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

May 29, 2015

Mr. David A. Heacock
President and Chief Nuclear Officer
Dominion Nuclear Connecticut, Inc.
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 2 – REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM AND INSPECTION PLAN FOR LICENSE RENEWAL COMMITMENT NO. 13. (TAC NO. MF3402)

Dear Mr. Heacock:

By letter dated July 31, 2013, as supplemented by letters dated July 21, 2014, December 19, 2014, and January 26, 2015, Dominion Nuclear Connecticut, Inc. (the licensee), submitted a document entitled, "Aging Management Program Description: Inservice Inspection – Reactor Vessel Internals," in support of License Renewal (LR) Commitment No. 13 for Millstone Power Station, Unit 2. Specifically, the licensee submitted an updated Reactor Vessel Internals (RVI) Aging Management Program (AMP) and RVI Inspection Plan in accordance with topical report, "Material Reliability Program: Pressurized Water Reactor Inspection and Evaluation Guidelines (MRP-227-A).

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's RVI AMP and RVI Inspection Plan. Based on that review, the NRC staff concludes that the licensee's RVI AMP and RVI Inspection Plan are acceptable and the licensee has fulfilled LR Commitment No. 13.

If you have any questions, please contact me at (301) 415-1030 or Richard.Guzman@nrc.gov.

Sincerely,

RGuzman

Richard V. Guzman, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-336

Enclosure: Safety Evaluation

cc w/encl: Distribution via Listserv

UNITED STATES
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# SAFETY ASSESSMENT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# INFORMATION IN SUPPORT OF LICENSE RENEWAL COMMITMENT No. 13

# PROGRAM DESCRIPTION FOR REACTOR VESSEL INTERNALS INSPECTION

# DOMINION NUCLEAR CONNECTICUT, INC.

# MILLSTONE POWER STATION, UNIT 2

# DOCKET NO. 50-336

## 1.0 INTRODUCTION

By letter dated July 31, 2013 (Reference 1), as supplemented by letters dated July 21, 2014 (Reference 2), December 19, 2014 (Reference 3), and January 26, 2015 (Reference 4), Dominion Nuclear Connecticut (the licensee) submitted a document entitled "Aging Management Program Description: Inservice Inspection - Reactor Vessel Internals" (RVI Program description) in support of fulfillment of License Renewal (LR) Commitment No. 13 for Millstone Power Station, Unit 2 (MPS2). The submittal contains an updated RVI Aging Management Program (AMP) and RVI Inspection Plan in accordance with topical report "Material Reliability Program [MRP]: Pressurized Water Reactor Inspection and Evaluation Guidelines (MRP-227-A) (Reference 5).

## 2.0 REGULATORY REQUIREMENTS

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

Structures and components subject to an AMP shall encompass those structures and components that (1) perform an intended function, as described in 10 CFR 54.4, without moving parts or without a change in configuration or properties, and (2) are not subject to replacement based on a qualified life or specified time period. These structures and components are referred to as "passive" and "long-lived" structures and components, respectively. The scope of components considered for inspection under MRP-227-A includes core support structures (typically denoted as Examination Category B-N-3 by the

American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI) and those RVI components that serve an intended LR safety function pursuant to criteria in 10 CFR 54.4(a)(1). The scope of the program does not include non-long-lived components such as fuel assemblies, reactivity control assemblies, or active components such as nuclear instrumentation because these components are not typically within the scope of the components that are required to be subject to an aging management review, as defined by the criteria set in 10 CFR 54.21(a)(1).

On January 12, 2009, Electric Power Research Institute (EPRI) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review and approval the MRP Report 1016596 (MRP-227), Revision 0, "PWR Internals Inspection and Evaluation Guidelines" (Reference 6), which was intended as guidance for the use of applicants in developing their plant-specific AMP for RVI components.

Subsequent to the submittal of MRP-227 and prior to the issuance of the safety evaluation (SE) on MRP-227, NUREG-1801, "Generic Aging Lessons Learned Report," Revision 2 (the generic aging lessons learned (GALL) Report, Rev. 2), was issued providing new AMR line items and aging management guidance in AMP XI.M16A, "PWR Vessel Internals." This GALL AMP was based on NRC staff expectations for the guidance to be provided in modification/rework package (MRP)-227-A.

Revision 1 to the final SE regarding MRP-227, Revision 0, was issued on December 16, 2011 (Reference 7), with seven conditions and eight applicant/licensee action items (A/LAIs). The topical report conditions were specified to ensure that certain information was revised generically in the final NRC-approved version of MRP-227 (MPR-227-A) and the A/LAIs were specified for licensees to address plant-specific issues which could not be resolved generically in Revision 1 of the final SE on MRP-227-A. On January 9, 2012, EPRI published the NRC approved version of the topical report, "Materials Reliability Program: PWR [Pressurized Water Reactor] Internals Inspection and Evaluation Guidelines" (MRP-227-A), (Reference 5). In MRP-227-A, a discussion of the technical basis for the development of plant-specific AMPs for RVI components in PWR vessels was provided as well as the inspection and evaluation guidelines for PWR applicants to use in their plant-specific AMPs. MRP-227-A provides the basis for renewed license holders to develop plant-specific inspection plans to manage aging effects on RVI components, as described by their Final Safety Analysis Report (FSAR) commitment.

Since the GALL Report, Rev. 2 (Reference 8), was published prior to the issuance of the final SE of MRP-227-A, the NRC staff published License Renewal Interim NRC staff Guidance LR-ISG-2011-04: "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors" (Reference 9), which modifies the guidance of AMP XI.M16A to be consistent with MRP-227-A.

In the MPS2 LRA, Table 3.1.2-2, the "Inservice Inspection: Reactor Vessel Internals," AMP was credited with managing various aging effects for RVI components. The licensee made LR Commitment No. 13, documented in Appendix A of NUREG-1838, "Safety Evaluation Report Related to the License Renewal of the Millstone Power Station, Units 2 and 3" (Reference 10), which states that:

Millstone will follow the industry efforts on RVI regarding such issues as thermal or neutron irradiation embrittlement (loss of fracture toughness), void swelling (change in dimensions), stress corrosion cracking (PWSCC [primary water stress corrosion cracking] and IASCC [irradiation-assisted stress corrosion cracking]) and loss of pre-load for baffle and former assembly bolts and will implement the appropriate recommendations resulting from this guidance. The revised program description, including a description of the 10 elements of the NUREG-1801 Program, will be submitted to the NRC for approval.

The implementation schedule for this commitment is at least two years prior to period of extended operation. The period of extended operation (PEO) for MPS2 begins on July 31, 2015. Therefore, the licensee met the required schedule for submission of its revised RVI program description by submitting its RVI program description on July 31, 2013.

