



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 11, 2014

Vice President, Operations
Entergy Nuclear Operations, Inc.
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

SUBJECT: PALISADES NUCLEAR PLANT – STAFF ASSESSMENT RE: REVISED
PROGRAM PLAN FOR AGING MANAGEMENT OF REACTOR VESSEL
INTERNALS (TAC NO. ME9569)

Dear Sir or Madam:

By letter dated September 13, 2012, as supplemented by letters dated April 4, 2013, April 3, July 1, and August 14, 2014, Entergy Nuclear Operations, Inc. (the licensee), submitted a program plan "Palisades Reactor Vessel Internals Aging Management Program," for U.S. Nuclear Regulatory Commission (NRC) review. The Palisades Nuclear Plant (PNP) aging management plan (AMP) was developed based on the NRC staff approved topical report MRP-227-A, "Material Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." The submittal of this AMP was to fulfill a regulatory commitment that originates from license renewal activities as documented in NUREG-1871, "Safety Evaluation Report Related to the License Renewal of PNP Nuclear Plant."

The NRC staff has completed its review of the PNP reactor vessel internals AMP and concludes that it is acceptable because it is consistent with the inspection and evaluation guidelines of MRP-227-A, and the licensee has appropriately addressed all eight applicant/licensee action items specified in MRP-227-A.

Regulatory Commitment #33, as documented in Appendix A of NUREG-1871, and the revised commitment submitted to the NRC by letter dated June 20, 2012, is considered fulfilled. The NRC staff's approval of the PNP reactor vessel internals AMP does not reduce, alter, or otherwise affect current American Society of Engineers Code, Section XI inservice inspection requirements, or any PNP specific licensing requirements related to inservice inspection. The NRC staff notes that Section 7.0, "Implementation Requirements" of MRP-227-A requires that the NRC be notified of any deviations from the "Needed" requirements.

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If you have any questions concerning this matter, please contact the Project Manager, Jennivine Rankin at (301) 415-1530 or via e-mail at Jennivine.Rankin@nrc.gov.

Sincerely,

A handwritten signature in black ink, appearing to read 'D. Pelton', with a long horizontal line extending to the right.

David L. Pelton, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosure:
Staff Assessment

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UNITED STATES
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STAFF ASSESSMENT BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO PROGRAM PLAN FOR AGING MANAGEMENT
OF REACTOR VESSEL INTERNALS
ENTERGY NUCLEAR OPERATIONS, INC.
PALISADES NUCLEAR PLANT
DOCKET NO. 50-255

1.0 INTRODUCTION

By letter dated September 13, 2012 (Reference 1), Entergy Nuclear Operations, Inc. (the licensee), submitted a program plan "Palisades Reactor Vessel Internals Aging Management Program," for U.S. Nuclear Regulatory Commission (NRC) review. This plan will be referred to as the Palisades Nuclear Plant (PNP) Aging Management Program (AMP) throughout this staff assessment (SA). The PNP AMP was developed by the licensee based on the NRC staff approved Electric Power Research Institute topical report, "Material Reliability Program: Pressurized Water Reactor [PWR] Internals Inspection and Evaluation [I&E] Guidelines (MRP-227-A)" (Reference 2). The submittal of the PNP AMP was to fulfill a regulatory commitment originating from license renewal activities as documented in NUREG-1871, "Safety Evaluation Report Related to the License Renewal of PNP Nuclear Plant" (Reference 3).

Commitment #33 of Reference 3 states the following:

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

By letter dated March 23, 2009 (Reference 4), the licensee changed the due date for Commitment #33 to allow time for completion of the industry program. The licensee submitted a Reactor Vessel Internals Program by letter dated March 10, 2010 (Reference 5). By letter dated March 31, 2011 (Reference 6), the licensee withdrew the submittal of Reference 5. In accordance with NRC Regulatory Issue Summary (RIS) 2011-07 (Reference 7), by letter dated June 20, 2012 (Reference 8), the licensee changed their commitment to the following:

Enclosure

ENO will submit a revised program plan for aging management of reactor vessel internals, for PNP, in accordance with MRP-227-A.

Scheduled Completion Date: October 1, 2012

The NRC staff reviewed the PNP AMP inspection plan for the reactor vessel internals (RVI) components at PNP and the licensee's responses to NRC requests for additional information (RAIs) dated April 4, 2013 (Reference 9), April 3, 2014 (Reference 10), July 1, 2014 (Reference 11), and August 14, 2014 (Reference 12), to determine whether the licensee provided an inspection plan consistent with the I&E guidelines in MRP-227-A.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (CFR) Part 54 addresses the requirements for plant license renewal (LR). The regulation at 10 CFR Section 54.21 requires that each application for LR contain an integrated plant assessment (IPA) and an evaluation of time-limited aging analyses (TLAAs). The plant-specific IPA shall identify and list those structures and components subject to an aging management review (AMR) and demonstrate that the effects of aging (cracking, loss of material, loss of fracture toughness, dimensional changes, loss of preload) will be adequately managed so that their intended functions will be maintained consistent with the current licensing basis (CLB) for the period of extended operation (PEO) as required by 10 CFR 54.29(a). In addition, 10 CFR 54.22 requires that a LR application include any technical specification (TS) changes or additions necessary to manage the effects of aging during the PEO as part of the LR application.

Structures and components subject to an AMP shall encompass those structures and components that (1) perform an intended function, as described in 10 CFR 54.4, without moving parts or without a change in configuration or properties and (2) are not subject to replacement based on a qualified life or specified time period. These structures and components are referred to as "passive" and "long-lived" structures and components, respectively.

On January 12, 2009, the Electric Power Research Institute (EPRI) submitted for NRC staff review and approval report, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)" (Reference 13), which was intended as guidance for licensees in developing their plant-specific AMP for RVI components. Revision 1 to the final safety evaluation regarding MRP-227, Revision 0, was issued on December 16, 2011 (Reference 14). This SE contains specific conditions on the use of the topical report and applicant/licensee action items that must be addressed by those utilizing the topical report as the basis for a submittal to the NRC. On January 9, 2012, EPRI published the NRC-approved version of the topical report, designated MRP-227-A. MRP-227-A contains a discussion of the technical basis for the development of plant-specific AMPs for RVI components in PWR vessels and also provides inspection and evaluation guidelines for PWR applicants to use in their plant-specific AMPs. MRP-227-A provides the basis for renewed license holders to develop plant-specific inspection plans to manage aging effects on RVI components, as described by their FSAR commitment.

The scope of components considered for inspection under the guidance of MRP-227-A includes core support structures, which are typically denoted as Examination Category B-N-3 by

Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), and those RVI components that serve an intended safety function consistent with the criteria in 10 CFR 54.4(a)(1). The scope of the program does not include consumable items such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation because these components are not subject to an AMR, as defined in 10 CFR 54.21(a)(1).

Subsequent to the submittal of MRP-227 and prior to the issuance of the SE on MRP-227, NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report – Final Report" (the GALL Report, Rev. 2) (Reference 15) was issued, providing new AMR line items and aging management guidance in AMP XI.M16A, "PWR Vessel Internals." The PNP AMP was based on NRC staff expectations for the guidance to be provided in MRP-227-A. Since the GALL Report, Rev. 2 was published prior to the issuance of the final SE of MRP-227-A, the staff published LR Interim Staff Guidance LR-ISG-2011-04: "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors," (Reference 16) which modifies the guidance of AMP XI.M16A to be consistent with MRP-227-A.

