Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

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U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Response to Requests for Additional Information for the
Review of the South Texas Project, Units 1 and 2,
License Renewal Application – Set 28 (TAC Nos. ME4936 and ME4937)

References: 1. Letter from G. T. Powell, STP, to NRC Document Control Desk, "License Renewal Application", dated October 25, 2010 (NOC-AE-10002607) (ML103010257)

2. Letter from NRC to STP, "Requests for Additional Information for the Review of the South Texas Project, Units 1 and 2, License Renewal Application – Set 28", dated July 16, 2014, (TAC Nos. ME4936 and ME4937) (AE-NOC-14002550) (ML14183B719)

By Reference 1, STP Nuclear Operating Company (STP) submitted a License Renewal Application (LRA) for South Texas Project Units 1 and 2.

By Reference 2, the NRC staff requested additional information for their review of the STP LRA. Requested information is related to STPs Aging Management of Reactor Vessel Internals, MRP-227A, and Reactor vessel clevis assemblies. STPNOC's response to the requests for additional information is provided in Enclosure 1 to this letter. Changes to LRA pages described in Enclosure 1 are depicted as line-in/line-out pages provided in Enclosure 2.

Enclosure 3 provides a new LRA section Appendix C, Response to Applicant Action Items for Inspection and Evaluation Guidelines for PWR Internals and STP's Reactor Vessel Internals Program Inspection Plan.

There are no new regulatory commitments in this letter.

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NRR

Should you have any questions regarding this letter, please contact either Arden Aldridge, STP License Renewal Project Lead, at (361) 972-8243 or Rafael Gonzales, STP License Renewal Project regulatory point-of-contact, at (361) 972-4779.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on June 30, 2015
Date

G. T. Powell
Site Vice President

RJG

Enclosures:
1. STPNOC Response to Requests for Additional Information
2. STPNOC LRA Changes with Line-in/Line-out Annotations
3. New LRA Appendix C, Response to Applicant Action Items for Inspection and Evaluation Guidelines for PWR Internals and STP's Reactor Internals Program Inspection Plan

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Enclosure 1

STPNOC Response to Requests for Additional Information

SOUTH TEXAS PROJECT, UNITS 1 AND 2
REQUEST FOR ADDITIONAL INFORMATION, SET 28
(TAC NOS. ME4936 AND ME4937)

RAI B2.1.35-1 – Aging Management of Reactor Vessel Internals and MRP-227A

Background:

The license renewal application (LRA) for South Texas Project (STP), Units 1 and 2, proposed aging management for the reactor vessel internal (RVI) components based on a regulatory commitment in the LRA's Updated Final Safety Analysis Report (UFSAR) Supplement. The commitment stated that the applicant will develop an aging management program (AMP) and inspection plan based on augmented inspection activities for the components developed by the EPRI Materials Reliability Project (MRP), and that the inspection plan will be submitted for NRC review and approval at least two (2) years prior to entering into the period of extended operation for STP, Units 1 and 2.

The NRC's recommended AMP for Pressured Water Reactor (PWR) RVIs in Revision 2 of NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," is given in Section XI.M16A, "PWR Vessel Internals," which was issued in December 2010. On January 9, 2012, subsequent to the issuance of Revision 2 of the GALL Report, the EPRI MRP issued Technical Report No. 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," which included the NRC safety evaluation (SE) on the report's methodology dated December 16, 2011. On June 3, 2013, the staff revised AMP XI.M16A and the aging management review (AMR) items in the GALL Report for PWR RVI components to be consistent with the contents of the MRP-227-A report and issued them in License Renewal Interim Staff Guidance Document No. LR-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal Components of Pressurized Water Reactors."

On July 21, 2011, the NRC issued Regulatory Information Summary (RIS) 2011-07, "License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management," which provided updated NRC procedures for LRA reviews of PWR RVI AMPs. This RIS identifies Category C plants as those plants that have an LRA currently under review, and states that these applicants will be expected to revise their commitment for aging management of PWR vessel internals such that the information identified in the SE for MRP-227 would be submitted to the NRC for review and approval not later than two years after issuance of the renewed license or not later than two years before the plant enters the period of extended operation, whichever comes first. STP, Units 1 and 2, are Category C plants in accordance with the RIS.

Issue:

The categorization of STP, Units 1 and 2, and other plants in Category C of RIS 2011-07 was based on an expectation that the LRA would be reviewed and approved on a normal review schedule of 22 months, and that it would be an unreasonable burden to expect the applicant to address all aspects of the NRC's SE for MRP-227 within the LRA review. However, the applicant requested that the NRC place the LRA review on hold for the year 2013 to allow the applicant to address plant-specific technical issues. The staff noted that, as of the date of this RAI, the review of these technical issues is still on-going. Since the

resolution of these issues is not imminent and completion of the staff's review of the LRA is on-going, the staff has concluded that the applicant should provide an LRA update or amendment that includes updated AMP and AMR items for the RVI components, including responses to the applicable Applicant/License Actions Items identified in the staff's SE for MRP-227.

In addition to the issues identified above, the staff has noted that recent operating experience at one 4-loop PWR (in 2010) reported cracking and failures of some nickel alloy (Alloy X-750) clevis insert bolts in the clevis assemblies attached to the lower internal portion of its reactor vessel (see EPRI Technical Report No. 1022863, "Materials Reliability Program: Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines (MRP-227-A)," Appendix A). The staff further noted that some inspection routines (such as an ASME Section XI visual VT-3 inspection on a 10-year frequency) of the clevis insert assemblies may not be adequate to ensure the integrity of clevis insert assemblies during design basis events if multiple bolt failures occur prior to detection, and then the design basis event were to occur.

Request:

The staff requests that STPNOC provide the following:

1. MRP-227-A Applicant/Licensee Action Items (A/LAIs): Provide either an LRA amendment or update that includes an updated AMP, updated AMR items, and any applicable inspection plan(s) for the PWR RVI components at STP, Units 1 and 2, that are based on the guidance in LR-ISG-2011-04, including responses to applicable A/LAIs identified in the staff's SE for MRP-227, dated December 16, 2011.

STPNOC Response:

STP has updated the PWR Reactor Internals Program to address MRP-227-A Applicant/Licensee Action Items (A/LAIs), update AMR tables, and Aging Management Program, and provided a program inspection plan for NRC review and approval, based on the guidance in LR-ISG-2011-04 and the NRC staff's Safety Evaluation for MRP-227.

LRA Appendix B2.1.35 and LR Basis Document AMP PWRI, PWR Reactor Internals program are revised to align the Scope of Program Element 1 component names with MRP-227-A and LR-ISG-2011-04 Table 3.1-1.

LRA Table of Contents is revised to include Appendix C, Response to Applicant Action Items for Inspection and Evaluation Guidelines for PWR Internals.

LRA Section 3.1.2.1.1 is revised to add Neutron Flux to the list of environments.

Further Evaluations LRA Sections 3.1.2.2.6, 3.1.2.2.9, 3.1.2.2.12, and 3.1.2.2.15 are no longer applicable because STP PWR Reactor Internals program (B2.1.35) relies on implementation of the inspection and evaluation guidelines in EPRI TR-1022863 (MRP-227-A) and EPRI TR-1016609 (MRP-228) to manage the aging effects of the reactor vessel internal.

LRA Table 3.1.1 is revised to include the verbiage from LR-ISG-2011-04 for the Table 3.1.1 Item Numbers 3.1.1.22, 3.1.1.27, 3.1.1.30, 3.1.1.33, 3.1.1.37, 3.1.1.60, 3.1.1.63 and 3.1.1.80. The corresponding IDs from LR-ISG-2011-04 Table 3.1-1 are 59c, 59b, 53a, 59a, 53c, 54, 32, 53b.

The AMR lines from the original LRA Table 3.1.2-1 pertaining to the reactor vessel internals are deleted and new lines are added using LR-ISG-2011-04 Table IV.B2 Item numbers for the reactor vessel internals components applicable to STP.

Appendix C is added to provide the STP response to the Applicant Action Items for Inspection and Evaluation Guidelines for PWR Internals and STP's Reactor Vessel Internals Program Inspection Plan.

Enclosure 2 provides the line-in/line-out revision to LRA Appendix B2.1.35, LRA Table of Contents, LRA Sections 3.1.2.1.1, 3.1.2.2.6, 3.1.2.2.9, 3.1.2.2.12, and 3.1.2.2.15 and Tables 3.1.1 and 3.1.2-1.

Enclosure 3 provides a new LRA section Appendix C, Response to Applicant Action Items for Inspection and Evaluation Guidelines for PWR Internals and STP's Reactor Vessel Internals Program Inspection Plan.

Request:

2. Reactor vessel clevis assemblies: Please address the clevis insert bolt operating experience issue cited above in reference to STP:
   a) Describe the configuration of clevis insert assemblies at STP, including number of bolts in the assemblies. Specify the materials of fabrication, including any applicable heat treatments that were used for the design of the clevis insert bolts at STP.
   b) Discuss and justify whether the operating experience associated with cracking of the clevis insert bolts is applicable to the clevis insert assembly designs at STP.
   c) Describe the inspections that have been performed of the clevis insert bolts, including the type of inspection (e.g., VT-3). Discuss the visual inspection coverage that was achieved during these inspections. Clarify the ASME examination category that applies to inspections of the clevis insert bolts (and identify the applicable inspection method and frequency) and whether any past examinations have resulted in the detection of any indications of cracking or failures of the clevis insert bolts that are included in the clevis insert assembly designs. If so, provide the details of the inspection results and clarify the corrective actions that were taken at the facility to justify the structural integrity of the clevis insert assemblies and the intended safety function of the plant's core support structure and its components during plant operations.
   d) Based on your responses to Parts (a) through (c) of this request, clarify whether the 10-year ISI basis (or the current approach used for STP) for the clevis insert bolts is sufficient to manage cracking and wear of the bolts during the period of extended operation. Justify your response to this request.

STPNOC Response:

(a) The clevis insert assemblies at South Texas Units 1 and 2 are comprised of eight clevis insert cap screws fabricated from Inconel® X-750, SA-637, Grade 688, Type 2, two dowel pins fabricated from Alloy 600 and an Alloy 600 hard-faced U-shaped clevis insert that rests onto the clevis locations in the reactor vessel. In total there are six clevis locations (0, 60, 120, 180, 240 and 300 degrees). The clevis insert cap screws include a lock bar that is welded to the clevis insert face. The clevis insert cap screws are attached to the interior face of the clevis insert assembly. See Figure 2-1 for a comparison of the clevis insert assemblies for STP and the reference plant that had clevis insert cap screw cracking.

The South Texas material used for the clevis insert cap screws is similar to that used by the operating experience referenced plant where cracking was observed. The reference plant cap screw material heat treatment consisted of equalization heat treatment followed by a one-hour solution heat treatment. The precipitation hardening temperatures are the same, but the cooling time between temperatures for the reference plant is longer. For either type of heat treatment, susceptibility to primary water stress corrosion cracking is known to exist. STP specific heat treatment values for the above methods are Westinghouse proprietary information but can be made available upon request. The cap screws are of the same design, except that the South Texas cap screw shank length is slightly longer. The cap screws were installed with the same torque as that used for the referenced plant.

(b) The main function of the Lower Radial Support System (LRSS) is to prevent tangential or rotational motion of the lower internals assembly while permitting axial displacement and differential radial expansion. These supports are designed to prevent excessive lateral and rotational displacement of the lower internals during seismic and loss-of coolant accident (LOCA) conditions. The supports also limit displacements and misalignments in order to avoid overstressing the core barrel and to ensure that the control rods can be freely inserted. Therefore, assuming the clevis inserts remain in place as limited by the adjoining radial keys and support lugs, the design function of the LRSS will be maintained during seismic and LOCA conditions.

South Texas Units 1 and 2 have six radial supports spaced at 60-degree intervals around the circumference of the vessel. Although labeled as radial supports, the supports actually support the core barrel only in the tangential direction because the tangential clearances between the core barrel keys and the vessel clevis inserts are much smaller than the radial clearances. This basic arrangement is the same for South Texas units and the reference plant where clevis insert cap screw cracking was observed; however, the clevis insert designs are different. See Figure 2-1 for this comparison. The same number of eight cap screws is arranged in the same two vertical columns of four cap screws each. Two interference-fit dowel pins of the same size are located in-line with the cap screws in the same manner as the reference plant. The main design difference is that the South Texas reactor clevis insert is U-shaped, with the cap screws located inboard of the "U"; whereas the reference plant insert, while also being U-shaped, has flanges on either side where the cap screws are located. The tangential interference fit of the insert against the support lug is at the ends of these flanges for the reference plant design and on the sides of the "U" for the South Texas reactor design. Therefore, the tangential interference-fit compression stiffness of the two inserts is different.

The clevis insert cap screws for the South Texas units are of a similar design, of the same material with very similar heat treatment, torqued to the same degree, and operated at a slightly hotter T_{cold} inlet temperature as compared to the reference plant, it is possible that these cap screws can experience primary water stress corrosion cracking (PWSCC) similar to that of the reference plant. Therefore, the operating experience relative to cracked clevis insert cap screws is applicable to South Texas units.

The structural evaluations performed by the reference plant to justify continued operation in the as-found condition demonstrated safe operation for an additional fuel cycle. The concern was possible long-term effects, such as the potential for vibratory loads to eventually cause loosening and wear of the insert and the subsequent increase in gaps between the insert, radial key and support lug. A similar review of the structural adequacy of the South Texas clevis insert design was performed to determine if broken cap screws present a structural concern for safe operation. The structural aspects and loose parts assessment for the South Texas Units is discussed below:

Clevis Support Lug Primary Stress
The clevis insert, if completely loose to slide radially inward, is captured in a manner similar to the reference plant and is restrained by a similar radial gap before it contacts the radial key. With the clevis insert displaced fully inward, the primary stresses on the clevis support lugs remain acceptable relative to the reactor vessel original ASME 1971 Edition (through 1973 Summer Addenda) code of construction under plant-specific maximum upset and faulted condition loads due to seismic and LOCA conditions.

Clevis Insert Primary Plus Secondary Stress
The increase in insert stress due to broken cap screws remains acceptable because the fatigue analysis maintains an acceptable usage factor less than 1.0.