Overview of the MRP-227-A Process

As the initial step in the process for developing the inspection recommendations of MRP-227-A, components were screened for eight different aging mechanisms: stress corrosion cracking (SCC), irradiation-assisted stress corrosion cracking (IASCC), wear, fatigue, thermal aging embrittlement (TE), irradiation embrittlement (IE), irradiation-enhanced stress relaxation and creep, and void swelling. Screening inputs included chemical composition (material grade), neutron fluence, temperature history, and representative stress levels. Components determined to be below the screening criteria for all aging mechanisms were designated category “A” while those exceeding the criteria for at least one mechanism were designated “non A.” For the “non A” components, Failure Modes, Effects, and Criticality Analyses (FMECA) were then performed to categorize each component as category “A”, “B”, or “C”, with “A” being the least affected and “C” being the most affected. The components determined to be “A” in the initial screening were also reviewed by the FMECA expert panel to confirm their Category “A” status. Category “B” and “C” components were determined to need further evaluation and were subject to a functionality assessment using irradiated and aged material properties to determine the effects of the degradation mechanisms on functionality. As a result of the functionality assessment, each RVI component was assigned to one of four functional groups:

- Primary: Those PWR internals that are highly susceptible to the effects of at least one of the eight aging mechanisms were placed in the “Primary” group. The Primary group also includes components which have shown a degree of tolerance to a specific aging degradation effect, but for which no highly susceptible component exists or for which no highly susceptible component is accessible. MRP-227-A specifies the scope, methods, coverage and schedule of inspections of Primary components. Initial inspection of most Primary components is required within two refueling outages of the start of the PEO. For a few components, actions other than inspections are specified for aging management, such as analysis.

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

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

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

Aging management strategy development combined the results of functionality assessment with component accessibility, operating experience (OE), existing evaluations, and prior examination results to determine the appropriate aging management methodology, baseline examination timing, and the need for and the timing of subsequent inspections.

Augmented inspections recommendations are identified for each Primary and Expansion category component. The recommendations for the Primary components also identify timelines for the inspection. The inspection strategy generally employs visual testing (VT)-3 level visual examinations to evaluate general component condition, enhanced visual examination (EVT-1) level visual examinations to identify surface breaking flaws, and VT-1 level visual examination to identify surface discontinuities such as gaps. Cracking in baffle-former bolts and core shroud bolts is monitored with ultrasonic techniques (UT).

## 3.0 TECHNICAL EVALUATION

The NRC staff reviewed the RVI Program description to determine if it demonstrated that the effects of aging on the subject RVI components covered by the report would be adequately managed so that the components' intended functions would be maintained consistent with the CLB for the PEO in accordance with 10 CFR 54.21(a)(3). The final SE for MRP-227, Revision 0, concluded that the MRP-227, Revision 0, report, as modified by the conditions and limitation and A/LAIs of the SE, provides for the development of an acceptable AMP for PWR RVI components. Therefore, the NRC staff's technical evaluation of the RVI Program description (documented in this Safety Assessment (SA)) focused on program consistency with the recommendations of MRP-227-A (SA Section 3.1), consistency of the AMP elements with LR-ISG-2011-04 (SA Section 3.2), and determining whether the program adequately addresses the plant-specific A/LAIs (SA Section 3.3).

### 3.1 MPS2 RVI Program Implementation

This subsection of the SA focuses on the consistency of the implementation of the MPS2 RVI Program with the MRP-227-A topical report, including the inspections to be performed (scope, schedule, methods, and acceptance criteria), TLAAs, and handling of operating experience.

### 3.1.1 Consistency of the RVI Program Inspections with MRP-227-A

In Section 2.3 of the RVI Program description, the licensee stated that the ASME Code, Section XI, Examination Category B-N-3 governs inspections of RVI components unless MRP-227-A imposes more specific requirements. The licensee referred to Table 1, 2, and 3 of the RVI Program description which provide the inspection requirements for the MRP-227-A "Primary," "Expansion," and "Existing Programs" inspection categories. These tables include the extent (scope) of examination, examination methods, and schedule. The licensee noted that inspection of "Expansion" items is only required if invoked by the expansion criteria for the "Primary" items. The licensee noted there is one plant-specific augmented examination for the areas of the core barrel associated with the thermal shield support brackets.

The NRC staff reviewed the licensee's RVI Program description, particularly Tables 1, 2, and 3, which contain the information for the "Primary", "Expansion", and "Existing Programs" inspections for MPS2. The NRC staff also reviewed Table 4, which contains the acceptance criteria for the "Primary" and "Expansion" components and expansion criteria for the "Primary" components at MPS2, for consistency with the corresponding information in the following MRP-227-A tables: Table 4-2, "CE [Combustion Engineering] Plants Primary Components," Table 4-5, "CE Plants Expansion Components," Table 4-8, CE Plants existing Programs Components, and Table 5-2, "CE Plants Examination Acceptance and Expansion Criteria." The NRC staff notes that all ASME Section XI, Inservice Inspection requirements, continue to apply in addition to the MRP-227-A requirements unless a relief request is submitted and approved by the NRC staff in accordance with 10 CFR 50.55a.

In request for additional information (RAI) 9, the NRC staff requested that the licensee explain or correct the following inconsistencies between the MPS2 "Primary" and "Expansion" inspection category components identified in Table 2 and 3 of the RVI Program description, and Tables 4-2 and 4-5 of MRP-227-A:

1. Table 3 lists Core Shroud Assembly (Welded) - Remaining Axial Welds with applicability to plant designs with core shrouds assembled in two vertical sections. MRP-227-A Table 4-5 has Core Shroud Assembly (Welded) - Remaining axial welds, ribs and rings, but this component is only applicable to plant designs with core shrouds assembled with full-height shroud plates, and

2. Core Support Barrel Assembly - Core Barrel Assembly Axial Welds, "Expansion" component in Table 4-5 of MRP-227-A, is not included in Table 3 of the RVI Program Description.

In its response to RAI 9, Part 1, dated July 21, 2014 (Reference 2), the licensee stated the MPS2 AMP submittal Table 2 entry for the expansion examination of the Remaining Axial Welds is correct for welded core shrouds assembled in two vertical sections, which is the applicable design of MPS2. The response to RAI 9 further stated that the published version of MRP-227-A Table 4-5 had an error that inadvertently omitted the correct entry for designs like MPS2, and that EPRI has been notified of the publishing error. In its response to RAI 9, Part 2, dated July 21, 2014, the licensee stated that the MPS2 AMP submittal Table 3 entry for the expansion examination of the Core Support Barrel Assembly Axial Welds was inadvertently omitted from the submitted program for MPS2, but was included in internal plant documents. The licensee also provided the missing row of the table containing the information for this component. The NRC staff reviewed the information and confirmed it matches the corresponding information for the Core Support Barrel Assembly Axial Welds in MRP-227-A, Table 4-5. Since the licensee clarified the first discrepancy and corrected the second one, RAI 9 is resolved.

As amended by the response to RAI 9, the NRC staff found that Tables 1, 2, and 3 implement all the recommended inspections for "Primary", "Expansion", and "Existing Programs" components applicable to MPS2 (i.e., CE-design RVI with welded core shroud assembled in two vertical sections), and that these inspections are consistent with respect to schedule, frequency, examination method, and coverage, with Tables 4-2, "CE Plants Primary Components," 4-5, "CE Plants Expansion Components," and 4-8, "CE Plants existing Programs Components," of MRP-227-A. The NRC staff found the acceptance and expansion criteria listed in Table 4 of the RVI Program are consistent with the corresponding information of Table 5-2 of MRP-227-A, "CE Plants Examination Acceptance and Expansion Criteria."