The licensee stated in its September 13, 2012, letter that it was submitting the revised RVI AMP to complete the regulatory commitment documented by letter dated June 20, 2012, which originated from Commitment #33 in NUREG-1871. The September 13, 2012, submittal indicated the revised RVI AMP was developed in accordance with MRP-227-A.

The NRC staff's review of the PNP RVI AMP focused on determining whether the licensee met its license renewal commitment to incorporate the applicable recommendations for augmented inspections and techniques resulting from the industry effort on RVI (summarized in MRP-227-A) into the PNP RVI AMP. Therefore, the most important aspects of the staff's review were the augmented inspections specified in the PNP RVI AMP, and the licensee's responses to the Applicant/Licensee Action Items (A/LAIs) from the NRC staff's final SE of MRP-227, Rev. 0. The NRC staff assessment discussed below indicates sections of the AMP used to conclude the PNP AMP is consistent with the I&E guidelines of MRP-227-A, and the licensee has appropriately addressed A/LAIs specified in MRP-227-A applicable to PNP.

3.0 TECHNICAL EVALUATION

3.1 Organization of the PNP RVI AMP

Section 1.0, "Introduction," of the AMP describes the objective, license renewal background, and provides a high level description of the program elements. Section 2.0 of the AMP contains an explanation of the mechanisms of age-related degradation in PWR internals which the program was designed to monitor. It also contains discussion of the aging management strategy of the program that comes directly from MRP-227-A. Finally, this section identifies and confirms compliance with the 10 AMP Program Elements of NUREG-1801. Section 3.0, "Palisades Reactor Vessel Internals Design and Operating Experience," describes PNP-specific operating experience (OE) related to RVI components. Section 4.0, "Program Description," outlines the structure of the AMP including preventive actions, method of evaluating OE, the basis for the component evaluation and inspections of the AMP, comparison of PNP-specific RVI components to the generic MRP-227-A components, and contains references to the tables listing the specific examination scope, method, and acceptance criteria for PNP RVI components. Section 5.0, "Examination Acceptance and Expansion Criteria," provides details of

the various examination techniques, expansion criteria, and evaluation, repair and replacement strategy if conditions not meeting the acceptance criteria are found during examinations. Section 6.0, "Operating Experience and Additional Considerations," describes how the AMP will review, evaluate, document internal and external OE, and how these results will be communicated to the industry. Section 7.0, "Responses to the NRC Safety Evaluation Report Applicant/Licensee Action Items," contains the licensee's responses to the A/LAIs from the staff's final SE of MRP-227, Rev.0. Section 8.0 contains the references. Attachment A lists all the ASME Code, Section XI, Inspection Category B-N-2 and B-N-3 inspections applicable to the PNP RVI. Attachment B contains a cross-index between the RVI components in the PNP LRA and the inspections specified by the AMP. Attachment C contains the planned scope of all existing and augmented RVI inspections at PNP.

3.2 Reactor Vessel Internals Aging Management Program Attributes Licensee Evaluation

3.2.1 Licensee Evaluation

By letter dated September 13, 2012, the licensee provided its evaluation of each of the 10 AMP Program Elements (subsections 2.3.1-2.3.10 of the PNP AMP), against the corresponding elements in Section XI.M16A, "PWR Vessel Internals," of the GALL Report, Revision 2. The licensee determined that its AMP is consistent, with no exceptions, with the corresponding element in the GALL Report, Revision 2.

3.2.2 Staff Evaluation

The NRC staff reviewed the licensee's AMP against the 10 elements of the revised version of GALL Section XI.M16A as provided in LR Interim Staff Guidance LR-ISG-2011-04. The staff found that the 10 elements of the PNP RVI AMP are consistent with the 10 elements described in LR-ISG-2011-04. Therefore, the implementation of the 10 AMP elements is acceptable for PNP.

3.3 PNP Reactor Vessel Internals Design and Operating Experience

3.3.1 Licensee Evaluation

Sections 3.0-3.6 of the PNP AMP contain descriptive information regarding the nature and configuration of the PNP RVIs. Section 3.7 "C-E Design Plants/Non-Relevant Operating Experience," indicates PNP has not had any design modifications beyond general industry guidance or original vendor recommendations. Section 3.7 also summarizes the licensee's review of OE related to PWR RVI degradation for relevance to PNP. The licensee evaluated the following items:

- Thermal shield fasteners and thermal shields have experienced fatigue and wear resulting from flow induced vibrations. SCC has also affected thermal shields. The thermal shield was removed from the PNP reactor vessel during early years of operation.

- Incore instrumentation tubes (thimble tubes) have experienced fatigue and wear damage caused by flow-induced vibrations. PNP uses top-mounted instrument guide tubes, which are not susceptible to these aging mechanisms.
- Hold-down springs have become permanently deformed, resulting in a loss of spring force, in some cases causing damage to the vessel flange and mating internals components. [As detailed in Section 2.3.10 of the RVI inspection plan, PNP's hold-down device was replaced early in plant life with a modified design due to problems with wear because of insufficient hold-down force of the core barrel keys and core barrel ledge in the RPV flange.] PNP's new hold-down device does not use Hold-down Springs, and is not susceptible to this aging mechanism. Surveillance after plant modification has shown that vibration fatigue does not occur in the core barrel.
- PNP is one of two C-E plants without susceptibility to cracking of the baffle-to-former plate (i.e., core shroud) bolts. In response to NRC's response to request for additional information (RAI No. 14) dated August 27, 2005, PNP clarified that it is one of only two C-E designed plants that uses bolts to attach the core shroud panels (i.e., the baffle or core shroud plates) to the former plates. [Compared to bolts that cracked in Westinghouse-design RVI], [i]t was determined that these bolts are less susceptible to IASCC [irradiation assisted stress corrosion cracking] because: (1) the material used in these bolts is annealed Type 316 stainless steel, which is not cold worked; (2) the bolt stress from preload, as percentage of yield strength, is much less than that of the susceptible plants; (3) the differential pressure across the core shroud panels does not result in tensile loads on the panel (i.e., the baffle or core shroud); and (4) the core shroud panel design allows for some flexing of the former plate relative to the core barrel, thus reducing the load on the panel (i.e., core shroud) bolts. [However, the NRC staff notes that the licensee plans to inspect the core shroud bolts as "Primary" components as specified by MRP-227-A.]

In addition, Section 4.2, "Operational Experience" of the PNP AMP states (Section 6.0 contains similar information) that OE related to degradation of RVI would be reviewed on a periodic basis as input to the AMP, including the inspection results summarized in a biennial MRP Inspection Data Survey. The biennial reports would then be used to assist in review of OE and trends for the RVI AMP.