Cap Screw Primary Plus Secondary Stress
Where one column of cap screws is entirely broken, the resulting cap screw stress produced by this prying load on the insert is acceptable with four intact screws. During heatup and steady-state, the clevis insert remains preloaded against the support lug, and this type of loading on the intact screws will not occur.

Clevis Insert Restraining Force (No Cap Screws)
If all of the cap screws are broken, and no restraint by the dowel pins is assumed, the clevis insert will maintain preload during steady-state operation and will maintain its ability to perform its intended function. Long-term wear between the insert and support lug is not expected or would be insignificant. The clevis insert is restrained tangentially by the support lugs and restrained radially by the limited clearance with the radial key. In addition, the insert has a thick upper flange that prevents it from falling downward, and the downward force from the downcomer flow will prevent it from working upward. The design installed at South Texas Units 1 and 2, which maintains a greater amount of preload between the insert and support lug as compared to the referenced plant, longer operation can be maintained before discernable degradation occurs.

During core barrel removal at cold conditions, the interference fit of the insert provides greater frictional force than the applied frictional force produced by the key sliding upward against the insert. The two dowel pins will also provide additional vertical constraint of the insert. Therefore, the clevis insert design prevents separation of the insert during core barrel removal operations if the cap screws (and dowel pins) are nonfunctional.

Loose Parts Assessment
The insert cap screws have the same head design and locking device design as the reference plant. A lock bar is installed in a groove in the cap screw head, and the bar is welded to the insert counterbore where the cap screw is inserted. If a cap screw head should separate, the lock bar can wear and separate over time, causing the cap screw head to be loose in the counterbore recess. The maximum design gaps between the core barrel radial keys and the inserts are less than the height of the cap screw heads. Therefore, the cap screw heads remain captured, unless, over a long period of time, wear of the heads reduces the height of the heads by this amount. The cap screw head wear is expected to be small because the cap screw material is much harder than the clevis insert and radial key material.

The potential for loose parts due to wear-related degradation of the lock bars related to failed clevis insert cap screws at the degraded cap screw locations was evaluated. South Texas and the reference plant have different lower internal designs; however, the effects of where these loose parts would be captured or would impact against the lower internals are the same. Therefore, no significant degradation of mechanical components is expected as a result of the potential presence of loose parts from the lock bars in the primary system.

(c) STP performs remote visual examinations on 100% of accessible ASME Section XI Category B-N-1, B-N-2 and B-N-3 examination areas in accordance with ASME Section XI B-N-2 code categorization. As part of these examinations the clevis insert bolting, pins, and welded lock bars are verified intact. The examinations are performed with a submersible mini-submarine (mini-sub) with an attached color camera and associated support equipment. The most recent inspections were performed for Unit 1 in 2009 during 1RE015 and for Unit 2 in 2010 during 2RE014. 100% of the clevis insert bolt heads, pins, and lock bars were observed in conjunction with the B-N-2 inspection and no degradation or damage was identified. There are no STP inspection results that identified similar clevis insert bolt head detachment, as was identified at the referenced plant.

(d) Based on the structural evaluations above and operation with potential loose parts of the type and quantities that are no different than have already been evaluated, safe operation of the reactors and primary systems at South Texas Units 1 and 2 is assured. The ability of the Lower Radial Support System (LRSS) to perform its intended design function under seismic and LOCA condition loadings is unrelated to the integrity of the cap screws and dowel pins that are used to hold the clevis insert in place. Even if all of the cap screws and dowel pins separate, complete disengagement of one of the clevis inserts will not occur because of the small size of the gaps between the clevis inserts and radial keys. Wear or some degradation of a key might occur, but the key would still be expected to maintain functionality. Taken as a whole, the core barrel and LRSS are expected to maintain their design function with degraded clevis insert bolts.

Augmentation of the reactor internals inspection program to include crack detection prior to cap screw failure is not required due to inherent design redundancy as discussed above. The effects of wear and/or looseness of the insert if the cap screws should become degraded, should not significantly affect the preload between the insert and support lug during steady-state operation. The Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A) categorization for wear-only is based on the primary concern for clevis insert looseness and wear of the clevis insert and radial key interfacing surfaces that could potentially lead to increased motion at the bottom end of the core barrel, rather than bolt material cracking. PWSCC was considered and screened in the Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191). Actions to address PWSCC are included in MRP-227-A. Manifestation of cap screw cracking is identified as a result of the observation of wear (see note 2 of Table 4-9 in MRP-227-A).

Existing inspections are already in place to account for this concern. Qualified personnel at South Texas performing video camera inspections at 10-year intervals, as specified in ASME Code Section XI and MRP-227-A, are capable of identifying wear or dislodged components of the clevis insert cap screws or dowel pins at any location.

Visual inspection at 10-year intervals can also detect wear and displacement of the clevis insert. Inspection of the insert and key contact surfaces can detect wear in adjacent non-contact surfaces. If cap screw heads are observed to be loose, any displacement of the insert relative to the vessel support lug can be easily observed. During the last in-service inspections at the South Texas units in 2010, no indications of loosening or adverse wear were observed. Based on the above considerations and observations, it is concluded that the current in-service inspections to examine the clevis insert (and LRSS) with VT-3 is adequate without augmentation.

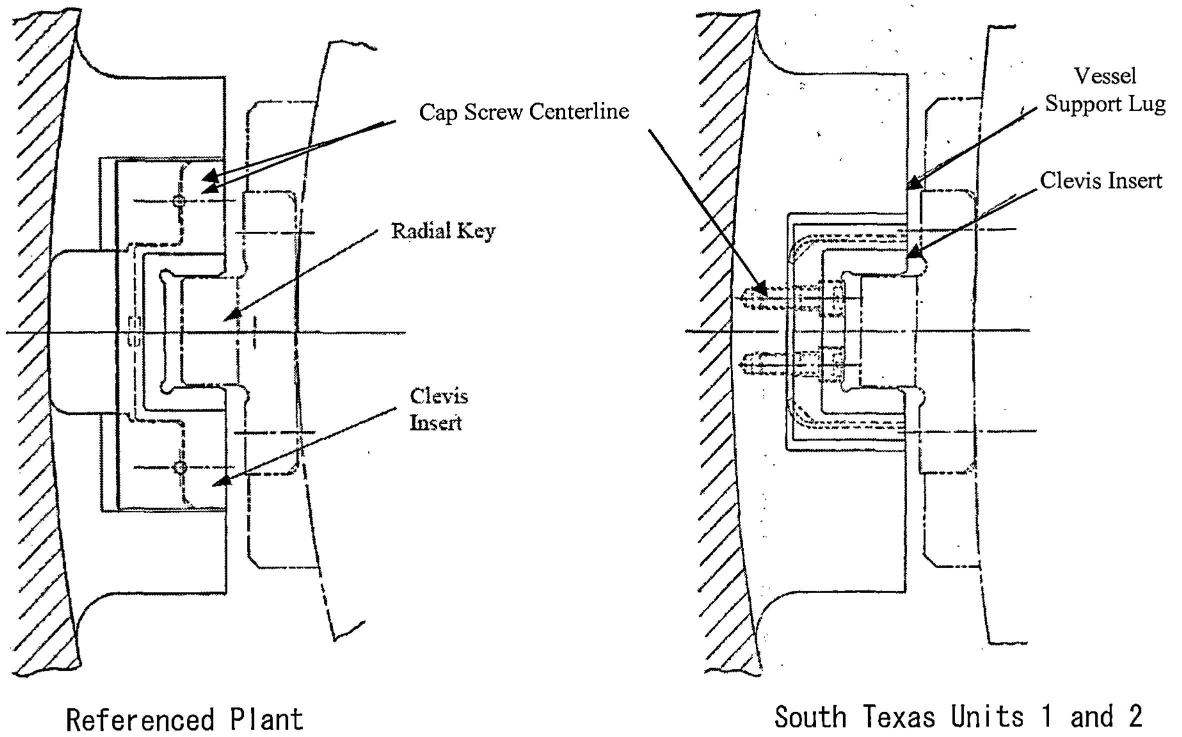

Figure 2-1: Lower Radial Support Comparison

Enclosure 2

STPNOC LRA Changes with Line-in/Line-out Annotations

B2.1.35 PWR Reactor Internals

Program Description

The PWR Reactor Internals program manages cracking, loss of material, loss of fracture toughness, dimensional changes, and loss of preload for reactor vessel components that provide a core structural support intended function. The program implements the guidance of EPRI 1022863, *PWR Internals Inspection and Evaluation Guideline* (MRP-227-A, Rev .0) and EPRI 1016609, *Inspection Standard for PWR Internals* (MRP-228, Rev. 0). The program manages aging consistent with the inspection guidance for Westinghouse designated primary components in Table 4-3 of MRP-227-A and Westinghouse designated expansion components in Table 4-6 of MRP-227-A, and the Westinghouse designated existing components in Table 4-9 of MRP-227-A. Primary components are expected to show the leading indications of the degradation effects. The expansion components are specified to expand the primary component sample should the indications of the sample be more severe than anticipated. The aging effects of a third set of MRP-227-A internals locations are deemed to be adequately managed by existing program components whose aging is managed consistent with ASME Section XI Table IWB-2500-1, Examination Category B-N-3.

Program examination methods include visual examination (VT-3), enhanced visual examination (EVT-1), volumetric examination, and physical measurements. Bolting ultrasonic examination technical justifications in MRP-228 have demonstrated the indication detection capability to detect loss of integrity of PWR internals bolts, pins, and fasteners, such as baffle-former bolting. For some components, the MRP-227-A methodology specifies a focused visual (VT-3) examination, similar to the current ASME Code, Section XI, Examination Category B-N-3 examinations, in order to determine the general mechanical and structural condition of the internals 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. In some cases, VT-3 visual methods are used for the detection of surface cracking when the component material has been shown to be tolerant of easily detected large flaws. In some cases, where even more stringent examinations are required, enhanced visual (EVT-1) examinations or ultrasonic methods of volumetric inspection, are specified for certain selected components and locations.

The program provides both examination acceptance criteria for conditions detected as a result of monitoring the primary components, as well as criteria for expanding examinations to the expansion components when warranted by the level of degradation detected in the primary components. Based on the identified aging effect, and supplemental examinations if required, the disposition process results in an evaluation and determination of whether to accept the condition until the next examination or implement corrective actions. Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection.

The PWR Vessel Internals program is a new program that has been implemented. The program will include future industry operating experience, as it is incorporated into the future revisions of MRP-227-A, to provide reasonable assurance for long-term integrity of the reactor internals. The reactor vessel internals included in the scope of the PWR Reactor Internals program are identified in Element 1. The scope of the program does not include welded attachments to the internal surface of the reactor vessel because these components are managed by the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program (B2.1.1) (exam category B-N-2) and /or the Nickel-Alloy Aging Management Program (B2.1.34). The scope of the program also does not include BMI flux thimble tubes which are managed by the Flux Thimble Tube Inspection program (B2.1.21).

Aging Management Program Elements

The results of an evaluation of each element against the 10 elements described in Appendix A of NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants* are provided below.

Scope of Program – Element 1

The scope of the program applies the guidance in MRP-227-A which provides augmented inspection and flaw evaluation methodology for assuring the functional integrity of Westinghouse reactor vessel internals. The scope of the PWR Reactor Internals program includes components that provide a core structural support intended function and are managed by the Westinghouse designated primary components in Table 4-3 of MRP-227-A and Westinghouse designated expansion components in Table 4-6 of MRP-227-A and applicable MRP-227-A methodology license renewal applicant action items. MRP-227-A Table 4-9 also identifies existing program components whose aging is managed consistent with ASME Section XI Table IWB-2500-1, Examination Category B-N-3.

Primary components are expected to show the leading indications of the degradation effects. The expansion components are specified to expand the primary component sample should the indications of the sample be more severe than anticipated. The aging effects of a third set of MRP-227-A internals locations are deemed to be adequately managed by existing program components whose aging is managed consistent with ASME Section XI Table IWB-2500-1, Examination Category B-N-3.

The STP reactor vessel internals are divided into the following major component groups: the lower core support assembly (including the entire core barrel assembly, baffle-former assembly, neutron shield panel, core support plate, and energy absorber assembly), the upper core support (UCS) assembly (including the upper support plate, support column, control rod guide tube assembly, upper core plate, and protective skirt), the incore instrumentation support structures (including the instrumentation columns (exit thermocouples), upper/lower tie plates, and instrumentation columns (BMI)), and miscellaneous alignment/interface components (including internals hold-down spring, upper core plate guide pins, and radial support keys including clevis inserts).

The following reactor vessel internals are included in the scope of the PWR Reactor Internals program:

1. Control rod guide tube assembly ~~and Bolting~~

- Guide plate (cards) [Primary component]
- Lower flange welds and adjacent base metal ~~(Addressed in AMR by Component Type of "RVI Control Rod Guide Tube Assembly")~~ [Primary component]
- Guide Tube Support Pins ~~(Split Pins) (Addressed in AMR by Component Type of "RVI Control Rod Guide Tube Bolting")~~ [Existing programs component]

2. Core barrel assembly

- Upper core barrel flange weld and adjacent base metal ~~(Addressed in AMR by Component Types of "RVI Core Barrel Assembly")~~ [Primary component]
- ~~Core barrel assembly–former bolting [Expansion component]~~
- Core barrel flange ~~(Addressed in AMR by Component Types of "RVI Core Barrel Assembly")~~ [Expansion component and Existing programs component]
- Core barrel vertical axial welds and adjacent base metal [Expansion component]
- Core barrel circumferential girth welds and adjacent base metal [Primary component]
- Core barrel outlet nozzle welds and adjacent base metal [Expansion component]
- Lower core barrel flange weld and adjacent base metal Addressed in AMR by Component Types of "RVI Core Barrel Assembly") [Primary component]

3. Baffle–former assembly ~~and bolting~~

- ~~Baffle-edge bolting [Primary component]~~
- Baffle–former bolting [Primary component]
- Barrel to Former Bolting [Expansion component]
- Baffle-former assembly baffle and former plates [Primary component]

4. Alignment and interfacing components

- Internals hold-down spring [Primary component]
- ~~Radial support key~~ Clevis insert bolts [Existing programs component]
- Upper core plate ~~guide~~ alignment pins [Existing programs component]

5. Bottom Mounted Instrumentation (BMI) Column assembly ~~support structures~~

- BMI ~~Instrumentation~~ columns bodies ~~– BMI~~ [Expansion component]

6. Upper ~~core support~~ Internals assembly

- Upper core support ~~protective~~ skirt [Existing programs component]
- Upper Core Plate [Expansion component]

7. Lower internal assembly ~~Core Support Structure~~

- XL lower core ~~Support~~ plate ~~Forging~~ [Expansion component]

The scope of the program also does not include welded attachments to the internal surface of the reactor vessel because these components are managed by the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD program (B2.1.1) (exam category B-N-2) and /or the Nickel-Alloy Aging Management Program (B2.1.34). The scope of the program also does not include BMI flux thimble tubes which are managed by the Flux Thimble Tube Inspection program (B2.1.21).

