected weld and adjacent base metal. (Note 2) See Figure 4
Lower Support Assembly Lower support column bodies (non cast)	Based on results of Primary Link Components	Cracking (IASCC) Aging Management (IE)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection	100% of accessible surfaces. (Note 2) See Figures 13 and 14
Lower Support Assembly Lower support column bodies (cast)	N/A – Point Beach lower support column bodies are forgings.	Cracking (IASCC) including the detection of fractured support columns Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection	100% of accessible surfaces. (Note 2) See Figures 13 and 14
Bottom Mounted Instrumentation System Bottom-mounted instrumentation (BMI) column bodies	Based on results of Primary Link Components	Cracking (Fatigue) including the detection of completely fractured column bodies Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (VT-3) examination of BMI column bodies as indicated by difficulty of insertion/withdrawal of flux thimbles. Re-inspection every 10 years following initial inspection Flux thimble insertion/withdrawal to be monitored at each inspection interval.	100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal. See Figures 13 and 15

Notes:

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

ATTACHMENT D
WESTINGHOUSE PLANTS EXISTING PROGRAM COMPONENTS

Item	Schedule	Effect (Mechanism)	Reference	Examination Method	Examination Coverage
Core Barrel Assembly Core barrel flange	Unit 1 – Spring 2010 Unit 2 – Fall 2009	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination to determine general condition for excessive wear.	All accessible surfaces at specified frequency.
Upper Internals Assembly Upper support ring or skirt	Unit 1 – Spring 2010 Unit 2 – Fall 2009	Cracking (SCC, Fatigue)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces at specified frequency.
Lower Internals Assembly Lower core plate XL lower core plate (XL= 14ft. Core) (Note 1)	Unit 1 – Spring 2010 Unit 2 – Fall 2009	Cracking (IASCC, Fatigue) Aging Management (IE)	ASME Code Section XI	Visual (VT-3) examination of the lower core plates to detect evidence of distortion and/or loss of bolt integrity.	All accessible surfaces at specified frequency.
Lower Internals Assembly Lower core plate XL lower core plate (XL= 14ft. Core) (Note 1)	Unit 1 – Spring 2010 Unit 2 – Fall 2009	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces at specified frequency.
Bottom Mounted Instrumentation System Flux thimble tubes	Unit 1 – Spring 2010 Unit 2 – Fall 2009	Loss of material (Wear)	NUREG-1801 Rev. 1	Surface (ET) examination	Eddy current surface examination as defined in plant response to IEB 88-09.
Alignment and Interfacing Components Clevis insert bolts	Unit 1 – Spring 2010 Unit 2 – Fall 2009	Loss of material (Wear) (Note 2)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces at specified frequency.

ATTACHMENT D
WESTINGHOUSE PLANTS EXISTING PROGRAM COMPONENTS

Item	Schedule	Effect (Mechanism)	Reference	Examination Method	Examination Coverage
Alignment and Interfacing Components Upper core plate alignment pins	Unit 1 – Spring 2010 Unit 2 – Fall 2009	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces at specified frequency.

Notes:

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

ATTACHMENT E
EXAMINATION ACCEPTANCE AND EXPANSION CRITERIA

Item	Shown In	Examination Acceptance Criteria (Note 1)	Expansion Links(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Control Rod Guide Tube Assembly Guide plates (cards)	Figure 2	Visual (VT-3) examination The specific relevant condition is wear that could lead to loss of control rod alignment and impede control assembly insertion.	None	N/A	N/A
Control Rod Guide Tube Assembly Lower flange welds	Figure 3	Enhanced visual (EVT-1) examination The specific relevant condition is a detectable crack-like surface indication.	a. Bottom-mounted instrumentation (BMI) column bodies b. Lower support column bodies (cast), upper core plate and lower support forging or casting	a. Confirmation of surface-breaking indications in two or more CRGT lower flange welds, combined with flux thimble insertion/withdrawal difficulty, shall require visual (VT-3) examination of BMI column bodies by the completion of the next refueling outage. b. Confirmation of surface-breaking indications in two or more CRGT lower flange welds shall require EVT-1 examination of cast lower support column bodies, upper core plate and lower support forging/casting within three fuel cycles following the initial observation.	a. For BMI column bodies, the specific relevant condition for the VT-3 examination is completely fractured column bodies. b. For cast lower support column bodies, upper core plate and lower support forging/casting, the specific relevant condition is a detectable crack-like surface indication.