3.3.2 NRC Staff Evaluation

The staff evaluated the licensee's information on OE by comparing the information presented in the licensee's submittal to the experience with RVI degradation in CE-designed RVI as described in Appendix A of MRP-227-A, to ensure the licensee independently evaluated this OE. The staff also evaluated the licensee's description of its process for reviewing and reporting RVI related OE to the process described in Section 7.6, "Aging Management Program Results Requirement" of MRP-227-A, which requires an NEI 03-08, "Guideline for the Management of Materials Issues," (Reference 17) "Needed" requirement that each licensee compile and submit its OE to MRP for inclusion in the biennial report.

The NRC staff compared the design-specific OE evaluated by the licensee to the OE

documented in Appendix A to MRP-227-A, and finds that the licensee has evaluated all the OE specific to CE-designed RVI. Specifically, the licensee evaluated and took action to modify the RVI to correct issues with thermal shield degradation due to vibration and hold-down spring deformation and loss of force. For issues with thimble tubes, the licensee evaluated the OE and determined its design is not susceptible to the types of degradation described in MRP-227-A, Appendix A due to design differences. The licensee also compared its core shroud bolt susceptibility to IASCC to the IASCC susceptibility of bolting in Westinghouse designed RVI, and found the PNP bolting less susceptible. However, the licensee is still complying with the recommended Primary inspection requirements for core shroud bolts from MRP-227-A. Therefore, the staff finds that the licensee has appropriately evaluated OE relevant to the PNP RVI design.

The NRC staff finds the licensee's description of its OE review process to be acceptable because it is consistent with the implementation guidance in Section 7.6 of MRP-227-A.

3.4 Examinations of RVI

3.4.1 Licensee Evaluation

In PNP AMP Section 4.4, "Examination of Reactor Vessel Internals," the licensee discussed the existing ASME Code Section XI inspections to be supplemented by the AMP inspections. The PNP AMP was further refined by the licensee to account for plant-specific component designs as listed in Appendix B to Attachment 1 of the submittal dated September 13, 2012. The staff evaluation of the licensee's handling of plant-specific components is located in Section 3.6.2 of this SA.

Attachment A contains a table listing the ASME Code, Section XI examinations of Examination Category B-N-2 and B-N-3 components, and Attachment C includes all the RVI component examinations, including MRP-227-A examinations, Section XI examinations, and plant-specific examinations, including the exam initial schedule, frequency, scope, method, and aging effect. Tables 2, 3, and 4 of the PNP AMP (discussed in PNP AMP Sections 4.5, 4.6, and 4.7) list "C-E Plants Primary Category Components from MRP-227-A," "C-E Plants Expansion Category Components from MRP-227," and "C-E Plants Existing Program Components Credited in MRP-227-A," respectively and are essentially identical to Tables 4-2, 4-5, and 4-8 of MRP-227-A, with comments specific to PNP. By letter dated April 4, 2013, the licensee submitted a response to RAI-8, which was transmitted by email dated March 4, 2013 (Reference 18). The response corrects Note 5 on Table 2, to include the minimum required coverage of 75% of the total population of Lower Support Structure - Core Support Column Welds.

PNP AMP Section 4.8, "Examination Systems (per MRP-228)" states the following:

Equipment, techniques, procedures and personnel used to perform examinations required under this program shall be consistent with the requirements of MRP-228 Section 7.2.

Examination results that do not meet the examination acceptance criteria (defined in MRP-227-A, Section 5) shall be recorded and entered into the ENO Corrective Action Program and dispositioned. This is "Needed" requirement 7.5

under MRP-227-A.

3.4.2 NRC Staff Evaluation

The staff evaluated the licensee's proposed augmented examinations of RVI against the information for the recommended "Primary," "Expansion," and "Existing Programs" examinations for CE designed RVI with a bolted core shroud design (the design variation applicable to PNP) from Tables 4-2, 4-5, and 4-8 of MRP-227-A. This information includes examination method, initial schedule and frequency, coverage, and Primary and Expansion links.

The comments in Table 2 indicate all the applicable "Primary" components would be inspected in 2013, with the exception of the core shroud bolts, which will be inspected in 2022 at approximately 34 effective full-power years (EFPYs). This is acceptable because MRP-227-A specifies the baseline volumetric ultrasonic testing examination of the core shroud bolts should take place between 25-35 EFPY.

In Table 2 of the PNP AMP, the component "Core Support Barrel Assembly – Lower Flange Weld," has a note stating "Not Applicable to Palisades." Also, in Table 3 of the PNP AMP, the component "Core Support Barrel Assembly – Lower Core Barrel Flange," has a note stating "Not Applicable to Palisades." However, MRP-227-A Table 4-2, "CE Plants Primary Components," lists the Core Support Barrel Assembly – Lower Flange Weld, as applicable to "All Plants," and MRP-227-A, Table 4-5, "CE Plants Expansion Components," lists the "Core Support Barrel Assembly – Lower Core Barrel Flange," as applicable to "All Plants."

Therefore, by email dated August 1, 2014 (Reference 19), the NRC staff requested the licensee to confirm that the weld joining the very bottom of the core support barrel to the plate the forms the bottom of the lower core support structure (Figure B), is considered a "Core Support Barrel – Lower Girth Weld," at PNP and is subject to EVT-1 visual examination as a "Primary" inspection category component. The staff also requested that the licensee confirm that it believes that an error exists in MRP-227-A, Tables 4-2 and 4-5, in that both the "Core Support Barrel – Lower Flange Weld," and "Core Support Barrel – Lower Core Barrel Flange," should have been identified as not applicable to PNP.

By letter dated August 14, 2014, the licensee provided a response to the request for information and stated the following:

The weld joining the bottom of the core support barrel to the plate that forms the bottom of the lower core support structure is considered a "Core Support Barrel Lower Girth Weld" at PNP, and is subject to EVT-1 visual examination as a "Primary" inspection category component.

As discussed in the RAI, Figure 14, "Palisades Reactor Vessel Internals Inspection Plan," of Reference 1 shows an elevation cross section of the PNP core support barrel and has four circled areas corresponding to portions to be inspected. The figure caption states the following:

Primary: EVT-1 examination of core of upper flange weld and lower cylinder girth welds no later than 2 RFOs [refueling outages] from the

beginning of the LR [license renewal] period (ten year intervals)

Coverage: 100% of accessible surfaces of upper flange weld and lower cylinder girth welds

The lower-most weld circled in the figure is the weld that joins the bottom of the core support barrel to the plate that forms the bottom of the lower core support structure. This weld is considered to be a "Core Support Barrel Lower Girth Weld" and is subject to EVT-1 visual examination as a "Primary" inspection category component.

Additionally, the licensee stated in its response to the RAI dated August 14, 2014, the following:

ENO understands that there is an error in MRP-227-A, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" (Reference 2), Tables 4-2, "CE plants Primary components," and 4-5, "CE plants Expansion components." In these two tables, the "Core Support Barrel Assembly, Lower flange weld" and the "Core Support Barrel Assembly, Lower core barrel flange" should have been identified as not applicable to PNP. This error has been documented in the PNP corrective action program.