The STP reactor vessel internals configuration does not include the lower internals assembly (lower support column bodies and lower core plate) noted in MRP-227-A.

The PWR Reactor Internals program is consistent with the following MRP-227-A assumptions (determination of applicability) which are based on PWR representative internals configurations and operational histories.

(1) STP has operated for less than 30 years of operation with high leakage core loading patterns. Operation with high leakage core loading was followed by implementation of a low leakage fuel management pattern for the remaining operating life.

(2) STP operates at fixed power levels and does not usually vary power based on calendar or load demand schedule.

(3) STP has not implemented any design changes beyond those identified in industry guidance or recommended by Westinghouse.

Preventive Actions – Element 2

The PWR Reactor Internals program does not prevent degradation due to aging effects, but provides measures for monitoring to detect the degradation prior to loss of intended function. Preventive measures to mitigate aging effects such as loss of material and cracking include monitoring and maintaining reactor coolant water chemistry consistent with the guidelines of EPRI TR 1014986, *PWR Primary Water Chemistry Guidelines*, Volume 1. The primary water chemistry program is described separately in the Water Chemistry program (B2.1.2).

Parameters Monitored or Inspected – Element 3

The PWR Reactor Internals program monitors the following aging effects by inspection in accordance with the guidance of MRP-227-A or ASME Section XI Category B-N-3:

(1). Cracking
Cracking is due to stress corrosion cracking (SCC), primary water stress corrosion cracking (PWSCC), irradiation assisted stress corrosion cracking (IASCC), or fatigue

/cyclical loading. Cracking is monitored with a visual inspection for evidence of surface breaking linear discontinuities or a volumetric examination. Surface examinations may also be used to supplement visual examinations for detection and sizing of surface-breaking discontinuities.

(2). Loss of Material
Loss of Material is due to wear. Loss of material is monitored with a visual inspection for gross or abnormal surface conditions.

(3). Loss of Fracture Toughness
Loss of Fracture Toughness is due to thermal aging or neutron irradiation embrittlement. The impact of loss of fracture toughness is indirectly monitored by using visual or volumetric examination techniques to monitor for cracking and by applying applicable reduced fracture toughness properties in the flaw evaluations if cracking is detected and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation.

(4). Dimensional Changes
Dimensional Changes are due to void swelling and irradiation growth, distortion or deflection. The program supplements visual inspection with physical measurements to monitor for any dimensional changes due to void swelling, irradiation growth, distortion, or deflection.

(5). Loss of Preload
Loss of Preload is caused by thermal and irradiation-enhanced stress relaxation or creep. Loss of preload is monitored with a visual inspection for gross surface conditions that may be indicative of loosening in applicable bolted, fastened, keyed, or pinned connections.

The PWR Reactor Internals program manages the aging effects noted above consistent with the inspection guidance for Westinghouse designated primary components in Table 4-3 of MRP-227-A and Westinghouse designated expansion components in Table 4-6 of MRP-227-A. MRP-227-A also identifies Existing Program components whose aging is managed consistent with ASME Section XI Table IWB-2500-1, Examination Category B-N-3. See the component list in element 1 to identify Primary, Expansion, and Existing components.

Detection of Aging Effects – Element 4

The PWR Reactor Internals program detects aging effects through the implementation of the parameters monitored or inspected criteria and bases for Westinghouse designated Primary Components in Table 4-3 of MRP-227-A and for Westinghouse designated Expansion Components in Table 4-6 of MRP-227-A. The aging effects of a third set of MRP-227-A internals locations identified in Table 4-9 of MRP-227-A are deemed to be adequately managed by existing program components whose aging is managed consistent with ASME Section XI Table IWB-2500-1, Examination Category B-N-3.

One hundred percent of the accessible volume/area of each component will be examined for the Primary and Expansion components inspection category components. The minimum examination coverage for primary and expansion inspection categories is 75 percent of the component's total (accessible plus inaccessible) inspection area/volume be examined. When addressing a set of like components (e.g. bolting), the minimum

examination coverage for primary and expansion inspection categories is 75 percent of the component's total population of like components (accessible plus inaccessible).

If defects are discovered during the examination, STP enters the information into the STP corrective action program and evaluates whether the results of the examination ensure that the component (or set of components) will continue to meet its intended function under all licensing basis conditions of operation until the next scheduled examination. Engineering evaluations that demonstrate the acceptability of a detected condition will be performed consistent with WCAP-17096-NP.

Monitoring and Trending – Element 5

The program provides both examination acceptance criteria (See Element 6) for conditions detected as a result of monitoring the primary components as described in Element 4, as well as criteria for expanding examinations to the expansion components when warranted by the level of degradation detected in the primary components. Based on the identified aging effect, and supplemental examinations if required, the disposition process results in an evaluation and determination of whether to accept the condition until the next examination or implement corrective actions. Any detected conditions that do not satisfy the examination acceptance criteria (See Element 6) are required to be dispositioned through the corrective action program (See Element 7), which may require repair, replacement, or analytical evaluation for continued service until the next inspection.

Acceptance Criteria – Element 6

Examination acceptance for the primary and expansion component examinations are consistent with Section 5 of MRP-227-A. ASME Section XI section IWB-3500 acceptance criteria apply to Existing Programs components. The following examination acceptance criteria apply to the STP reactor vessel internals:

Visual examination (VT-3) and enhanced visual examination (EVT-1)

For existing program components, the ASME Code Section XI, Examination Category B-N-3 provides the following general relevant conditions for the visual (VT-3) examination of removable core support structures.

(1) Structural distortion or displacement of parts to the extent that component function may be impaired,

(2) Loose, missing, cracked, or fractured parts, bolting, or fasteners,

(3) Corrosion or erosion that reduces the nominal section thickness by more than 5 percent,

(4) Wear of mating surfaces that may lead to loss of function; and

(5) Structural degradation of interior attachments such that the original cross-sectional area is reduced more than 5 percent.

In addition, for the visual examinations (VT-3) of Primary and Expansion components, the PWR Reactor Internals program is consistent with the more specific descriptions of relevant conditions provided in Table 5-3 of MRP-227-A. EVT-1 examinations are used

for detecting small surface breaking cracks and surface crack length sizing when used in conjunction with sizing aids. EVT- 1 examination has been selected to be the appropriate NDE method for detection of cracking in plates or their welded joints. The relevant condition applied for EVT-1 examination is the same as found for cracking in ASME Section XI section 3500 which is crack-like surface breaking indications.

Volumetric examination

Individual bolts are accepted (pass/fail acceptance) based on the detection of relevant indications established as part of the examination technical justification. When a relevant indication is detected in the cross-sectional area of the bolt, it is assumed to be non-functional and the indication is recorded. Bolted assemblies are evaluated for acceptance based on meeting a specified number and distribution of functional bolts. Acceptance criteria for volumetric examination of STP reactor internals bolting are consistent with Table 5-3 of MRP-227-A.

Physical Measurements

~~Continued functionality of the internals hold down spring is confirmed by direct physical measurement. The examination acceptance criterion for this measurement is consistent with Table 5-3 of MRP-227-A and requires that the remaining compressible height of the spring shall provide hold-down forces within the plant-specific design tolerance.~~ Physical measurement of the internals hold down spring is not required because STP internals hold down spring is fabricated from 403 stainless steel.

Corrective Actions – Element 7

The following corrective actions are available for the disposition of detected conditions that exceed the examination acceptance criteria:

(1) Supplemental examinations to further characterize and potentially dispose of a detected condition consistent with Section 5.0 of MRP-227-A;

(2) Engineering evaluation that demonstrates the acceptability of a detected condition consistent with WCAP-17096-NP;

(3) Repair, in order to restore a component with a detected condition to acceptable status (ASME Section XI); or

(4) Replacement of a component with an unacceptable detected condition (ASME Section XI)

(5) Other alternative corrective action bases if previously approved or endorsed by the NRC.

Relevant indications failing to meet applicable acceptance criteria are repaired or replaced in accordance with plant procedures. Appropriate codes and standards are specified in both the "ASME Section XI Repair, Replacement, and Post-Maintenance Pressure Testing" procedure and in design drawings. Quality assurance requirements for repair and replacement activities are also included in the STP Operations Quality Assurance Plan.

STP site QA procedures, review and approval process, and administrative controls are implemented in accordance with the requirements of 10 CFR 50 Appendix B and are acceptable in addressing corrective actions. The QA program includes elements of corrective action, confirmation process and administrative controls, and is applicable to the safety-related and non-safety related systems, structures, and components that are subject to aging management review.

Confirmation Process – Element 8

STP site QA procedures, review and approval process, and administrative controls are implemented in accordance with the requirements of 10 CFR 50 Appendix B and are acceptable in addressing the confirmation process. The QA program includes elements of corrective action, confirmation process and administrative controls and is applicable to the safety-related and non-safety related systems, structures and components that are subject to aging management review.

Administrative Controls – Element 9

STP site QA procedures, review and approval process, and administrative controls are implemented in accordance with the requirements of 10 CFR 50 Appendix B and are acceptable in addressing administrative controls. The QA program includes elements of corrective action, confirmation process and administrative controls and is applicable to the safety-related and non-safety related systems, structures and components that are subject to aging management review.

Operating Experience — Element 10

Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. However, a considerable amount of PWR internals aging degradation has been observed in European PWRs, with emphasis on cracking of baffle-former bolting. The experience reviewed includes NRC Information Notice 84-18, Stress Corrosion Cracking (SCC) in PWR Systems and NRC Information Notice 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.

Several other items with existing or suspected material degradation concerns that have been identified for PWR components are wear in thimble tubes and potentially in control guide cards and observed cracking in some high-strength bolting and in control rod guide tube alignment (split) pins. The latter are conditions that have been corrected primarily through bolt replacement with less susceptible material and improved control of pre-load.

Based on industry operating experience, STP replaced the Alloy-750 guide tube support pins (split pins) with strained hardened (cold worked) 316 stainless steel pins during Refueling Outage 1RE12 (Spring 2005) for Unit 1 and Refueling Outage 2RE11 (Fall 2005) for Unit 2. The replacement was conducted to reduce the susceptibility for stress corrosion cracking in the split pins. There were no cracked Alloy X-750 pins discovered during the replacement process.

The ASME Code, Section XI, Examination Category B-N-3 examinations of core support structures conducted during Refueling Outage 1RE15 (Fall 2009) for Unit 1, and Refueling Outage 2RE14 (Spring 2010) for Unit 2, did not identify any conditions that required repair, replacement or evaluation.

The ISI Program portion of the PWR Reactor Internals program at STP is updated to account for industry operating experience. ASME Section XI is also revised every three years and addenda issued in the interim, which allows the code to be updated to reflect operating experience. The requirement to update the ISI Program to reference more recent editions of ASME Section XI at the end of each inspection interval ensures the ISI Program reflects enhancements due to operating experience that have been incorporated into ASME Section XI.

With exception of the ASME Section ISI portions, the PWR Reactor Internals program will be a new program and has no direct programmatic history. A key element of the MRP-227-A program is the reporting of aging of reactor vessel components. STP, through its participation in PWR Owners Group 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.

As additional Industry and applicable plant-specific operating experience become available, the OE will be evaluated and appropriately incorporated into the program through the STP Corrective Action and Operating Experience Programs. This ongoing review of OE will continue throughout the period of extended operation, and the results will be maintained on site. This process will confirm the effectiveness of this new license renewal aging management program by incorporating applicable OE and performing self assessments of the program.

Conclusion

The implementation of the PWR Reactor Internals program provides reasonable assurance that aging effects will be adequately managed such that the systems and components within the scope of this program will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

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Appendix C ~~Not Used~~ **Response to Applicant Action Items for Inspection and Evaluation Guidelines for PWR Internals**

3.1.2.1.1 Reactor Vessel and Internals

Materials

The materials of construction for the reactor vessel and internals component types are:

- Carbon Steel
- Carbon Steel with Stainless Steel Cladding
- High Strength Low Alloy Steel (Bolting)
- Nickel-Alloys
- Stainless Steel
- Stainless Steel Cast Austenitic (CASS)

Environment

The reactor vessel and internals components are exposed to the following environments:

- Borated Water Leakage
- Reactor Coolant
- Neutron Flux

Aging Effects Requiring Management

The following reactor vessel and internals aging effects require management:

- Changes in dimensions
- Cracking
- Loss of fracture toughness
- Loss of material
- Loss of preload

Aging Management Programs

The following aging management programs manage the aging effects for the reactor vessel and internals component types:

- ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)
- Boric Acid Corrosion (B2.1.4)
- Flux Thimble Tube Inspection (B2.1.21)
- Nickel-Alloy Aging Management (B2.1.34)
- Nickel-Alloy Penetration Nozzles Welded to the Upper Reactor Vessel Closure Heads of Pressurized Water Reactors (B2.1.5)

- PWR Reactor Internals (B2.1.35)
- Reactor Head Closure Studs (B2.1.3)
- Reactor Vessel Surveillance (B2.1.15)
- Water Chemistry (B2.1.2)

For Reactor Coolant System Nickel-Alloy Pressure Boundary Components, STP will:

(1) Implement applicable NRC Orders, Bulletins and Generic Letters associated with nickel-alloys; (2) implement staff-accepted industry guidelines, (3) participate in the industry initiatives, such as owners group programs and the EPRI Materials Reliability Program, for managing aging effects associated with nickel-alloys, and (4) upon completion of these programs, but not less than 24 months before entering the period of extended operation, STP will submit an inspection plan for reactor coolant system nickel-alloy pressure boundary components to the NRC for review and approval.