3.1.2 Time-Limited Aging Analyses

The NRC staff reviewed the MPS2 LRA and the safety evaluation report (SER) (Reference 10) and found no TLAAs related to RVI. For two of the "Primary" inspection category components applicable to MPS2, MRP-227-A permits a demonstration of fatigue life via a TLAA instead of inspection. These components are the Core Support Barrel Assembly – Lower Flange Weld, which is referred to as the Core Support Barrel-Lower Support Structure Flexure Weld at MPS2 (flexure weld), and the Lower Support Structure – Core Support Plate (CSP). For each of these components, Note 6 to Table 2 of the RVI Program description states this item originally screened in for fatigue by MRP-227-A, but as permitted by MRP-227-A for this item a plant specific fatigue evaluation has been performed and demonstrated acceptable fatigue life. Therefore, the EVT-1 specified by MRP-227-A Table 4-2 is not required and this item is subject to the normal ASME Code, Section XI, Examination Category B-N-3 inspection requirements. Under "Ongoing Activities" in the RVI Program description, the licensee stated for the MPS2 CE design, evaluations were required by MRP-227-A to determine the actual inspection requirement for some components. The licensee further stated that these have been completed, with the result that the fatigue life of the lower core plate and the lower flange (flexure weld) have been demonstrated to be below the MRP-175 screening criteria of 0.1 for the projected 60 years of operation and, therefore, that no inspection for fatigue cracking of these locations is required, and a notation to this effect has been added to Table 1.

Details of these plant-specific fatigue evaluations were not provided in the RVI Program description. Further, the NRC staff was concerned that these evaluations may be new TLAAs not identified in the MPS2 LRA, and thus the MPS2 CLB may need to be modified to reflect these new TLAAs.

Therefore, in RAI 8, the NRC staff requested that the licensee: (1) provide the plant-specific fatigue evaluations for the RVI components for which fatigue evaluations are being credited in lieu of inspections, and (2) discuss the need to update the MPS2 FSAR to reflect the fatigue analyses for the three components.

In its response to Item No. 1 of RAI 8, dated July 21, 2014, the licensee indicated that a plant-specific evaluation for this MRP-227-A activity was not performed, but that the evaluation for

MPS2 was developed based on comparisons and scaling of analyses performed by the vendor for another plant (the reference plant). The RAI response described the evaluation based on the reference plant in which it was demonstrated that the stresses resulting from the various transients are bounded by the reference plant stresses. The same fatigue analysis methodology and stresses for the reference plant were then used with the MPS2-specific numbers of transient cycles for 60 years to determine the cumulative usage factor for the MPS2 components.

The licensee stated that the results of the evaluation identified that the analyzed components had a cumulative usage factor that did not exceed the screening value allowed by MRP-191, "Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design" (Reference 11), and that the MPS2 CSP and flexure welds screened out for fatigue cracking concerns under MRP-227-A. The licensee further stated that existing ASME Section XI program requirements remain applicable.

In response to Item No. 2 of RAI 8, the licensees stated that no FSAR description of the evaluation is required to be included in the FSAR because the evaluation is not required to demonstrate compliance with a MPS2 licensing basis requirement. In support of this, the licensee stated that the fatigue evaluation was performed as a permitted option in the implementation of MRP-227-A in lieu of inspection. The licensee further elaborated that because MRP-191 conservatively screened in certain components based on a lack of plant specific fatigue susceptibility information and operational history, MRP-227-A permitted the option to re-evaluate the fatigue screening of those components on a plant-specific basis. The licensee also stated that the MPS2 CSP and flexure weld are not newly identified components for aging management review, but were considered in the development of the LRA.

Upon further review, the NRC staff determined that the fatigue evaluation described in the RAI 8 response appears to meet all the criteria for a TLAA from 10 CFR 54.3, except (6) are contained or incorporated by reference in the CLB. Since a fatigue evaluation was not part of the original CLB when the LRA for MPS2 was submitted, the evaluation described in the RAI 8 response does not meet the definition of a TLAA, since analyses must meet all six criteria of 10 CFR 54.3 to be considered a TLAA. The NRC staff's concern of RAI 8, Item No. 2, is therefore resolved. However, the NRC staff considers these evaluations now to be part of the CLB and must be evaluated as TLAAs for any subsequent LRA.

The NRC staff was concerned that the licensee's response to Item No. 1 of RAI 8 did not provide enough information for the NRC staff to determine whether the licensee's fatigue evaluation met the criteria of MRP-227-A for a plant-specific evaluation. In addition, although the consideration of the effects of the reactor coolant environment on fatigue is not required by the design code of record for the MPS2 reactor coolant system, A/LAI 8, Item 5 requires TLAA analyses for fatigue to include the effects of the reactor coolant system water environment. The NRC staff notes that A/LAI 8, Item 5, is only required for current license renewal applicants that submit an RVI aging management program in accordance with MRP-227-A, not for licensees that issued renewed licenses prior to the publication of MRP-227-A. However, the NRC staff considers the consideration of environmental effects for RVI fatigue analyses to be a good practice.

Therefore, in follow-up RAI 2-2, the NRC staff requested additional details on the fatigue evaluation for the two components at MPS2, including: (1) a reference for the fatigue analysis methodology;( 2) a description of the method by which the effects of the reactor coolant system water environment were considered in the fatigue analysis, or a justification for not considering environmental effects; (3) The exact cumulative usage factor (CUF) calculated for the Core Support Barrel Assembly – Lower Flange Weld and the Lower Support Structure – Core Support Plate at MPS2.

In its response to RAI 2-2 dated January 26, 2015 (Reference 4), the licensee stated that the fatigue evaluations were performed in accordance with the ASME Code, Section III, 1971 Edition with the Winter 1973 Addendum, which incorporated draft Subsection NG. The licensee stated that the fatigue evaluation described in the response to RAI 8 was based on traditional ASME Code calculations and did not include environmental effects. The licensee further stated that, as described in MRP-175, Appendix D, the 0.1 screening value used in the fatigue evaluation includes appropriate consideration of potential environmental effects. The licensee provided the evaluated fatigue usage for 60 years of operation of 0.052 for the Lower Support Structure – Core Support Plate and 0.005 for the Core Support Barrel/Lower Support Structure Flexure Weld (equivalent to the Core Support Barrel Assembly – Lower Flange Weld).

The NRC staff finds the methodology used by the licensees for the fatigue life evaluation of the CSP and flexure weld to be acceptable because the licensee used established ASME Code techniques to evaluate fatigue, demonstrated the stresses were bounded by the reference plant, and used MPS2-specific numbers of transients. The NRC staff's concern with RAI 8, Item No. 1, and RAI 2-2, Item No. 1, are therefore resolved.

In order to determine if environmental effects could be significant for the RVI components subject to fatigue analyses, the NRC staff calculated the environmental correction factor ($F_{en}$) in accordance with Regulatory Guide 1.207, "Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors (Reference 12)," which references NUREG/CR-6909, Rev. 0, "Effect of Light Water Reactor (LWR) Coolant Environments on the Fatigue Life of Reactor Materials," Final Report (Reference 13). The NRC staff calculated a $F_{en}$ value of slightly less than 10 for both components, which would result in environmentally-corrected CUF's for these components of 0.5 for the core support plate and 0.05 for the flexure weld. Therefore, the NRC staff finds that although the licensee did not explicitly consider the effects of the environment in its fatigue evaluation, the fatigue CUF would remain acceptable if maximum environmental effects were considered. Since both CUF's would be far less than the ASME acceptance criteria of 1.0, when adjusted for environmental effects, the NRC staff concluded that fatigue cracking is not expected. The NRC staff thus finds the licensee's conclusion that no inspection of the CSP and flexure weld for fatigue is necessary, is acceptable. Therefore, the NRC staff's concern of RAI 2-2, Item No. 2, is resolved.

### 3.1.3 Operating Experience

In Section 3.10 of the RVI Program description, the licensee described MPS2-specific OE related to RVI. The MPS2-specific OE includes an event in 1983 requiring removal of the thermal shield due to an issue with flow-induced vibration. See Section 3.3.2 for details of the

NRC staff's evaluation of this issue. In 2009, flux thimble tubes were replaced due to irradiation-induced growth. The NRC staff's evaluation of this OE is located in Section 3.3.3.