The NRC staff finds the licensee's response to the RAI dated August 14, 2014, acceptable because the licensee has confirmed the following:

1. The PNP RVI do not have a core barrel lower flange weld,
2. Tables 4-2 and 4-5 of MRP-227-A contain an error where the "Core Support Barrel – Lower flange weld" and "Core Support Barrel – Lower flange" should have been identified as not applicable to PNP, and
3. The PNP RVI weld that performs the analogous function to the "Core Support Barrel – Lower flange weld" is considered a "Core Support Barrel – Lower cylinder girth weld," thus is subject to inspection as a "Primary" component consistent with the recommendations of MRP-227-A.

The NRC staff finds the licensee's specified examinations of the RVI acceptable because the PNP AMP will implement all the "Primary", "Expansion", and "Existing Programs" inspections recommended for CE-design RVI with a bolted core shroud design¹ in MRP-227-A, with the exception of those components erroneously listed in in MRP-227-A as applicable to PNP, and because the equipment, techniques, procedures, and personnel used for the RVI inspections will be consistent with the requirements of MRP-228, which is consistent with LR-ISG-2011-04.

3.5 Examination Acceptance and Expansion Criteria

3.5.1 Licensee Evaluation

Section 5.0 of the PNP AMP is nearly identical to Section 5 of MRP-227-A. This section

¹ The PNP RVI has a bolted core shroud, so inspections of certain components in Table 4-2 of MRP-227-A are not applicable to the PNP design. In addition, MRP-227-A Table 4-2 also specifies certain inspections for plants with core shrouds assembled with full-height shroud plates, which are also not applicable to PNP's design.

discusses the examination techniques used in the AMP: VT-3, VT-1, surface examination, volumetric examination, and physical measurements. Expansion criteria are noted to be listed in Table 5 of Attachment 1 of the application. Evaluation, repair, and replacement strategy is discussed and defined as being "in accordance with an NRC approved evaluation methodology [as appropriate]." Finally reporting requirements are described as "consistent with MRP-227-A and the [PNP] Corrective Action Program."

3.5.2 NRC Staff Evaluation

The NRC staff reviewed the licensee's examination acceptance and expansion criteria against the information in Table 5-2, "CE Plants Examination Acceptance and Expansion Criteria," of MRP-227-A. The NRC staff reviewed the information in Section 5.0 of the PNP AMP and found it acceptable because the information is consistent with the examination acceptance and expansion criteria of MRP-227-A, Table 5-2.

3.6 Applicant/Licensee Action Items

Section 4.2, "Plant-Specific Action Items" of Revision 1 of the NRC staff's final SE of MRP-227, Rev. 0 contained eight A/LAIs that must be addressed by an applicant or licensee referencing MRP-227-A as the basis of its plant-specific RVI AMP. The licensee addressed these eight action items in Section 7.0 of the AMP. A/LAIs 4 and 6 are applicable only to Babcock & Wilcox (B&W) designed RVI; therefore were not evaluated in the PNP AMP.

3.6.1 A/LAI 1 – Plant-specific Applicability Verification of MRP-227-A

Section 4.2.1 of Revision 1 of the NRC staff's final SE of MRP-227, Rev.0 states the following:

... each applicant/licensee is responsible for assessing its plant's design and operating history and demonstrating that the approved version of MRP-227 is applicable to the facility. Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE, or B&W) which support MRP-227 and describe the process used for determining plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection categories. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227.

3.6.1.1 Licensee Evaluation

In Section 7.1 of the PNP AMP, the licensee stated the following:

ENO has assessed its plant design and operating history and has determined that MRP-227-A is applicable to the facility. The assumptions regarding plant design and operating history made in MRP-191 are appropriate for PNP and there are no differences in component inspection at PNP. PNP operated the first 7 cycles of operation with a high leakage core loading Pattern. The FMECA [failure modes, effects, and criticality analyses] and functionality analyses were

based on the assumption of 30 years of operation with high leakage core loading patterns; therefore, PNP is bounded by the assumption in MRP-191.

The licensee also confirmed that PNP conform to the general assumptions in Section 2.4 of MRP-227-A, which are:

- 30 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 30 years of operation;
- base load operation, i.e., typically operates at fixed power levels and does not usually vary power on a calendar or load demand schedule; and
- no design changes beyond those identified in general industry guidance or recommended by the original vendors.

Additionally, Section 3.8 describes a 1.4% power uprate approved on July 14, 2004.

3.6.1.2 NRC Staff Evaluation

The information provided by the licensee confirmed that PNP switched to a low-leakage core loading pattern prior to 30 calendar years of operation, has always operated as a base-loaded unit, and no plant-unique modifications, consistent with the three assumptions of the FMECA and functionality analyses supporting MRP-227-A, listed in Section 2.4 of MRP-227-A.

To resolve the generic issue of the information needed from licensees to resolve A/LAI 1, a series of proprietary and public meetings were conducted (References 20, 21, 22, 23, and 24), at which the NRC, Westinghouse, the EPRI, and utility representatives discussed regulatory concerns and determined a path for a comprehensive and consistent utility response to demonstrate applicability of MRP-227-A, specifically for Westinghouse and CE-design PWR RVI. A summary of the proprietary meeting presentations and supporting proprietary generic design basis information is contained in Westinghouse proprietary report WCAP-17780-P (Reference 25). WCAP-17780-P provides background proprietary design information regarding variances in stress, fluence, and temperature in the plants designed by Westinghouse and CE to support NRC reviews of utility submittals to demonstrate plant-specific applicability of MRP-227-A.

As a result of the technical discussions with the NRC staff, the basis for a plant to respond to the NRC's RAI to demonstrate compliance with MRP-227-A for originally licensed and uprated conditions was determined to be satisfied with plant-specific responses to the following two questions (References 22 and 24):

1. Does the plant have non-weld or bolting austenitic stainless steel components with 20 percent cold work or greater, and if so, do the affected components have operating stresses greater than 30 kilopounds per square inch (ksi)? (If both conditions are true, additional components may need to be screened in for stress corrosion cracking).

2. (New Question 2):

Does the plant have atypical fuel design or fuel management that could render the assumptions of MRP-227-A, regarding core loading/core design, non-representative for that plant? [Reference 22 indicated this question covers power uprates as well as other core design and fuel management aspects.]

By MRP Letter No. 2013-025 dated October 14, 2013 (Reference 26), EPRI provided to licensees "MRP-227-A Applicability Guidelines for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs," (MRP-227-A Applicability Guidelines), a non-proprietary document containing guidance for responding to the two questions above. With respect to Question 1, MRP-227-A Applicability Guidelines provides guidance for licensees to assess whether RVI components at their plant, other than those identified in the generic evaluation, have the potential for cold work greater than 20%. With respect to Question 2, the MRP-227-A Applicability Guidelines provide quantitative criteria to allow a licensee to assess whether a particular plant has atypical fuel design or fuel management. For a CE design plant such as PNP, these criteria are:

- (1) The heat generation rate must be $\leq 68 \text{ Watts/cm}^3$.
- (2) The maximum average core power density must be less than 110 Watts/cm^3 .
- (3) The active fuel to fuel alignment plate (FAP) distance must be greater than 12.4 inches.