For Reactor Vessel Internals, STP will:

(1) Participate in the industry programs for investigating and managing aging effects on reactor internals; (2) evaluate and implement the results of the industry programs as applicable to the reactor internals. ~~; and (3) upon completion of these programs, but not less than 24 months before entering the period of extended operation, STP will submit an inspection plan for reactor internals to the NRC for review and approval.~~

3.1.2.2.6 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement and Void Swelling

Not applicable, the STP PWR Reactor Internals program (B2.1.35) relies on implementation of the inspection and evaluation guidelines in EPRI TR-1022863 (MRP-227-A) and EPRI TR-1016609 to manage the aging effects of the reactor vessel internal components.

~~Loss of fracture toughness due to neutron irradiation embrittlement and void swelling for stainless steel reactor internals components exposed to reactor coolant is managed by the plant-specific PWR Reactor Internals program (B2.1.35) based on the guidelines provided in EPRI 1022863 (MRP-227-A). Consistent with EPRI 1022863 (MRP-227-A), loss of fracture toughness is not an applicable aging effect requiring management for the RVI neutron shield panel.~~

3.1.2.2.9 Loss of Preload due to Stress Relaxation

Not applicable, the STP PWR Reactor Internals program (B2.1.35) relies on implementation of the inspection and evaluation guidelines in EPRI TR-1022863 (MRP-227-A) and EPRI TR-1016609 to manage the aging effects of the reactor vessel internal components.

~~Loss of preload due to stress relaxation for nickel-alloy and stainless steel reactor internals components exposed to reactor coolant is managed by the plant-specific PWR Reactor Internals program (B2.1.35) based on the guidelines provided in EPRI 1022863 (MRP-227-A). Consistent with EPRI 1022863 (MRP-227-A), loss of preload is not an applicable aging effect requiring management for the RVI Lower Core Support-Clevis Insert Bolting and RVI Upper Support Column Bolting.~~

3.1.2.2.12 Cracking due to Stress Corrosion Cracking and Irradiation-Assisted Stress Corrosion Cracking (IASCC)

Not applicable, the STP PWR Reactor Internals program (B2.1.35) relies on implementation of the inspection and evaluation guidelines in EPRI TR-1022863 (MRP-227-A) and EPRI TR-1016609 to manage the aging effects of the reactor vessel internal components.

~~For managing the aging effect of cracking due to stress corrosion cracking and irradiation-assisted stress corrosion cracking of stainless steel reactor internals components exposed to reactor coolant, Water Chemistry (B2.1.2) is augmented by the plant-specific PWR Reactor Internals program (B2.1.35) based on the guidelines provided in EPRI 1022863 (MRP-227-A). Consistent with EPRI 1022863 (MRP-227-A), PWR Reactor Internals (B2.1.35) is not an applicable aging program for managing cracking of for the following components. Instead, cracking is managed by ASME Section XI Inservice Inspection (B2.1.1):~~

~~- RVI Hold Down Spring~~
~~- RVI Neutron Shield Panel~~

- ~~RVI Upper Core Support-Upper Support Column~~
- ~~RVI Upper Core Support-Upper Support Column Base~~
- ~~RVI Upper Core Support-Upper Support Plate~~
- ~~RVI Control Rod Guide Tube Guide Plates~~
- ~~RVI ICI Support Structures - Exit Thermocouples~~
- ~~RVI ICI Support Structures - Upper/Lower Tie Plates~~
- ~~RVI Irradiation Specimen Basket~~
- ~~RVI Lower Core Support - Energy Absorber Assembly~~

3.1.2.2.15 Changes in dimensions due to Void Swelling

Not applicable, the STP PWR Reactor Internals program (B2.1.35) relies on implementation of the inspection and evaluation guidelines in EPRI TR-1022863 (MRP-227-A) and EPRI TR-1016609 to manage the aging effects of the reactor vessel internal components.

~~Changes in dimensions due to void swelling for stainless steel reactor internals components exposed to reactor coolant will be managed by the plant-specific PWR Reactor Internals program (B2.1.35) based on the guidelines provided in EPRI 1022863 (MRP-227-A). Consistent with EPRI 1022863 (MRP-227-A), changes in dimension is not an applicable aging effect requiring management for the following components:~~

- ~~RVI Control Rod Guide Tube Assembly~~
- ~~RVI Control Rod Guide Tube Bolting~~
- ~~RVI Control Rod Guide Tube Guide Plates~~
- ~~RVI Core Barrel Assembly~~
- ~~RVI Hold Down Spring~~
- ~~RVI ICI Support Structures-Instrument Column (BMI)~~
- ~~RVI ICI Support Structures-Upper/Lower Tie Plates~~
- ~~RVI Lower Core Support Bolts~~
- ~~RVI Lower Core Support-Clevis Insert Bolting~~
- ~~RVI Lower Core Support-Core Support Plate Forging~~
- ~~RVI Neutron Shield Panel~~
- ~~RVI Radial Support Keys and Clevis Inserts~~
- ~~RVI Upper Core Plate Guide Pins~~
- ~~RVI Upper Core Support-Protective Skirt~~
- ~~RVI Upper Core Support-Upper Core Plate~~
- ~~RVI Upper Core Support-Upper Support Column~~
- ~~RVI Upper Core Support-Upper Support Column Base~~
- ~~RVI Upper Core Support-Upper Support Plate~~
- ~~RVI Upper Support Column Bolting~~

Table 3.1.1 Summary of Aging Management Evaluations in Chapter IV of NUREG-1801 for Reactor Vessel, Internals, and Reactor Coolant System

Item Number	Component Type	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
~~3.1.1.22~~	~~Stainless steel and nickel alloy reactor vessel internals components exposed to reactor coolant and neutron flux~~	~~Loss of fracture toughness due to neutron irradiation embrittlement, void swelling~~	~~FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation.~~	~~No~~	~~Consistent with NUREG-1801 for material, environment, and aging effect, but a different aging management program PWR Reactor Internals (B2.1.35) is credited, with exception of the RVI neutron shield panel. See further evaluation in Section 3.1.2.2.6.~~
3.1.1.22	Stainless steel SS, including CASS, PH SS or martensitic SS) or nickel alloy Westinghouse reactor vessel internal "Existing Programs" components	Loss of fracture toughness due to thermal aging and neutron irradiation embrittlement, and for CASS, martensitic SS, and PH SS due to thermal aging embrittlement; or changes in dimensions due to void swelling or distortion; or loss of preload due to thermal and irradiation enhanced stress relaxation or creep; or loss of material due to wear	PWR Vessel Internals (B2.1.35)	No	Consistent with NUREG-1801, Revision 2 and LR-ISG 2011-04
~~3.1.1.27~~	~~Stainless steel and nickel alloy reactor vessel internals screws, bolts, tie rods, and hold-down springs~~	~~Loss of preload due to stress relaxation~~	~~FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation.~~	~~No~~	~~Consistent with NUREG-1801 for materiál, environment, and aging effect, but a different aging management program PWR Reactor Internals (B2.1.35) is credited, with exception of the RVI Lower Core Support clevis insert bolting and RVI upper support column~~

Item Number	Component Type	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
					~~bolting.~~ ~~See further evaluation in Section 3.1.2.2.9.~~
3.1.1.27	Stainless steel (SS, including CASS, PH SS or martensitic SS) or and nickel alloy Westinghouse reactor vessel internal "Expansion" components	Loss of fracture toughness due to neutron irradiation embrittlement and for CASS, martensitic SS, and PH SS due to thermal aging embrittlement; or changes in dimensions due to void swelling or distortion; or loss of preload due to thermal and irradiation enhanced stress relaxation or creep; or loss of material due to wear	PWR Vessel Internals (B2.1.35)	No	Consistent with NUREG-1801 Revision 2 and LR-ISG-2011-04
~~3.1.1.30~~	~~Stainless steel reactor vessel internals components (e.g., Upper internals assembly, RCCA guide tube assemblies, Baffle/former assembly, Lower internal assembly, shroud assemblies, Plenum cover and plenum cylinder, Upper grid assembly, Control rod guide tube (CRGT) assembly, Core support shield assembly, Core barrel assembly, Lower grid assembly, Flow distributor assembly, Thermal shield,~~	~~Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking~~	~~Water Chemistry (B2.1.2) and FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation.~~	~~No~~	~~Consistent with NUREG 1801 for material, environment, and aging effect, but different AMPs are credited: Water Chemistry program (B2.1.2) augmented by the plant-specific aging management program PWR Reactor Internals (B2.1.35). Consistent with EPRI 1022863 (MRP-227-A), cracking is managed by ASME Section XI Inservice Inspection for selected components.~~ ~~See further evaluation in Section 3.1.2.2.12.~~

Item Number	Component Type	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
	~~Instrumentation support structures)~~				
3.1.1.30	Stainless steel or nickel alloy Westinghouse reactor vessel internal "Primary"components (	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.35)	No	Consistent with NUREG 1801 Revision 2 and LR-ISG-2011-04.
~~3.1.1.33~~	~~Stainless steel and nickel alloy reactor vessel internals components~~	~~Changes in dimensions due to void swelling~~	~~FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an RVI inspection plan based on industry recommendation.~~	~~No~~	~~Exception to NUREG-1801. Aging effect in NUREG-1801 for this material and environment combination is not applicable for selected components.~~ ~~See further evaluation in Section 3.1.2.2.15.~~
3.1.1.33	Stainless steel (SS, including CASS, PH SS or martensitic SS) or nickel alloy Westinghouse reactor vessel internal "Primary" components	Loss of fracture toughness due to neutron irradiation embrittlement and for CASS, martensitic SS, and PH SS due to thermal aging embrittlement; or changes in dimensions due to void swelling or distortion; or loss of preload due to thermal and irradiation enhanced stress relaxation or creep; or loss of material due to wear	PWR Vessel Internals (B2.1.35)	No	Consistent with NUREG 1801 Revision 2 and LR-ISG-2011-04
~~3.1.1.37~~	~~Stainless steel and nickel alloy reactor vessel internals components (e.g., Upper internals assembly, RCCA guide tube assemblies, Lower internal~~	~~Cracking due to stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking~~	~~Water Chemistry (B2.1.2) and FSAR supplement commitment to (1) participate in industry RVI aging programs (2) implement applicable results (3) submit for NRC approval > 24 months before the extended period an~~	~~No~~	~~Consistent with NUREG-1801 for material, environment, and aging effect, but different AMPs are credited: Water Chemistry program (B2.1.2) augmented by the plant-specific aging management program PWR~~

Item Number	Component Type	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
	~~assembly, CEA shroud assemblies, Core shroud assembly, Core support shield assembly, Core barrel assembly, Lower grid assembly, Flow distributor assembly)~~		~~RVI inspection plan based on industry recommendation.~~		~~Reactor Internals (B2.1.35). Consistent with EPRI 1022863 (MRP-227-A), cracking is managed by ASME Section XI Inservice Inspection for selected components.~~ ~~See further evaluation in Section 3.1.2.2.17.~~
3.1.1.37	Stainless steel or nickel alloy Westinghouse reactor vessel internal "Existing Programs" components	Cracking due to stress corrosion cracking, primary water stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.35)	No	Consistent with NUREG-1801 Revision 2 and LR-ISG-2011-04.
3.1.1.60	Stainless steel flux thimble tubes (with or without chrome plating) "Existing Programs" component	Loss of material due to Wear	Flux Thimble Tube Inspection (B2.1.21)	No	Consistent with NUREG-1801 NUREG-1801 Revision 2 and LR-ISG-2011-04..
~~3.1.1.63~~	~~Steel reactor vessel flange, stainless steel and nickel alloy reactor vessel internals exposed to reactor coolant (e.g., upper and lower internals assembly, CEA shroud assembly, core support barrel, upper grid assembly, core support shield assembly, lower grid assembly)~~	~~Loss of material due to Wear~~	~~Inservice Inspection (IWB, IWC, and IWD) (B2.1.1)~~	~~No~~	~~Consistent with NUREG-1801 except for aging management of the clevis insert bolting, control rod guide tube bolting, core barrel assembly, control rod guide tube guide plates and upper core plate guide pins, for which the material, environment, and aging effect are consistent with NUREG-1801 but a different aging management program PWR Reactor Internals (B2.1.35) is credited.~~

Item Number	Component Type	Aging Effect / Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.1.1.63	Stainless steel, nickel alloy or CASS reactor vessel internals, core support structure components in MRP-227-A, exposed to reactor coolant and neutron flux	Cracking or Loss of material due to Wear	Inservice Inspection (IWB, IWC, and IWD) (B2.1.1) or PWR Vessel Internals (B2.1.35)	No	Consistent with NUREG-1801 Revision 2 and LR-ISG-2011-04
~~3.1.1.80~~	~~Cast austenitic stainless steel reactor vessel internals (e.g., upper internals assembly, lower internal assembly, CEA shroud assemblies, control rod guide tube assembly, core support shield assembly, lower grid assembly)~~	~~Loss of fracture toughness due to thermal aging and neutron irradiation embrittlement~~	~~Thermal Aging and Neutron Irradiation Embrittlement of CASS~~	~~No~~	~~Exception to NUREG-1801. Aging effect in NUREG-1801 for this material and environment combination is not applicable based on EPRI 1016596 (MRP-227).~~
3.1.1.80	Stainless steel Westinghouse reactor vessel internal "Expansion" components	Cracking due to stress corrosion cracking, irradiation-assisted stress corrosion cracking, or fatigue	Water Chemistry (B2.1.2) and PWR Vessel Internals (B2.1.35)	No	Consistent with NUREG-1801 Revision 2 and LR-ISG-2011-04

Table 3.1.2-1 Reactor Vessel, Internals, and Reactor Coolant System – Summary of Aging Management Evaluation – Reactor Vessel and Internals