The licensee also indicated it has reviewed the industry operating experience with RVI degradation compiled in Appendix A to MRP-227-A for applicability to MPS2. The licensee also stated it reviewed the latest revision of the report of industry OE published by EPRI, MRP-219, Rev. 8 (Reference 14), finding no OE relevant to MPS2. The licensee noted that industry OE related to baffle-former bolts is not applicable to MPS2 since it has a welded core shroud design. Finally, the licensee noted that it reviewed the recommendations of LR-ISG-2011-04 but that no changes to the MPS2 RVI Program were needed.

The NRC staff finds that the licensee's handling of OE is acceptable because it reviewed and dispositioned industry OE related to RVI with respect to its plant and took appropriate corrective actions with respect to OE involving component degradation of the MPS2 RVI, as detailed in SA Sections 3.3.2 and 3.3.3.

## 3.2 RVI Aging Management Program Evaluation

### Licensee Evaluation

In Section 3.0 of the RVI Program description, the licensee provided a description of the 10 AMP elements of the RVI Program. The licensee indicated the comparison was made to Section XI.M16A, PWR Vessel Internals, of the GALL Report, Rev. 2, rather than NUREG-1801, "Generic Aging Lessons Learner Report," Revision 0, that was referenced in the LRA for MPS2. In the "Operating Experience" section of the licensee's AMP description, the licensee stated that LR Interim NRC Staff Guidance LR-ISG-2011-04: "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors" (Reference 9), has been reviewed for relevant OE, and that there were no changes to this AMP required as a result of this review. In Section 4.0 of the RVI Program description, the licensee provided a summary of their comparison with the GALL Report, Rev. 2, Section XI.M16A, and identified no exceptions from the GALL Report, Rev. 2, and one enhancement. The enhancement is the augmented examination of the mitigated flaws from the previous thermal shield attachment bracket areas (see SA Section 3.3.2). The licensee identified this as an enhancement to the "Scope" and "Detection of Aging Effects" program elements.

### NRC Staff Evaluation

The NRC staff found that the licensee's description of the 10 elements of its RVI AMP is consistent with the criteria of LR-ISG-2011-04 (Reference 9), which represents the most current NRC guidance on aging management of RVI. Therefore, the NRC staff finds the 10 elements of the licensee's RVI AMP are acceptable.

## 3.3 Applicant/Licensee Action Items (A/LAIs) from SE of MRP-227, Revision 0

The NRC staff's final SE of MRP-227, Revision 0 (Reference 7), contained eight plant-specific A/LAIs. The NRC staff determined that A/LAIs 1, 2, 3, 5, 7, and 8 are applicable to MPS2 and A/LAIs 4 and 6 are not applicable to MPS2 because these A/LAIs are applicable only to Babcock & Wilcox (B&W)–design RVI.

3.3.1 A/LAI 1 – Applicability of FMECA and Functionality Analysis Assumptions (plant-specific applicability verification of MRP-227-A)

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

Licensee Evaluation

In its response to A/LAI 1, contained in Enclosure 1 to the RVI Inspection Plan, the licensee described its process and results for providing assurance that MPS2 is reasonably represented by the generic industry program (with regard to neutron fluence, temperature, stress values, and materials) used in the development of MRP-227-A, as follows:

1. Identification of typical CE-designed PWR RVI components in Table 4-5 of MRP-191.

2. Identification of MPS2 PWR components.

3. Comparison of the typical CE-designed PWR RVI components to the MPS2 RVI components:

    a. Confirmation that no additional items were identified by this comparison (primarily supports A/LAI 2).

    b. Confirmation that the materials from Table 4-5 of MRP-191 are consistent with MPS2 RVI component materials.

    c. Confirmation that the design and fabrication of MPS2 RVI components are the same as, or equivalent to, the typical CE-designed PWR RVI components.

4. Confirmation that the MPS2 operating history is consistent with the assumptions in MRP-227-A regarding core loading patterns.

5. Confirmation that the MPS2 RVI materials operated at temperatures within the original design basis parameters.

6. Determination of stress values based on design basis documents.

7. Confirmation that any changes to the MPS2 RVI components do not impact the application of the generic aging management strategy in MRP-227-A.

The licensee stated that MPS2 RVI components are represented by the design and operating history assumptions regarding neutron fluence, temperature, materials, and stress values in the MRP-191 generic FMECA and in the MRP-232 functionality analysis based on the following:

1. MPS2 operating history is consistent with the assumptions in MRP-227-A with regard to neutron fluence.

    a. The FMECA and functionality analysis for MRP-227-A are based on the assumption of 30 years of operation with high-leakage core loading patterns followed by 30 years of low-leakage core fuel management strategy. As stated in the AMP, MPS2 completed its transition to a low-leakage core design in fuel cycle 10 (November 4, 1990) at 14.9 years of operation. Therefore, MPS2 meets the fluence and fuel management assumptions in MRP-191 and meets the requirements for application of MRP-227-A.

    b. MPS2 has operated under base load conditions over the life of the plant, as stated in the MPS2 AMP. Therefore, MPS2 satisfies the assumptions in MRP-227-A regarding base load operation.

2. The MPS2 RVI components operate between $T_{hot}$ and $T_{cold}$, which are, approximately, not less than 515 °F for $T_{cold}$ and not higher than 604 °F for $T_{hot}$. The design temperature for the vessel is 650 °F. MPS2 operating history is within original design basis parameters; therefore, it is consistent with the assumptions used to develop the aging management strategy (MRP-227-A) with regard to temperature operational parameters.

3. MPS2 RVI components and materials are covered by the list of generic CE-designed PWR RVI components in Table 4-5 of MRP-191, as summarized in a letter to the licensee from Westinghouse.

    a. No additional components are identified for MPS2 by the comparison of the listing of Millstone Unit 2 RVI components listed in NUREG-1838 to the listing in Table 4-5 of MRP-191.

    b. MPS2 RVI component materials are consistent with, or nearly equivalent to, those materials identified in Table 4-5 of MRP-191 for CE designed plants. Where differences exist, there is no impact on the MPS2 RVI.

    c. Design and fabrication of MPS2 RVI components are the same as, or equivalent to, the typical CE-designed PWR RVI components.

4. As stated in the AMP, modifications to the MPS2 RVI components made over the lifetime of the plant are those specifically directed by the original equipment manufacturer (OEM). The design has been maintained over the lifetime of the plant as specified by the OEM. Operational parameters with regard to fluence and temperature are compliant with requirements in MRP-227-A. The components and materials are equivalent to those considered in MRP-191. Therefore, the MPS2 RVI components are represented by the assumptions in MRP-191, MRP-227-A, and MRP-232, confirming the applicability of the generic FMECA.

Based on the above, the licensee therefore concluded that MPS2 complies with A/LAI 1 of the SE on MRP-227, Revision 0, and that, therefore, MPS2 meets the requirement for application of MRP-227-A as a strategy for managing age-related material degradation in RVI components.

NRC Staff Evaluation

The NRC staff notes that Items 1-3 of the licensee's process, and Item 3 of the licensee's conclusions above, overlap with A/LAI 2, and will be evaluated by the NRC staff under its evaluation of A/LAI 2.