The NRC staff's review of the criteria in the MRP-227-A Applicability Guidelines, and the supporting technical information in WCAP-17780-P, are documented in Reference 27, in which the staff concluded that if an applicant or licensee demonstrates that its plant(s) comply with the guidance in MRP Letter 2013-025, there is reasonable assurance that the I&E guidance of MRP-227-A will be applicable to the specific plant(s). The MRP-227-A Applicability Guidelines provide an acceptable basis for licensees to prepare responses to the generic RAI questions (Question 1 and 2 above). Reference 27 further states that the basis for the NRC staff's conclusion is that the recommended criteria provide 1) a systematic process for an applicant/licensee to assess whether it's RVI contain cold-worked materials that may be susceptible to SCC, and 2) quantitative measures of whether a plant is operating with a low-leakage core design as assumed by MRP-227-A.

The NRC staff also concluded in Reference 27 that the criteria and guidance for verification of plant-specific applicability are acceptable because 1) the information provided on evaluation of cold work in WCAP-17780-P provides an adequate technical basis for the guidance in the MRP-227-A Applicability Guidelines for responding to Question 1; 2) the sensitivity studies of variations in neutron fluence, RVI geometry and temperature, and the information on power uprate effects on fluence and temperature, documented in WCAP-17780-P, provide an acceptable technical basis for the guidance in the MRP-227-A Applicability Guidelines for responding to Question 2.

By email dated December 12, 2013 (Reference 28), the NRC staff requested that ENO respond to the following questions, which are essentially identical to the questions as defined in the guidance document:

1. Do the Palisades RVI have non-weld or bolting austenitic stainless steel components with 20% cold work or greater, and if so do the affected components have operating stresses greater than 30 ksi?
2. Has Palisades ever utilized atypical design or fuel management that could make the assumptions of MRP-227-A regarding core loading/core design non-representative for that plant, including power changes/uprates?

By letter dated April 3, 2014, the licensee responded to the RAI dated December 12, 2013, and stated the following:

No, Palisades' reactor vessel internals (RVI) do not contain any non-weld or bolting austenitic stainless steel components with 20% cold work or greater.

To support this conclusion, Electric Power Research Institute (EPRI) letter 2013-025, "MRP-227-A Applicability Guidelines for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs," response guidance was used as guidance to fully demonstrate PNP specific applicability by performing a detailed review of all RVI stainless steel component drawings, procurement specifications, and listed material standards. The review included a categorization and reexamination of the screening process that had been used under MRP-191, "Materials Reliability Program: Screening Categorization and Ranking of Reactor Internals Components Westinghouse and Combustion Engineering PWR Design [Reference 29]," and MRP-227-A, "Materials Reliability Program: Pressurized Water Reactors Internals Inspection and Evaluation Guidelines," to determine those components that may be most susceptible due to fabrication or replacement, and whether any additional components or locations may fall outside the criteria that were chosen for generic screening purposes. The review determined that PNP does not have materials with greater than 20% cold working.

The NRC staff finds the licensee's response to Question 1 to be acceptable because the licensee, using the criteria of the MRP-227-A Applicability Guidelines, confirmed that PNP RVI components are consistent with the generic component design and material of MRP-191, and had no modifications that induced greater than 20% cold work.

By letter dated April 3, 2014, the licensee responded to the second question in the RAI dated December 12, 2013, and stated the following:

No, Palisades has not utilized atypical design or fuel management, including power changes/uprates, which are non-representative of the assumptions of MRP-227-A.

To support this conclusion, the assumptions of MRP-227-A, along with the additional guidance provided by the MRP 2013-025, were evaluated. The assumptions of MRP-227-A were evaluated against fuel design changes, core designs, and plant operation. Palisades Nuclear Plant (PNP) highest calculated reactor core power density remains below the MRP-191 assumption of 83.0

W/cm³. Additionally, PNP operated 14 years with a high leakage core, followed by 25 plus years of low (or ultra-low) leakage cores, resulting in much less fluence than assumed in MRP-227-A. The MRP 2013-025 screening criteria for Combustion Engineering plants was also met, except for fuel cycles prior to fuel cycle 14. For fuel cycles prior to fuel cycle 14 the distance between the active fuel and the fuel alignment plate was slightly below the assumed 12.4 inches (by a maximum of 0.57" for fuel cycle 1). Since fuel cycle 14, this screening criterion has been met. Even though this screening criteria was not met prior to fuel cycle 14, PNP is still bounded by the MRP 2013-025 analysis because the screening criteria used a fuel power density that was 27% higher than PNP's fuel density during fuel cycle 1 when the maximum deviation from the screening criteria occurred. This 27% margin more than offsets the increased flux due to the active fuel being closer to the alignment plate. Therefore, even with the active fuel slightly closer to the upper alignment plate than analyzed for MRP 2013-025 for the first 13 fuel cycles, PNP reactor vessel upper internals would have received less fluence than analyzed in MRP 2013-025.

The NRC staff finds that PNP meets the criteria of the MRP-227-A Applicability Guidelines with respect to core power density. The staff also finds that PNP has met the active fuel to FAP dimensional requirement since Cycle 14. Although the dimensional criterion was not met prior to Fuel Cycle 14, the staff finds the licensee's argument that the reduced distance would be offset by the core power limit, which is much less than the criterion in the MRP-227-A Applicability Guidelines acceptable. The licensee did not provide the specific value or range of the heat generation rate figure of merit ("F") which did not exceed 68 Watts/cm³. Therefore, by email dated June 9, 2014 (Reference 30), the NRC staff requested the licensee to provide the value of "F" applicable to the PNP RVI. By letter dated July 1, 2014, the licensee provided a diagram showing the calculations of "F" for the various locations around the core as defined in the MRP-227-A Applicability Guidelines, which shows the highest value of "F" for PNP is 48. For all three parameters, the licensee stated that these values were representative of anticipated future operation during the PEO.

Therefore, since the licensee's response indicates that PNP meets the numerical criteria of the MRP-227-A Applicability Guidelines, the NRC staff finds that PNP does not have atypical fuel design or fuel management that that could make the assumptions of MRP-227-A regarding core loading/core design non-representative for that plant. The staff's concerns documented in the RAIs dated December 12, 2013, and June 9, 2014, are thus resolved.

The licensee adequately addressed the two factors for which the NRC staff determined additional plant-specific information was necessary to verify applicability of MRP-227-A to PNP – (1) cold work induced stress; and, (2) fuel management, by confirming that PNP complies with the criteria defined in the MRP-227-A Applicability Guidelines. Furthermore, the licensee confirmed that PNP will continue to comply with these limits during the PEO. Therefore, the staff finds the licensee's response acceptable and Action Item 1 is resolved for PNP.

3.6.2 A/LAI 2 – RVI Components within the Scope of License Renewal

Section 4.2.2 of Revision 1 of the NRC staff's final SE of MRP-227, Rev. 0 states the following:

...consistent with the requirements addressed in 10 CFR 54.4, each applicant/licensee is responsible for identifying which RVI components are within the scope of LR for its facility. Applicants/licensees shall review the information in Tables 4-1 and 4-2 in MRP-189, Revision 1, and Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RVI components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. [Note: Table 4-5 of MRP-191 is the applicable table for CE-design RVI]. If the tables do not identify all the RVI components that are within the scope of LR for its facility, the applicant or licensee shall identify the missing component(s) and propose any necessary modifications to the program defined in MRP-227-A, as modified by this SE, when submitting its plant-specific AMP. The AMP shall provide assurance that the effects of aging on the missing component(s) will be managed for the period of extended operation.

