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

September 8, 2016

Mr. Joel P. Gebbie
Senior Vice President and
Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 – STAFF ASSESSMENT REGARDING PROGRAM PLAN FOR AGING MANAGEMENT FOR REACTOR VESSEL INTERNALS (CAC NOS. MF0050 AND MF0051)

Dear Mr. Gebbie:

By letter dated October 1, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12284A320), Indiana Michigan Power Company (I&M, the licensee) submitted an aging management program (AMP) for the Donald C. Cook Nuclear Plant (CNP), Units 1 and 2, reactor vessel internals (RVI). The submittal was supplemented by letters dated July 30, 2014, September 4, 2014, October 22, 2014, August 6, 2015, and October 30, 2015 (ADAMS Accession Nos. ML14216A497, ML14253A316, ML14316A449, ML15223A495, and ML15308A091, respectively). The CNP RVI AMP was developed based on the U.S. Nuclear Regulatory Commission (NRC)-approved topical report MRP-227-A, "Material Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." The RVI AMP was submitted to fulfill a regulatory commitment that originated from license renewal activities, as documented in NUREG-1831, "Safety Evaluation Report Related to License Renewal of the Donald C. Cook Nuclear Plant, Units 1 and 2" (ADAMS Accession No. ML052230442). License Renewal Commitment No. 36 stated that I&M would submit the RVI AMP for NRC staff review and approval. License Renewal Commitment No. 36 for CNP was fulfilled upon submittal of the RVI AMP on October 1, 2012.

The NRC staff has completed its review of the CNP RVI AMP, and concludes that it is acceptable because it is consistent with the inspection and evaluation guidelines of MRP-227-A. The licensee has adequately addressed all eight action items specified in MRP-227-A.

The NRC staff's approval of the CNP RVI AMP does not reduce, alter, or otherwise affect the current ASME Code, Section XI inservice inspection requirements, or any CNP licensing basis requirements related to inservice inspection of structures, systems, and components. The staff notes that Section 7, "Implementation Requirements," of MRP-227-A, requires that the NRC be notified of any deviations from the "needed" requirements.

The NRC staff's assessment of the CNP RVI AMP is enclosed. If you have any questions concerning this matter, please contact the Project Manager, Allison Dietrich, at (301) 415-2846, or via e-mail at Allison.Dietrich@nrc.gov.

Sincerely,

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

Docket Nos. 50-315 and 50-316

Enclosure:
Staff Assessment

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UNITED STATES
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STAFF ASSESSMENT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE AGING MANAGEMENT PROGRAM

FOR REACTOR VESSEL INTERNALS

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By letter dated October 1, 2012 (Ref. 1), Indiana Michigan Power Company (I&M, the licensee) submitted an aging management program (AMP) for the Donald C. Cook Nuclear Plant (CNP), Units 1 and 2, reactor vessel internals (RVIs). The submittal was supplemented by letters dated July 30, 2014, September 4, 2014, October 22, 2014, August 6, 2015, and October 30, 2015 (Refs. 2-6, respectively). The CNP RVI AMP was developed based on the U.S. Nuclear Regulatory Commission (NRC)-approved Electric Power Research Institute (EPRI) topical report MRP-227-A, "Material Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines" (Ref. 7). The RVI AMP was submitted to fulfill a regulatory commitment that originated from license renewal activities as documented in Appendix A of NUREG-1831, "Safety Evaluation Report Related to License Renewal of the Donald C. Cook Nuclear Plant, Units 1 and 2" (Ref. 8). License Renewal Commitment No. 36 stated that I&M would submit the RVI AMP for NRC staff review and approval. License Renewal Commitment No. 36 for CNP was fulfilled upon submittal of the AMP on October 1, 2012.

By letter dated June 22, 2011 (Ref. 9), the NRC issued the first version of its safety evaluation (SE) for Rev. 0 of the MRP-227 report (Ref. 10). On July 21, 2011, the NRC issued Regulatory Issue Summary (RIS) 2011-07, "License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management" (Ref. 11) to provide guidance to pressurized water reactor (PWR) license renewal applicants and renewed license holders for the submittal of plant-specific AMPs for the RVI components. By letter dated September 1, 2011 (Ref. 12), the licensee submitted a revision to License Renewal Commitment No. 36 to allow for submittal of the CNP RVI AMP plan to the NRC by October 1, 2012, consistent with the guidance contained in RIS 2011-07. On December 16, 2011, the NRC issued Rev. 1 of its SE for the MRP-227 report, which is included in MRP-227-A (Ref. 7).

Enclosure

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 54 addresses the requirements for plant license renewal. Section 54.21 of 10 CFR requires that each application for license renewal contain an integrated plant assessment (IPA) and an evaluation of time-limited aging analyses. 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 will be adequately managed so that their intended functions will be maintained consistent with the current licensing basis for the period of extended operation (PEO), as required by 10 CFR 54.29(a). In addition, 10 CFR 54.22 requires that a license renewal application include any technical specification changes or additions necessary to manage the effects of aging during the PEO.

Structures and components subject to an AMR 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.

The NRC staff's final SE for MRP-227 specifies seven generic conditions for the topical report and eight action items that must be addressed on a plant-specific basis by those utilizing the topical report as the basis for an RVI AMP submittal to the NRC. On January 9, 2012, EPRI issued the NRC-approved version of the topical report, MRP-227-A (Ref. 7), which includes Rev. 1 of the final SE. MRP-227-A addresses the seven generic conditions established in the SE and provides the technical basis for the development of plant-specific AMPs for managing the effects of aging on RVI components. MRP-227-A also provides specific inspection and evaluation guidelines for PWR license renewal applicants and renewed license holders to use in their plant-specific AMPs. The aging management activities described in MRP-227-A are intended for use by licensees in meeting the conditions of the license renewal commitments related to aging management of the RVI components.

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). The scope also includes RVI components that serve an intended safety function consistent with the criteria in 10 CFR 54.4(a)(1), and any non-safety related RVI components whose failure could impact the intended functions of a safety related component that was included under 10 CFR 54.4(a)(1) and serve an intended function as defined in 10 CFR 54.4(b). 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).

In December 2010, the NRC published NUREG-1801, Rev. 2, "Generic Aging Lessons Learned (GALL) Report – Final Report" (Ref. 13), providing new generic AMR line items and generic AMP criteria in Chapter XI.M16A, "PWR Vessel Internals" (GALL Report AMP XI.M16A). GALL Report AMP XI.M16A was based on 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 SE for MRP-227-A, the NRC published license renewal Interim Staff Guidance (ISG) in LR-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized

Water Reactors" (Ref. 14), which modified the criteria of GALL Report AMP XI.M16A to be consistent with MRP-227-A.

3.0 TECHNICAL EVALUATION

The licensee's October 1, 2012, RVI AMP submittal for CNP provided the following information for NRC staff review and approval:

- A description of the RVI AMP attributes, based on the ten AMP elements described in the GALL Report AMP XI.M16A. This is addressed in Section 3.1 of this assessment.
- A description of the RVI AMP inspection plan and evaluation criteria, based on MRP-227-A. This is addressed in Section 3.2 of this assessment.
- A discussion of the plant-specific operating experience (OE) for the CNP RVI components. This is addressed in Section 3.3 of this assessment.
- A discussion of the plant-specific applicability of MRP-227-A to CNP based on the licensee's responses to the eight action items established in Section 4.2 of the MRP-227-A SE. This is addressed in Section 3.4 of this assessment.
- A discussion of other existing programs and activities that are related to aging management of the RVI components at CNP. This is addressed in Section 3.4 of this assessment, as it relates to the plant-specific evaluation required for Action Item 3.

As part of its review, the NRC staff issued a request for additional information (RAI), dated June 6, 2014 (Ref. 15), and a follow-up RAI, dated May 5, 2015 (Ref. 16), to address specific technical issues.

3.1 Reactor Vessel Internals Aging Management Program Attributes

3.1.1 Licensee Evaluation of Program Attributes

In Section 5 of the CNP RVI AMP submittal, the licensee evaluated each of the 10 AMP Program Elements against the corresponding elements in GALL Report AMP XI.M16A. The licensee determined that its RVI AMP for CNP is consistent with GALL Report AMP XI.M16A.

3.1.2 NRC Staff Evaluation of Program Attributes

The NRC staff reviewed the licensee's AMP against the 10 elements of the revised version of GALL Report AMP XI.M16A, as provided in LR-ISG-2011-04. The staff determined that the 10 elements of the CNP RVI AMP are consistent with the 10 elements described in LR-ISG-2011-04. Therefore, the staff finds the licensee's implementation of the 10 AMP elements acceptable for CNP.

3.2 Reactor Vessel Internals Inspection Plan and Evaluation Criteria

3.2.1 Licensee Evaluation of Inspection Plan and Evaluation Criteria

The licensee's inspection plan for managing the effects of aging on the CNP RVI components is described in Section 4 of its RVI AMP submittal and is based on the implementation of MRP-227-A. The licensee stated that its RVI component inspection criteria do not reduce, alter, or otherwise affect the current ASME Code, Section XI or plant-specific licensing basis inservice inspection (ISI) requirements. The licensee stated that the MRP developed the companion document MRP-228, "Materials Reliability Program: Inspection Standard for PWR Internals," which contains requirements specific to the inspection methodologies involved as well as requirements for qualification of the non-destructive examination (NDE) systems used to perform these inspections.

The licensee stated that all RVI components requiring aging management are categorized as Primary components, Expansion components, Existing Program components, or No Additional Measures components, based on the guidance of MRP-227-A. The licensee provided definitions of each of these RVI component inspection categories that are consistent with those provided in Section 3.3.1 of MRP-227-A, and referenced the applicable Westinghouse plant inspection tables contained in MRP-227-A for each inspection category. These tables are included in the licensee's RVI AMP submittal as Appendix A, Appendix B, and Appendix C, for Primary, Expansion, and Existing Program components respectively.

The applicable RVI examination acceptance and expansion criteria for Westinghouse plants are contained in Table 5-3 of MRP-227-A. This table is included as Appendix D of the RVI AMP submittal. The licensee stated that RVI examination results from the MRP-227-A inspections that do not meet the applicable acceptance criteria will be dispositioned in accordance with an NRC-approved methodology, or the methodology will be submitted for NRC review and approval prior to implementation.

3.2.2 NRC Staff Evaluation of Inspection Plan and Evaluation Criteria

The MRP-227-A inspection guidelines consider the effects of eight age-related degradation mechanisms on the integrity of the RVI components. The eight age-related degradation mechanisms are stress corrosion cracking (SCC), irradiation-assisted stress corrosion cracking (IASCC), fatigue, irradiation embrittlement (IE), thermal embrittlement (TE), wear, void swelling, and irradiation-assisted stress relaxation (ISR). The MRP-227-A inspection guidelines prescribe RVI component examinations to detect the various aging effects that are associated with the eight degradation mechanisms. The aging effects which are addressed by these examinations include cracking (due to SCC, IASCC, and/or fatigue), loss of material (due to wear), component distortion (due to void swelling), loss of fracture toughness (due to TE or IE), and loss of bolt preload (due to ISR).

MRP-227-A prescribes inspections of PWR RVI components based on a categorization of the components using the results of the MRP's Failure Modes, Effects, and Criticality Analyses (FMECA) for PWR internals. The FMECA for Westinghouse plants are documented in EPRI MRP Topical Report MRP-191, "Material Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Designs," dated November 2006 (Ref. 17). Topical Report MRP-191 screened the components

for susceptibility to the eight degradation mechanisms and evaluated the degree of tolerance of the components' safety-related function to the associated aging effects. The FMECA results were applied to the PWR RVI components for each of the three PWR designs in the U.S., which are Babcock & Wilcox, Combustion Engineering, and Westinghouse. Based on this categorization for each of the three PWR designs, MRP-227-A places the RVI components into one of four functional groups: Primary components, Expansion components, Existing Programs components, and No Additional Measures components. Each of these functional groups is defined in Section 3.3.1 of MRP-227-A, and the specific inspection requirements are tabulated for the Primary, Expansion, and Existing Programs component groups in Chapter 4 of MRP-227-A. Components in the No Additional Measures group require no augmented inspection activity because they were either generically determined to be not susceptible to any of the eight age-related degradation mechanisms, the effect of an aging mechanism was determined to have negligible impact on safety function, or it was determined that there was no significance of aging based on a 60-year assessment, per the FMECA.

For Westinghouse plants, the inspection requirements for Primary, Expansion, and Existing Programs components are provided in MRP-227-A Tables 4-3, 4-6, and 4-9, respectively. The NRC staff reviewed the licensee's RVI component inspection criteria, provided in Appendices A, B, and C of the CNP RVI AMP submittal, and determined that they are the same as those specified in MRP-227-A. These appendices also state that the CNP RVI AMP implements the inspection criteria provided in these tables. Therefore, the NRC staff finds the licensee's RVI component inspection criteria acceptable because they are consistent with MRP-227-A.

Table 4-3 of MRP-227-A, which is directly incorporated into Appendix A of the CNP RVI AMP, specifies generic guidelines for the initial inspection schedule for the Primary components. Specifically, for the baffle-former assembly components, the initial inspections must occur within the specified ranges of effective full power years (EFPY) of facility operation. For all other Primary components, the initial inspection schedule guidelines are specified in terms of the number of refueling cycles from the beginning of the PEO. The NRC staff requested in Follow-Up RAI-5, issued on May 5, 2015 (Ref. 16), that the licensee provide the plant-specific schedule in terms of calendar year and refueling outage for these initial Primary component inspections at CNP.

In its response to Follow-Up RAI-5 (Ref. 5), dated August 6, 2015, the licensee provided a table listing the initial Primary component inspection schedule for CNP in terms of refueling outage, calendar year, and EFPY. The first Primary component inspections are scheduled to occur during the fall 2017 refueling outage for CNP Unit 1 and the fall 2016 refueling outage for CNP Unit 2. The NRC staff reviewed this table and determined that the licensee's schedule for performing these initial Primary component inspections is consistent with the inspection schedule guidelines in Table 4-3 of MRP-227-A.

The NRC staff also reviewed the licensee's RVI examination acceptance and expansion criteria provided in Appendix D of the CNP RVI AMP submittal and determined that they are the same as those specified in MRP-227-A, Table 5-3. Appendix D also states that the CNP RVI AMP implements the examination acceptance and expansion criteria provided in this table. Therefore, the staff determined that the licensee's RVI examination acceptance and expansion criteria are acceptable because they are consistent with MRP-227-A.

3.2.3 NRC Staff Conclusion for Inspection Plan and Evaluation Criteria

Based on its review of the CNP RVI AMP inspection plan and examination acceptance and expansion criteria, as documented above, the staff finds the licensee's plan to implement the MRP-227-A inspection and evaluation guidelines acceptable. The CNP RVI AMP will implement inspections for all Primary, Expansion, and Existing Programs components in accordance with MRP-227-A, Tables 4-3, 4-6, and 4-9, respectively, and it will implement the examination acceptance and expansion criteria in accordance with MRP-227-A, Table 5-3.