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
RV Upper, Intermediate, Lower Shell and Welds	PB	Carbon Steel with Stainless Steel Cladding	Reactor Coolant (Int)	Loss of material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.A2-25	3.1.1.63	A
~~RVI Baffle-Edge Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Changes in dimensions~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-4~~	~~3.1.1.33~~	~~E, 3~~
~~RVI Baffle-Edge Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of preload~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-5~~	~~3.1.1.27~~	~~E, 3~~
~~RVI Baffle-Edge Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of fracture toughness~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-6~~	~~3.1.1.22~~	~~E, 3~~
~~RVI Baffle-Edge Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)~~	~~IV.B2-10~~	~~3.1.1.30~~	~~E, 3~~
~~RVI Baffle-Edge Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cumulative fatigue damage~~	~~Time-Limited Aging Analysis evaluated for the period of extended operation~~	~~IV.B2-31~~	~~3.1.1.05~~	~~A~~
~~RVI Baffle-Edge Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Baffle-Former Assembly~~	~~DF, SLD, SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Changes in dimensions~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-1~~	~~3.1.1.33~~	~~E, 3~~
~~RVI Baffle-Former Assembly~~	~~DF, SLD, SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)~~	~~IV.B2-2~~	~~3.1.1.30~~	~~E, 3~~
~~RVI Baffle-Former Assembly~~	~~DF, SLD, SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of fracture toughness~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-3~~	~~3.1.1.22~~	~~E, 3~~
~~RVI Baffle-Former Assembly~~	~~DF, SLD, SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
~~RVI Baffle-Former Assembly Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Changes in dimensions~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-4~~	~~3.1.1.33~~	~~E, 3~~
~~RVI Baffle-Former Assembly Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of preload~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-5~~	~~3.1.1.27~~	~~E, 3~~
~~RVI Baffle-Former Assembly Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of fracture toughness~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-6~~	~~3.1.1.22~~	~~E, 3~~
~~RVI Baffle-Former Assembly Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)~~	~~IV.B2-10~~	~~3.1.1.30~~	~~E, 3~~
~~RVI Baffle-Former Assembly Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cumulative fatigue damage~~	~~Time-Limited Aging Analysis evaluated for the period of extended operation~~	~~IV.B2-31~~	~~3.1.1.05~~	~~A~~
~~RVI Baffle-Former Assembly Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Control Rod Guide Tube Assembly~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of fracture toughness~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-9~~	~~3.1.1.22~~	~~E, 3~~
~~RVI Control Rod Guide Tube Assembly~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-29~~	~~3.1.1.33~~	~~I, 4~~
~~RVI Control Rod Guide Tube Assembly~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)~~	~~IV.B2-30~~	~~3.1.1.30~~	~~E, 3~~
~~RVI Control Rod Guide Tube Assembly~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Control Rod Guide Tube Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-27~~	~~3.1.1.33~~	~~I, 4~~

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
~~RVI Control Rod Guide Tube Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~Water Chemistry (B2.1.2)and PWR Reactor Internals (B2.1.35)~~	~~IV.B2-28~~	~~3.1.1.37~~	~~E, 3~~
~~RVI Control Rod Guide Tube Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Control Rod Guide Tube Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-34~~	~~3.1.1.63~~	~~E, 3~~
~~RVI Control Rod Guide Tube Guide Plates~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-29~~	~~3.1.1.33~~	~~I, 4~~
~~RVI Control Rod Guide Tube Guide Plates~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Control Rod Guide Tube Guide Plates~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~	~~IV.B2-30~~	~~3.1.1.30~~	~~E, 5~~
~~RVI Control Rod Guide Tube Guide Plates~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-34~~	~~3.1.1.63~~	~~E, 3~~
~~RVI Core Barrel Assembly~~	~~DF, SLD, SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-7~~	~~3.1.1.33~~	~~I, 4~~
~~RVI Core Barrel Assembly~~	~~DF, SLD, SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)~~	~~IV.B2-8~~	~~3.1.1.30~~	~~E, 3~~
~~RVI Core Barrel Assembly~~	~~DF, SLD, SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of fracture toughness~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-9~~	~~3.1.1.22~~	~~E, 3~~
~~RVI Core Barrel Assembly~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cumulative fatigue damage~~	~~Time-Limited Aging Analysis evaluated for the period of extended operation~~	~~IV.B2-31~~	~~3.1.1.05~~	~~A~~

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
~~RVI Core Barrel Assembly~~	~~DF, SLD, SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Core Barrel Assembly~~	~~DF, SLD, SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-34~~	~~3.1.1.63~~	~~E, 3~~
~~RVI Core Barrel Assembly-Former Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Changes in dimensions~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-4~~	~~3.1.1.33~~	~~E, 3~~
~~RVI Core Barrel Assembly-Former Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of preload~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-5~~	~~3.1.1.27~~	~~E, 3~~
~~RVI Core Barrel Assembly-Former Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of fracture toughness~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-6~~	~~3.1.1.22~~	~~E, 3~~
~~RVI Core Barrel Assembly-Former Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~Water Chemistry (B2.1.2)and PWR Reactor Internals (B2.1.35)~~	~~IV.B2-10~~	~~3.1.1.30~~	~~E, 3~~
~~RVI Core Barrel Assembly-Former Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cumulative fatigue damage~~	~~Time-Limited Aging Analysis evaluated for the period of extended operation~~	~~IV.B2-31~~	~~3.1.1.05~~	~~A~~
~~RVI Core Barrel Assembly-Former Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Hold Down Spring~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Hold Down Spring~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of preload~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-33~~	~~3.1.1.27~~	~~E, 3~~
~~RVI Hold Down Spring~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-41~~	~~3.1.1.33~~	~~I, 4~~
~~RVI Hold Down Spring~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~	~~IV.B2-42~~	~~3.1.1.30~~	~~E, 5~~

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
~~RVI Hold Down Spring~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~	~~IV.B2-34~~	~~3.1.1.63~~	~~C~~
~~RVI ICI Support Structures (Exit Thermocouple)~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI ICI Support Structures (Exit Thermocouple)~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~	~~IV.B2-12~~	~~3.1.1.30~~	~~E, 5~~
~~RVI ICI Support Structures-Instr Column (BMI)~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-11~~	~~3.1.1.33~~	~~I, 4~~
~~RVI ICI Support Structures-Instr Column (BMI)~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of fracture toughness~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-9~~	~~3.1.1.22~~	~~E, 3~~
~~RVI ICI Support Structures-Instr Column (BMI)~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)~~	~~IV.B2-12~~	~~3.1.1.30~~	~~E, 3~~
~~RVI ICI Support Structures-Instr Column (BMI)~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI ICI Support Structures-Upper/Lower Tie Plates~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-23~~	~~3.1.1.33~~	~~I, 4~~
~~RVI ICI Support Structures-Upper/Lower Tie Plates~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~	~~IV.B2-24~~	~~3.1.1.30~~	~~E, 5~~

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
~~RVI ICI Support Structures-Upper/Lower Tie Plates~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Irradiation Specimen Basket~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Irradiation Specimen Basket~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~	~~IV.B2-12~~	~~3.1.1.30~~	~~E, 5~~
~~RVI Lower Core Support Bolts~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-15~~	~~3.1.1.33~~	~~I, 4~~
~~RVI Lower Core Support Bolts~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~	~~IV.B2-16~~	~~3.1.1.37~~	~~E, 7~~
~~RVI Lower Core Support Bolts~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cumulative fatigue damage~~	~~Time-Limited Aging Analysis evaluated for the period of extended operation~~	~~IV.B2-31~~	~~3.1.1.05~~	~~A~~
~~RVI Lower Core Support Bolts~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Lower Core Support-Clevis Insert Bolting~~	~~SS~~	~~Nickel Alloys~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-14~~	~~3.1.1.27~~	~~I, 6~~
~~RVI Lower Core Support-Clevis Insert Bolting~~	~~SS~~	~~Nickel Alloys~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-15~~	~~3.1.1.33~~	~~I, 4~~
~~RVI Lower Core Support-Clevis Insert Bolting~~	~~SS~~	~~Nickel Alloys~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~	~~IV.B2-16~~	~~3.1.1.37~~	~~E, 7~~

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
~~RVI Lower Core Support-Clevis Insert Bolting~~	~~SS~~	~~Nickel Alloys~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Lower Core Support-Clevis Insert Bolting~~	~~SS~~	~~Nickel Alloys~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-34~~	~~3.1.1.63~~	~~E, 3~~
~~RVI Lower Core Support-Core Support Plate Forging~~	~~DF, SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of fracture toughness~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-22~~	~~3.1.1.22~~	~~E, 3~~
~~RVI Lower Core Support-Core Support Plate Forging~~	~~DF, SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-23~~	~~3.1.1.33~~	~~I, 4~~
~~RVI Lower Core Support-Core Support Plate Forging~~	~~DF, SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)~~	~~IV.B2-24~~	~~3.1.1.30~~	~~E, 3~~
~~RVI Lower Core Support-Core Support Plate Forging~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cumulative fatigue damage~~	~~Time-Limited Aging Analysis evaluated for the period of extended operation~~	~~IV.B2-31~~	~~3.1.1.05~~	~~A~~
~~RVI Lower Core Support-Core Support Plate Forging~~	~~DF, SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Lower Core Support-Energy Absorber Assembly~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Lower Core Support-Energy Absorber Assembly~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~	~~IV.B2-24~~	~~3.1.1.30~~	~~E, 5~~

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
~~RVI Neutron Shield Panel~~	~~SLD~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-7~~	~~3.1.1.33~~	~~I, 4~~
~~RVI Neutron Shield Panel~~	~~SLD~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~	~~IV.B2-8~~	~~3.1.1.30~~	~~E, 5~~
~~RVI Neutron Shield Panel~~	~~SLD~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-9~~	~~3.1.1.22~~	~~I, 8~~
~~RVI Neutron Shield Panel~~	~~SLD~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Radial Support Keys and Clevis Inserts~~	~~SS~~	~~Nickel Alloys~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-19~~	~~3.1.1.33~~	~~I, 4~~
~~RVI Radial Support Keys and Clevis Inserts~~	~~SS~~	~~Nickel Alloys~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~	~~IV.B2-20~~	~~3.1.1.37~~	~~E, 7~~
~~RVI Radial Support Keys and Clevis Inserts~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~	~~IV.B2-34~~	~~3.1.1.63~~	~~C~~
~~RVI Radial Support Keys and Clevis Inserts~~	~~SS~~	~~Nickel Alloys~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Upper Core Plate Guide Pins~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Upper Core Plate Guide Pins~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-34~~	~~3.1.1.63~~	~~E, 3~~
~~RVI Upper Core Plate Guide Pins~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-39~~	~~3.1.1.33~~	~~I, 4~~
~~RVI Upper Core Plate Guide Pins~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)~~	~~IV.B2-40~~	~~3.1.1.37~~	~~E, 3~~

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
~~RVI Upper Core Support-Protective Skirt~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cumulative fatigue damage~~	~~Time-Limited Aging Analysis evaluated for the period of extended operation~~	~~IV.B2-31~~	~~3.1.1.05~~	~~A~~
~~RVI Upper Core Support-Protective Skirt~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Upper Core Support-Protective Skirt~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-41~~	~~3.1.1.33~~	~~I, 4~~
~~RVI Upper Core Support-Upper Core Plate~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~PWR Reactor Internals (B2.1.35)~~	~~IV.B2-34~~	~~3.1.1.63~~	~~E, 3~~
~~RVI Upper Core Support-Protective Skirt~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)~~	~~IV.B2-42~~	~~3.1.1.30~~	~~E, 3~~
~~RVI Upper Core Support-Upper Core Plate~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cumulative fatigue damage~~	~~Time-Limited Aging Analysis evaluated for the period of extended operation~~	~~IV.B2-31~~	~~3.1.1.05~~	~~A~~
~~RVI Upper Core Support-Upper Core Plate~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Upper Core Support-Upper Core Plate~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-41~~	~~3.1.1.33~~	~~I, 4~~
~~RVI Upper Core Support-Upper Core Plate~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~	~~IV.B2-42~~	~~3.1.1.30~~	~~E, 3~~

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
~~RVI Upper Core Support-Upper Support Column~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cumulative fatigue damage~~	~~Time-Limited Aging Analysis evaluated for the period of extended operation~~	~~IV.B2-31~~	~~3.1.1.05~~	~~A~~
~~RVI Upper Core Support-Upper Support Column~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Upper Core Support-Upper Support Column~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-35~~	~~3.1.1.33~~	~~I, 4~~
~~RVI Upper Core Support-Upper Support Column~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~	~~IV.B2-36~~	~~3.1.1.30~~	~~E, 5~~
~~RVI Upper Core Support-Upper Support Column Base~~	~~SS~~	~~Stainless Steel Cast Austenitic~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Upper Core Support-Upper Support Column Base~~	~~SS~~	~~Stainless Steel Cast Austenitic~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-35~~	~~3.1.1.33~~	~~I, 4~~
~~RVI Upper Core Support-Upper Support Column Base~~	~~SS~~	~~Stainless Steel Cast Austenitic~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~	~~IV.B2-36~~	~~3.1.1.30~~	~~E, 5~~
~~RVI Upper Core Support-Upper Support Column Base~~	~~SS~~	~~Stainless Steel Cast Austenitic~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-37~~	~~3.1.1.80~~	~~I, 9~~
~~RVI Upper Core Support-Upper Support Plate~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
~~RVI Upper Core Support-Upper Support Plate~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-41~~	~~3.1.1.33~~	~~I, 4~~
~~RVI Upper Core Support-Upper Support Plate~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~	~~IV.B2-42~~	~~3.1.1.30~~	~~E, 5~~
~~RVI Upper Support Column Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Loss of material~~	~~Water Chemistry (B2.1.2)~~	~~IV.B2-32~~	~~3.1.1.83~~	~~A~~
~~RVI Upper Support Column Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-38~~	~~3.1.1.27~~	~~I, 6~~
~~RVI Upper Support Column Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~None~~	~~None~~	~~IV.B2-39~~	~~3.1.1.33~~	~~I, 4~~
~~RVI Upper Support Column Bolting~~	~~SS~~	~~Stainless Steel~~	~~Reactor Coolant (Ext)~~	~~Cracking~~	~~ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)~~	~~IV.B2-40~~	~~3.1.1.37~~	~~E, 7~~
RVI Baffle Former Assembly Lock Device	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A
RVI Baffle Former Assembly Lock Device	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material (wear)	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A
RVI Baffle-Former Assembly (All Components)	DF, SLD, SS	Stainless Steel	Reactor Coolant (Ext)	Loss of material	Water Chemistry (B2.1.2)	IV.B2-32	3.1.1.83	A
RVI Baffle-Former Assembly Baffle-Former Bolting	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Changes in dimensions	PWR Reactor Internals (B2.1.35)	IV.B2.RP-272	3.1.1.33	A, 3
RVI Baffle-Former Assembly Baffle-Former Bolting	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of fracture toughness	PWR Reactor Internals (B2.1.35)	IV.B2.RP-272	3.1.1.33	A, 3