The information provided by the licensee confirmed that MPS2 switched to a low-leakage core loading pattern prior to 30 calendar years of operation, has always operated as a base-loaded unit, and has had no plant-unique modifications consistent with the three assumptions of the FMECA and functionality analyses supporting MRP-227-A listed in Section 2.4 of MRP-227-A. To resolve the generic issue of the information needed from licensees to resolve A/LAI 1, a series of proprietary and public meetings were conducted (References 15, 16, 17, 18, and 19), at which the NRC, Westinghouse, EPRI, and utility representatives discussed regulatory concerns and determined a path for a comprehensive and consistent utility response to demonstrate applicability of MRP-227-A, specifically for Westinghouse and CE-design PWR RVI. A summary of the proprietary meeting presentations and supporting proprietary generic design basis information is contained in Westinghouse proprietary report, WCAP-17780-P (Reference 20). WCAP-17780-P provides background proprietary design information regarding variances in stress, fluence, and temperature in the plants designed by Westinghouse and CE to support NRC reviews of utility submittals to demonstrate plant-specific applicability of MRP-227-A.

As a result of the technical discussions with the NRC staff, the basis for a plant to respond to the NRC's RAI to demonstrate compliance with MRP-227-A for originally licensed and uprated conditions was determined to be satisfied with plant-specific responses to the following two questions (Reference 17):

Question 1: Does the plant have non-weld or bolting austenitic stainless steel (SS) components with 20 percent cold work or greater, and if so, do the affected components have operating stresses greater than 30 ksi (If both conditions are true, additional components may need to be screened in for stress corrosion cracking (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/core design, non-representative for that plant (Reference 17 indicated this question covers power uprates as well as other core design and fuel management aspects)?

In MRP Letter 2013-025, dated October 14, 2013 (Reference 21), EPRI provided to licensees a non-proprietary document containing guidance for responding to the two questions above. With respect to Question 1, MRP Letter 2013-025 provides guidance for licensees to assess whether RVI components at their plant, other than those identified in the generic evaluation, have the potential for cold work greater than 20 percent. With respect to Question 2, MRP Letter 2013-025 provides quantitative criteria to allow a licensee to assess whether a particular plant has atypical fuel design or fuel management. For a CE-designed plant such as MPS2, these criteria are:

1. The heat generation rate must be ≤ 68 Watts/$cm^3$ for both CE and Westinghouse plants.

2. The maximum average core power density must be less than 110 Watts/$cm^3$ for a CE plant.

3. The active fuel to fuel alignment plate (FAP) distance must be greater than 12.4 inches for a CE plant.

The NRC staff's review of MRP 2013-025 and the supporting technical information in WCAP-17780-P are documented in Reference 21. In Reference 22, the NRC staff concluded that the information provided on evaluation of cold work in WCAP-17780-P provides an adequate technical basis for the guidance in MRP Letter 2013-025 for responding to Question 1. The NRC staff further concluded in Reference 21 that the sensitivity studies of variations in neutron fluence, RVI geometry and temperature, and the information on power uprate effects on fluence and temperature, documented in WCAP-17780-P, provide an acceptable technical basis for the guidance in MRP Letter 2013-025 for responding to Question 2.

In RAI 1, the NRC staff requested that the licensee respond to the following questions, which are essentially identical to the questions as defined in the guidance document:

1. Do the MPS2 reactor vessel internals (RVI) have non-weld or bolting austenitic stainless steel components with 20 percent cold work or greater, and if so do the affected components have operating stresses greater than 30 ksi? If so, perform a plant-specific evaluation to determine the aging management requirements for the affected components.

2. Has MPS2 ever utilized atypical fuel design or fuel management that could make the assumptions of MRP-227-A regarding core loading/core design non-representative for that plant, including power changes/uprates?

In its response to RAI 1, Question 1, dated December 19, 2014 (Reference 3), the licensee stated that it performed a plant-specific review of documentation associated with austenitic stainless steel cold work in accordance with the current industry guidance issued in EPRI letter MRP 2013-025. The licensee further stated that the review considered both original fabrication of reactor vessel internals and modifications completed through November 2014. The licensee stated that following the process described in MRP 2013-025, the materials of the reactor vessel internals components were binned into five categories, including "cold worked austenitic stainless steel (without subsequent solution annealing)." Finally, the licensee stated that the review determined that the RVI at MPS2 have no non-weld, non-bolting austenitic stainless steel components with 20 percent or greater cold work. The NRC staff finds the licensee's response to RAI 1, Question 1, to be acceptable because the licensee, using the NRC-approved guidance of MRP Letter 2013-025, confirmed that MPS2's RVI components contain no non-weld, non-bolting, austenitic stainless steel components with 20 percent or greater cold work.

In its response to RAI 1, Question 2, dated July 21, 2014, the licensee stated that the average core power density for MPS2 for the last six operating cycles has been 83 Watts/$cm^3$, which is less than the limit of 110 Watts/$cm^3$ from MRP 2013-025. The licensee also stated the calculated heat generation rate figure of merit (HGR-FOM) for the peripheral fuel assemblies did not exceed 58 Watts/$cm^3$ during the last six fuel cycles, which is less than the limit of 68 Watts/$cm^3$ from MRP 2013-025. The licensee also stated that the minimum distance between the top of the active fuel to the FAP, considering past and current fuel designs, is 13.16 inches which is greater than the limit of 12.4 inches for a CE plant from MRP Letter 2013-025. For all three parameters, the licensee stated that these core design and management parameters are expected to remain representative for future plant operations within the currently licensed thermal power limit. Therefore, since the licensee's response indicates that MPS2 meets the numerical criteria of MRP Letter 2013-025, the NRC staff finds that MPS2 does not have atypical fuel design or fuel management that that could make the assumptions of MRP-227-A regarding core loading/core design non-representative for that plant.

The licensee demonstrated MPS2 meets the three basic criteria of MRP-227-A, Section 2.4, for applicability of the guidelines. In Section 2.1 of the RVI Program description, "Applicability of MRP-227-A," the licensee provided a summary of MPS2 compliance with the criteria of MRP-227-A, Section 2.4. The licensee adequately addressed the two factors for which the NRC staff determined additional plant-specific information was necessary to verify applicability of MRP-227-A to MPS2: (1) cold work induced stress; and, (2) fuel management, by confirming that MPS2 complies with the criteria defined in MRP 2013-025. Furthermore, the licensee stated that these core design and management parameters are expected to remain representative for future plant operations within the currently licensed thermal power limit. Therefore the NRC staff finds the licensee's response acceptable, and that the licensee has adequately addressed A/LAI 1 for MPS2.

3.3.2 A/LAI 2 PWR Vessel Internal Components within the Scope of License Renewal

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

Licensee Evaluation

In its response to A/LAI 2, the licensee stated that several components had a different material than that specified in MRP-191 (Reference 11), but those differences have no effect on the recommended MRP aging management strategy, and therefore, that no changes to the program details of MRP-227-A need to be proposed. The licensee's response further stated that this supports the requirement that the AMP shall provide assurance that the effects of aging on the

MPS2 RVI components within the scope of license renewal, but not included in the generic CE-designed PWR RVI components from Table 4-5 of MRP-191, will be managed for the period of extended operation. The generic scoping and screening of the RVI components, as summarized in MRP-191 and MRP-232, "Material Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals" (Reference 23), to support the inspection sampling approach for aging management of the RVI components specified in MRP-227-A, are applicable to MPS2 with no modifications. The licensee concluded that MPS2 complies with A/LAI 2 of the SE on MRP-227, Revision 0, and that therefore; MPS2 meets the requirement for application of MRP-227-A as a strategy for managing age-related material degradation in RVI components.