3.3 Plant-Specific Reactor Vessel Internals Operating Experience

3.3.1 Licensee Evaluation of Operating Experience

In Section 2 of the CNP RVI AMP submittal, the licensee provided a discussion of the plant-specific RVI OE for each unit. The licensee stated that CNP Units 1 and 2 are both Westinghouse 4-loop reactors in the downflow configuration, and indicated that the RVI component configurations are the same as those described for Westinghouse reactors in MRP-227-A. The licensee stated that a review of plant records was performed to locate unit-specific RVI OE and design changes. The licensee provided the following results of the RVI OE.

3.3.1.1 CNP Unit 1 Operating Experience

Control Rod Guide Tube (CRGT) and Support Pin Replacement
The CRGT support pins were replaced in 1985 with an improved stress design fabricated from Alloy X-750. The licensee stated that installation was expedited by replacing CRGT assemblies with spares available from a CNP Unit 2 modification, which is discussed in Section 3.3.1.2 of this assessment.

Barrel-Former Bolt Partial Replacement
A barrel-former bolt was found on the lower core plate in 1994. All barrel-former bolts were visually inspected in 1995, and a sample of bolts was mechanically agitated to reveal that two bolts adjacent to the vacant location were loose. A total of three bolts were replaced with oversized bolts. Replacement required three holes to be machined into the core barrel for tool access. This work was completed in 1997.

Clevis Insert Bolt Degradation and Replacement
Indications were discovered in the clevis insert bolts of the lower radial support system (LRSS) while performing the ASME Code, Section XI, 10-year ISIs in 2010. At the time of the RVI AMP submittal in October 2012, a repair methodology and associated tooling was under development. In its response to RAI-1 (Ref. 2), dated July 30, 2014, the licensee provided an updated summary of the corrective actions for the CNP Unit 1 LRSS clevis insert bolt degradation, indicating that a minimum bolt pattern was installed on each of the six LRSS clevis inserts in 2013.

Reactor Coolant System (RCS) Pressure and Temperature Changes
RCS operating pressure and temperature were reduced in 1989 to support aging management for the original steam generators (SGs). The RCS operating pressure and temperature will return to their original values in 2016 based on replacement of the SGs. The change to RCS

operating pressure and temperature was approved by the NRC in the issuance of license amendment No. 329 for CNP Unit 1.

3.3.1.2 CNP Unit 2 Operating Experience

Modification from "15X15" to "17X17" Fuel Assembly Design
Modifications at CNP Unit 2 were made to accept "17X17" fuel assemblies. The licensee stated that appropriate changes were made to the RVIs at the manufacturer's shop prior to operation. Spare parts generated from the modification were retained by I&M. These spare CRGT assemblies were later installed in CNP Unit 1 as described in Section 3.3.1.1 of this assessment.

Control Rod Guide Tube Support Pin Replacement
A small number of the CRGT support pins failed at CNP Unit 2 and were retrieved from two SGs in 1985. The original pins were replaced during the following refueling outage with an improved stress design fabricated from Alloy X-750. This work was completed in 1986.

Control Rod Guide Tube Cap Screw Modification
Each CRGT has four hold down socket head cap screws fastening it to the support plate. During the CRGT support pin replacement project for CNP Unit 2, two of the screws were broken during untorquing, leaving the threaded portion of the screws in the tapped support plate holes. In addition, two threaded holes were damaged. Each of the four damaged locations was on a different CRGT assembly. These four cap screw locations were abandoned, as supported by analysis from the original equipment manufacturer. Three high strength bolts were installed at the remaining available locations on these four CRGT assemblies. This work was completed in 1986.

Baffle-Former Bolt Partial Replacement
A number of baffle-former bolts were discovered on the lower core plate in 2010. Visual inspection revealed 18 failed bolts in a local area on the large south baffle plate. The licensee replaced bolts with visual indications, and adjacent bolts, in the large south baffle plate to bound the edge of the local degradation. Bolt samples were removed from the other three large baffle plates and inspected to ensure degradation was not occurring at symmetric locations. A total of 52 bolts were replaced, with two locations left vacant.

Reactor Pressure Vessel Closure Head Replacement
The original reactor pressure vessel (RPV) closure head was replaced in 2007. Flow restrictors were installed on part-length CRGT assemblies to accommodate the new head design.

3.3.2 NRC Staff Evaluation of Operating Experience

Based on its review of the plant-specific RVI OE, the NRC staff determined that additional information would be needed regarding the findings, root cause, and corrective actions for the RVI component degradation. The degraded RVI components that could potentially impact the efficacy of the CNP RVI AMP are the CNP CRGT support pins, CNP Unit 1 LRSS clevis inserts bolts, the CNP Unit 1 barrel-former bolts, and the CNP Unit 2 baffle-former bolts. The RAIs and the NRC staff's evaluation of RAI responses related to aging management of the clevis insert bolts, barrel-former bolts, and baffle-former bolts are addressed below. A plant-specific

analysis of the aging management requirements for the CRGT support pins is addressed in Subsection 3.4.3 of this staff assessment for Action Item 3.

3.3.2.1 Barrel-Former Bolt Failures and Baffle-Former Bolt Failures

Regarding the CNP Unit 1 barrel-former bolt failures and CNP Unit 2 baffle-former bolt failures, the NRC staff requested in RAI-7 (Ref. 15), that the licensee discuss the root cause of the failed bolts and, given the extent of these bolt failures at CNP beyond the general industry OE for these bolts, justify the use of the generic MRP-227-A guidelines for bolt inspection.

In its response to RAI-7 (Ref. 2), dated July 30, 2014, the licensee stated that a root cause evaluation for the three failed barrel-former bolts at CNP Unit 1 was performed following discovery of the bolt failures in 1995. The root cause evaluation included a metallurgical evaluation of the failed bolts, an as-built flow-induced vibration model of the core barrel and thermal shield in the region of the failed bolts, and a steady state thermal analysis. No single root cause was identified. Contributing causes were determined to be elevated bolt stress near the thermal shield support block and bending stress on the bolts during normal steady state operation. The licensee specifically noted that SCC was not a factor in the mode of failure.

The CNP Unit 1 barrel-former bolts were returned to a fully qualified condition by replacing the three failed bolts, and the system was returned to its former monitoring requirements following bolt replacement. All barrel-former bolts were inspected at the time of discovery of the three failures, and inspections at symmetrical locations in the lower internals assembly revealed no other bolt failures. According to the licensee, this indicates an isolated, not a generic issue. The licensee stated that I&M has not observed any abnormal conditions or symptoms related to the CNP Unit 1 barrel-former bolts following the permanent repair completed in 1997. The licensee concluded that the barrel-former bolts will be adequately managed as Expansion components under the MRP-227-A inspection guidelines and by existing monitoring and aging management programs already in place. Therefore the licensee concluded that no augmentation of inspection criteria is necessary for these components.

For the baffle-former bolt failures at CNP Unit 2, the licensee stated in its response to RAI-7 (Ref. 2), dated July 30, 2014, that a root cause evaluation was completed following the discovery of the failed baffle-former bolts in 2010. The licensee indicated that the root cause of the failed baffle-former bolts was IASCC, in conjunction with loss of preload in several bolts. The licensee noted that other potential contributing causes included embrittlement, high cycle fatigue, overload, and a steady-state pressure gradient across the baffle plates.

The licensee stated that the CNP Unit 2 baffle-former bolts were returned to a fully qualified condition in 2010 by replacing the 18 failed bolts, and adjacent bolts, in the large south baffle plate. A total of 52 bolts were replaced in the large south baffle plate, with two locations left vacant. Bolt integrity was confirmed at symmetric locations on the other three large baffle plates. The licensee noted that this approach justified the return of the unit to normal operation without additional monitoring requirements, although a voluntary visual inspection of the CNP Unit 2 baffle-former bolts was performed during the following refueling outage in 2012. I&M has not observed any abnormal conditions for these bolts following the repair completed in 2010. The licensee concluded that the baffle-former bolts will be adequately managed as Primary components under the MRP-227-A inspection guidelines and by existing monitoring

and aging management programs already in place. Therefore the licensee concluded that no augmentation of inspection criteria is necessary for these components.

The NRC staff reviewed the licensee's July 30, 2014, response to RAI-7 for the barrel-former and baffle-former bolt failures, and determined that additional information was needed regarding the root cause and corrective actions for the bolt failures. Therefore, the staff issued Follow-Up RAI-4 (Ref. 16), dated May 5, 2015, consisting of parts (a) through (h). The licensee's response (Ref. 5), dated August 6, 2015, to Follow-Up RAI-4, parts (a) through (h), is summarized below.

4(a): The NRC staff determined that the three barrel-former bolts failures at CNP Unit 1 may have resulted from abnormal loading conditions. Therefore, the staff requested clarification on the corrective actions taken to resolve the root cause of the abnormal loading on the barrel-former bolts.

Response: The licensee stated that thermal shield attachments to the core barrel caused differential thermal expansion between the core barrel and former plates, resulting in bending loads in the three failed barrel-former bolts that could have affected bolt preload. The bending loads, combined with elevated vibration loads due to the proximity to a thermal shield support block, may have provided the driver for wear by fretting on the bottom of the bolt threads. The licensee stated that wear led to the eventual thread disengagement of one bolt, and looseness in two other bolts. The licensee explained that these concerns were addressed for the replacement bolts by selecting a higher yield strength material, allowing more elastic strain to be stored in the form of bolt preload. The higher preload reduces the effects of the small high-cycle vibration loads.

4(b): The NRC staff requested that the licensee state the original and replacement bolt materials and confirm whether the three barrel-former bolts with indications of failure were the only bolts replaced at CNP Unit 1.

Response: The licensee stated that only the three barrel-former bolts were replaced because these were the only locations with indications of looseness or failure. The original bolt material for CNP Unit 1 was Type 347 stainless steel (SS). The replacement bolt material is Type 316 SS.

4(c): The NRC staff requested that the licensee describe the existing monitoring and aging management programs currently in place that are applicable to the barrel-former bolts.

Response: The licensee stated that CNP has an impact monitoring system for detecting loose parts in the RCS in the form of free or unrestrained barrel-former bolts or bolt fragments. The licensee stated that a comprehensive foreign material inspection is performed during every outage. The inspection includes the lower core plate and a sample of accessible areas below the lower core plate. Such inspections could detect barrel-former bolts or bolt fragments displaced from their installed location. Access was gained for replacement of the three barrel-former bolts by machining a hole in the thermal shield at each bolt location. The holes provide accessibility for continued monitoring. A visual inspection was performed on the three replacement bolts concurrent with the 2010 ASME Code, Section XI, Category B-N-3 inspections in 2010. The inspection confirmed that the replacement bolts are still in place with no indications.

4(d): The NRC staff requested that the licensee elaborate on the reason for leaving two bolt locations vacant at CNP Unit 2, and briefly discuss whether any analysis was performed to ensure continued functionality of the baffle-former assembly with the two vacancies.

Response: The licensee stated that two additional baffle-former bolts with visual indications were identified during an inspection following bolt replacement. These bolts were located along the edges of the degradation zone. The bolts were removed and the holes were abandoned. The licensee performed a bounding evaluation to justify the reduction in the number of required baffle-former bolts, which excluded credit for eight bolts (two holes vacant plus six uncredited bolts in place) along the edges of the degradation zone. The evaluation concluded that the baffle-former assembly would continue to perform its design function. The licensee performed a separate evaluation of the two vacancies, which concluded that acceptable margins existed to resume operation with the two vacancies.

4(e): The NRC staff requested that the licensee state the total number of baffle-former bolts at CNP Unit 2, the total number of baffle-former bolts examined at symmetrical locations in the other three baffle plates in 2010, and the method of examination that was used for the sampled bolts at the symmetrical locations.

Response: The licensee stated that the total original design number of baffle-former bolts at CNP Unit 2 was 832, and with the two vacant locations there are now 830 baffle-former bolts. A VT-3 visual inspection was performed on all baffle-former bolts in the four large baffle plates at CNP Unit 2 in 2010, and indications were found only on the bolts in the large south baffle plate. One sample bolt was removed from the north, east, and west large baffle plates and tensile tested to a maximum load. NDE was performed on each bolt pre- and post-tensile test. The three sampled bolts were tested for chemical composition. One bolt was sectioned, and cross-sectional metallography and hardness testing were performed. Higher chromium content was observed in the bolts from the east and west wall. The licensee stated that this is beneficial for SCC resistance of SS in a PWR RCS environment, and that it is not an indication of increased susceptibility. The licensee stated that no other anomalies or indications were identified.

4(f): The NRC staff requested that the licensee indicate whether the cracked bolts conformed to any pattern related to neutron exposure.

Response: The licensee stated that the highest neutron exposure is on the re-entrant corners of the baffle-former assembly where no failures were observed. Therefore, the pattern of failed bolts did not conform to the areas of highest neutron exposure. The licensee noted that neutron exposure for all of the baffle-former bolts exceeded three displacements per atom (dpa), which means that they are all susceptible to IASCC.

4(g): The NRC staff requested that the licensee identify the original baffle-former bolt material and the replacement bolt material for CNP Unit 2.

Response: The licensee stated that the original CNP Unit 2 baffle-former bolts are Type 347 SS, and the replacement bolts are Type 316 SS.

4(h): Given that the 18 baffle-former bolt failures occurred in a local area in the south baffle plate, the baffle-former bolts may have greater susceptibility to IASCC at this location and thus may require the baseline ultrasonic testing (UT) examination sooner than the 35 EFPY deadline specified in MRP-227-A. Therefore, taking into consideration the inspection schedule for these Primary components, the NRC staff requested that the licensee justify the adequacy of performing the baseline UT examination of these bolts between 25 and 35 EFPY, as specified in MRP-227-A.

Response: The licensee stated that the baffle-former bolts will receive UT examinations during the 2019 refueling outage at a projected 29.7 EFPY. The licensee also performed a voluntary VT-3 visual examination during the 2012 refueling outage with no indications. The licensee stated that the original UT inspection date of 2022 was specifically rescheduled to 2019 based on the plant-specific OE with baffle-former bolt degradation, and that no other inspections are planned between the 2012 visual exam and the initial UT exam in 2019. The licensee also stated that I&M implements the impact monitoring system for detecting loose parts during plant operation and performs comprehensive foreign material inspections of the RPV during each refueling outage. The licensee confirmed through the tests described above that bolt degradation occurred only on the large south baffle plate. The licensee stated that the design, fabrication, and installation of the replacement bolts met the applicable ASME Code requirements. Therefore, the licensee concluded that this bolt replacement activity addressed the root cause of baffle-former bolt degradation at CNP Unit 2.