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
RVI Baffle-Former Assembly Baffle-Former Bolting	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of preload	PWR Reactor Internals (B2.1.35)	IV.B2.RP-272	3.1.1.33	A, 3
RVI Baffle-Former Assembly Baffle-Former Bolting	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)	IV.B2.RP-271	3.1.1.30	A, 3
RVI Baffle-Former Assembly Baffle-Former Bolting	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cumulative fatigue damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.B2.RP-303	3.1.1.05	A
RVI Baffle-Former Assembly Barrel to Former Bolting	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Changes in dimensions	PWR Reactor Internals (B2.1.35)	IV.B2.RP-274	3.1.1.27	A, 3
RVI Baffle-Former Assembly Barrel to Former Bolting	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	Water Chemistry (B2.1.2)and PWR Reactor Internals (B2.1.35)	IV.B2.RP-273	3.1.1.80	A, 3
RVI Baffle-Former Assembly Barrel to Former Bolting	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cumulative fatigue damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.B2.RP-303	3.1.1.05	A
RVI Baffle-Former Assembly Barrel to Former Bolting	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of fracture toughness	PWR Reactor Internals (B2.1.35)	IV.B2.RP-274	3.1.1.27	A, 3
RVI Baffle-Former Assembly Barrel to Former Bolting	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of preload	PWR Reactor Internals (B2.1.35)	IV.B2.RP-274	3.1.1.27	A, 3
RVI Baffle-Former Assembly - baffle and former Plates	DF, SLD, SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Changes in dimensions	PWR Reactor Internals (B2.1.35)	IV.B2.RP-270	3.1.1.33	A, 3

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
RVI Baffle-Former Assembly-baffle and former plates	DF, SLD, SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)	IV.B2.RP-270a	3.1.1.30	A, 3
RVI Bottom Mounted Instrumentation System	SS	Stainless Steel	Reactor Coolant (Ext)	Cumulative fatigue damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.B2.RP-303	3.1.1.05	A
RVI Bottom Mounted Instrumentation System (All Components)	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material	Water Chemistry (B2.1.2)	IV.B2.RP-24	3.1.1.83	A
RVI Bottom Mounted Instrumentation System BMI Column Bodies	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)	IV.B2.RP-293	3.1.1.80	A, 3
RVI Bottom Mounted Instrumentation System BMI Column Bodies	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of fracture toughness	PWR Reactor Internals (B2.1.35)	IV.B2.RP-292	3.1.1.27	A, 3
RVI Bottom Mounted Instrumentation System Flux Thimble Tubes	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of Material (wear)	Flux Thimble Tube Inspection (B2.1.21)	IV.B2.RP-284	3.1.1.60	A, 5

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
RVI Bottom Mounted Instrumentation System Column Collars, Extension Bars And Extension Tubes	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	None	None	IV.B2.RP-265	NA	A
RVI Bottom Mounted Instrumentation System Column Cruciforms	SS	Stainless Steel Cast Austenitic	Reactor Coolant and neutron flux (Ext)	None	None	IV.B2.RP-265	NA	A
RVI Bottom Mounted Instrumentation System Locking Devices	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A
RVI Bottom Mounted Instrumentation System Locking Devices	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material (wear)	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A
RVI Clevis Insert (All Components)	SS	Nickel Alloys	Reactor Coolant and neutron flux (Ext)	Loss of material	Water Chemistry (B2.1.2)	IV.B2.RP-24	3.1.1.83	A
RVI Clevis Insert Bolting	SS	Nickel Alloys	Reactor Coolant and neutron flux (Ext)	Cracking	Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)	IV.B2.RP-399	3.1.1.37	A, 3
RVI Clevis Insert Bolting	SS	Nickel Alloys	Reactor Coolant and neutron flux (Ext)	Loss of material (wear)	PWR Reactor Internals (B2.1.35)	IV.B2.RP-285	3.1.1.22	A, 3

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
RVI Clevis Insert Bolting	SS	Nickel Alloys	Reactor Coolant and neutron flux (Ext)	Loss of preload	Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)	IV.B2.RP-285	3.1.1.22	A, 3
RVI Clevis Inserts and Lock Keys	SS	Nickel Alloys	Reactor Coolant and neutron flux (Ext)	Loss of material	Water Chemistry (B2.1.2)	IV.B2.RP-24	3.1.1.83	A
RVI Control Rod Guide Tube Assembly and Flow Downcommers (All Components)	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material	Water Chemistry (B2.1.2)	IV.B2.RP-24	3.1.1.83	A
RVI Control Rod Guide Tube Assembly C Tubes	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	None	None	IV.B2.RP-265	NA	A
RVI Control Rod Guide Tube Assembly Guide Plates (Cards)	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material (wear)	PWR Reactor Internals (B2.1.35)	IV.B2.RP-296	3.1.1.33	A, 3
RVI Control Rod Guide Tube Assembly Guide Tube Support Pins	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)	IV.B2.RP-355	3.1.1.37	A, 3
RVI Control Rod Guide Tube Assembly Guide Tube Support Pins	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material (wear)	PWR Reactor Internals (B2.1.35)	IV.B2.RP-356	3.1.1.22	A, 3
RVI Control Rod Guide Tube Assembly Inserts and Screw Locking Devices	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A,

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
RVI Control Rod Guide Tube Assembly Inserts and Screw Locking Devices (ISI Items No 10	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material (wear)	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A
RVI Control Rod Guide Tube Assembly Sheaths	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	None	None	IV.B2.RP-265	NA	A
RVI Control Rod Guide Tube Assembly Lower Flange Welds	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of fracture toughness	Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)	IV.B2.RP-297	3.1.1.33	A, 3
RVI Control Rod Guide Tube Assembly Lower Flange Welds	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)	IV.B2.RP-298	3.1.1.30	A, 3
RVI Core Barrel Assembly	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cumulative fatigue damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.B.RP-303	3.1.1.05	A
RVI Core Barrel Assembly (All Components)	DF, SLD, SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material	Water Chemistry (B2.1.2)	IV.B2.RP-24	3.1.1.83	A
RVI Core Barrel Assembly Core Barrel Flange	DF, SLD, SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material (wear)	PWR Reactor Internals (B2.1.35)	IV.B2.RP-345	3.1.1.22	A, 3
RVI Core Barrel Assembly Core Barrel Outlet Nozzle Welds	DF, SLD, SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)	IV.B2.RP-278	3.1.1.80	A, 3

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
RVI Core Barrel Assembly Core Barrel Outlet Nozzle Welds	DF, SLD, SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of fracture toughness	Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)	IV.B2.RP-278a	3.1.1.27	A, 3
RVI Core Barrel Assembly Lower Core Barrel Flange Welds	DF, SLD, SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)	IV.B2.RP-280	3.1.1.30	A, 3
RVI Core Barrel Assembly Upper Core Barrel Flange Welds	DF, SLD, SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)	IV.B2.RP-276	3.1.1.30	A, 3
RVI Core Barrel Assembly Upper Core Barrel and Lower Core Barrel Circumferential (girth) Welds	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)	IV.B2.RP-387	3.1.1.30	A, 3
RVI Core Barrel Assembly Upper Core Barrel and Lower Core Barrel Circumferential (girth) Welds	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of fracture toughness	PWR Reactor Internals (B2.1.35)	IV.B2.RP-388	3.1.1.33	A, 3
RVI Core Barrel Assembly Upper Core Barrel and Lower Core Barrel Vertical (axial) Welds	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)	IV.B2.RP-387a	3.1.1.30	A, 3

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
RVI Core Barrel Assembly Upper Core Barrel and Lower Core Barrel Vertical (axial) Welds	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of fracture toughness	PWR Reactor Internals (B2.1.35)	IV.B2.RP-388a	3.1.1.27	A, 3
RVI Fuel Pin to Core Support Locking Device	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material (wear)	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A
RVI Fuel Pin to Core Support Locking Device	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A
RVI Hold Down Spring	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	None	None	IV.B2.RP-265	NA	I, 4
RVI Irradiation Specimen Guides (All Components)	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material	Water Chemistry (B2.1.2)	IV.B2.RP-24	3.1.1.83	A
RVI Irradiation Specimen Guides Screw Locking Device and Dowel Pins	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A
RVI Lower Internal Assembly-XL Lower Core Plate	DF, SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)	IV.B2.RP-289	3.1.1.37	A, 3
RVI Lower Internal Assembly-XL Lower Core Plate	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cumulative fatigue damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.B2.RP-303	3.1.1.05	A

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
RVI Lower Internal Assembly-XL Lower Core Plate	DF, SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material	Water Chemistry (B2.1.2)	IV.B2.RP-24	3.1.1.83	A
RVI Lower Internal Assembly-XL Lower Core Plate	DF, SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material (wear)	PWR Reactor Internals (B2.1.35)	IV.B2.RP-288	3.1.1.27	A, 3
RVI Lower Internal Assembly-XL Lower Core Plate	DF, SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of fracture toughness	PWR Reactor Internals (B2.1.35)	IV.B2.RP-288	3.1.1.27	A, 3
RVI Neutron Shield Panel Screw Locking Devices (ISI Item No 22)	SLD	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material (wear)	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A
RVI Neutron Shield Panel Screw Locking Devices	SLD	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A
RVI Radial Support Key Lock Keys and Clevis Insert Lock Keys	SS	Nickel Alloys	Reactor Coolant and neutron flux (Ext)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A
RVI Radial Support Key Lock Keys and Clevis Insert Lock Keys	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material (wear)	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A
RVI Secondary Core Support (SCS) Assembly SCS locking Device	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A
RVI Secondary Core Support (SCS) Assembly SCS locking Device	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material (wear)	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
RVI Secondary Core Support SCS) Assembly (All Components)	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material	Water Chemistry (B2.1.2)	IV.B2.RP-24	3.1.1.83	A
RVI Shield Assembly (All Components)	SLD	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material	Water Chemistry (B2.1.2)	IV.B2.RP-24	3.1.1.83	A
RVI Upper Core Assembly – Upper Support Column Assemblies (all components)	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material	Water Chemistry (B2.1.2)	IV.B2.RP-24	3.1.1.83	A
RVI Upper Core Assembly Thermocouple Clamps, Conduit Swaglock Fittings, Bandings, and Tab Locks	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A,
RVI Upper Core Assembly Thermocouple Clamps, Conduit Swaglock Fittings, Bandings, and Tab Locks	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material (wear)	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B2.1.1)	IV.B2.RP-382	3.1.1.63	A
RVI Upper Core Assembly -Upper Core Plate	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cumulative fatigue damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.B2.RP-303	3.1.1.05	A
RVI Upper Core Assembly -Upper Core Plate	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material (wear)	PWR Reactor Internals (B2.1.35)	IV.B2.RP-290b	3.1.1.27	A, 3

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
RVI Upper Core Assembly-Upper Core Plate	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	PWR Reactor Internals (B2.1.35)	IV.B2.RP-291b	3.1.1.80	A, 3
RVI Upper Core Plate Alignment Pins	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)	IV.B2.RP-301	3.1.1.37	A, 3
RVI Upper Core Plate Alignment Pins	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material	Water Chemistry (B2.1.2)	IV.B2.RP-24	3.1.1.83	A
RVI Upper Core Plate Alignment Pins	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material (wear)	PWR Reactor Internals (B2.1.35)	IV.B2.RP-299	3.1.1.22	A, 3
RVI Upper Core Support-Upper Support Column	SS	Stainless Steel	Reactor Coolant (Ext)	Cumulative fatigue damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.B2.RP-303	3.1.1.05	A
RVI Upper Core Support-Upper Support Plate Assembly (All Components)	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material	Water Chemistry (B2.1.2)	IV.B2.RP-24	3.1.1.83	A
RVI Upper Instrument Conduit and Supports (all components)	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material	Water Chemistry (B2.1.2)	IV.B2.RP-24	3.1.1.83	A
RVI Upper Instrumentation Conduit and Supports (All Components)	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material	Water Chemistry (B2.1.2)	IV.B2.RP-24	3.1.1.83	A
RVI Upper Support Plate Assembly (All Components)	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Loss of material	Water Chemistry (B2.1.2)	IV.B2.RP-24	3.1.1.83	A

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Vol. 2 Item	Table 1 Item	Notes
RVI Upper Support Plate Assembly - Upper Support Skirt	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cracking	Water Chemistry (B2.1.2) and PWR Reactor Internals (B2.1.35)	IV.B2.RP-346	3.1.1.37	A, 3
RVI Upper Support Plate Assembly- Upper Support Skirt	SS	Stainless Steel	Reactor Coolant and neutron flux (Ext)	Cumulative fatigue damage	Time-Limited Aging Analysis evaluated for the period of extended operation	IV.B2-303	3.1.1.05	A
Seal Table	PB	Stainless Steel	Borated Water Leakage (Ext)	None	None	IV.E-3	3.1.1.86	A

Notes for Table 3.1.2-1:

Standard Notes:

A Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.

B Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP.

C Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.

E Consistent with NUREG-1801 for material, environment, and aging effect, but a different aging management program is credited or NUREG-1801 identifies a plant-specific aging management program.

G Environment not in NUREG-1801 for this component and material.

I Aging effect in NUREG-1801 for this component, material and environment combination is not applicable.

Plant Specific Notes:

1 Includes the plant specific Nickel-Alloy Aging Management Program (B2.1.34) in addition to the programs identified in NUREG-1801.

2 NUREG-1801 does not address the aging effect of nickel-alloys in borated water leakage. Nickel-alloys subject to an air with borated water leakage environment are similar to stainless steel in a borated water leakage environment and do not experience aging effects due to borated water leakage.