3.6.2.1 Licensee Evaluation

The licensee stated that it reviewed the information in Table 4-5 of MRP-191 and determined that this table contains all of the RVI components that are within the scope of license renewal. This is shown in the Table presented in Attachment B of the PNP AMP. The licensee also discussed the process and results in Section 4.4 of the PNP AMP.

The licensee identified a number of components in Attachment B of the PNP AMP which were not included in Table 4-5 of MRP-191. The licensee stated that these components required screening and judgment to determine if any additional inspections were required.

The licensee stated the following in the PNP AMP regarding the PNP control rod element assembly (CEA) shrouds:

PNP has control rod shrouds in the upper guide structure where other C-E plants have CEA shrouds. MRP-191 only screens the CEA shroud configurations to determine if additional exams are required. The control rod shroud components were included in the license renewal application as being susceptible to stress corrosion cracking (in welds), void swelling (changes in dimensions), and irradiation-induced stress relaxation (loss of preload). MRP-191 classified all the components susceptible to stress corrosion cracking as Category A for the CEA shrouds. Since the control rod shroud assemblies are above the core they are not expected to see any effects due to void swelling or stress relaxation. Therefore, these components would fall into category A and not require any augmented examinations.

Another example of a PNP specific component is the incore instrumentation (ICI) guide tube assemblies which were not specifically listed in MRP-191 or MRP-227-A. The licensee stated the following in the PNP AMP:

These components were identified as being susceptible to loss of preload (irradiation-induced stress relaxation), changes in dimensions (void swelling), cracking (stress corrosion cracking), loss of material (wear), and reduction in fracture toughness (neutron irradiation embrittlement). The FMECA evaluation contained in MRP-191 found every item in the ICI assembly to be classified as Category A. Therefore, there would be no additional examinations required for these components.

The licensee also identified tie rods in the core support beams which are not explicitly listed in MRP- 191. The licensee indicated that the tie rods are subject to changes in dimensions, cracking, and reduction in fracture toughness, but that this location is in a low fluence area and hence is rated as Category A. Thus, the licensee concluded that no augmented examination would be required.

Lastly, the licensee identified a spacer shim in the upper internals guide structure (UGS) as a PNP specific component, which was installed as a result of operating experience with excessive wear noticed in the mid-1990's in the upper flange area of the core barrel. The licensee stated that due to the prior operating experience with wear and structure functional support (function of the shim), future component examination would ensure no additional wear is present. Attachment B identifies augmented inspection requirements to examine the shim by VT-3 to determine if any additional wear has occurred, and that this should be done as part of the ASME Code, Section XI examination of the UGS. The spacer shim is identified in the table in Attachment C, "Planned Scope Of Existing And Augmented Reactor Vessel Internals Examinations of the AMP," as subject to VT-3 examination for loss of material due to wear with subsequent examinations on a ten-year interval under a plant-specific program.

3.6.2.2 NRC Staff Evaluation

During its review of the PNP RVI components subject to aging management review during license renewal, the licensee identified a number of components unique to PNP. The licensee performed a screening process for applicable aging effects and evaluated these components by comparison of the general location and function in the RVI to the equivalent components in MRP-191. The licensee then assigned them to Category A, B, or C similar to the initial categorization system in MRP-191 and determined if augmented inspections are required. The licensee determined all the plant-unique components required no augmented inspections except the UGS spacer shim. The NRC staff finds inclusion of the spacer shim as a plant-specific component to be acceptable since it ensures that an inspection will be conducted periodically to look for the specific aging effect identified by the OE.

Although the PNP AMP stated that there are no Cast Austenitic Stainless Steel (CASS), martensitic or precipitation-hardenable components in the PNP RVI, by RAI email dated March 4, 2013, the NRC staff requested confirmation that the PNP RVI did not contain some specific grades of nickel-based alloys and precipitation-hardenable alloys for which PWR operating experience has shown a history of stress corrosion cracking (RAI-5). By letter dated April 4, 2013, the licensee confirmed there were no RVI components made from nickel based alloys. However, the licensee indicated that it found through its review of RVI materials that Alloy A-286 (a precipitation-hardenable iron-nickel-chromium alloy) is used in two components related to the hold-down spring assembly, the plunger assemblies and jackscrews. The

licensee indicated these components are not subject to tensile stresses in service, therefore are not susceptible to stress corrosion cracking. The licensee also indicated that 17-4 PH stainless steel is used in the grid ring bushings in the upper guide structure. The licensee stated that thermal aging would not cause a loss of function of the bushings because strength was the primary parameter of importance, which is increased by thermal aging, and also because the bushings are inspected for thread damage under a PNP refueling procedure. Since these components do not appear to be listed in Attachment B to the PNP AMP, unless under a different name, by RAI email dated June 9, 2014, the NRC staff requested the licensee clarify whether the components identified in the RAI response dated April 4, 2013, are included in Attachment B under different names, or revise Attachment B to include these components (RAI-3.2). By letter dated July 1, 2014, the licensee stated that the components are listed in Attachment B and stated the following clarifying the location in Attachment B:

The plunger assemblies and jack screws are listed under "Component/Commodity" "Upper Internal Assembly Hold-down Ring Plunger." The column labeled "Additional Description" states what is included in each component and for this component, it lists the "Hold-Down Ring Plungers including washers and jack screws."

The 17-4 PH stainless steel grid ring bushings are included in Attachment B under the "Component/Commodity" "Upper Internal Assembly Brace Grid Beam" grouping, as they are part of this assembly.

The licensee clarified under which line item in Attachment B the A-286 and 17-4 PH components are included; therefore, the NRC staff finds the response to RAI-3.2 acceptable. As supplemented by the response to RAI-3.2, the staff found the licensee's response to RAI-5 acceptable because the licensee evaluated the components and determined no additional aging management activities were necessary. The staff's concerns in RAI-5 and RAI-3.2 are therefore resolved.

The process used by the licensee to evaluate the PNP unique components is similar to that used in MRP-191. Also, many of the plant-unique components have the same location and function as generic CE components in MRP-191, and similar environmental conditions (such as fluence) to the generic components against which they were compared. Therefore, the staff finds the process used by the licensee to evaluate the PNP unique components to be acceptable. The staff finds the recommended augmented inspections for the PNP unique components to be acceptable. Therefore, the NRC staff finds the licensee has adequately addressed A/LAI 2.

3.6.3 A/LAI 3 – Evaluation of the Adequacy of Plant-Specific Existing Programs

Section 4.2.3 of Revision 1 of the NRC staff's final SE of MRP-227, Rev. 0 states the following:

... applicants/licensees of CE and Westinghouse are required to perform plant-specific analysis either to justify the acceptability of an applicant's/licensee's existing programs, or to identify changes to the programs that should be implemented to manage the aging of these components for the period of extended operation. The results of this plant-specific analyses and a description

of the plant-specific programs being relied on to manage aging of these components shall be submitted as part of the applicant's/licensee's AMP application. The CE and Westinghouse components identified for this type of plant-specific evaluation include: CE thermal shield positioning pins and CE in-core instrumentation thimble tubes (Section 4.3.2 in MRP-227), and Westinghouse guide tube support pins (split pins) (Section 4.3.3 in MRP-227).