3.3.2.1.1 NRC Staff Conclusion for Barrel-Former and Baffle-Former Bolt Failures

For the three barrel-former bolt failures at CNP Unit 1, the NRC staff reviewed the information provided by the licensee in response to Follow-Up RAI-4, parts (a), (b), and (c). The staff determined that the licensee took the appropriate corrective actions to resolve the root cause of these three bolt failures. Specifically, the staff determined that the licensee's description of the elevated bending loads and vibration loads at the location of the three failed bolts demonstrates appropriate identification of the root cause of the bolt failures. Furthermore, the staff determined that the licensee took the appropriate corrective actions to resolve the root cause because it selected a higher yield strength material and utilized an oversized bolt design, which allowed for a higher bolt preload. The staff verified that the design and materials for the replacement bolting would be effective in allowing for greater bolt preload, which would more effectively restrain the bolts under the vibration loads described by the licensee and thereby mitigate any potential fretting of bolt threads. The staff also confirmed that the selection of Type 316 SS for the replacement bolt material is more effective than the original bolt material, Type 347 SS, in maintaining a greater bolt preload. Also, the staff found that the licensee has existing monitoring programs in place for detecting bolt failures. Based on its review of the licensee's corrective actions and monitoring programs, the staff determined that the licensee's implementation of the generic MRP-227-A inspection criteria for Expansion components, without augmentation, is adequate for aging management of the barrel-former bolts. It will provide adequate assurance of Unit 1 core barrel assembly functionality during the PEO.

For the baffle-former bolt failures at CNP Unit 2, the NRC staff reviewed the information provided by the licensee in response to Follow-Up RAI-4 parts (d), (e), (f), (g), and (h). The staff determined that the licensee took the appropriate corrective actions to resolve the root cause of the failed baffle-former bolts at CNP Unit 2, and adequately addressed the concern regarding the implementation of the MRP-227-A inspection guidelines for these Primary

components. Specifically, the staff determined that the licensee's response to part (d) adequately described the analysis for ensuring functionality of the baffle-former assembly with the two vacant bolt holes. The staff determined that the licensee's response to part (e) adequately described the methods of examination and testing that were used to ensure that baffle-former bolt degradation was restricted to the large south baffle plate. The staff determined that the information provided in response to part (f) demonstrates that the licensee comprehensively evaluated all of the aging mechanisms that could have contributed to the baffle-former bolt failures at CNP Unit 2. Based on its review of the responses to parts (g) and (h), the staff determined that the licensee implemented the appropriate corrective actions to address all of the possible aging mechanisms that may have contributed to the bolt degradation.

The MRP-227-A guidelines require a UT examination of a minimum of 75 percent of the total baffle-former bolt population, and the Unit 2 bolts were scheduled to receive this examination in 2019, per the licensee's response to Follow-Up RAI-4 part (h). However, the NRC staff noted that more recent (spring 2016) industry OE with significant baffle-former bolt failures at other Westinghouse 4-loop plants with RVI assemblies designed for downward coolant flow (i.e., the "downflow" configuration) has resulted in the need for additional interim measures to address this aging degradation concern. By letter dated July 27, 2016 (Ref. 18), the EPRI MRP provided interim guidelines for inspection of baffle-former bolts as "needed" under the Nuclear Energy Institute (NEI) 03-08 (Ref. 19) implementation protocol. These interim guidelines direct Westinghouse 4-loop plants with RVI assemblies in the downflow configuration (referred to as "Tier 1a" plants in the EPRI letter, which includes CNP Unit 1 and Unit 2) to perform the UT examination of the full population of baffle-former bolts at the next scheduled refueling outage from the date of the EPRI letter. Therefore, CNP Unit 1 and Unit 2 will be performing the initial UT examination of the full population of the baffle-former bolts during the fall 2017 and fall 2016 refueling outages, respectively, which are next scheduled refueling outages for each unit. All participating plants are obligated per industry protocols to implement the NEI 03-08 "needed" elements, however the NEI 03-08 initiative is not considered a licensing basis requirement by the NRC. These interim guidelines are intended to supplement the current MRP-227-A inspection guidelines to provide for more immediate aging management of the baffle-former bolts based on the most recent industry OE, and will be incorporated into subsequent revisions of the generic MRP-227 program for NRC staff approval. Based on its review of the licensee's corrective actions and monitoring programs, the staff determined that implementation of the MRP-227-A Primary components inspection guidelines, and EPRI interim inspection guidelines in Ref. 18, will be adequate for aging management of the baffle-former bolts during the PEO.

3.3.2.2 LRSS Clevis Insert Bolt Degradation

Section 2.1.3 of I&M's submittal discusses indications of age-related degradation for the LRSS clevis insert bolts at CNP Unit 1 in 2010. In RAI-8 (Ref. 15), the NRC staff requested additional information related to these findings, and the planned aging management activity for these components considering the plant-specific OE with clevis insert bolt degradation. The licensee's response (Ref. 3), dated September 4, 2014, to RAI-8, parts (a) through (d), is summarized below.

8(a): The NRC staff requested that the licensee discuss the safety significance of the LRSS clevis insert bolts, specifically taking into account the potential impact of clevis insert bolting failures on the capability to safely shut down the reactor.

Response: The licensee stated that the degraded LRSS clevis insert bolts did not result in the loss of the ability of the LRSS to perform its intended design function, or impede the ability to shut down the reactor. The licensee indicated that, even with a postulated failure of all bolts and dowel pins, the clevis inserts are expected to remain functional. The licensee explained that LRSS clevis inserts have redundant means of attachment. The licensee stated that bolt failures do not challenge the ability of the LRSS to perform its design function, however they are important to the plant from a commercial perspective. If the LRSS clevis insert bolts, dowel pins, and interference fit are defeated, a clevis insert may not be fully restrained if the core barrel is removed for maintenance. Displacement of a clevis insert during core barrel removal would result in the need for repair or replacement, which is economically challenging.

8(b): The NRC staff requested that the licensee provide a summary of previous activities related to aging management of the clevis insert bolts, dowel pins, or other clevis insert components that have exhibited age-related degradation.

Response: The licensee stated that the CNP Unit 1 LRSS clevis insert bolts and dowel pins received a VT-3 visual examination during the spring 2010 refueling outage. This examination revealed that 7 of 48 LRSS clevis insert bolts showed wear between the bolt head and dowel pin, indicating failure. Additionally, 1 of 12 LRSS dowel pins had broken tack welds, resulting in rotation and displacement of the pin. Based on an analysis performed by the original equipment manufacturer, the licensee replaced the commercial minimum bolt pattern in each of the LRSS clevis inserts at CNP Unit 1 in the spring of 2013. The commercial minimum bolt pattern is the minimum number of bolts required to ensure the proper function of the clevis insert bolting, assuming that the original bolts left in place had failed. The commercial minimum of 28 of the 48 clevis insert bolts were replaced at CNP Unit 1. The analysis supporting the 28 bolt replacement took no credit for the 20 original bolts left in place. The replacement bolts, fabricated from Alloy X-750, included a crimp cup locking device fabricated from Type 304L SS attached to the head of the bolt. The materials selected for the replacement bolting satisfied the applicable requirements of the ASME Code, Sections II and III. Following the replacement activity, a VT-3 preservice examination was performed on the clevis inserts before returning the system to service.

8(c): The NRC staff requested that the licensee state whether subsequent inspections of clevis insert components were performed during the 2013 outage, when clevis insert component replacement activities occurred. The staff requested that the licensee discuss the examination method and results, noting whether any additional degradation to the clevis insert components was found beyond that identified in 2010.

In addition, the NRC staff requested that the licensee address the aging management approach for the clevis insert components, and identify whether I&M will continue to perform inspections of the bolts in accordance with the MRP-227-A guidelines invoking ASME Code, Section XI requirements, or whether a more comprehensive examination method and/or shorter examination cycle will be used. If the clevis insert inspection criterion delineated in MRP-227-A is not modified for CNP, the staff requested that the licensee justify the adequacy of the current VT-3 visual examination method to detect bolt degradation before it results in component failure.

Response: The licensee stated that a VT-3 visual examination was performed on the LRSS clevis inserts prior to actual bolt replacement in the spring of 2013. The pre-maintenance VT-3 revealed that one additional (for a total of 8 of 48) LRSS clevis insert bolt had failed since the

2010 discovery of the 7 bolt failures and 1 dowel pin failure. The licensee stated that LRSS wear surfaces were inspected, and that no abnormal or excessive wear was observed on any surfaces. Also, the licensee found no dislocation or shifting of components, and all inserts appeared to be fully seated in their originally installed location.

The licensee stated that the aging management approach for the LRSS clevis inserts will continue to follow MRP-227-A guidelines for Existing Programs components, which specify VT-3 visual examinations per the ASME Code, Section XI. The licensee explained that, considering the low safety significance of LRSS clevis insert bolt and dowel pin failures, no augmentation has been made to the inspection requirements for these components.

8(d): The NRC staff requested that the licensee state when the last ASME Code, Section XI ISI was performed for the clevis inserts at CNP Unit 2, and discuss whether there were any findings of age-related degradation. The staff requested that the licensee address whether any repair/replacement activities were performed for potentially susceptible clevis insert bolts at CNP Unit 2, or provide an explanation for not replacing susceptible bolting.

Response: The licensee stated that the last ASME Code, Section XI ISI of the clevis inserts at CNP Unit 2 was performed in 1996 with no relevant indications. The licensee stated that relief was granted for CNP Unit 2 from the requirements of the ASME Code, Section XI for the 2009 ISI of RVI components that require removal of the core barrel (Ref. 20). Therefore, the LRSS was not inspected during the 2009 refueling outage. However, the core barrel was removed for other work in 2010, so the opportunity was taken to perform a VT-3 examination of the CNP Unit 2 LRSS clevis inserts. No indications of bolt degradation were identified during this inspection. The licensee stated that the low safety significance of clevis insert bolt failure does not warrant pre-emptive bolt replacement at CNP Unit 2.

3.3.2.2.1 NRC Staff Conclusion for LRSS Clevis Insert Bolt Degradation

Based on its review of the licensee's response to RAI-8, parts (a) though (d), the NRC staff determined that the licensee adequately demonstrated that the MRP-227-A Existing Programs inspection guidelines will remain adequate for aging management of the LRSS clevis insert bolts. The staff's determination is based on its findings that these inspection criteria will continue to provide adequate assurance that the essential core support and alignment function of the LRSS clevis inserts will be maintained for the PEO.

Regarding the issue raised in RAI-8(a), concerning the safety significance of the LRSS clevis insert bolts, multiple redundancies are incorporated into the safety-related design function of the LRSS clevis inserts. The failures of the clevis insert bolts at CNP Unit 1 did not have a significant impact on plant safety due to the fact that the core support and alignment function of the LRSS clevis inserts was adequately maintained. The root cause evaluation supports the licensee's statement that even if all clevis insert bolts and dowel pins failed, LRSS functionality would still be maintained during plant operation because the clevis inserts are adequately restrained by their interference fit in the RPV interior-attached lugs, and because of the specific geometry of the LRSS, which inherently restricts any movement of the LRSS clevis insert during plant operation. The staff verified the licensee's statement that the only concerns stemming from the failure of all bolts and dowel pins are inherently commercial due to the potential for the clevis inserts to back out of the interior lugs during removal of the core barrel. This potential only exists during core barrel removal, and core barrel removal only occurs during plant

maintenance activities when the reactor is in cold shutdown. Based on its review of these facts, the staff determined that the licensee's explanation of the significance of the clevis insert bolts is acceptable for resolving its concerns with the safety significance of the bolt degradation that occurred at CNP Unit 1.

Regarding the issues raised in RAI-8(b) and RAI-8(c), the NRC staff determined that the licensee took the appropriate corrective actions by replacing a total of 28 of the 48 clevis insert bolts at CNP Unit 1. The replacement was based on a minimum bolt pattern analysis performed by the original equipment manufacturer, taking no structural credit for the 20 original bolts remaining in place. Further, the staff noted that the safety-related alignment function of the LRSS clevis inserts is also ensured through the structural redundancies discussed above. Therefore, the staff determined that the licensee's corrective actions are acceptable for aging management of the LRSS clevis insert bolts.

Regarding the issues raised in RAI-8(d), concerning the adequacy of the MRP-227-A Existing Programs guidelines given the OE with clevis insert bolt failures at CNP Unit 1, the staff noted that the FMECA results summarized in MRP-191, Table 6-5 show that these components screened in for SCC and wear and are determined to have a medium likelihood of component failure. The component failures would result in a low likelihood of core damage. Accordingly, the clevis insert bolts are binned in MRP-191 FMECA Group 1, "low significance." However, it should be noted that the medium likelihood of failure is based on having no known failures for the component in the Westinghouse PWR fleet at the time the FMECA were developed. If known failures occur, the components must be considered to have a high likelihood of failure, in which case the bolts would be binned in FMECA Group 2, "medium significance," under the MRP-191 FMECA groupings. Given the high level of redundancy in the design of the LRSS clevis inserts, and the fact that the integrity of the bolts is not required to maintain the core support and alignment function of the LRSS clevis inserts, the Existing Programs inspection criteria invoking the ASME Code, Section XI ISI VT-3 visual examinations would still be appropriate for inspection of the bolts. Furthermore, the staff determined that the VT-3 examinations of the accessible surfaces for loss of material due to wear, as required by Existing Programs inspection criteria, would detect any movement of the clevis inserts and would be effective in detecting failures, as demonstrated by the previous VT-3 examinations performed in 2010 and 2013. Given the redundancies in the design of the LRSS clevis inserts, the efficacy of the VT-3 visual examinations in demonstrating proper alignment and detecting bolt failures, and the replacement of the 28 clevis insert bolts at CNP Unit 1, the staff determined that the licensee's implementation of the MRP-227-A guidelines invoking the existing ASME Code, Section XI ISI program requirements will provide adequate assurance of LRSS clevis insert functionality during the PEO.

Regarding the issue raised in RAI-8(e), concerning inspection results and aging management for the clevis inserts at CNP Unit 2, the NRC staff determined that clevis insert bolt replacement at CNP Unit 2 is not necessary because no indications of bolt degradation were identified during the most recent VT-3 visual examinations of the clevis inserts in 2010. Furthermore, the high level of redundancy in the design of the LRSS clevis inserts, the fact that the integrity of the bolts is not required to maintain the core support and alignment function of the LRSS clevis inserts, and the efficacy of the VT-3 visual examinations, demonstrate that the Existing Program guidelines invoking the ASME Code, Section XI ISI program requirements for VT-3 visual examinations will provide adequate assurance of clevis insert functionality during the PEO.

3.3.3 NRC Staff Conclusion for RVI Operating Experience

The NRC staff determined that the licensee performed the necessary corrective actions to address the root cause of the RVI component degradations discussed above. Furthermore, considering the licensee's corrective actions for the RVI component degradation, the licensee adequately demonstrated that its plant-specific implementation of MRP-227-A guidelines will provide adequate assurance of functionality for these RVI components during the PEO. The staff's findings specifically apply to those CNP RVI components that have exhibited age-related degradation, as documented in the licensee's description of its plant-specific OE. Those components are the CNP CRGT support pins, CNP Unit 1 LRSS clevis inserts bolts, the CNP Unit 1 barrel-former bolts, and the CNP Unit 2 baffle-former bolts. A detailed evaluation of the licensee's aging management activities for the CRGT support pins is addressed in Subsection 3.4.3 of this staff assessment for Action Item 3.