NRC Staff Evaluation

To verify the licensee adequately addressed A/LAI 2, the NRC staff: (1) compared the RVI components subject to AMR in the MPS2 LRA to the corresponding components addressed by MRP-227-A in order to verify that all components that credited the Inservice Inspection Program: Reactor Vessel Internals for aging management are covered by MRP-227-A (since the commitment was to update this program to account for the industry program), and (2) reviewed the licensee's evaluation of MPS2 components fabricated from different materials than the generic MRP-191 material to ensure the licensee made changes if necessary to aging management requirements.

There were two RVI components listed in Table 3.1.2-2 of the MPS2 LRA, the Expansion Compensating Ring and the Fuel Alignment Plate Guide Lugs and Guide Lug Inserts, which the NRC staff could not match to an equivalent component in MRP-191. Therefore, to resolve this discrepancy, in RAI 4, the NRC staff requested the licensee provide the MRP-191 equivalent component name for these two components.

In its response to RAI 4, dated July 21, 2014 (Reference 2), the licensee stated that "[t]he MRP-191 equivalent component name for the Expansion Compensating Ring is the "hold-down ring" listed in Table 4-5, Page 4-11 of MRP-191. The licensee also stated that "MRP-191 lists the Fuel Alignment Plate "Guide Lugs" and "Guide Lug Inserts" in Table 4-5, Page 4-13."

The NRC staff finds the licensee's response to RAI 4 is acceptable because it clarified the identity of the two components listed in LRA Table 3.1.2-2 that the NRC staff could not match to the equivalent component name in MRP-227-A. Therefore, the NRC staff's concern in RAI 4 was resolved.

In RAI 3, the NRC staff requested for the components with material differences identified in the A/LAI 2 evaluation, that the licensee identify the component, material type at MPS2, and the generic material from MRP-227-A; as well as to describe how the difference in material was evaluated and explain why no changes to the aging management requirements were needed for those components.

In its response to RAI 3, dated July 21, 2014, the licensee identified two components with material differences from the corresponding generic component in MRP-191. These are the in-core instrumentation (ICI) guide tubes which are Type 304 stainless steel at MPS2 versus the generic MRP-191 material of Type 316 stainless steel, and the shaft retention pin and retention block which are subcomponents of the control element assembly (CEA) shrouds. The licensee stated that Type 304 stainless steel and Type 316 stainless steel fall under the same austenitic stainless steel category; therefore, the same degradation mechanisms listed in MRP-191 for the ICI guide tubes are applicable to both materials (IASCC and IE). The licensee stated that the shaft retention pin and block are not specifically identified in MRP-191, and that the CEA shrouds are either Type 304 stainless steel, or CF8 or CF8M copper accelerated acetic acid salt spray (CASS) per MRP-191. The response to RAI 3 indicated the aging mechanism of concern for the CEA shrouds is SCC, and that the retention pin and block material is Type 304 or 304L stainless steel, which are equally susceptible to SCC as the generic MRP-191 material and are not susceptible to any additional aging mechanisms.

The NRC staff reviewed the licensee's response to RAI 3 and agrees with the licensee's assessment that found the same aging mechanisms are applicable to the MPS2-specific materials. No additional inspections are needed.

The NRC staff notes that Sections 2.1 and 3.10 of the AMP describe a modification in which the thermal shield was removed due to fatigue cracking caused by localized high-cycle fatigue loading at the thermal shield support brackets welded to the core barrel. The licensee stated the localized degraded condition of the core barrel was corrected by a combination of flaw removal and bounding by crack stop holes drilled in the core barrel at the crack ends to prevent linear crack propagation. The licensee stated they reinspected these areas twice in 1994 and 2008 and observed no crack growth. The licensee further stated that these locations will be subject to a one-time reinspection by EVT-1 under the RVI Program, and if new indications are detected they will be evaluated for flaw growth and tolerance in accordance with the principles of topical report WCAP-17096, Rev. 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements" (Reference 24).

The licensee indicated that the basis for planning to perform this examination only one additional time is that the inspection exceeds the vendor recommendation, and the direct cause of the original degradation (thermal shield attachments) has been removed. The licensee further stated that the enhanced visual examination EVT-1 for this additional inspection provides greater resolution than prior inspections and hence provides one final confirmation that no cracking has occurred, and that this augmentation of the MRP-227-A Program is included as a primary component line item in Table 1.

The NRC staff considers the crack stop hole region of the core barrel to be a plant-specific component for MPS2.

Since the crack stop hole areas are treated as a plant-specific component, in RAI 7 the NRC staff requested the licensee to describe how the applicable aging mechanisms and effects for the period of extended operation were determined, and to provide a justification for the process used if different than the process described in MRP-227-A, Section 2. The NRC staff also requested details of the evaluation that determined that fatigue cracking was the only aging effect requiring management for these locations, and an explanation of why the other aging effects/mechanisms generically evaluated in MRP-227-A are not applicable to these locations. The response to RAI 7, dated July 21, 2014, indicates the licensee used the criteria of MRP-191 to screen for applicable aging effects at the crack stop hole areas, and that aging mechanisms other than fatigue cracking screened out. The RAI 7 response also indicates that the licensee confirmed that the crack stop hole locations were known and considered during the screening and FMECA process for the core support barrel during the development of MRP-227-A. The NRC staff finds the licensee's response to RAI 7 acceptable because the response indicates the crack stop hole locations were subject to the same process to determine applicable aging effects and aging management requirements that was used for the generic RVI components during the development of MRP-227-A. The NRC staff's concern in RAI 7 has been resolved.

Therefore, since the NRC staff verified that all the RVI components within the scope of LR for MPS2 that credit the RVI AMP for aging management are either addressed by MRP-191, a plant-specific inspection, or have been evaluated to require no changes in aging management, the NRC staff finds that the licensee has adequately addressed A/LAI 2.

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

This A/LAI states that licensees of CE and Westinghouse are required to perform plant-specific analysis either to justify the acceptability of the licensee existing programs, or to identify changes to the programs that should be implemented to manage the aging of these components for the period of extended operation. The A/LAI further states that the results of this plant-specific analysis and a description of the plant-specific programs being relied on to manage aging of these components shall be submitted as part of the licensee AMP application, and that the CE and Westinghouse components identified for this type of plant-specific evaluation include: CE thermal shield positioning pins, CE ICI thimble tubes (Section 4.3.2 in MRP-227-A), and Westinghouse guide tube support pins (split pins) (Section 4.3.3 in MRP-227-A).

Licensee Evaluation

With respect to MPS2 compliance with A/LAI 3, the licensee stated that the one component applicable to MPS2 in this A/LAI is the CE ICI thimble tubes. The thermal shield positioning pins are no longer applicable because the thermal shield was removed in 1983. The licensee further stated that all ICI flux thimble tubes for MPS2 were replaced during refueling outage 2R19 (Fall 2009) after 22.65 effective full-power years of operation. The licensee indicated that the original Zircaloy thimbles had approximately a 4" gap allowance for Zircaloy growth before contacting the lower end of the fuel assembly guide tube fitting, while the new replacement thimbles have almost a 14" gap for future growth. The licensee stated that this factor of greater than three on available length for thimble growth effectively removes Zircaloy flux thimble growth as an operational issue requiring further active management, and that the functionality of the incore detector instrumentation is maintained in accordance with Technical Specification 4.3.3.2.

The licensee thus concluded that MPS2 has adequately addressed A/LAI 3 and that MPS2 meets the requirement for application of MRP-227-A as a strategy for managing age-related material degradation in RVI components.