3.6.3.1 Licensee Evaluation

The licensee stated that A/LAI 3 is not applicable to PNP because it does not have thermal shield positioning pins or ICI thimble tubes.

3.6.3.2 NRC Staff Evaluation

Plant-specific programs for CE-design RVI for ICI thimble tubes are related to irradiation-induced growth of zirconium-alloy ICI thimble tubes. The NRC staff reviewed the information in the PNP LRA and the AMP, and confirmed PNP does not have ICI thimble tubes of the type addressed by A/LAI 3. Therefore, the staff finds that A/LAI 3 does not apply to PNP.

3.6.4 A/LAI 4 – B&W Core Support Structure Upper Flange Stress Relief

A/LAI 4 is not applicable to CE-design RVI such as PNP.

3.6.5 A/LAI 5 – Application of Physical Measurements as part of I&E Guidelines for B&W, CE, and Westinghouse RVI Components

Section 4.2.5 of Revision 1 of the NRC staff's final SE of MRP-227, Rev. 0 states the following:

... applicants/licensees to identify plant-specific acceptance criteria to be applied when performing the physical measurements required by the NRC-approved version of MRP-227 for loss of compressibility for Westinghouse hold down springs, and for distortion in the gap between the top and bottom core shroud segments in CE units with core barrel shrouds assembled in two vertical sections. The applicant/licensee shall include its proposed acceptance criteria and an explanation of how the proposed acceptance criteria are consistent with the plants' licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation during the period of extended operation as part of their submittal to apply the approved version of MRP-227.

3.6.5.1 Licensee Evaluation

The licensee stated that this requirement is not applicable as the PNP core shroud is bolted and is not assembled in two vertical sections.

3.6.5.2 NRC Staff Evaluation

The NRC staff reviewed the design of the PNP RVI with respect to applicability of A/LAI 5, and finds that A/LAI 5 does not apply to PNP as the PNP core shroud is bolted and not assembled in two vertical sections.

3.6.6 A/LAI 6 – Evaluation of Inaccessible B&W Components

This action item does not apply to C-E designed units such as PNP.

3.6.7 A/LAI 7 Plant-Specific Evaluation of CASS Materials

Section 4.2.7 of Revision 1 of the NRC staff's final SE of MRP-227, Rev. 0 states the following:

... applicants/licensees of B&W, CE, and Westinghouse reactors to develop plant-specific analyses to be applied for their facilities to demonstrate that B&W IMI guide tube assembly spiders and CRGT assembly spacer castings, CE lower support columns, and Westinghouse lower support column bodies will maintain their functionality during the period of extended operation or for additional RVI components that may be fabricated from CASS, martensitic or precipitation hardened PH stainless steel materials. These analyses should also consider the possible loss of fracture toughness in these components due to thermal embrittlement and irradiation embrittlement....

The plant-specific analysis shall be consistent with the plant's licensing basis and the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation. The applicants/licensees shall include the plant-specific analysis as part of their submittal to apply the approved version of MRP-227.

3.6.7.1 Licensee Evaluation

The licensee stated that this requirement is not applicable to PNP as CASS, martensitic stainless steel or precipitation hardenable stainless steel materials are not present in the reactor vessel internal lower support structures.

3.6.7.2 NRC Staff Evaluation

Since the licensee's evaluation of A/LAI 7 only mentioned the lower support structures, by RAI email dated March 4, 2013, the NRC staff requested the licensee to confirm that no CASS is present in any of the locations covered by the PNP AMP, or provide a discussion of how the PNP AMP adequately addresses the requirements specified in GALL AMP, XI.M12, "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)," and GALL AMP XI.M13, "Thermal Aging and Neutron Embrittlement of Cast Austenitic Stainless Steel (CASS)," for CASS materials used in PWR RVI components (RAI-6). In its April 4, 2013 response, the licensee stated that it performed a confirmatory review of PNP drawings and the aging management evaluation of the reactor vessel internals developed for LR, and that the review confirmed CASS is not present in any of the reactor internal locations covered by the AMP.

Therefore, the staff finds that A/LAI 7 does not apply to PNP.

3.6.8 A/LAI 8 – Submittal of Information for Staff Review and Approval

Section 4.2.8 of Revision 1 of the NRC staff's final SE of MRP-227, Rev. 0 states the following:

... applicants/licensees shall make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SE, as an AMP for the RVI components at their facility. This submittal shall include the information identified in Section 3.5.1 of this SE.

Section 3.5.1 of Revision 1 of the NRC staff's final SE of MRP-227, Rev. 0 states, in part, the following:

In addition to the implementation of MRP-227 in accordance with NEI 03-08, applicants/licensees whose licensing basis contains a commitment to submit a PWR RVI AMP and/or inspection program shall also make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SE. An applicant/licensee's application to implement MRP-227, as amended by this SE shall include the following items (1) and (2).

1. An AMP for the facility that addresses the 10 program elements as defined in NUREG-1801, Revision 2, AMP XI.M16A.
2. To ensure the MRP-227 program and the 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.

Applicants that submit applications for LR after the issuance of the MRP-227, Rev. 0 final SE, Revision 1 are required to submit additional information items. The NRC staff notes that since the PNP LRA was submitted prior to the issuance of Revision 1 of the NRC staff's final SE related to MRP-227, the licensee is only required to submit the above two information items.

3.6.8.1 Licensee Evaluation

This A/LAI requires that the applicant establish that the AMP fulfills the ten attributes in the GALL report. The licensee reviewed these ten attributes in Sections 2.3.1- 2.3.10 of the PNP AMP and further states that it reviewed the design bases for the plant's core support structure for license renewal and identified no TLAA for fatigue of internals. Further review per MRP-227-A identified no further TLAA's for fatigue. The licensee stated the following in regards to fatigue of internals:

However, fatigue damage for primary and (potentially) expansion internals components such as the core support plate will be managed by performing visual examinations, including any required periodic enhanced visual (EVT-1) examinations. No reduction in examination coverage by plant-specific analysis will be requested. Therefore, this approach is essentially equivalent to managing the effects of fatigue on reactor internals components with fatigue analyses during the period of extended operation through 10 CFR 24.21 (c)(1)(iii). As an example, for the 2013 outage, the core support plate has been identified as a Primary component requiring enhanced visual inspection (EVT-1) to manage the effects of fatigue, as shown in Table 2. PNP is not requesting any deviations from the guidance provided in MRP-227-A, as approved by the NRC.

3.6.8.2 NRC Staff Evaluation

The licensee provided the information for Item 1 of A/LAI 8 via Sections 2.3.1-2.3.10 of the PNP AMP, as discussed in Section 3.2 of this SA. The licensee provided the information required by Item 2 of A/LAI 8 via the PNP AMP, including evaluation of the A/LAIs, as supplemented by its April 14, 2013, April 13, 2014, July 1, 2014, and August 14, 2014, RAI responses. The NRC staff also reviewed the PNP LRA and LR SE and did not note any TLAA's related to the RVI. Therefore, the staff finds that the licensee has adequately addressed A/LAI 8.