3.4 Applicant/Licensee Action Items

Section 4.2, "Plant-Specific Action Items," of the SE for MRP-227-A, provides eight action items that must be addressed on a plant-specific basis by PWR licensees and license renewal applicants submitting RVI AMPs for NRC staff review and approval. These action items concern topics related to the implementation of MRP-227-A which could not be effectively addressed on a generic basis in MRP-227-A. The licensee addressed these eight action items in Section 4.4.2 of the CNP RVI AMP.

3.4.1 Action Item 1 – Plant-Specific Applicability of FMECA Assumptions for MRP-227-A

As addressed in Section 4.2.1 of the SE for MRP-227-A, this action item specifies that each licensee is responsible for assessing its plant's design and operating history and demonstrating that MRP-227-A is applicable to the facility. Each licensee shall refer to the assumptions regarding plant design and operating history made in the FMECA and functionality analyses for reactors of their design. Licensees shall also 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 licensee shall submit this evaluation for NRC review and approval as part of its application to implement MRP-227-A.

The EPRI Letter MRP 2013-025 (Ref. 21) provides a generic template for use by licensees to demonstrate the plant-specific applicability of the FMECA assumptions for MRP-227-A when responding to NRC staff RAI questions concerning Action Item 1. For Action Item 1, the NRC staff determined that the basis for a plant to demonstrate plant-specific consistency with the underlying assumptions of MRP-227-A may be satisfied with plant-specific responses to the following generic questions:

Question 1: Does the plant have non-weld or bolting austenitic SS components with 20 percent cold work or greater? 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 SCC.

Question 2: Does the plant have atypical fuel design or fuel management that could render the assumptions of MRP-227-A regarding core loading and core design non-representative for that plant?

With respect to Question 1, the MRP 2013-025 guidelines state that licensees must complete the following actions:

- Confirm that plant-specific RVI components identified for aging management as part of license renewal were included in the MRP-191 component reviews. This action primarily supports Action Item 2.

- Confirm that the design and operating history of these components are consistent with MRP-191.

- Confirm that modifications or plant-specific activities performed on the component do not introduce a cold-worked condition.

MRP 2013-025 specifies that the plant-specific RVI component materials shall be binned according to the following categories:

Category 1: Cast Austenitic Stainless Steel (CASS)
Category 2: Hot-Formed Austenitic Stainless Steel
Category 3: Annealed Austenitic Stainless Steel
Category 4: Austenitic Stainless Steel Fasteners
Category 5: Cold-Formed Austenitic Stainless Steel, Without Subsequent Solution Annealing

The MRP 2013-025 guidelines state that component material Categories 1, 2, and 3 are generically considered as not cold-worked, and therefore are not considered susceptible to SCC in the generic assessments. Components in Categories 4 and 5 are generically identified as cold-worked. However, the manufacturer's specifications for Category 4 fastener materials may include limits on yield or tensile strength that would preclude cold work greater than 20 percent. The MRP 2013-025 guidelines state that licensees are responsible for determining whether plant-specific Category 4 components exceed the 20 percent cold-worked limit. MRP 2013-025 states that if a new component subject to SCC degradation is confirmed, a plant-specific evaluation of the components must be performed.

With respect to Question 2 above, the MRP 2013-025 guidelines state that to demonstrate plant-specific applicability of the MRP-227-A inspection guidelines, licensees must demonstrate that the criteria of MRP-227-A Section 2.4 are met, and that the neutron fluence and heat generation rates are within the allowable ranges. For Westinghouse plants, the limiting threshold values are as follows:

- Active fuel to upper core plate distance greater than 12.2 inches.
- Average core power density less than 124 Watts per cubic centimeter (W/cm^3).
- Heat generation figure of merit (HG-FOM) less than or equal to 68 W/cm^3.

MRP 2013-025 states that plants that exceed these thresholds may require additional evaluations to fully demonstrate plant-specific applicability of MRP-227-A. The limiting core thermal power values apply to operation going forward, and are based on plant operation for the PEO. A plant that maintains core loading patterns that meet the limits specified above would

satisfy the MRP-227-A applicability requirement relative to core loading and core design. MRP 2013-025 also states that operating above these limits for periods of fewer than 2 years would not invalidate the MRP-227-A Section 2.4 requirement to maintain the low neutron leakage core loading pattern. However, plants that exceed the limits for more than 2 years of operation would need to provide further evaluations to demonstrate compliance with this applicability requirement.

In its evaluation (Ref. 22) of the MRP-227-A applicability guidelines contained in EPRI Letter MRP 2013-025, the NRC staff determined that the guidance contained therein provides an acceptable basis for licensees to prepare responses to the generic RAI questions above.

3.4.1.1 Licensee Evaluation of Action Item 1

In Section 4 of the CNP RVI AMP submittal, the licensee addressed the three general assumptions of MRP-227-A Section 2.4 that were used to develop the MRP-227-A inspection guidelines, and described how CNP Units 1 and 2 are bounded by the three general assumptions.

Assumption 1: MRP-227-A assumes a maximum of 30 years of operation with high neutron leakage core loading patterns followed by implementation of a low neutron leakage fuel management strategy for the remainder of the 60-year extended license term. The licensee confirmed that CNP Units 1 and 2 changed from a high neutron leakage to a low neutron leakage core loading pattern prior to 30 years of operation.

Assumption 2: MRP-227-A assumes base load operation, which refers to plant operation at fixed power levels and does not involve variations in power level on a calendar or load demand schedule. The licensee confirmed that CNP Units 1 and 2 are both base load units, which operate at a fixed power level.

Assumption 3: MRP-227-A assumes no plant-specific design changes beyond those identified in general industry guidance or recommended by the original vendors. The licensee addressed its plant-specific OE and associated RVI component modifications and determined that CNP Units 1 and 2 remain bounded by this assumption.

In Section 4.4.2.1 of the CNP RVI AMP submittal, the licensee stated that I&M is participating in PWR Owners Group (PWROG) Project PA-MSC-0938, "Support for Applicant/Licensee Action Items 1, 2, and 7 from the Final Safety Evaluation on MRP-227," to address Action Item 1, and the results of the evaluation would be provided to the NRC prior to the PEO for each unit.

3.4.1.2 NRC Staff Evaluation of Action Item 1

Regarding the three general assumptions of MRP-227-A Section 2.4, the NRC staff noted that the licensee indicated that it switched to a low-leakage core loading pattern prior to 30 calendar years of operation, has always operated as a base load unit, and has no plant-specific design changes beyond those identified in general industry guidance or recommended by the original vendors. However, for Westinghouse and Combustion Engineering plants, a more detailed evaluation of the plant-specific design and operating criteria for addressing Action Item 1 is required, per the MRP 2013-025 guidelines discussed above.

Based on the licensee's statement in Section 4.4.2.1 of the CNP RVI AMP submittal that I&M is currently participating in PWROG Project PA-MSC-0938, the NRC staff requested in RAI-2 (Ref. 15) that the licensee justify that the results of PWROG Project PA-MSC-0983 are fully bounding for CNP, or identify any plant-specific non-conservatisms relative to these generic results for Action Item 1.

In its response to RAI-2 (Ref. 4), dated October 22, 2014, the licensee stated that I&M has assessed the design and operating history for CNP Units 1 and 2 to demonstrate that MRP-227-A is applicable to both units. The licensee stated that these evaluations considered the assumptions regarding plant-specific design and operating history made in the Westinghouse FMECA and functionality analyses supporting MRP-227-A. Based on these evaluations, the licensee determined that no changes are required for the CNP RVI AMP.

Enclosure 6 of the October 22, 2014, RAI response provides Westinghouse Letter LTR-RIAM-14-24, Rev. 1, "Reports for CNP Units 1 and 2 for PWROG PA-MSC-0983 Cafeteria Tasks 3, 4, and 5 Deliverables," dated October 3, 2014. Attachments 1 and 2 of Westinghouse Letter LTR-RIAM-14-24, Rev. 1 provide the plant-specific reports addressing Action Items 1 and 2 for CNP under PWROG Project PA-MSC-0983. Attachments 1 and 2 of LTR-RIAM-14-24, Rev. 1 are hereafter referred to as the Action Item 1 and Action Item 2 reports in this assessment.

The licensee's Action Item 1 report provided the following information:

- The CNP operating histories are consistent with the assumptions in MRP-227-A with regard to neutron fluence, based on the fact that these units switched to a low neutron leakage core design prior to 30 years of operation. However, the licensee noted that CNP Unit 1 is evaluating operating history to confirm compliance with the numerical limits regarding core loading/core design in MRP 2013-025.

- The CNP RVI components operate at temperatures that are consistent with the FMECA assumptions for MRP-227-A.

- Most of the CNP RVI component materials are consistent with the generic Westinghouse RVI component materials assumed for MRP-227-A, however some differences do exist. The licensee concluded that the MRP-227-A guidelines are still applicable based on the FMECA results for these plant-specific RVI component materials.

- Other than the specific material differences addressed for Action Item 2, the design and fabrication of the CNP RVI components are equivalent to the generic Westinghouse-designed PWR RVI components as established in MRP-191. Operating stress values for the CNP RVI components are consistent with those assumed in the FMECA, as established in MRP-191.

- The modifications to the RVI components over the lifetime of the plant are identified in general industry guidance or specifically directed by the original equipment manufacturer. The licensee and Westinghouse evaluated these modifications and determined that the CNP RVI components have remained within the original structural design configuration.

In RAI-3 (Ref. 15), the NRC staff requested that the licensee address the plant-specific questions identified in the MRP 2013-025 for demonstrating consistency with the FMECA assumptions of MRP-227-A.

(a) Address whether the CNP Units 1 and 2 RVI have non-weld or bolting austenitic SS components with 20 percent cold work or greater that are subject to operating stresses greater than 30 ksi (that were not generically evaluated as such in MRP-191). If both conditions are true, perform a plant-specific evaluation to determine the aging management requirements for the affected component, taking into consideration the susceptibility of the component to SCC.

(b) Address whether CNP Units 1 and 2 have ever utilized atypical fuel design or fuel management that could make the assumptions of MRP-227-A regarding core design and fuel loading patterns non-representative for the plant, taking into consideration power uprates. If so, describe how the differences were reconciled with the assumptions of MRP-227-A, or provide plant-specific aging management criteria for the affected components, as appropriate.

In its response to RAI-3(a) (Ref. 4), dated October 22, 2014, the licensee stated that I&M has evaluated the CNP RVI components according to the MRP-191 generic component listings and screening criteria, including consideration of cold work. The licensee identified differences in plant-specific RVI component material type relative to the generic RVI material types specified in MRP-191, Table 4-4 for Westinghouse plants. These material differences are listed in Tables 3-1 and 3-2 of the licensee's response to RAI-3(a) for CNP Units 1 and 2, respectively, and they include the 20 percent cold work assessment. The licensee stated that, with the exception of the components listed in Tables 3-1 and 3-2 of the RAI-3(a) response, all other CNP RVI components are the same as those listed in MRP-191, Table 4-4.

The licensee stated that differences in plant-specific RVI component material type from the CASS to wrought product family were resolved based on expert elicitation, consistent with the process outlined in MRP-191. The licensee stated that the remaining material differences are within the wrought austenitic SS family, for which the MRP-191 Section 3 screening criteria remain the same. The licensee concluded that these evaluations do not affect any component inspection categories, and no changes are required to the CNP RVI AMP as a result of these evaluations.

Tables 3-1 and 3-2 of the licensee's response to RAI-3(a) list differences in plant-specific RVI component materials, for CNP Units 1 and 2, respectively, from the Westinghouse generic RVI component materials that were analyzed in the MRP-191 FMECA. The tables include the 20 percent or greater cold work assessment requested in RAI-3(a). The staff's evaluation of differences in plant-specific RVI component materials from the MRP-191 generic material types, not specifically related to cold work, is addressed under Action Item 2 in Section 3.4.2 of this assessment.

MRP-191 generically evaluated a number of non-weld or bolting austenitic SS RVI components that are susceptible to SCC based on having 20 percent or greater cold work and subject to operating stresses greater than 30 ksi. Therefore, the NRC staff's evaluation of the licensee's response to RAI-3(a) specifically addresses the licensee's cold work assessments for the non-

weld or bolting austenitic SS components that are of a different material type than that assumed in the MRP-191 FMECA.

For most of the plant-specific RVI components that are of a different material type than those identified in MRP-191, the licensee confirmed that the amount of cold work is less than 20 percent. Therefore, since the MRP-191 screening criteria and MRP 2013-025 guidelines indicate that SCC is not an aging degradation concern if the amount of cold work is less than 20 percent, the NRC staff determined that these components are not susceptible to SCC.

The only plant-specific RVI components that are of a different material type than those in MRP-191, and that were identified by the licensee as having 20 percent or greater cold work, are the CNP Unit 1 replacement clevis insert bolt lock keys and the CNP Unit 2 thermal shield dowels. As discussed in Subsection 3.3 of this assessment, the lock keys on the CNP Unit 1 replacement clevis insert bolts are a crimp cup locking device fabricated from Type 304L SS, attached to the head of the bolt. MRP-191 identifies the clevis insert bolt lock keys as Type 316 SS or Alloy 600. The CNP Unit 2 thermal shield dowels with 20 percent or greater cold work are Type 302 or Type 304 SS, whereas MRP-191 identifies them as Type 316 SS. These components are all identified by the licensee as Category 4, wrought austenitic SS fasteners. The NRC staff noted that they are not screened as susceptible to SCC in the MRP-191 generic evaluation. Enclosure 2 of the licensee's RAI response, dated October 22, 2014, regarding the plant-specific cold work assessments, indicates that these Category 4 components were already assumed to have the potential for cold work in MRP-191. The Enclosure 2 cold work assessment also states that the CNP plant-specific material fabrication and design are consistent with the MRP-191 basis relative to cold work, and that the MRP-227-A inspection criteria are therefore directly applicable to CNP.

The NRC staff reviewed the statements in Enclosure 2 of the RAI response and verified that the MRP 2013-025 applicability guidelines indicate that all Category 4 and 5 RVI components are generically evaluated as having been cold-worked in the MRP-191 component evaluations. Furthermore, the staff verified that the Category 4 components identified above are not subject to operating stresses greater than 30 ksi, based on the operating stress screening results listed for these components in Appendix A of MRP-191. Therefore, the above Category 4 components are not susceptible to SCC, consistent with the MRP-191 screening results. The staff also confirmed that the cold work evaluation documented in Enclosure 2 of the RAI response determined that the Category 1, 2, and 3 RVI components at CNP all have cold work less than 20 percent based on a review of material specification and process information. Thus, the MRP-191 FMECA grouping, categorization, and MRP-227-A inspection criteria remain applicable for all RVI components relative to the consideration of cold work, and for the plant-specific components identified above with cold work greater than 20 percent.