NRC Staff Evaluation

The licensee identified the CE ICI thimble tubes as the only MPS2 RVI components needing a plant-specific AMP. The ICI flux thimble tubes are not included in MPS2 LRA Table 3.1.2-2, "Reactor Vessel, Internals, and Reactor Coolant System – Reactor Vessel Internals –Aging

Management Evaluation." This component is also not included in the GALL Report, Rev. 0. However, LR Interim NRC staff Guidance LR-ISG-2011-04: "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors, (Reference 9)," Table IVB3, "Reactor Vessel , Internals, and Reactor Coolant System – Reactor Vessel Internals (PWR) – Combustion Engineering," does have line item IV.B3.RP-357 for Incore Instruments: ICI thimble tubes - lower. However, the applicable aging effect/mechanism is loss of material due to wear, not change in dimensions.

Therefore, in RAI 5, the NRC staff requested the licensee clarify the following: (1) Were the ICI flux thimble tubes identified as a component subject to aging management under LR for MPS2? (2) If so, what aging effects and mechanisms were determined to require aging management and which program(s) were credited for managing aging of the ICI flux thimble tubes? (3) If not identified in the LRA as subject to AMR, what is the basis for including the ICI flux thimbles as a component required to be addressed under MRP-227-A A/LAI 3?

In its response to RAI 5, dated July 21, 2014, the licensee indicated that the ICI flux thimble tubes were not included in the LRA as a component subject to aging management because the MPS2 ICI flux thimble design does not have a pressure boundary function. The response to RAI 5 further indicated that there is no existing program to manage wear of the ICI flux thimble tubes at MPS2 because the MPS2 design does not have the interface or features that led to wear in a different CE design for flux thimble tubes, and that the existing program at MPS2 was for managing dimensional changes and the requirements of this program were completed via the replacement of the tubes in 2009. Therefore, the RAI 5 response indicates that there is no ongoing requirement for monitoring under the program. The licensee's evaluation of A/LAI 3 in the RVI Inspection Plan described the MPS2 plant-specific program for managing change of dimensions of the ICI flux thimble tubes for completeness.

Based on the licensee's evaluation of A/LAI 3 and its response to RAI 5, the NRC staff finds that a plant-specific program is not needed for aging management of the ICI flux thimble tubes at MPS2, because the tubes are not subject to AMR. Therefore, A/LAI 3 is not applicable because MPS2 does not have any RVI components subject to AMR to which plant-specific programs are applicable. The NRC staff's concern in RAI 5 has been resolved.

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

A/LAI 4 is applicable only to B&W-design RVI, therefore is not applicable to CE-design RVI such as MPS2.

### 3.3.5 A/LAI 5 – Application of Physical Measurements as Part of Information and Education (I&E) Guidelines for B&W, CE, and Westinghouse RVI

This A/LAI requires applicants/licensees to 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, and for distortion in the gap between the top and bottom core shroud segments in CE units with core barrel shrouds assembled in two vertical sections. The licensee shall include its proposed acceptance criteria and an explanation of how the proposed acceptance criteria are consistent with the plants' licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation as part of their submittal to apply MRP-227-A.

Licensee Evaluation

The licensee indicated its effort related to physical measurements focused on the core shroud. The licensee described proposed physical measurement acceptance criteria for the gap between the upper and lower sections of the core shroud, between which a gap is predicted to develop due to void swelling of the interfacing bottom plate of the upper section and upper plate of the bottom section. Thermal expansion due to gamma heating also causes a temporary increase in the gap width during operation, but during cold shutdown when physical measurements would be conducted, this increase would not be present. Therefore, the licensee calculated the predicted gap due to void swelling. The licensee calculated a maximum gap at the innermost corners of shroud (the location where the corner of the interfacing shroud plates is closest to the core) and away from the innermost corner (the location where the corner of the interfacing shroud plates are furthest from the core). These predicted gaps were 1/8 inch and 1/16 inch. The licensee stated the structural and functional effects of these gaps had been evaluated and found to be acceptable. The licensee referenced a Westinghouse letter with regard to the gap values and the structural and functional effects.

NRC Staff Evaluation

The NRC staff reviewed the licensee's physical acceptance criteria for the gap between the upper and lower core shroud sections. The severity of void swelling is a function of the neutron fluence received (the flux and the temperatures). The licensee's maximum allowable gap corresponds to a linear growth of 4 percent or a volumetric swelling amount of 12 percent. This value is conservative compared to void swelling predicted for former plates in NUREG/CR-7027, "Degradation of LWR Core Internal Materials due to Neutron Irradiation" (Reference 25), of around 5 percent.

Since it was not clear how the licensee evaluated the structural and functional effects associated with the presence of these gaps and determined the gaps to be acceptable, in RAI 6, the NRC staff requested the following additional information:

1. How are the acceptance criteria consistent with the licensing basis of MPS2?

2. Other than distortion, what structural effects are expected to occur (for example, increased stresses), and how were these determined to be acceptable?

3. How is the function of the core shroud affected, if at all, by the maximum allowable swelling?

4. How were the effects on the core shroud functions determined to be acceptable and what is the source of the functionality acceptance criteria for the core shroud?

The licensee's response to Part 1 of RAI 6, dated July 21, 2014, indicated that the licensee evaluated the potential for increased stresses due to the maximum postulated gaps due to void swelling. The licensee stated that stresses due to void swelling have not been calculated for any CE plant, but that based on the stress classifications defined in Section III, Subsection NG, of the ASME B&PV Code (prescribed in the MPS2 FSAR) stresses due to void swelling are considered secondary stresses because such stresses are self-limiting. The licensee further stated that the main impact of secondary stresses in structural analysis is on fatigue analysis, but since void swelling stresses are not cyclical, such stresses would have no impact on cumulative usage factors. Therefore, the licensee stated that it is reasonable to conclude that void swelling stresses need not be included in the structural evaluation of the core shroud, and thus would not adversely impact the satisfaction of the MPS2 licensing basis acceptance criteria.

The licensee's response also discussed the possibility that the increase in thickness of the interfacing horizontal plates of the upper and lower core shroud subassemblies could result in increased stresses in the core shroud tie rods which hold the two subassemblies together. However, the licensee determined the increase in plate thickness would be very small compared to the overall height of the core shroud assembly, so the increase in stress is expected to be negligible.

In response to RAI 6, Item 3, the licensee described the design functions from the MPS2 FSAR of the core shroud which are to provide an envelope for the core and limits the amount of coolant bypass flow. The licensee stated the potential adverse effects of the maximum core shroud gaps on these functions are: (a) increase in core shroud-to-core support barrel (CSB) bypass coolant flow, and (b) inward deflection of core shroud plates encroaching on fuel space.

In response to RAI 6, Item 4, the licensee provided the results of its evaluations of these two effects. With respect to core shroud to CSB bypass coolant flow, the licensee stated coolant flow jetting through the gaps between the interfacing horizontal plates of the upper and lower core shroud subassemblies would increase the core shroud-to-CSB bypass coolant flow, and that this increase in core shroud-to-CSB bypass flow was conservatively estimated and added to the existing total bypass flow. The licensee stated that the resulting increased total bypass flow was less than the allowable value prescribed in the MPS2 FSAR and that therefore, the effect of the core shroud gaps on core bypass flow was determined to be acceptable and bounded with regard to the MPS2 licensing basis. With respect to the inward deflection of the core shroud plates, the licensee stated that the maximum gaps between the interfacing plates of the core shroud upper and lower subassemblies result from the vertical deflection of one plate relative to the other, and that there is also a horizontal (inward) component to this deflection. The licensee stated that the maximum inward deflection is less than the dimensional tolerance on the lateral position of the core shroud plates relative to the core centerline. The licensee further stated that this inward deflection, which would occur at core mid-height (approximately), and could be accommodated by the lateral flexibility of the fuel assemblies. Therefore, the licensee concluded that the inward deflection of the core shroud due to irradiation-induced void swelling would not have a significant adverse effect on the fuel assemblies.