4.0 CONCLUSION

The NRC staff has reviewed the AMP for the PNP RVI components and concludes that the PNP AMP is acceptable because it is consistent with the I&E guidelines of MRP-227-A, and, the licensee has appropriately addressed A/LAIs specified in MRP-227-A applicable to PNP.

Consequently, Regulatory Commitment #33, as documented in Appendix A of NUREG-1871, and the revised commitment submitted to the NRC by letter dated June 20, 2012, is considered fulfilled. The NRC staff's approval of the PNP RVI AMP does not reduce, alter, or otherwise affect current ASME Code, Section XI ISI requirements, or any PNP specific licensing requirements related to ISI. The staff notes that Section 7.0, "Implementation Requirements" of MRP-227-A requires that the NRC be notified of any deviations from the "Needed" requirements.

5.0 REFERENCES

1. Palisades Nuclear Plant, Revised Program Plan for Aging Management of Reactor Vessel Internals, dated September 13, 2012 (ADAMS Accession No. ML12257A352)
2. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A) 1022863 Final Report, December 2011 (ADAMS Accession No. ML120170453) – Transmitted to NRC by MRP letter MRP-2011-036 dated January 9, 2012
3. NUREG-1871, "Safety Evaluation Report Related to the License Renewal of PNP Nuclear Plant" (ADAMS Accession No. ML062710074)
4. Palisades, Reactor Vessel Internals Program Submittal Commitment Date Change, March 23, 2009 (ADAMS Accession No. ML090830419)
5. Entergy Nuclear Operations, Inc. letter dated March 10, 2010, Program Plan for Aging Management of Reactor Vessel Internals (ADAMS Accession No. ML100710083)
6. Entergy Nuclear Operations, Inc. letter dated March 31, 2011, Withdrawal and New Commitment for Program Plan for Aging Management of Reactor Vessel Internals (ADAMS Accession No. ML110910461)
7. NRC Regulatory Issue Summary 2011-07, License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management, dated July 21, 2011 (ADAMS Accession No. ML111990086)
8. Entergy Nuclear Operations, Inc. letter dated June 20, 2012, Revised Commitment Date for Program Plan for Aging Management of Reactor Vessel Internals Submittal (ADAMS Accession No. ML12173A219)
9. Palisades, Response to Request for Additional Information - Revised Program Plan for Aging Management of Reactor Vessel Internals, dated April 4, 2013 (ADAMS Accession No. ML13094A414)
10. Palisades, Response to Second Request for Additional Information Reactor Vessel Internals - ME9569, dated April 3, 2014 (ADAMS Accession No. ML14097A376)
11. Palisades, Response to Third Request for Additional Information - Reactor Vessel Internals (ME9569), July 1, 2014 (ADAMS Accession No. ML14183A014)
12. Palisades, Response to Fourth Request for Additional Information - Reactor Vessel Internals (ME9569), August 14, 2014 (ADAMS Accession No. ML14226A807)
13. "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, (MRP-227-Rev. 0)," 1016596 Final Report, December, 2008,

(ADAMS Accession No. ML090160212) - Transmitted to NRC by MRP letter number MRP 2009-04 dated January 12, 2009

14. Letter from Robert Nelson, NRC, to Neil Wilmshurst, EPRI dated December 16, 2011; Subject: Revision 1 of the Final Safety Evaluation of EPRI Report, Materials Reliability Program Report 1016596 (MRP-227), Revision 0, "PWR (PWR) Internals Inspection and Evaluation Guidelines" (TAC NO. ME0680) (ADAMS Accession No. ML11308A770)
15. NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," Final Report, December 31, 2010 (ADAMS Accession No. ML103490041)
16. LR Interim Staff Guidance LR-ISG-2011-04: Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors. May 28, 2013 (ADAMS Accession No. ML12270A251)
17. Nuclear Energy Institute 0308, Revision 2, "Guideline for the Management of Material Issues," January 2010 (ADAMS Accession No. ML101050337)
18. Email from NRC, to Entergy Nuclear Operations, Inc., "Palisades – Request for Additional Information – ME9569 – Revised Program Plan for Aging Management of Reactor Vessel internals," March 4, 2013 (ADAMS Accession No. ML13063A318)
19. Email from NRC, to Entergy Nuclear Operations, Inc., "Request for Additional Information: Palisades Aging Management Program for Reactor Vessel Internals (ME9569)," August 1, 2014 (ADAMS Accession No. ML14217A011)
20. U. S. Nuclear Regulatory Commission Letter, "Summary of November 28, 2012, Category II Public Meeting with the Electric Power Research Institute and Industry Representatives," January 29, 2013. (ADAMS: ML13009A066)
21. U. S. Nuclear Regulatory Commission Letter, "Summary of January 22-23, 2013, Closed Meeting with the Electric Power Research Institute and Westinghouse," February 21, 2013. (ADAMS Accession Nos. ML13042A048 and ML13043A062, respectively)
22. U. S. Nuclear Regulatory Commission Letter, "Summary of February 25, 2013, Telecom with the Electric Power Research Institute and Westinghouse Electric Company," March 15, 2013. (ADAMS Accession No. ML13067A262)
23. U. S. Nuclear Regulatory Commission Letter, "Summary of May 21, 2013, Public Meeting Regarding Pressurized Water Reactor (PWR) Vessel Internals Inspections," June 24, 2013. (ADAMS Accession No. ML13164A126)
24. U. S. Nuclear Regulatory Commission Presentation: "Status of MRP-227-A Action Items 1 and 7," June 5, 2013 (ADAMS Accession No. ML13154A152).

25. Westinghouse Report, WCAP-17780-P, Rev. 0, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs," June 2013 (ADAMS Accession No. ML13183A372)
26. MRP-227-A Applicability Guidelines for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs, Enclosure to MRP Letter 2013-025, October 14, 2013, transmitted via email from K. Amberge to J. Holonich, November 15, 2013 (ADAMS Accession No. ML13322A454)
27. "Office of Nuclear Reactor Regulation Evaluation of WCAP-17780-P, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering [(CE)] and Westinghouse [Electric Company (Westinghouse)] Pressurized Water Reactor Designs," and MRP-227-A, Applicability Guidelines for CE and Westinghouse Pressurized Water Reactor Designs," November 7, 2014 (ADAMS Accession No. ML14309A484)
28. Email from NRC, to Entergy Nuclear Operations, Inc., "Request for Additional Information – Reactor Vessel Internals – ME9569 – Email Resend," December 12, 2013 (ADAMS Accession No. ML13346A644)
29. 1013234, "Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191)," dated November 30, 2006 (ADAMS Accession No. ML091910130)
30. Email from NRC, to Entergy Nuclear Operations, Inc., "Request for Additional Information – Palisades – Aging Management Program for Reactor Vessel Internals – ME9569," June 9, 2014 (ADAMS Accession No. ML14160B264)

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If you have any questions concerning this matter, please contact the Project Manager, Jennivine Rankin at (301) 415-1530 or via e-mail at Jennivine.Rankin@nrc.gov.

Sincerely,

/RA/

David L. Pelton, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-255

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