In its response to RAI-3(b) (Ref. 4), dated October 22, 2014, the licensee stated that I&M has evaluated the CNP fuel design and fuel management and compared them with the assumptions of MRP-227-A. The licensee stated that CNP Units 1 and 2 both converted to a low neutron leakage fuel management strategy in the first 30 years of operation. However, the licensee identified that the CNP Unit 1 HG-FOM exceeded 68 W/cm^3 during Cycle 25, which started in the spring of 2013 and ended in the fall of 2014. The licensee stated that the CNP Unit 1 HG-FOM also exceeds the 68 W/cm^3 threshold during Cycle 26, which started in the fall of 2014 and ends in the spring of 2016. Thus, CNP Unit 1 will exceed 2 years of operation outside this HG-FOM threshold provided in the MRP 2013-025 guidelines. The licensee stated that no other

MRP 2013-025 thresholds have been exceeded, or are projected to be exceeded, for CNP. The licensee stated that no changes are required to its RVI AMP as a result of these evaluations; however, they are currently evaluating the interaction of the current fuel management strategy and the RVI aging management strategy.

Based on its review of the licensee's response to RAI-3(b) and the information provided in PWROG-14049-P, the NRC staff confirmed that two of the three MRP 2013-025 core design/thermal power thresholds are satisfied for all RVI components at CNP Unit 1, and all three of the MRP 2013-025 thresholds are satisfied for all RVI components at CNP Unit 2. Specifically, this information demonstrates that the active fuel to upper core plate distance is greater than 12.2 inches, and the average core thermal power density is less than 124 W/cm^3, for both CNP Units 1 and 2. The HG-FOM is less than or equal to 68 W/cm^3 for CNP Unit 2. However, the staff determined that additional information would be needed concerning the total amount of operating time, after the initial 30 years of plant operation, during which CNP Unit 1 is projected to exceed the 68 W/cm^3 threshold for HG-FOM. Therefore, the staff issued Follow-Up RAI-1 (Ref. 16), consisting of parts (a) and (b) stated below:

(a) The NRC staff noted a discrepancy between the licensee's response to RAI-3(b) and the information in PWROG-14049-NP, regarding the operating period when the CNP Unit 1 HG-FOM exceeded and/or will exceed the MRP-227-A applicability threshold of 68 W/cm^3. The staff requested that the licensee resolve this discrepancy by stating the actual operating period(s) during which the CNP Unit 1 HG-FOM exceeded and/or will exceed this applicability limit.

(b) The NRC staff requested that the licensee discuss any fuel management strategies that will be implemented for CNP Unit 1 to ensure that the HG-FOM will not exceed 68 W/cm^3 during future operation beyond the current fuel cycle, Cycle 26. If there are no fuel management strategies that would bring the CNP Unit 1 HG-FOM to within this limit, the staff requested that the licensee submit a plant-specific evaluation to demonstrate that the MRP-227-A guidelines and supporting MRP-191 FMECA are applicable to CNP Unit 1.

In its response to Follow-Up RAI-1(a) (Ref. 5), dated August 6, 2015, the licensee clarified that CNP Unit 1 exceeded the HG-FOM screening limit during Cycle 24, and is exceeding the HG-FOM screening limit during the current fuel cycle, Cycle 26. The licensee stated that no other fuel cycles have exceeded this limit after the first 30 years of operation. However, the CNP Unit 1 HG-FOM will exceed the 68 W/cm^3 threshold for more than 2 years during the current cycle. The licensee stated that CNP Unit 1 Cycle 24 ran from October 25, 2011, to March 27, 2013, for a total of 1.4 EFPY, and CNP Unit 1 Cycle 26 started on October 23, 2014, and is scheduled to complete on March 22, 2016. Cycle 26 is expected to operate for only 1.2 EFPY due to a forced maintenance outage.

In its response to Follow-Up RAI-1(b) (Ref. 5), dated August 6, 2015, the licensee stated that a fuel management strategy has been implemented to ensure that the HG-FOM will not exceed the 68 W/cm^3 screening limit beyond Cycle 26. The licensee stated that the site reactor core design procedure, EHI-4300, "Reactor Core Design," has been revised to include a requirement to observe screening limits outlined in MRP 2013-025, including the HG-FOM screening limit of 68 W/cm^3, and it includes a specific engineering stakeholder responsibility for this purpose.

Based on its review of the licensee's response to Follow-Up RAI-1(a), the NRC staff determined that CNP Unit 1 will operate for a total of 2.6 EFPY after the first 30 years of plant operation with a HG-FOM greater than the 68 W/cm^3 threshold established in MRP 2013-025. Based on its review of the response to Follow-Up RAI-1(b), the staff determined that the licensee provided an acceptable plan for ensuring that the HG-FOM will not exceed 68 W/cm^3 during future operation beyond the current fuel cycle. The licensee cited specific changes to relevant site procedures to ensure that it is bounded by this key MRP-227-A applicability threshold during future operation.

Plant operation for more than 2 years after the first 30 years with HG-FOM greater than the 68 W/cm^3 threshold would potentially warrant a more detailed analysis to demonstrate plant-specific applicability of the MRP-227-A guidelines, as recommended by MRP 2013-025. However, the NRC staff determined that for CNP Unit 1, the additional 0.6 EFPY of facility operation in excess of 2 years does not violate the MRP-227-A Section 2.4 requirement to maintain the low neutron leakage core loading pattern. Assuming a worst-case scenario, where all of the additional 0.6 EFPY results in high neutron leakage, the additional 0.6 EFPY would result in a maximum increase of approximately one percent of the projected neutron fluence assumed in the MRP-191 generic FMECA. In addition, neutron embrittlement trends in ferritic alloys follow a saturation curve, which means that additional neutron fluence beyond the 2-year threshold would have a progressively diminishing impact on the embrittlement of RVI components. Therefore, for CNP Unit 1, the exceedance of the 2-year HG-FOM threshold does not necessitate a detailed plant-specific analysis to demonstrate applicability of FMECA assumptions, because the additional 0.6 EFPY operating time will have a negligible impact on the susceptibility of RVI components to neutron fluence-dependent aging mechanisms. The MRP-227-A inspection guidelines will provide for adequate aging management and adequate assurance of RVI component functionality at CNP Unit 1.

3.4.1.3 NRC Staff Conclusion for Action Item 1

Based on its review of the licensee's RAI responses for addressing the MRP-227-A applicability guidelines established in MRP 2013-025, as documented above, the NRC staff determined that the licensee has adequately demonstrated that the FMECA assumptions of MRP-227-A are applicable to CNP Units 1 and 2, and has demonstrated that it is bounded by the three general assumptions of MRP-227-A Section 2.4. Therefore, the staff determined that the licensee has satisfied the criteria of Action Item 1 for CNP.

3.4.2 Action Item 2 – RVI Components within the Scope of License Renewal

As addressed in Section 4.2.2 of the SE for MRP-227-A, this action item specifies that pursuant to 10 CFR 54.4, each licensee is responsible for identifying which RVI components are within the scope of license renewal for its facility. Specifically, licensees for Westinghouse plants shall review the information in Table 4-4 in MRP-191, and identify whether this table contains all of the RVI components that are within the scope of license renewal for their facilities in accordance with 10 CFR 54.4. If this table does not identify all the RVI components that are within the scope of license renewal for its facility, the licensee shall identify the missing component(s) and propose any necessary modifications to the program defined in MRP-227-A. The AMP shall provide assurance that the effects of aging on the missing component(s) will be managed for the PEO.

3.4.2.1 Licensee Evaluation of Action Item 2

In Section 4.4.2.2 of the CNP RVI AMP submittal, the licensee stated that I&M is participating in PWROG Project PA-MSC-0938 to address this action item, and the results of the evaluation will be provided to the NRC prior to the PEO for each unit.

3.4.2.2 NRC Staff Evaluation of Action Item 2

The NRC staff requested in RAI-4 (Ref. 15) that the licensee provide the results of the plant-specific evaluation required to satisfy Action Item 2. In its response to RAI-4 (Ref. 4), dated October 22, 2014, the licensee stated that I&M has compared the plant-specific RVI components for CNP with those listed in MRP-191, Table 4-4. The licensee stated that all plant-specific RVI components for CNP are represented in MRP-191, Table 4-4, but some material differences do exist. The licensee indicated that a list of the plant-specific RVI components with materials that differ from the generic Westinghouse material types identified in MRP-191, Table 4-4 are provided in Tables 3-1 and 3-2 of the RAI-3(a) response for CNP, respectively, and the information in Westinghouse Letter LTR-RIAM-14-24, Rev. 1 describes how these materials differences are reconciled.

The licensee's report for Action Item 2 is included in Westinghouse Letter LTR-RIAM-14-24, Rev. 1, which is provided in Enclosure 6 of the October 22, 2014, RAI response. The Action Item 2 report states that the differences in plant-specific RVI component materials from those specified in MRP-191 Table 4-4 were considered, and it was determined that those plant-specific CASS components that are generically evaluated as non-CASS (wrought) in the FMECA required further evaluation. The report states that a FMECA expert review, applying the same methodology as used in the development of MRP-191, was conducted for these plant-specific CASS components. The review concluded that the MRP-227-A guidelines are still applicable to CNP based on the FMECA results. The Action Item 2 report states that the evaluation of these differences in plant-specific material type, relative to those listed in MRP-191 Table 4-4, ensures that the effects of aging on the plant-specific RVI components will be adequately managed for the PEO.

The plant-specific components that were originally established as being within the scope of license renewal for the facility, and subject to aging management for the PEO, are identified in the plant's license renewal application AMR tables. This license renewal application information was subject to NRC review and approval, as documented in the final license renewal safety evaluation report (SER). The AMR table for the CNP RVI components is provided in Table 3.1.2-2 of the CNP license renewal application dated October 2003 (Ref. 23). As documented in the final license renewal SER (Ref. 8) for CNP, the staff's review of license renewal application Table 3.1.2-2 did not revise or otherwise alter the scope of components and corresponding materials established therein. Therefore, for this review of the licensee's Action Item 2 evaluation, the staff compared the generic Westinghouse RVI components and materials identified in MRP-191, Table 4-4, with the RVI components and materials listed in license renewal application Table 3.1.2-2, specifically taking into consideration the plant-specific RVI component material differences. Based on this comparison, the staff determined that there are no plant-specific RVI components that were identified as being within the scope of license renewal for CNP that are not captured in MRP-191. Furthermore, the staff determined that Tables 3-1 and 3-2 of the licensee's RAI-3(a) response comprehensively identified all plant-

specific RVI components within the scope of license renewal that have different materials from those identified in MRP-191, Table 4-4.

The NRC staff reviewed the licensee's evaluation of the impact of its plant-specific material differences on the FMECA underlying the implementation of the MRP-227-A guidelines for the CNP RVI AMP. Based on its review of Tables 3-1 and 3-2 of the RAI-3(a) response, the NRC staff noted that there are three types of differences in plant-specific RVI component materials relative to the MRP-191, Table 4-4 generic RVI component materials:

- Plant-specific CASS components that are analyzed as non-CASS (i.e., wrought components) in MRP-191;
- Plant-specific wrought components that are analyzed as CASS in MRP-191; and
- Plant-specific RVI component material differences within the wrought product family.

For the plant-specific CASS components that were generically evaluated as wrought product in MRP-191, the NRC staff noted that there is one additional aging degradation mechanism – TE – that must be considered for CASS, and one aging degradation mechanism – IE – that has a lower neutron irradiation screening threshold for CASS than for wrought product forms. TE and IE could pose a plant-specific aging degradation concern if these CASS components' material properties and neutron exposure exceed the screening thresholds. The staff's evaluation of the licensee's analysis for these plant-specific CASS components is addressed in Subsection 3.4.7 of this staff assessment for Action Item 7.

For the plant-specific wrought components that were generically evaluated as CASS in MRP-191, there exists one aging mechanism – TE – that does not need to be considered for these components because it is uniquely applicable to CASS and precipitation-hardenable SS RVI components. Furthermore, there are no additional aging mechanisms that would need to be considered for these plant-specific wrought components, since SCC of the base metal was screened out for these under Action Item 1. Therefore, the net effect of the above components being plant-specific wrought product, rather than CASS, is that there are fewer aging mechanisms, thereby demonstrating that the FMECA grouping and MRP-227-A inspection criteria are conservative and appropriate for these plant-specific wrought components. Accordingly, the NRC staff determined that Action Item 2 is resolved for these components.

For plant-specific material differences within the wrought product family, the NRC staff confirmed that the difference between plant-specific Type 304 SS (or 304L SS) and generic Type 316 SS, does not have any impact on these components' screening results for the eight age-related degradation mechanisms addressed in MRP-191. Therefore, the staff determined that these material differences will not affect the FMECA results that underlie the MRP-227-A inspection and evaluation guidelines for these components. Accordingly, the staff determined that the FMECA grouping and MRP-227-A inspection criteria remain the same for the plant-specific components that are of a different wrought material type than the wrought material type analyzed in MRP-191. Therefore, the staff determined that Action Item 2 is resolved for these components.

3.4.2.3 NRC Staff Conclusion for Action Item 2

Based on its review of the licensee's RAI responses for addressing Action Item 2, as documented above, the NRC staff determined that all plant-specific RVI components that are

within the scope of license renewal are represented in MRP-191, Table 4-4, taking into consideration the plant-specific material differences evaluated above. Furthermore, the staff determined that the licensee adequately resolved differences in plant-specific RVI component materials from those evaluated in MRP-191. The licensee demonstrated that the MRP-227-A guidelines remain applicable for these components. Therefore, the NRC staff determined that the licensee has satisfied the criteria of Action Item 2 for CNP.

3.4.3 Action Item 3 – Evaluation of the Adequacy of Plant-Specific Existing Programs

As addressed in Section 4.2.3 of the MRP-227-A SE, this action item specifies that licensees for Combustion Engineering and Westinghouse plants are required to perform a plant-specific analysis either to justify the acceptability of their existing programs, or to identify changes to the programs that should be implemented to manage the aging of the applicable RVI components for the PEO. The results of this plant-specific analysis, and a description of the plant-specific programs relied on to manage aging of these components, shall be submitted as part of the licensee's AMP application. The Westinghouse components identified for this type of plant-specific evaluation are the CRGT support pins.

3.4.3.1 Licensee Evaluation of Action Item 3

In Section 3 of the CNP RVI AMP submittal, the licensee identified four existing plant programs that are credited for aging management of the RVI components.

<u>ASME Code, Section XI Inservice Inspection Program</u>
The licensee credits this program for periodic ISI to detect aging degradation in RVI components and disposition of ISI results in accordance with the acceptance criteria provided in the ASME Code, Section XI. The licensee noted that this program has been effective for managing RVI component aging.