3 Plant-specific aging management program PWR Reactor Internals (B2.1.35), is credited to manage this aging effect.

4 ~~See Further Evaluation in Section 3.1.2.2.15.~~ STP uses 403 SS Hold Down Springs which do not require aging management.

5 ~~See Further Evaluation in Section 3.1.2.2.12.~~ Aging management program Flux Thimble Tube Inspection (B2.1.21), is credited to manage this aging effect.

~~6 See Further Evaluation in Section 3.1.2.2.9.~~

~~7 See Further Evaluation in Section 3.1.2.2.17.~~

~~8 See Further Evaluation in Section 3.1.2.2.6.~~

~~9 Consistent with EPRI 1016596 (MRP-227), loss of fracture toughness is not an applicable aging effect requiring management for the upper support column base.~~

Enclosure 3

New LRA Appendix C, Response to Applicant Action Items for Inspection and Evaluation Guidelines for PWR Internals and STP's Reactor Internals Program Inspection Plan

Electric Power Research Institute (EPRI) has published the NRC-approved version of Materials Reliability Program (MRP) Report 1022863 (MRP-227-A), "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines." This report was developed to provide inspection and evaluation guidelines as part of an aging management program for PWR reactor vessel internal components.

The NRC safety evaluation for MRP-227 is included in MRP-227-A, in which the NRC staff determined that MRP-227 is acceptable for referencing in license renewal applications for PWR internals inspection and evaluation. The safety evaluation includes eight plant-specific applicant action items. Included in these items is a request to provide a plant-specific reactor vessel internals inspection plan.

Appendix C includes the following:

1.0 STP conformance with MRP-227 Assumptions
2.0 Topical Report Conditions and Licensee Action Items
3.0 Plant-specific reactor vessel internals inspection plan.

1.0 Conformance with MRP-227 Assumptions

1.1 **Assumption 1: Thirty years of operation with high leakage core loading patterns (fresh fuel assemblies loaded in peripheral locations) followed by implementation of a low-leakage fuel management strategy for the remaining years of operation.**

STPNOC Response: *STPNOC implemented a full low-leakage fuel pattern during Cycle 6 for STP Unit 1 and Cycle 5 for STP Unit 2. The assumption of 30 years of operation with a high-leakage core is bounded for STP.*

1.2 **Assumption 2: Base load operation, i.e., typically operates at full power levels and does not usually vary power on a calendar or load demand schedule.**

STPNOC Response: *STP Unit 1 and Unit 2 are considered base load units.*

1.3 **Assumption 3: No design changes beyond those identified in general industry guidance or recommended by the original vendors.**

STPNOC Response: *STP Unit 1 and Unit 2 have had no identified changes beyond those recommended by OEM.*

2.0 Topical Report Conditions and Licensee Action Items

The final NRC Safety Evaluation (SE) for MRP-227-A contains seven Topical Report Conditions and eight Applicant/Licensee Action Items. This section provides the STPNOC responses to the Topical Report Conditions and Applicant/Licensee Action Items that are incorporated in the inspection plan.

2.1 Topical Report Conditions

2.1.1 **SE Section 4.1.1, Topical Report Condition 1: Moving components from "No Additional Measures" to "Expansion" category.**

STPNOC Response: *In accordance with SE Section 4.1.1, the (1) Upper Core Plate and (2) Lower Support Forging or Casting have been added to the STPNOC "Expansion" category and are contained in Table 4.6. The components are linked to, and examination method is consistent with, the "Primary" components Control Rod Guide Tube Assembly Lower Flange Welds.*

2.1.2 **SE Section 4.1.2, Topical Report Condition 2: Inspection of components subject to irradiation stress corrosion cracking.**

STPNOC Response: *In accordance with SE Section 4.1.2, the (1) Upper and Lower Core barrel Girth Welds and the (2) Lower Core Barrel Flange Welds have been added to the STPNOC "Primary" inspection category and are contained in table 4-3. The examination method is consistent with the MRP recommendations for these components, the exam coverage conforms to the criteria described in Section 3.3.1 of the NRC SE, and the re-examination period is on a 10-year interval consistent with other "Primary" inspection category components.*

2.1.3 **SE Section 4.1.3, Topical Report Condition 3: Inspection of high consequence components subject to multiple degradation mechanisms.**

STPNOC Response: *This condition and limitation is only applicable to B&W and CE designed reactors. NOT applicable to STPNOC (Westinghouse Design Plants).*

2.1.4 **SE Section 4.1.4, Topical Report Condition 4: Imposition of minimum examination coverage criteria for "Expansion" inspection category components.**

STPNOC Response: *One hundred percent of the volume/area of each accessible components will be examined. The minimum examination coverage for "Primary" and "Expansion" inspection categories is seventy-five percent of the component's total (accessible plus inaccessible) inspection volume/area be examined or, when addressing a set of like components, that the inspection examine a minimum sample size of seventy-five percent of the total population of like components. Defects shall be entered into the Corrective Action Program and evaluated whether the component, or set of like components, will continue to meet their intended function under all licensing basis conditions of operation until the next scheduled examination.*

2.1.5 **SE Section 4.1.5, Topical Report Condition 5: Examination frequencies for baffle former bolts and core shroud bolts.**

STPNOC Response: *The frequency of examination for baffle former bolts has been revised to a ten-year interval following the initial or baseline examination.*

2.1.6 **SE Section 4.1.6, Topical Report Condition 6: Periodicity of the re-examination of "Expansion" inspection category components.**

STPNOC Response: *In accordance with SE Section 4.1.6, Table 4-6 has been revised to require a 10-year re-examination frequency for all "Expansion" inspection category components once degradation is identified in the associated "Primary" inspection category component.*

2.1.7 **SE Section 4.1.7, Topical Report Condition 7: Updating of industry guideline MRP-227-A, Appendix A to include a reference to AMP XI.M16A in NUREG 1801, Revision 2.**

STPNOC Response: *STPNOC conforms to the recommended program element criteria in AMP XI.M16A, Revision 2. No plant specific actions are required.*

2.2 Applicant/Licensee Action Items (A/LAI)

2.2.1 **SE Section 4.2.1, Applicant/Licensee Action Item 1: Applicability of FMECA and Functionality Analysis Assumptions.**

STPNOC Response: *The STPNOC Units 1 and 2 Reactor Vessel Internals (RVI) components are reasonably represented by the design and operating history assumptions regarding neutron fluence, temperature, materials, and stress values in the MRP-191 generic Failure Modes, Effects and Criticality Analysis (FMECA) and in the MRP-232 functionality analysis. The STPNOC Units1 and 2 comply with A/LAI 1 of the NRC SE regarding MRP-227-A. Therefore the requirements are met for application of MRP-227-A as a strategy for managing age-related material degradation in the RVI components. (Ref. PWROG-15001-P)*

2.2.2 **SE Section 4.2.2, Applicant/Licensee Action Item 2: PWR Vessel Internals Components within the Scope of License Renewal.**

STPNOC Response: *The generic scoping and screening of the RVI, as summarized in MRP-191 and MRP-232, to support the inspection sampling approach for aging management of the RVI specified in MRP-227-A are applicable to STPNOC Units 1 and 2 with no modifications for the components. STPNOC Units 1 and 2 comply with A/LAI 2 of the NRC SE in MRP-227-A for all components. (Ref. PWROG-15001-P).*

2.2.3 **SE Section 4.2.3, Applicant/Licensee Action Item 3: Evaluation of the Adequacy of Plant-Specific Existing Programs.**

STPNOC Response: *STPNOC replaced the X-750 guide tube support pins (split pins) with cold worked (CW) 316 stainless steel during refueling outages 1RE12 and 2RE11 (2005). The new split pins were qualified for a 60 year design objective at a 100% capacity factor (WCAP-16620-P, Rev.0). Potential aging effects were evaluated including those identified in MRP-191 Table 5-1. No additional inspection requirements were established for the control rod guide tube support pins in the design change packages that installed them based on the following:*

- *Cold-worked Type 316 SS split pins have been installed at other plants since 1997 and none of these plants have experienced any failures.*
- *Since other plants have installed split pins since 1997 and STPNOC did not install them until 2005 for Units 1 and 2, the other plants will provide a leading indicator.*

At STPNOC the effects of aging on these components will be managed in the period of extended operation based on operating experience.

2.2.4 **SE Section 4.2.4, Applicant/Licensee Action Item 4: B&W Core Support Structure Upper Flange Stress Relief.**

STPNOC Response: *This condition and limitation is only applicable to B&W designed reactors. NOT applicable to STPNOC (Westinghouse Design Plants).*

2.2.5 **SE Section 4.2.5, Applicant/Licensee Action Item 5: Application of Physical Measurements as Part of I&E Guidelines for B&W, CE, and Westinghouse RVI Components.**

STPNOC Response: *The reactor internals hold down spring is fabricated with Type 403 stainless steel. The type 403 stainless steel reactor internals hold down spring is screened as FMECA Group 1, Category A in MRP-191, Table 7-2. The Category A assignment is applied to components that are deemed with "low probability of failure".*

2.2.6 **SE Section 4.2.6, Applicant/Licensee Action Item 6: Evaluation of Inaccessible and Non-Inspectable B&W Components.**

STPNOC Response: *This condition and limitation is only applicable to B&W designed reactors. NOT applicable to STPNOC (Westinghouse Design Plants).*

2.2.7 **SE Section 4.2.7, Applicant/Licensee Action Item 7: Plant Specific Evaluation of CASS Materials.**

<u>STPNOC Response:</u> A/LAI 7, from the NRC final SE on MRP-227 [3], states that, for assessment of CASS materials, the applicant/licensee for license renewal may apply the criteria in the NRC letter of May 19, 2000, "License Renewal Issue No. 98-0030, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components,"" as the basis for determining whether the CASS materials are susceptible to the thermal aging mechanism. If the application of the applicable screening criteria for the components material demonstrates that the components are not susceptible to either TE or IE, or to the synergistic effects of TE and IE combined, then no other evaluation would be necessary.

Both the STP Units 1 and 2 upper internals assembly subcomponents a) upper support columns, and b) column base, are comprised of CASS, Grade CF8 material. Accounting for the elemental percentages from the chemical data retrieved from the certified material test report (CMTRs), calculated percentage delta ferrite and potential for thermal embrittlement (TE), it is concluded that continued application of the MRP-227-A strategy will meet the requirement for managing age-related degradation of the STP Units 1 and 2 CASS reactor vessel internals components (Ref. PWROG-15001-P).

2.2.8 **SE Section 4.2.8, Applicant/Licensee Action Item 8: Submittal of Information for Staff Review and Approval.**

<u>STPNOC Response:</u> In accordance with the NUREG-1801, Revision 2, the following information is provided for the following items (1) through (5) for staff review and approval.

1. **An AMP for the facility that addresses the 10 program elements as defined in NUREG-1801, Revision 2, AMP XI.M16A.**

   <u>STPNOC Response:</u> STPNOC AMP B2.1.35, South Texas Project License Renewal Program Evaluation address the 10 program elements as defined NUREG-1801 (GALL Report), Revision 2, Chapter XI.M16A, "PWR Vessel Internals" as modified by LR-ISG-2011-04.

2. **To ensure the MRP-227 program and plant-specific action items will be carried out by applicants/licensees, applicants/licensees are to submit an inspection plan which addresses the identified plant-specific action items for staff review and approval consistent with the licensing basis for the plant. If an applicant/licensee plans to implement an AMP which deviates from the guidance provided in MRP-227, as approved by the NRC, the applicant/licensee shall identify where their program deviates from the**

recommendations of MRP-227, as approved by the NRC, and shall provide a justification for any deviation which includes a consideration of how the deviation affects both "Primary" and "Expansion" inspection category components.

<u>STPNOC Response:</u> The PWR Vessel Internals program (B2.1.35) is described in LRA Section A1.35 and LRA Section B2.1.35. The STPNOC Inspection Plan is included in LRA Appendix C and addresses plant specific action items and does NOT identify any deviations to MRP-227-A.

3. **The regulation at 10 CFR 54.21(d) requires that an FSAR supplement for the facility contain a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAAs for the period of extended operation. Those applicants for LR referencing MRP-227, as approved by the NRC, for their RVI component AMP shall ensure that the programs and activities specified as necessary in MRP-227, as approved by the NRC, are summarily described in the FSAR supplement.**

   <u>STPNOC Response:</u> The UFSAR Supplement is included in LRA Appendix A, Section A1.35 and includes a summary of the program and activities specified as necessary for the PWR Reactor Internals (B2.1.35) program.

4. **The regulation at 10 CFR 54.22 requires each applicant for LR to submit any TS changes (and the justification for the changes) that are necessary to manage the effects of aging during the period of extended operation as part of its LR application (LRA). For the plant CLBs that include mandated inspection or analysis requirements for RVI either in the operating license for the facility or in the facility TS, the applicant/licensee shall compare the mandated requirements with the recommendations in the NRC-approved version of MRP-227. If the mandated requirements differ from the recommended criteria in MRP-227, as approved by the NRC, the conditions in the applicable license conditions or TS requirements take precedence over the MRP recommendations and shall be complied with.**

   <u>STPNOC Response:</u> No changes to the Technical Specifications (TS) are required.

5. **Pursuant to 10 CFR 54.21(c)(1), the applicant is required to identify all analyses in the CLB for their RVI components that conform to the definition of a TLAA in 10 CFR 54.3 and shall identify these analyses as TLAAs for the application in accordance with the TLAA identification requirement in 10 CFR 54.21(c)(1). MRP-227 does not specifically address the resolution of TLAAs that may apply to applicant/licensee RVI components. Hence, applicants/licensees who implement MRP-227, as approved by the NRC, shall still evaluate the CLB for their facilities to determine if they have plant-specific TLAAs that shall be addressed. If so, the applicant's/licensee's TLAA shall be submitted for NRC review along with the applicant's/licensee's application to implement the NRC- approved version of MRP-227.**

   STPNOC Response: Reactor Internals TLAAs are addressed in LRA Section 4.3.3 and does not credit the PWR Reactor Vessel Internals program (B2.1.35).

3.0 PWR Vessel Internals Inspection Plan

The PWR Vessel Internals Inspection Plan is provided in Tables A through E.