ATTACHMENT E
EXAMINATION ACCEPTANCE AND EXPANSION CRITERIA

Item	Shown In	Examination Acceptance Criteria (Note 1)	Expansion Links(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Upper core barrel flange weld	Figure 4	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	a. Core barrel outlet nozzle welds b. Lower support column bodies (non cast)	a. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel flange weld shall require that the EVT-1 examination be expanded to include the core barrel outlet nozzle welds by the completion of the next refueling outage. b. If extensive cracking in the core barrel outlet nozzle welds is detected, EVT-1 examination shall be expanded to include the upper six inches of the accessible surfaces of the non-cast lower support column bodies within three fuel cycles following the initial observation.	a and b. The specific relevant condition for the expansion core barrel outlet nozzle weld and lower support column body examination is a detectable crack-like surface indication.
Core Barrel Assembly Lower core barrel flange weld	Figure 4	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	None	None	None

ATTACHMENT E
EXAMINATION ACCEPTANCE AND EXPANSION CRITERIA

Item	Shown In	Examination Acceptance Criteria (Note 1)	Expansion Links(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Upper core barrel cylinder girth welds	Figure 4	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Upper core barrel cylinder axial welds	The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the upper core barrel cylinder axial welds by the completion of the next refueling outage.	The specific relevant condition for the expansion upper core barrel cylinder axial weld examination is a detectable crack-like surface indication.
Core Barrel Assembly Lower core barrel cylinder girth welds	Figure 4	Periodic enhanced visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Lower core barrel cylinder axial welds	The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the lower core barrel cylinder axial welds by the completion of the next refueling outage.	The specific relevant condition for the expansion upper core barrel cylinder axial weld examination is a detectable crack-like surface indication.
Baffle-Former Assembly Baffle-edge bolt	Figures 5 and 6	Visual (VT-3) examination. The specific relevant conditions are missing or broken locking devices, failed or missing bolts, and protrusion of bolt heads.	None	N/A	N/A

ATTACHMENT E
EXAMINATION ACCEPTANCE AND EXPANSION CRITERIA

Item	Shown In	Examination Acceptance Criteria (Note 1)	Expansion Links(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Baffle-Former Assembly Baffle-former bolts	Figures 5 and 6	Volumetric (UT) examination. The examination acceptance criteria for the UT of the baffle-former bolts shall be established as part of the examination technical justification.	a. Lower support column bolts b. Baffle-former bolts	a. Confirmation that more than 5% of the baffle-former bolts actually examined on the four baffle plates at the largest distance from the core (presumed to be the lowest dose locations) contain unacceptable indications shall require UT examination of the lower support column bolts within the next three fuel cycles. b. Confirmation that more than 5% of the lower support column bolts actually examined contain unacceptable indications shall require UT examination of the baffle-former bolts.	a and b. The examination acceptance criteria for the UT of the lower support column bolts and the baffle-former bolts shall be established as part of the examination technical justification.
Baffle-Former Assembly Assembly	Figures 5, 6, 7, 8 and 9	Visual (VT-3) examination. The specific relevant conditions are evidence of abnormal interaction with fuel assemblies, gaps along high fluence shroud plate joints, vertical displacement of shroud plates near high fluence joints, and broken or damaged edge bolt locking systems along high fluence baffle plate joints.	None	N/A	N/A

ATTACHMENT E
EXAMINATION ACCEPTANCE AND EXPANSION CRITERIA

Item	Shown In	Examination Acceptance Criteria (Note 1)	Expansion Links(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Alignment and Interfacing Components Internals hold down spring	N/A - Point Beach hold down springs are 403 SS.	Direct physical measurement or spring height.	None	N/A	N/A
Thermal Shield Assembly Thermal shield flexures	Figure 11	Visual (VT-3) examination. The specific relevant conditions for thermal shield flexures are excessive wear, fracture or complete separation.	None	N/A	N/A

Notes:

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