<u>Primary Water Chemistry Program</u>
The licensee credits this program for control of primary water chemistry to mitigate RVI component degradation due to corrosive contaminants. The licensee stated that this program maintains concentrations of primary water chemical species within the system-specific tolerance, and follows guidance provided in the EPRI PWR Water Chemistry Guidelines. The licensee noted that this program has been effective for controlling water chemistry to mitigate RVI component degradation.

<u>Incore Instrumentation Thimble Tube Multi-Frequency Eddy Current Inspection</u>
The licensee credits this program for inspections of the thimble tubes for loss of material due to wear in accordance with NRC Bulletin 88-09. The licensee noted that this program has been effective in identifying loss of material due to wear prior to leakage for the thimble tubes.

<u>Industry Involvement</u>
The licensee stated that CNP participates in industry activities related to aging management of RVI, such as the PWROG and the EPRI MRP.

In Section 4.4.2.3 of the CNP RVI AMP submittal, the licensee stated that CNP Units 1 and 2 both have Alloy X-750 CRGT support pins, and project requests have been initiated to

investigate support pin replacement for each unit. The licensee also stated that I&M will provide the NRC with the strategy for managing support pins prior to the PEO for each unit.

3.4.3.2 NRC Staff Evaluation of Action Item 3

Appendix C of the CNP RVI AMP submittal identifies the inspection criteria for the RVI components that receive aging management under the existing plant programs described above. These inspection criteria are consistent with those specified in Table 4-9 of MRP-227-A for Existing Programs components.

The LRSS clevis insert bolts are designated Existing Programs components in Appendix C of the RVI AMP and MRP-227-A Table 4-9, and they receive VT-3 visual examinations of all accessible surfaces once every 10-year ISI interval. As discussed in staff assessment Section 3.3, the staff determined that the licensee's ASME Code, Section XI ISI program would remain adequate for aging management of the LRSS clevis inserts taking into consideration the plant-specific OE with clevis insert bolt degradation at CNP Unit 1. Therefore, considering this plant-specific aging degradation, the staff determined that the MRP-227-A Existing Programs RVI component inspection criteria are acceptable for aging management of these components at CNP. Furthermore, other than the CRGT support pins, there are no other RVI components assigned to the Existing Programs inspection category that require plant-specific evaluation under this action item.

Regarding the CRGT support pins and associated OE with support pin failures at CNP Unit 2, the NRC staff determined that the licensee must provide more specific information concerning its plans for managing aging degradation of the support pins during the PEO. Therefore, in RAI-5 (Ref. 15), the staff requested that the licensee provide the following information:

(a) The NRC staff requested that the licensee provide a summary of the previous inspections that were performed on the support pins including the type of inspections, frequency of inspections, and the results of the inspections. The staff also requested that the licensee confirm whether the support pins are binned under ASME Code, Section XI, Examination Category B-N-3 for the core support structure.

(b) If SCC occurred in the support pins, the NRC staff requested that the licensee describe the corrective action that was taken to prevent recurrence of the aging degradation. If future replacement of the support pins is deemed necessary, the staff requested that the licensee state the type of material that will be used for replacement.

In its response to RAI-5(a) (Ref. 2), dated July 30, 2014, the licensee stated that the support pins are not binned under the ASME Code, Section XI, Examination Category B-N-3 because these items are not part of the core support structure. The licensee stated that no inspections are performed on the support pins, however they will be replaced as described in response to RAI-5(b).

In its response to RAI-5(b), the licensee stated that all support pins at CNP were replaced in the mid-1980s with an improved stress design fabricated from Alloy X-750 with a modified heat treatment. Replacement was performed for asset management due to industry and plant-specific support pin failures at that time. The licensee indicated that two original support pins fabricated from Alloy X-750 failed at CNP Unit 2 during operation in 1985.

The licensee stated that I&M is in the process of competitively bidding support pin replacement for CNP, and the material type will be selected by I&M once a vendor is chosen. The licensee provided a schedule and regulatory commitments that specify replacement of the support pins at CNP during the refueling outages scheduled for the fall of 2017 at Unit 1 and the fall of 2016 at Unit 2. These commitments also specify that information regarding the replacement support pin material will be provided to the NRC prior to support pin replacement at each unit.

The NRC staff finds that reasonable controls for the implementation of the above regulatory commitments are best provided through the licensee's administrative processes, including its AMP as described in the updated final safety analysis report (UFSAR) and its commitment management program. The above regulatory commitments do not warrant the creation of regulatory requirements, and are not relied upon for the approval of the CNP RVI AMP.

Based on its review of the licensee's response to RAI-5(b), the NRC staff noted that the current support pins that were placed into service in the mid-1980s are fabricated from Alloy X-750 with a modified heat treatment, which is susceptible to cracking. Therefore, the staff requested in Follow-Up RAI-3 (Ref. 16) that the licensee state whether a more cracking-resistant type of material, such as Type 316 SS, will be selected for the replacement pins scheduled for installation in 2016 and 2017. Additionally, based on its review of the licensee's statement that no inspections are performed for the pins, the staff also requested in Follow-Up RAI-3 that the licensee justify not performing any VT-3 visual inspections on accessible replacement pins during the PEO.

In its response to Follow-up RAI-3 (Ref. 5), dated August 6, 2015, the licensee stated that the Alloy X-750 support pins will be replaced with pins fabricated from Type 316 SS. The licensee noted that support pins fabricated from Type 316 SS are not susceptible to primary water SCC, which is the primary failure mechanism for Alloy X-750 support pins. The licensee stated that the MRP-227-A Section 4.4.3 guidance for CRGT support pins is limited to plant-specific recommendations, which instruct owners to review and follow the supplier recommendations for aging management and subsequent performance monitoring. The licensee noted that MRP-227-A, Table 3-3, "Final disposition of Westinghouse internals," does not identify Type 316 SS support pins as requiring aging management, and MRP-227-A does not include Type 316 SS support pins as a line item. The licensee emphasized that I&M is proactively replacing these items at CNP with more SCC-resistant Type 316 SS pins, in accordance with the supplier instructions. Further, the licensee noted that the Type 316 SS replacement pin supplier does not recommend inspection following installation and return to service. The licensee stated that the design of the replacement Type 316 SS support pins for CNP will be based on a program developed through the PWROG, which assessed the effects of wear, fatigue, stress relaxation, creep, SCC, void swelling, and embrittlement. This will address the degradation mechanisms identified in MRP-191. The licensee indicated that Type 316 SS pins were categorized as a Category A, "No Additional Measures" component. This licensee also indicated that the Type 316 support pin design first entered service in 1997, and no failures or adverse OE have been observed to date in the Type 316 SS support pin design. The licensee concluded that the CNP existing programs comply with the supplier recommendations and MRP-227-A requirements, and no inspections are required for Type 316 SS support pins.

The NRC staff reviewed the licensee's response to Follow-Up RAI-3 concerning its future aging management activities for the CRGT support pins. The licensee's schedule for replacing the

support pins at CNP during the refueling outages scheduled for the fall of 2017 at Unit 1, and the fall of 2016 at Unit 2, is consistent with the initial inspection schedules for Primary components required by MRP-227-A. The RAI response indicates that the support pin replacement activity will occur as soon as practical for the licensee. It should be noted that the support pins are not Primary components, however the NRC staff determined that the timing of the licensee's planned replacement of these items is consistent with the MRP-227-A aging management strategy. Therefore, it is acceptable.

The NRC staff determined that the licensee's selection of Type 316 SS for the replacement pins provides a sufficient basis for not performing any inspections of these components after they are replaced. The staff's determination is based on its verification of the MRP-191 FMECA results for Type 316 SS support pins. Specifically, Type 316 SS CRGT support pins are assigned to FMECA Group 1, "low significance." MRP-191 also indicates that the three screened-in degradation mechanisms of wear, fatigue, and stress relaxation would have minimal likelihood to cause failure, based on their assignment to FMECA "Category A," defined as "minimal aging degradation significance." The staff also confirmed that, in the final disposition of Westinghouse RVI components for developing the MRP-227-A inspection guidelines, Type 316 SS support pins were determined not to require any specific inspection activity. Therefore, based on its confirmation of FMECA results for Type 316 SS CRGT support pins, the staff determined that no inspections will be required during the PEO for the Type 316 SS replacement support pins.

3.4.3.3 NRC Staff Conclusion for Action Item 3

Based on its review of the licensee's RAI responses concerning plant-specific aging management of the CNP CRGT support pins, as documented above, the NRC staff determined that the licensee has demonstrated that its existing programs will be adequate for aging management of the RVI components assigned to the Existing Programs inspection category in MRP-227-A, and these inspection guidelines will provide adequate assurance of functionality during the PEO for the Existing Programs components at CNP. Therefore, the staff determined that the licensee has satisfied the criteria of Action Item 3 for CNP.

3.4.4 Action Item 4 – Babcock & Wilcox Core Support Structure Upper Flange Stress Relief

Action Item 4 is only applicable to the RVI for Babcock & Wilcox plants. Since CNP Units 1 and 2 are Westinghouse reactors, this action item requires no evaluation from the licensee.

3.4.5 Action Item 5 – Application of Physical Measurements for Babcock & Wilcox, Combustion Engineering, and Westinghouse RVI Components

As addressed in Section 4.2.5 of the MRP-227-A SE, this action item specifies that licensees shall identify plant-specific acceptance criteria to be applied when performing the physical measurements required by MRP-227-A for loss of compressibility for Westinghouse hold down springs. This action item also states that the licensee shall include in its proposed acceptance criteria an explanation of how the criteria are consistent with the plant's licensing basis and the need to maintain the functionality of the component under all conditions of operation during the PEO.

3.4.5.1 Licensee Evaluation of Action Item 5

In Section 4.4.2.5 of the CNP RVI AMP submittal, the licensee stated that CNP Units 1 and 2 both have Type 304 SS hold down springs and that physical measurement of the hold down springs is required by MRP-227-A. In Enclosure 2 of its RVI AMP submittal, the licensee provided a regulatory commitment stating that plant-specific acceptance criteria will be developed and submitted to the NRC prior to the first required physical measurement, and if acceptance criteria are not developed, the hold down springs will be replaced in lieu of performing the first required physical measurement. This commitment is applicable to CNP Units 1 and 2, and it specifies a scheduled completion date of "[p]rior to the first required physical measurement" for each unit.

3.4.5.2 NRC Staff Evaluation of Action Item 5

This action item requires that licensees provide acceptance criteria for physical measurement of the hold down springs. The licensee has provided a regulatory commitment for CNP that requires plant-specific acceptance criteria to be submitted to the NRC, or the replacement of the hold down springs, prior to the first required physical measurement. Table 4-3 of MRP-227-A specifies that direct measurement of hold down springs shall be performed within three cycles of the beginning of the license renewal period. Therefore, the first required physical measurement for CNP Unit 1 is the Cycle 29 refueling outage scheduled for 2019. The first required physical measurement for CNP Unit 2 is the Cycle 27 refueling outage scheduled for 2022. Therefore, the staff expects that the activity specified in this commitment will be completed no later than the Cycle 29 refueling outage for CNP Unit 1 and the Cycle 27 refueling outage for CNP Unit 2.

The NRC staff finds that reasonable control for the implementation of the above regulatory commitment is best provided through the licensee's administrative processes, including its AMP as described in the UFSAR and its commitment management program. The above regulatory commitment does not warrant the creation of a regulatory requirement, and is not relied upon for the approval of the CNP RVI AMP.

3.4.5.3 NRC Staff Conclusion for Action Item 5

The NRC staff has determined that the licensee's aging management activity for the hold down springs will provide adequate assurance of component functionality during the PEO for CNP. Accordingly, the staff determined that the licensee has satisfied the criteria of Action Item 5 for CNP.

3.4.6 Action Item 6 – Evaluation of Inaccessible Babcock & Wilcox Components

Action Item 6 is only applicable to the RVI for Babcock & Wilcox plants. Since CNP Units 1 and 2 are Westinghouse reactors, this action item requires no evaluation from the licensee.

3.4.7 Action Item 7 – Plant-Specific Evaluation of CASS Materials

As addressed in Section 4.2.7 of the MRP-227-A SE, this action item specifies that licensees shall develop plant-specific analyses to demonstrate that Westinghouse CASS lower support column (LSC) castings (also referred to as LSC bodies) will maintain their functionality during

the PEO. This action item also applies for additional RVI components that may be fabricated from CASS, martensitic SS, or precipitation hardened SS materials that were not generically evaluated as such in MRP-191, and for which a licensee has determined that aging management is required. These analyses shall consider the possible loss of fracture toughness in these components due to TE and IE, and may also need to consider limitations on accessibility for inspection and the resolution and sensitivity of the inspection techniques. The licensee shall include this plant-specific analysis as part of their submittal to implement MRP-227-A.

3.4.7.1 Licensee Evaluation of Action Item 7

In Section 4.4.2.7 of the CNP RVI AMP submittal, the licensee stated that I&M is participating in PWROG Project PA-MSC-0938 to address this action item for CNP, and the results of the evaluation will be provided to the NRC prior to the PEO.

3.4.7.2 NRC Staff Evaluation of Action Item 7

The NRC staff requested in RAI-6 (Ref. 15) that the licensee provide the following additional information:

(a) The NRC staff requested that the licensee identify all RVI components for which plant-specific functionality analyses are required under Action Item 7.

(b) Considering that CASS LSC castings are potentially susceptible to TE and IE, the NRC staff requested that the licensee provide the ferrite content and casting method for each LSC casting at CNP.

In its response to RAI-6(a) (Ref. 4), dated October 22, 2014, the licensee stated that evaluations of the CNP CASS components' susceptibility to TE and IE were performed, and a number of components in both units were found to be potentially susceptible. For those plant-specific CASS RVI components that are susceptible to TE and IE, and not generically analyzed as CASS in MRP-191, the licensee stated that a FMECA expert review was performed for these plant-specific CASS components. The licensee stated that the LSC castings at CNP are the only CASS components for which further analysis is being pursued under Action Item 7.

In its response to RAI-6(b), the licensee stated that all LSCs in both units were manufactured with static casting methods. The licensee stated that no Certified Material Test Reports (CMTRs) or specific fabrication records were located for the CNP Unit 1 LSC castings, therefore, the delta ferrite content could not be calculated. Thus, the licensee conservatively assumed that the CNP Unit 1 LSC castings have delta ferrite content greater than 20 percent, and are therefore considered potentially susceptible to TE. The licensee stated that the CNP Unit 2 LSC castings' CMTRs were located and used to calculate a delta ferrite content less than 20 percent. Since the CNP Unit 2 LSC castings have ferrite content less than 20 percent, they are considered not to be susceptible to TE.

At this time, there is no established methodology for assessing the impact of non-functional LSCs. The licensee stated the following in its RAI-6(b) response:

> I&M is participating in a PWROG project to develop a LSC functionality methodology to demonstrate sufficient redundancy within the lower support structure to tolerate a number of LSC failures. Once the methodology is developed, then it may be applied to representative plants, to specific plants, or otherwise used as required to satisfy NRC staff questions and concerns.