- Table A specifies the vessel internal components classified as Primary components and is based on MRP-227-A, Table 4.3.
- Table B specifies the vessel internal components classified as expansion components and is based on MRP-227-A, Table 4.6.
- Table C specifies the examination acceptance and expansion criteria and is based on MRP-227-A, Table 5.3.
- Table D specifies the components that are classified as Existing Program components.
- Table E provides the Examination Plan Summary

Table A: Westinghouse/STP Primary components

Item / ISG 2011-04	**Applicability**	**Effect (Mechanism)**	**Expansion Link (Note 1)**	**Examination Method/Frequency (Note 1)**	**Examination Coverage** (see Addendum 5 for figures)
Control Rod Guide Tube Assembly Guide plates (cards) IV.B2.RP-296	All plants	Loss of Material (Wear)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period, and no earlier than two refueling outages prior to the start of the license renewal period. Subsequent examinations are required on a ten-year interval.	20% examination of the number of CRGT assemblies, with all guide cards within each selected CRGT assembly examined. See Figure 4-20
Control Rod Guide Tube Assembly Lower flange welds IV.B2.RP-297 IV.B2.RP-298	All plants	Cracking (SCC, Fatigue) Loss of Fracture Toughness (IE)	Bottom-mounted instrumentation (BMI) column bodies, Upper core plate	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	100% of outer (accessible) CRGT lower flange weld surfaces and adjacent base metal on the individual periphery CRGT assemblies. (Note 5) See Figure 4-21.
Core Barrel Assembly Upper core barrel flange weld IV.B2.RP-276	All plants	Cracking (SCC IASCC, Fatigue)	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 4-22.

Table A: Westinghouse/STP Primary components					
Item / ISG 2011-04	**Applicability**	**Effect (Mechanism)**	**Expansion Link (Note 1)**	**Examination Method/Frequency (Note 1)**	**Examination Coverage** (see Addendum 5 for figures)
Core Barrel Assembly Upper and lower core barrel Circumferential girth welds IV.B2.RP-387 IV.B2.RP-388	All plants	Cracking (SCC, IASCC, Fatigue) Loss of Fracture Toughness (IE)	Upper and lower core barrel vertical 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.	Bolts and locking devices on high fluence seams.100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 3). See Figure 4-22.
Core Barrel Assembly Lower core barrel flange weld (Note 4) IV.B2.RP-280	All plants	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).
Baffle-Former Assembly Baffle-edge bolts	All plants with baffle-edge bolts N/A STP does not have baffle-edge bolts				
Baffle-Former Assembly Baffle-former bolts IV.B2.RP-271 IV.B2.RP-272	All plants	Cracking (SCC, IASCC, Fatigue) Loss of Fracture Toughness (IE and Loss of Preload (ISR)	Barrel-former bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination 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 4-23 and 4-24.

Table A: Westinghouse/STP Primary components					
Item / ISG 2011-04	**Applicability**	**Effect (Mechanism)**	**Expansion Link (Note 1)**	**Examination Method/Frequency (Note 1)**	**Examination Coverage** (see Addendum 5 for figures)
Baffle-Former Assembly Assembly (Includes: Baffle plates, and indirect effects of void swelling in former plates) IV.B2.RP-270 IV.B2.RP-270a	All plants	Change in Dimensions (Distortion or Void Swelling), or Cracking (IASCC), Fatigue) that results in • Abnormal interaction with fuel assemblies • Gaps along high fluence baffle joint • Vertical displacement of baffle plates near high fluence joint	None	Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval.	Core side surface as indicated. See Figures 4-24, 4-25, 4-26 and 4-27.
Alignment and Interfacing Components Internals hold down spring IV.B2.RP-300	All plants with 304 stainless steel hold down springs STP utilizes 403 SS hold down springs				
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields	N/A STP does not utilize flexure design			

Notes Table A:

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

2. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-3, 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 5-3, 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.

Table B: Westinghouse/STP Expansion components					
Item / ISG-2011-04	**Applicability**	**Effect (Mechanism)**	**Primary Link (Note 1)**	**Examination Method (Note 1)**	**Examination Coverage** (see Addendum 5 for figures)
Upper Internals Assembly Upper core plate IV.B2.RP-290b IV.B2.RP-291b	All plants	Cracking (Fatigue), Loss of Material (Wear)	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 Lower support forging or castings	All plants N/A STP has an extended core does not have a Lower Support Assembly				
Baffle-Former Assembly Barrel to former bolts IV.B2.RP-273 IV.B2.RP-274	All plants	Cracking (IASCC, Fatigue) Loss of Facture Toughness `(IE), Change in Dimensions (Distortion or Void Swelling) and Loss of Preload (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 or neutron pads (Note 2). See Figure 4-23.
Lower Support Assembly Lower support column bolts	All plants N/A STP has an extended core does not have a Lower Support Assembly				
Core Barrel Assembly Core barrel outlet nozzle welds IV.B2.RP-278 IV.B2.RP-278a	All plants	Cracking (SCC, Fatigue) Facture Toughness (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 4-22.
Core Barrel Assembly Upper and lower core barrel vertical axial welds IV.B2.RP-387a IV.B2.RP-388a	All plants	Cracking (SCC, IASCC, Fatigue) Facture Toughness (IE)	Upper and lower core barrel Circumferential 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 4-22.

Table B: Westinghouse/STP Expansion components					
Item / ISG-2011-04	**Applicability**	**Effect (Mechanism)**	**Primary Link (Note 1)**	**Examination Method (Note 1)**	**Examination Coverage** (see Addendum 5 for figures)
Lower Support Assembly Lower support column bodies (non cast)	All plants N/A STP has an extended core does not have Lower Support Assembly				
Lower Support Assembly Lower support column bodies (cast)	All plants N/A STP has an extended core does not have a Lower Support Assembly				
Bottom Mounted Instrumentation System Bottom-mounted instrumentation (BMI) column bodies IV.B2.RP-292 IV.B2.RP-293	All plants	Cracking (Fatigue) including the detection of completely fractured column bodies Facture Toughness (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 Figure 4-35.

Notes to Table B:

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

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).

Table C: Westinghouse/STP examination acceptance and expansion criteria					
Item	**Applicability**	**Examination Acceptance Criteria (Note 1)**	**Expansion Link(s)**	**Expansion Criteria**	**Additional Examination Acceptance Criteria**
Control Rod Guide Tube Assembly Guide plates (cards)	All plants	Visual (VT-3) examination The specific relevant condition is wear that could lead to loss of control rod alignment and impede control assembly insertion.	None	N/A	N/A
Control Rod Guide Tube Assembly Lower flange welds	All plants	Enhanced visual (EVT-1) examination The specific relevant condition is a detectable crack-like surface indication.	a. Bottom-mounted instrumentation (BMI) column bodies b. upper core plate	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 s, upper core plate within three fuel cycles following the initial observation.	a. For BMI column bodies, the specific relevant condition for the VT-3 examination is completely fractured column bodies. b. For cast lower support column bodies, upper core plate and lower support forging/castings, the specific relevant condition is a detectable crack-like surface indication.

Table C: Westinghouse/STP examination acceptance and expansion criteria					
Item	**Applicability**	**Examination Acceptance Criteria (Note 1)**	**Expansion Link(s)**	**Expansion Criteria**	**Additional Examination Acceptance Criteria**
Core Barrel Assembly Upper core barrel flange weld	All plants	Periodic enhanced visual (EVT-1) examination The specific relevant condition is a detectable crack-like surface indication.	a. Core barrel outlet nozzle welds	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.	a and b. The specific relevant condition for the expansion core barrel outlet nozzle weld and lower support column body examination is a detectable crack-like surface indication.
Core Barrel Assembly Lower core barrel flange weld (Note 2)	All plants	Periodic enhanced visual (EVT-1) examination The specific relevant condition is a detectable crack-like surface indication.	None	None	None
Core Barrel Assembly Upper core barrel circumferential girth welds	All plants	Periodic enhanced visual (EVT-1) examination The specific relevant condition is a detectable crack-like surface indication.	Upper core barrel vertical axial welds	The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel circumferential girth welds shall require that the EVT-1 examination be expanded to include the upper core barrel vertical axial welds by the completion of the next refueling outage.	The specific relevant condition for the expansion upper core barrel vertical axial weld examination is a detectable crack-like surface indication.

Table C: Westinghouse/STP examination acceptance and expansion criteria

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Lower core barrel circumferential girth welds	All plants	Periodic enhanced visual (EVT-1) examination The specific relevant condition is a detectable crack-like surface indication.	Lower core barrel vertical axial welds	The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the lower core barrel circumferential girth welds shall require that the EVT-1 examination be expanded to include the lower core barrel vertical axial welds by the completion of the next refueling outage.	The specific relevant condition for the expansion lower core barrel vertical axial weld examination is a detectable crack-like surface indication.
Baffle-Former Assembly Baffle-edge bolts	All plants with baffle-edge bolts N/A STP does not have baffle-edge bolts				
Baffle-Former Assembly Baffle-former bolts	All plants	Volumetric (UT) examination The examination acceptance criteria for the UT of the baffle-former bolts shall be established as part of the examination technical justification.	a. Barrel-former bolts	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 barrel-former bolts bolts within the next three fuel cycles.	The examination acceptance criteria for the UT of the barrel-former bolts shall be established as part of the examination technical justification.

Table C: Westinghouse/STP examination acceptance and expansion criteria					
Item	**Applicability**	**Examination Acceptance Criteria (Note 1)**	**Expansion Link(s)**	**Expansion Criteria**	**Additional Examination Acceptance Criteria**
Baffle-Former Assembly Assembly	All plants	Visual (VT-3) examination The specific relevant conditions are evidence of abnormal interaction with fuel assemblies, gaps along high fluence shroud plate joints, vertical displacement of shroud plates near high fluence joints, and broken or damaged edge bolt locking systems along high fluence baffle plate joints.	None	N/A	N/A
Alignment and Interfacing Components Internals hold down spring	All plants with 304 stainless steel hold down springs STP utilizes 403 SS hold down springs	None			
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields N/A STP does not utilize flexure design				

Notes to Table C:

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.

Table D: Westinghouse/STP Existing Programs components					
Item	**Applicability**	**Effect (Mechanism)**	**Reference**	**Examination Method**	**Examination Coverage**
Core Barrel Assembly Core barrel flange	All plants	Loss of material (Wear)	PWR Reactor Internals (B2.1.35) IV.B2.RP-345	Visual (VT-3) examination to determine general condition for excessive wear	All accessible surfaces at specified frequency
Upper Internals Assembly Upper support skirt	All plants Also includes bolts, locking devices,	Cracking (SCC, Fatigue)	ASME Code Section XI Items 12 and 13 and PWR Reactor Internals (B2.1.35) IV.B2.RP-346	Visual (VT-3) examination	All accessible surfaces at specified frequency
Lower Internals Assembly Lower core plate XL lower core plate (Note 1)	All plants STP has the XL lower core plate	Cracking (IASCC, Fatigue) Facture Toughness (IE) and Loss of Material (Wear) (SR)	PWR Reactor Internals (B2.1.35) IV.B2.RP-288 and IV.B2.RP-289	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
Bottom Mounted Instrumentation System Flux thimble tubes	All plants	Loss of material (Wear)	Flux Thimble Tube Inspection (B2.1.21)	Surface (ET) examination	Eddy current surface examination as defined in plant response to IEB 88-09. (ref. 8.1.22)
Alignment and Interfacing Components Clevis insert bolts	All plants	Cracking (SCC, IASCC, Fatigue Loss of material (Wear) Loss of Preload (SR) (Note 2)	PWR Reactor Internals (B2.1.35) IV.B2.RP-399 and IV.B2.RP-285	Visual (VT-3) examination	All accessible surfaces at specified frequency
Alignment and Interfacing Components Upper core plate alignment pins	All plants	Loss of material (Wear) Cracking (SCC)	PWR Reactor Internals (B2.1.35) IV.B2.RP-299 and IV.B2.RP-301	Visual (VT-3) examination	All accessible surfaces at specified frequency
RVI Control Rod Guide Tube Assembly Guide Tube Support Pins	All plants	Loss of material (wear)	PWR Reactor Internals (B2.1.35) IV.B2.RP-356	Visual (VT-3) examination	All accessible surfaces at specified frequency

Notes to Table D:

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.

Table E: Examination Plan Summary Table

Primary Component	Expansion Links	Inspection Schedule		Comments
		STPNOC Unit 1	STPNOC Unit 2	
Control Rod Guide Tube Assembly Guide plates (cards)	None	1RE26 (Spring 2026)	2RE25 (Spring 2027)	See tables 4-3, 4-6, and 5-3 for primary/expansion components, and acceptance criteria
Control Rod Guide Tube Assembly Lower flange welds	Bottom-mounted Instrumentation (BMI) column bodies, Upper core plate	1RE26 (Spring 2026)	2RE25 (Spring 2027)	See tables 4-3, 4-6, and 5-3 for primary/expansion components, and acceptance criteria
Core Barrel Assembly Upper core barrel flange weld	Core barrel outlet nozzle welds	1RE26 (Spring 2026)	2RE25 (Spring 2027)	See tables 4-3, 4-6, and 5-3 for primary/expansion components, and acceptance criteria
Core Barrel Assembly Upper and lower core barrel Circumferential girth welds	Upper and lower core barrel vertical axial welds	1RE26 (Spring 2026)	2RE25 (Spring 2027)	See tables 4-3, 4-6, and 5-3 for primary/expansion components, and acceptance criteria
Core Barrel Assembly Lower core barrel flange weld	None	1RE26 (Spring 2026)	2RE25 (Spring 2027)	See tables 4-3, 4-6, and 5-3 for primary/expansion components, and acceptance criteria
Baffle-Former Assembly Baffle-former bolts	Barrel-former bolts	1RE26 (Spring 2026)	2RE25 (Spring 2027)	See tables 4-3, 4-6, and 5-3 for primary/expansion components, and acceptance criteria
Baffle-Former Assembly (Includes: Baffle plates, and indirect effects of void swelling in former plates)	None	1RE26 (Spring 2026)	2RE25 (Spring 2027)	See tables 4-3, 4-6, and 5-3 for primary/expansion components, and acceptance criteria