Based on its ongoing participation in PWROG activities for demonstrating LSC casting functionality, the licensee stated that I&M will provide a supplemental response to the NRC on RAI-6(b) when an acceptable methodology is developed by the PWROG.

The licensee's Action Item 7 reports for the plant-specific evaluation of CASS RVI components are provided in Attachments 3 and 4 of Westinghouse Letter LTR-RIAM-14-24, Rev. 1, in Enclosure 6 of the RAI response (Ref. 4), dated October 22, 2014. The licensee screened its CASS components for susceptibility to TE using the criteria established for non-irradiated CASS in the NRC letter from Christopher Grimes, Office of Nuclear Reactor Regulation, dated May 19, 2000 (Ref. 24, the C. Grimes letter). The licensee screened its CASS components for susceptibility to IE using the generic neutron exposure screening threshold for CASS components established in MRP-191.

Table 1 summarizes the licensee's determination of its CASS components' susceptibility to TE and IE, and the components' generic material and FMECA classification from MRP-191.

Table 1 – CNP CASS RVI Components

CASS Component	TE Susceptibility Based on May 2000 C. Grimes Letter	IE Susceptibility Based on MRP-191	Generic MRP-191 Material and FMECA[3] Group
CNP Unit 1			
CRGT[1] Assembly - Guide Plates/Cards	YES[2]	NO	304 SS, Group 3
CRGT[1] Assembly - Housing Plates	YES[2]	NO	304 SS, Group 0
Upper Instrumentation Conduit and Supports - Brackets, Clamps, Terminal Blocks, Conduit Straps	YES[2]	NO	304 SS, Group 0
Lower Support Column Assemblies - Lower Support Column Bodies	YES[2]	YES	CF8 CASS, Group 1
CNP Unit 2			
Upper Instrumentation Conduit and Supports - Brackets, Clamps, Terminal Blocks, Conduit Straps	YES[2]	NO	304 SS, Group 0
Upper Support Plate Assembly - Plate, Flange, and Upper Support Ring or Skirt	NO	NO	304 SS, Group 0 – Plate and Flange Group 2 – Ring /Skirt
Lower Support Column Assemblies - Lower Support Column Bodies	NO	YES	CF8 CASS, Group 1

Notes:
[1] CRGT components are assumed to be CASS by the licensee since documentation of constructional materials was not located.
[2] Susceptibility to TE is due to lack of CMTR data and assumed delta ferrite greater than 20 percent.
[3] Generic FMECA results for Westinghouse RVI are provided in MRP-191, Table 6-5.

On May 5, 2015 (Ref. 16), the NRC staff issued Follow-Up RAI-2 (a), (b), and (c), regarding the components in Table 1. Follow-up RAI-2 and the licensee's RAI response (Ref. 5), dated August 6, 2015, are discussed below.

(a) For the plant-specific CASS RVI components other than the CRGT guide cards and LSC castings (which are specifically addressed in parts (b) and (c) below), the NRC staff requested that the licensee provide justification for its determination that the MRP-227-A guidelines are still applicable based on the MRP-191 FMECA methodology. The staff requested that this justification include the plant-specific screening results for all aging mechanisms, an explanation of the likelihood of component failure, the likelihood of core damage, the resulting FMECA group for the components, and a discussion of how the final aging management strategy was determined.

(b) The CRGT guide cards are generically analyzed as 304 SS steel in MRP-191 and assigned to FMECA Group 3 in that report based on the screening results indicating susceptibility to SCC of the welds, wear, and fatigue. MRP-227-A Table 4-3 specifies that the guide cards are to be inspected for loss of material, also referred to as wear, using the VT-3 visual examination method every 10 years. The required examination coverage is 20 percent of the CRGT assemblies, with all guide cards in each selected CRGT assembly to receive the VT-3 examination. At CNP Unit 1, the CRGT guide cards are assumed to be CASS and susceptible to TE, based on the lack of CMTR data or other plant-specific records indicating otherwise. Therefore, in Follow-Up RAI-2(b) the NRC staff requested that the licensee provide an evaluation of the susceptibility of potentially CASS guide cards to cracking that considers the lower fracture toughness of TE-susceptible material. If the guide cards are susceptible to cracking, the staff requested that the licensee propose plant-specific inspection criteria for these components that would be sufficient for detecting cracking, or provide an evaluation of the components justifying that the VT-3 visual examination prescribed by MRP-227-A is adequate considering their greater susceptibility to cracking.

(c) Based on the lack of CMTR data for determining CNP Unit 1 LSC casting ferrite content, the NRC staff confirmed that the CNP Unit 1 LSC castings should be considered susceptible to TE and IE and therefore require analysis to demonstrate their functionality during the PEO. The CMTR data for CNP Unit 2 LSC castings shows that they have ferrite content less than 20 percent; however if ferrite is greater than 15 percent, the staff considers the synergistic effects of both TE and IE to be potentially applicable. Therefore, in Follow-Up RAI-2(c), the staff requested that the licensee submit the analysis to demonstrate the functionality of the CNP LSC castings during the PEO, considering aging degradation due to the synergistic effects of TE and IE, or propose a plant-specific change to the inspection criteria for these components.

In its response to Follow-Up RAI-2(a) (Ref. 5), dated August 6, 2015, the licensee stated that a plant-specific FMECA was performed for these CASS components using guidance in MRP-191. The FMECA determined that these CASS components screened in as susceptible to TE, but this did not change the generic MRP-191 classification of low susceptibility for any of these items. The licensee stated that even with the addition of TE, the MRP-227-A guidelines remain appropriate for these plant-specific CASS components. The licensee explained that these CASS components receive neutron irradiation below the IE screening thresholds defined in MRP-191. The licensee concluded that no changes are required for the RVI AMP concerning

these components. The licensee provided additional details regarding this determination in Enclosure 2 of its response to Follow-Up RAI-2, PWROG-15066-NP, "Responses to Follow-Up NRC RAI on the Donald C. Cook Nuclear Plant Units 1 and 2 Reactor Internals Aging Management Program."

The NRC staff reviewed the licensee's response to Follow-Up RAI-2(a) and confirmed that these plant-specific CASS components would not be susceptible to IE under any circumstances because their estimated neutron fluence range of less than 10^{20} n/cm^2 is far below the corresponding IE screening thresholds in MRP-191. Therefore, the licensee's use of the TE screening criteria in the C. Grimes letter (Ref. 24) are valid for these CASS components. Accordingly, the staff determined that the CNP Unit 2 Upper Support Ring/Skirt would not be susceptible to TE or IE because these Type CF8 CASS parts have low Molybdenum content, and ferrite content less than 20 percent, based on CMTR data. They also have a low neutron fluence of less than 10^{20} n/cm^2. For the rest of the plant-specific CASS components addressed in Follow-Up RAI-2(a), the staff confirmed that as plant-specific CASS, they screen as susceptible only to TE. The MRP-191 generic evaluation for these components as non-CASS/wrought material determined that these were not susceptible to any of the eight aging mechanisms. Therefore, based on the FMECA methodology of MRP-191, the staff confirmed that the FMECA classification for these plant-specific CASS components would be FMECA Group 1 (low significance) rather than the generically-determined FMECA Group 0 (no significance based on no susceptibility to any aging mechanism). Both FMECA Group 1 and FMECA Group 0 components are generally dispositioned as requiring No Additional Measures, based on the final categorization and ranking process used for developing the MRP-227-A inspection criteria. Therefore, the staff determined that the licensee's conclusion that no changes are required for the RVI AMP concerning these components is valid.

3.4.7.2.1 CASS Components - CRGT Guide Cards

In its response to Follow-Up RAI-2(b) (Ref. 5), dated August 6, 2015, the licensee stated that a plant-specific FMECA was performed for potentially CASS guide cards at CNP Unit 1 using guidance in MRP-191. The FMECA determined that CASS guide cards screened in as susceptible to TE. The licensee stated that even with the addition of TE as an aging mechanism for CASS guide cards, the MRP-227-A guidelines remain appropriate for these plant-specific CASS components. The licensee indicated that the guide cards receive neutron irradiation below the IE screening thresholds defined in both MRP-191 and the NRC staff's June 2014 revised screening criteria. The licensee stated that cracking is not a greater concern for guide cards fabricated from CASS than for guide cards fabricated from Type 304 SS based on the following justifications:

- Stress, function, and geometry of the part remain the same regardless of material;
- Guide card welds are similar for Type CF8 CASS and Type 304 SS base metal;
- TE does not result in a complete loss of fracture toughness;
- The current MRP-227-A VT-3 visual examination method would detect gross failures;
- Guide card redundancy requires multiple card failures to prevent control rod insertion; and
- Periodic monitoring of control rod functionality is performed per plant procedures.

The licensee concluded that no changes are required for the RVI AMP concerning the guide cards at CNP Unit 1.

The NRC staff reviewed the license's response to Follow-Up RAI-2(b) and confirmed that CASS guide cards would not be susceptible to IE under any circumstances because their estimated neutron fluence range of less than 10^{20} n/cm^2 is far below the corresponding IE screening thresholds. Therefore, the guide cards only need to be considered susceptible to TE, based on the criteria of the C. Grimes letter. Also, the staff verified that the guide card welds are already analyzed as susceptible to SCC in MRP-191 FMECA, and the welds are not any more susceptible to SCC if they adjoin CASS base metal. The susceptibility of CASS guide card base metal to formation of new cracks by SCC is not a concern, and CASS guide card base metal is no more susceptible to formation of new fatigue cracks than wrought SS base metal. Therefore, the staff determined that the initiation of new cracks for CASS guide cards is generally not more of a concern than for wrought guide cards. However, the staff noted that if pre-existing flaws are present, thermally embrittled CASS guide card material would be more susceptible to flaw propagation and fracture due to its lower fracture toughness. Therefore, the staff determined that further justification was needed regarding the adequacy of performing VT-3 visual examinations of CASS guide cards, taking into consideration the lower fracture toughness and corresponding lower flaw tolerance of the thermally embrittled material. Therefore, in a public meeting held on September 23, 2015, and documented in a public meeting summary (Ref. 25), dated October 5, 2015, the staff requested that the licensee provide supplemental information regarding the following four statements made in response to Follow-Up RAI-2(b):

1. TE does not result in a complete loss of fracture toughness for CASS guide cards;
2. The current MRP-227-A VT-3 visual examination method would detect gross failures;
3. Guide card redundancy requires multiple card failures to prevent control rod insertion; and
4. Periodic monitoring of control rod functionality is performed per plant procedures.

In its supplemental response to Follow-Up RAI-2(b) (Ref. 6), dated October 30, 2015, the licensee provided the requested supplemental information concerning these four items. The supplemental response and the staff's evaluation of the response for each item are addressed below.

<u>Item 1 - Reduction in Fracture Toughness due to TE</u>
The licensee stated that a meeting was held between the PWROG and NRC on September 16, 2015, to discuss a statistical approach to resolving fracture toughness concerns with CASS components with unknown ferrite content. The licensee stated that the results of the PWROG work indicate that RVI materials fabricated from Type CF8 CASS have TE saturation fracture toughness greater than the 255 kilojoules per square meter (kJ/m^2) fracture toughness screening criterion provided in the C. Grimes Letter, based on industry-wide data. The licensee also provided further information in a supplement to its PWROG report for addressing Follow-Up RAI-2(b) on the guide cards – PWROG-15066-NP, Rev. 1, is provided in Enclosure 2 to the October 30, 2015, supplemental response (Ref. 6) to Follow-Up RAI-2(b). The supplemental information in PWROG-15066-NP, Rev. 1 states that industry-wide fracture toughness data shows that the minimum TE saturation fracture toughness for PWR RVI components manufactured from Type CF8 CASS is 100 kJ/m^2 above the 255 kJ/m^2 fracture toughness screening criterion provided in the C. Grimes letter. The licensee proposed to use

this industry-wide fracture toughness data in lieu of plant-specific information as a statistical basis for identification of the TE saturation fracture toughness for the potentially CASS guide cards at CNP Unit 1. Invoking this data, the licensee indicated that the CNP Unit 1 guide cards would not require additional aging management due to TE.

NRC Staff Evaluation of Item 1

The NRC staff reviewed the statements provided by the licensee concerning the statistical evidence for adequate TE saturation fracture toughness of Type CF8 CASS. The staff determined that, within the context of this plant-specific AMP review, it cannot render a technical finding on the acceptability of this industry-wide data as a statistical basis for establishing a generic saturated fracture toughness for Type CF8 CASS material. However, the information provided in the licensee's supplemental RAI response concerning the minimum TE saturation fracture toughness for Type CF8 CASS RVI components is consistent with the PWROG's generic evaluation of the industry-wide fracture toughness data for Type CF8 CASS. Specifically, the PWROG report, PWROG-15032-NP, Rev. 0 "PA-MSC-1288, Statistical Assessment of PWR RV Internals CASS Materials," November 2015, demonstrates that the minimum TE saturation fracture toughness for manufacturers of Type CF8 CASS PWR RVI components is 364 kJ/m^2, with a "95/95" tolerance limit. This is more than 100 kJ/m^2 above the 255 kJ/m^2 fracture toughness screening criterion provided in the C. Grimes letter. The staff noted that the "95/95" tolerance limit, which is defined as 95 percent confidence that at least 95 percent of the population is bounded by the tolerance limit, per NUREG-1475, Rev. 1 (Ref. 26), was applied to the saturation fracture toughness data for each manufacturer of Type CF8 CASS RVI components. The lowest of these manufacturer-based distributions corresponds to the minimum value of 364 kJ/m^2. Therefore, the evidence cited by the licensee for adequate TE saturation fracture toughness of Type CF8 CASS is sufficient to provide a partial basis to support the licensee's overall assertion that the MRP-227-A VT-3 visual examinations would provide adequate assurance of guide card functionality during the PEO. This fracture toughness basis is being considered in connection with the staff's review of guide card aging management for CNP Unit 1, and specifically in connection with the licensee's supplemental information concerning the other three statements below. Therefore, the staff determined that the licensee's supplemental response on this item is acceptable.

Item 2 - MRP-227-A VT-3 Visual Examination Detection Capabilities

The licensee stated that in addition to the MRP-227-A guidelines, CNP will also follow the interim guidance for guide card inspections provided in EPRI letter MRP 2014-006. The interim guidance directs utilities to follow proprietary guidance provided in WCAP-17451-P, "Reactor Internals Guide Tube Wear Westinghouse Domestic Fleet Operational Projections," Rev. 1. The licensee stated that I&M plans to inspect, at a minimum, all active CRGTs as recommended in WCAP-17451-P, which exceeds the existing MRP-227-A requirements for examining 20 percent of the number of CRGT assemblies, with all cards in each assembly examined. The licensee noted that actual inspection coverage may vary depending upon accessibility conditions. The licensee stated this inspection strategy is adequate to detect cracking prior to loss of guide card function, and, while the inspection will be primarily used to evaluate wear, it will also detect damage such as deformed card sections and missing card ligaments. The licensee emphasized that VT-3 inspection is not intended to detect fine cracking; however the components remain functional until cracking becomes extensive enough that it is detectable by VT-3 inspection.

NRC Staff Evaluation of Item 2

The NRC staff reviewed the interim guidance provided in EPRI letter MRP 2014-006 and WCAP-17451-P, and confirmed that it provides for more comprehensive inspection coverage for the CRGT guide cards, compared to the inspection coverage required by MRP-227-A. Since this interim guidance has not been reviewed by the NRC staff for implementation by licensees, the staff considers the licensee's implementation of this guidance at CNP Unit 1 to be voluntary. However, the staff confirmed the licensee's statement that, although VT-3 examinations are not intended to detect fine cracking, these examinations would be capable of detecting missing guide card ligaments. VT-3 examination within the CRGT assembly would most likely be capable of detecting significant cracking prior to loss of guide card function. Very fine cracking, which is undetectable by VT-3, is often associated with crack initiation due to SCC or fatigue, for which CASS guide cards are no more susceptible than wrought guide cards generically analyzed in MRP-191. Therefore, the staff determined that VT-3 examinations of CASS guide cards would provide the detection capabilities required to identify the type of significant cracking and fracture associated with more brittle CASS material. This determination, in combination with the evidence that adequate fracture toughness may exist for Type CF8 CASS that is saturated with respect to TE, provides adequate assurance that VT-3 examinations would provide the aging management necessary for ensuring guide card functionality. Therefore, the staff determined that the licensee's supplemental response on this item is acceptable.

Item 3 - CRGT Guide Card Redundancy

The licensee stated that a discussion of the redundancy of the CRGT guide cards can be found in WCAP-17451-P, which demonstrates that there are a number of allowable consecutive worn-through guide cards for a CRGT assembly to remain functional. The licensee also stated that shutdown margin is maintained as described in the TS Section 1.1 definition of "Shutdown Margin." The TS definition of shutdown margin assumes that all rod control cluster assemblies (RCCAs) are fully inserted, except for the single RCCA of highest reactivity worth, which is assumed to remain fully withdrawn.

NRC Staff Evaluation of Item 3

The NRC staff reviewed the information in WCAP-17451-P regarding the guide card redundancy. This information provides evidence that the control rod guidance function of the CRGT assembly is maintained, even if some of the guide cards become non-functional. As with the generic implications for the industry-wide TE saturation fracture toughness data addressed above, the staff cannot render a definitive finding regarding the degree of guide card redundancy. However, this information is sufficient to support the licensee's plant-specific argument that there is, in fact, some guide card redundancy within the CRGT assemblies at CNP Unit 1. Taking into consideration the evidence for guide card redundancy at CNP Unit 1, the staff noted that the MRP-227-A inspection guidelines specify VT-3 visual examinations for the detection of cracking in other Westinghouse Primary components that are redundant, such as the baffle-edge bolts and the thermal shield flexures. Therefore, the evidence for guide card redundancy is relevant to support the licensee's assertion that the MRP-227-A VT-3 visual examinations would provide adequate assurance of CASS guide card functionality during the PEO. The staff also verified the licensee's statement that the necessary reactor shutdown margin is maintained per TS requirements even if any single RCCA fails to insert and remains fully withdrawn. Therefore, the staff determined that the licensee's supplemental response provides sufficient evidence of the guide card redundancy necessary to justify the adequacy of VT-3 visual examination for guide card aging management. Accordingly, the NRC staff determined that the licensee's supplemental response on this item is acceptable.

Item 4 - Requirements for Monitoring Control Rod Functionality
The licensee stated that control rod functionality and operability are monitored through testing, which provides evidence of the absence of damage impacting guide card functionality. The licensee identified the following tests:

- Control rod drag testing is performed every time an RCCA is latched to a drive shaft, following each refueling outage. The licensee stated that drag testing demonstrates that predetermined resistance load acceptance criteria are met.

- Control rod drop testing is performed, at a minimum, each time the RPV closure head is removed, prior to reactor criticality, in accordance with TS SR 3.1.4.3. The licensee stated that drop testing demonstrates control rod drop times are within predetermined acceptance criteria following maintenance and/or refueling.

- Control rod operability surveillance is performed quarterly in accordance with TS SR 3.1.4.2. The licensee stated that the operability testing moves each rod which is not fully inserted by a minimum number of steps in either direction to validate that behavior is within predetermined acceptance criteria.

NRC Staff Evaluation of Item 4
The NRC staff confirmed that control rod drop testing and quarterly control rod operability testing are performed in accordance with the TS requirements cited above. The staff verified that this testing provides adequate assurance of control rod operability, which is sufficient to indicate that significant guide card failures impacting control rod functionality have not occurred. Therefore, the staff determined that the licensee's supplemental response on this item is acceptable.

NRC Staff Conclusion for CRGT Guide Cards
The NRC staff noted that the concerns documented above regarding the potential susceptibility of CASS guide card material to fracture are also relevant to the potential generation of loose parts in the CRGT assembly from the fracture and separation of guide card ligaments. The staff noted that the generation of loose parts from fractured guide cards could be a potential concern for some embrittled CASS materials. However, the staff determined that, for the CNP Unit 1 guide cards, the industry-wide data from the PWROG study on the TE saturation fracture toughness for Type CF8 CASS does provide sufficient evidence of adequate fracture toughness for the staff to determine that fracture and separation of guide card pieces due to propagation of undetected preexisting flaws is not likely to occur during the PEO. In addition, crack initiation is not more likely if the CNP Unit 1 guide cards are Type CF8 CASS rather than Type 304 wrought SS. Furthermore, if the fracture and separation of guide card pieces were to create loose parts that blocked control rod insertion, this would be evident based on the control rod operability testing identified for Item No. 4. The MRP-227-A VT-3 examinations alone would be sufficient to detect missing guide card pieces, thereby ascertaining the presence of loose parts within the CRGT. Therefore, the staff determined that the potential for the generation of loose parts as the result of guide card fracture is not a sufficiently viable concern to warrant a change to the MRP-227-A inspection criteria for the guide cards at CNP Unit 1.

Based on its review of the supplemental information provided by the licensee concerning the implementation of MRP-227-A VT-3 visual examinations for aging management of potentially

CASS guide cards, the NRC staff determined that these inspection criteria will provide adequate assurance of guide card functionality during the PEO.

3.4.7.2.2 CASS Components - LSC Castings

In its response to Follow-Up RAI-2(c) (Ref. 5), dated August 6, 2015, the licensee stated that CMTRs for CNP Unit 1 LSC castings were not located; therefore, they are conservatively assumed to be potentially susceptible to TE. The licensee determined that ferrite content for the CNP Unit 2 LSC castings is less than 15 percent based on CMTR data; therefore, they are not susceptible to TE using either the screening criteria in the C. Grimes letter or the NRC proposed screening criteria dated June 2014. LSC castings at both units screen as potentially susceptible to IE. The licensee stated that I&M is participating in the PWROG project to develop a methodology which will form the basis for a generic or plant-specific functionality analysis of the lower support structure. The licensee stated that this project has a projected completion date of 2017. In Enclosure 3 of its response to Follow-Up RAI-2, the licensee provided a regulatory commitment related to CNP LSC casting functionality, with a completion date of August 31, 2017. The commitment states:

> I&M will continue to participate in the Pressurized Water Reactor Owners Group (PWROG) project for lower support columns. I&M will provide a supplemental response to the NRC on this request for additional information when an acceptable methodology is developed by the PWROG project.

The NRC staff reviewed the licensee's response to Follow-Up RAI-2(c) and the associated commitment related to LSC casting functionality. Based on its review, the staff determined that the CNP Unit 1 LSC castings require an analysis to demonstrate that the castings will remain functional during the PEO taking into consideration loss of fracture toughness due to the effects of TE and IE. For the CNP Unit 2 LSC castings, the staff confirmed that these castings would not be susceptible to TE during the PEO because the licensee identified that ferrite content is less than 15 percent based on CMTR data. However, the functionality of the CNP Unit 2 LSC castings would still need to be evaluated for the PEO taking into consideration their potential for loss of fracture toughness due to IE alone, per the requirements of Action Item 7.

<u>NRC Staff Conclusion for LSC Castings</u>
The NRC staff determined that the licensee's commitment to provide, by August 31, 2017, the plant-specific demonstration of LSC casting functionality for CNP, taking into consideration the loss of fracture toughness due to the effects of TE and/or IE, as applicable to each unit, is acceptable based on two indications that the licensee will be able to fulfill this commitment. Firstly, it is established that the licensee is an active participant in the PWROG project for demonstrating LSC casting functionality for the PEO, taking into consideration potential aging degradation of LSC castings due to TE and IE. Secondly, the NRC staff has been actively engaged with the PWROG on the issue of LSC casting functionality. Notably, the PWROG developed a proprietary technical report, PWROG-14048-P, "Functionality Analysis: Lower Support Columns," Rev. 0 that addresses the generic methodology for evaluating the functionality of LSC castings. This report was submitted to the NRC for information only, therefore the staff did not perform a formal review of the methodology. However, the staff did perform a summary assessment (Ref. 27) of this PWROG generic methodology, wherein the staff determined that the PWROG generic methodology may be used for guidance in performing

plant-specific functionality evaluations of LSC castings, using plant-specific parameters and conditions as inputs into the analytical evaluations.

The NRC staff finds that reasonable control for the implementation of the above regulatory commitment is best provided through the licensee's administrative processes, including its AMP as described in the UFSAR and its commitment management program. The above regulatory commitment does not warrant the creation of a regulatory requirement, and is not relied upon for the approval of the CNP RVI AMP.

3.4.7.3 NRC Staff Conclusion for Action Item 7

The NRC staff determined that the licensee has adequately evaluated all plant-specific CASS RVI components that were not generically analyzed as such in the MRP-191 FMECA, taking into consideration the components' susceptibility to TE and IE. Therefore, the staff determined that the licensee has satisfied the criteria of Action Item 7 for CNP Units 1 and 2.

3.4.8 Action Item 8 – Submittal of RVI AMP Information for NRC Staff Review and Approval

This action item requires licensees to submit a plant-specific RVI AMP for NRC review and approval to credit their implementation of MRP-227-A guidelines for aging management of the RVI components at their facility. This submittal shall include the information identified in Section 3.5.1 of the MRP-227-A SE. Section 3.5.1 of the MRP-227-A SE specifies that a licensee's RVI AMP submittal to implement MRP-227-A must include the following items:

1. An AMP that addresses the ten program elements as defined in Rev. 2 of the GALL Report AMP XI.M16A (Ref. 13).

2. An RVI AMP inspection plan that addresses the plant-specific action items. If a license renewal applicant or licensee plans to implement an AMP that deviates from the MRP-227-A guidelines, the license renewal applicant or licensee shall identify where their program deviates from MRP-227-A. The license renewal applicant or licensee shall provide a justification for any deviation, including a consideration of how the deviation affects both "Primary" and "Expansion" components.

Applicants that submit applications for license renewal after the issuance of the MRP-227-A SE are required to submit additional information items. The NRC staff notes that since the CNP license renewal application (Ref. 23) was submitted prior to the issuance of the MRP-227-A SE, the licensee is only required to submit the above two information items.

3.4.8.1 Licensee Evaluation of Action Item 8

Section 4.4.2.8 of the CNP RVI AMP submittal addresses the two plant-specific RVI AMP information requirements identified in Section 3.5.1 of the MRP-227-A SE. For AMP information item 1, the licensee stated that the CNP RVI AMP addresses the ten program elements as defined in Rev. 2 of the GALL Report AMP XI.M16A (Ref. 13). The ten program elements defined in GALL Report AMP XI.M16A are specifically addressed in Section 5 of its RVI AMP submittal. For AMP information item 2, the licensee stated that the CNP RVI AMP addresses the plant-specific action items, and the AMP does not deviate from MRP-227-A guidelines.

3.4.8.2 NRC Staff Evaluation of Action Item 8

The NRC staff determined that the licensee provided the information required by item 1 of Action Item 8 because it provided an AMP that addressed the ten elements of the revised version of GALL Report AMP XI.M16A, as provided in LR-ISG-2011-04 (Ref. 14). The staff's review determined that the ten elements of the CNP RVI AMP are consistent with the ten elements described in LR-ISG-2011-04, as documented in Section 3.1 of this staff assessment. The staff determined that the licensee provided the information required by item 2 of Action Item 8 because the licensee has satisfied all plant-specific action items. The staff also determined that the CNP RVI AMP does not deviate from the MRP-227-A guidelines; therefore there are no plant-specific AMP deviations, relative to MRP-227-A, that need be evaluated for CNP. Thus, the staff determined that the licensee has satisfied the criteria of Action Item 8 for CNP.

4.0 CONCLUSION

The NRC staff has reviewed the CNP RVI AMP, and concludes that it is acceptable because it is consistent with the MRP-227-A inspection and evaluation guidelines for RVI components. The licensee has adequately addressed all eight action items established in Section 4.2 of the MRP-227-A SE.

The NRC staff's approval of the CNP RVI AMP does not reduce, alter, or otherwise affect the current ASME Code, Section XI ISI requirements, or any CNP licensing basis requirements related to ISI of structures, systems, and components. The staff notes that Section 7, "Implementation Requirements," of MRP-227-A, requires that the NRC be notified of any deviations from the "needed" requirements.

5.0 REFERENCES

1. Letter from J.P. Gebbie, I&M to NRC, "Transmittal of Reactor Vessel Internals Aging Management Program," October 1, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12284A320).

2. Letter from J.P. Gebbie, I&M to NRC, "First Response to Request for Additional Information Concerning the Reactor Vessel Internals Aging Management Program," July 30, 2014 (ADAMS Accession No. ML14216A497).

3. Letter from J.P. Gebbie, I&M to NRC, "Second Response to Request for Additional Information Concerning the Reactor Vessel Internals Aging Management Program," September 4, 2014 (ADAMS Accession No. ML14253A316).

4. Letter from J.P. Gebbie, I&M to NRC, "Final Response to Request for Additional Information Concerning the Reactor Vessel Internals Aging Management Program," October 22, 2014 (ADAMS Accession No. ML14316A449).

5. Letter from J.P. Gebbie, I&M to NRC, "Response to Follow-Up Request for Additional Information Concerning the Reactor Vessel Internals Aging Management Program," August 6, 2015 (ADAMS Accession Nos. ML15223A435 and ML15223A436).

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The NRC staff's assessment of the CNP RVI AMP is enclosed. If you have any questions concerning this matter, please contact the Project Manager, Allison Dietrich, at (301) 415-2846, or via e-mail at Allison.Dietrich@nrc.gov.

Sincerely,

/RA/

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

Docket Nos. 50-315 and 50-316

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