The NRC staff reviewed the licensee's response to RAI 6, and finds that the licensee has clarified how the physical measurement acceptance criteria for the upper-to-lower core shroud section gap are related to the MPS2 design basis. The licensee also clarified that the expected stress increase due to void swelling would be inconsequential and would not affect the design function of the core shroud. The licensee's response also clarified the two design basis parameters on which the acceptance criteria are based, core shroud-to-CSB bypass flow and inward deflection of core shroud plates upon the fuel, and how it was determined these parameters would remain acceptable. Therefore, the NRC staff finds the licensee's response to RAI 6 acceptable. The NRC staff's concern in RAI 6 has been resolved.

The NRC staff finds that the licensee has adequately addressed A/LAI 5 because it has described the acceptance criteria for physical measurements applicable to the MPS2 RVI, and has explained how these criteria are consistent with the licensing basis for MPS2 and the need to maintain the functionality of the RVI.

3.3.6 A/LAI 6 Evaluation of Inaccessible B&W Components

This A/LAI is applicable only to B&W-design RVI, therefore is not applicable to CE-design RVI such as at MPS2.

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

This A/LAI requires the applicants/licensees of B&W, CE, and Westinghouse reactors to develop plant-specific analyses to be applied for their facilities to demonstrate that B&W incore monitoring instrumentation guide tube assembly spiders control rod guide tube assembly spacer castings, CE lower support columns, and Westinghouse lower support column bodies, or additional RVI components that may be fabricated from CASS, martensitic or precipitation hardened stainless steel, will maintain their functionality during the PEO. These analyses should also consider the possible loss of fracture toughness in these components due to TE and IE. The plant-specific analysis shall be consistent with the plant's licensing basis and the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation. The licensee shall include the plant-specific analysis as part of their submittal to apply the approved version of MRP-227.

Licensee Evaluation

In Attachment 2 to the submittal, in its response to A/LAI 7, the licensee described its plant-specific evaluation of CASS RVI. The licensee's evaluation used a screening approach based on the criteria of NRC Letter, "License Renewal Issue No. 98-0030, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components" (Reference 26). The result of the screening was that 63 of 68 core support columns were determined to be nonsusceptible (screened out) for TE based on the ferrite content calculated from the chemical composition from the certified material test reports (CMTRs). For 5 of 68 columns, the CMTR could not be located so the licensee conservatively assumed these 5 columns are susceptible to TE. For the columns with CMTRs available, the maximum calculated ferrite content is 9.48 percent. The licensee assumed those columns without CMTRs to have ferrite content greater than 20 percent.

The licensee concluded that the results of this core support column CASS evaluation do not conflict with the MRP-227-A strategy for aging management of RVI. The licensee stated that although there are core support columns potentially susceptible to TE, these components had previously been screened in for this age-related degradation mechanism in the MRP-227-A process, and that basis assumptions for the components of MRP-227-A methodology are still valid. The licensee concluded that continued application of the strategy of MRP-227-A will meet the requirement for managing age-related degradation of the MPS2 CASS core support columns.

NRC Staff Evaluation

The NRC staff has developed interim guidance for screening of CASS materials that are susceptible to both TE and IE (Reference 27). Under the interim guidance, low-molybdenum CASS materials (such as type CF8) that receive neutron fluences greater than 0.45 displacements per atom (dpa) ($3x10^{20}$ n/cm$^2$) are considered to be susceptible to TE and IE if the ferrite content is greater than 15 percent, while if the ferrite content is less than or equal to 15 percent, these materials are only susceptible to IE at neutron fluences greater than 1.5 dpa ($1x10^{21}$ n/cm$^2$). Using the NRC staff's interim guidance, 63 of 68 columns would screen out for TE.

The columns screen-in for IE based on the peak neutron fluence to the columns. The NRC staff's understanding is that although the core support column welds are required to be inspected as "a primary category" component by MRP-227-A, the weld inspection is conducted from above the CSP; therefore, no portion of the core support columns would be viewed during the visual inspection of the core support column welds.

However, considering the potential susceptibility of some columns to both TE and IE, the NRC staff was concerned that the MRP-227-A aging strategy alone was not sufficient to manage aging of the columns, because the actual columns are not inspected as "Primary," "Expansion," or "Existing Programs" components. Only the column welds are inspected with no link to the columns as an "Expansion" component.

Therefore, in RAI 2, the NRC staff requested the licensee provide the following:

1. Clarify the scope of the core support column weld inspection. Specifically, is any portion of the column viewed during the visual examination of the core support column welds? If not, can any information regarding the integrity of the columns be gained from the results of the core support column weld visual examination?

2. If the core support columns (other than the welds) are not inspected,

    a. Provide a functionality evaluation of the lower support structure considering the effects of IE, and IE plus TE for those columns for which TE could not be screened out; or

    b. Modify the RVI inspection program to include inspections of the core support columns as a Primary or Expansion component. Appropriate Primary link(s) to components with higher susceptibility to TE and IE would need to be identified if the columns are added as Expansion components.

In its response to RAI 2, Part 1, dated July 21, 2014, the licensee stated that due to the small diameter of most of the flow holes (less than 2-inch except for one 3-inch flow hole per quadrant) and relative thick plate (approximately 2-inch), the portions of the core support columns below the core support plate are not accessible for inspection in conjunction with the inspection of the core support column welds.

In addition, in its response to RAI 2, Part 1, the licensee stated that during core support column weld visual examination, loss of integrity or material degradation would be detectable via evidence of changes to the surface at the weld location, and that cracking (SCC, IASCC, and fatigue including damaged material) or structural distortion of the embrittled material at the weld would appear as a disruption of the normally smooth machined surface of the lower core support plate.

In its response to RAI 2, Part 2, dated July 21, 2014, the licensee indicated it performed a statistical analysis demonstrating that the probability of the ferrite content of any of the five columns exceeding 12.68 percent (5 standard deviations above the mean) is extremely low (6 in ten million). The licensee's analysis was based on the standard normal distribution, the mean calculated delta ferrite of the columns of known composition of 5.53 percent, and the standard deviation of 1.43 percent. The NRC staff reviewed the licensee's statistical analysis, and agrees with the results. The licensee's analysis demonstrates that it is extremely unlikely for any of the columns with unknown composition to have a ferrite content exceeding 15 percent that would make the columns screen-in for TE at the expected neutron fluence for the columns.

The licensee also stated in its response to RAI 2, Part 2, that since the core support columns screen out for TE, the CASS material in the upper portions of the columns may be considered equivalent to wrought austenitic stainless steel material with respect to IE, and that each arm of the core support columns has a peg that is inserted into and welded to the core support plate. The licensee further stated that this location exposes the CASS material to a bounding level of irradiation such that the primary inspection of this weld area required by MRP-227-A constitutes adequate management for potential IE of the core support columns.

The NRC staff finds the licensee's assertion that the inspection of the core support column weld area represents an inspection of the bounding portion of the core support column with respect to IE, is reasonable. However, if cracking was detected in the core support column weld area during the "Primary" inspection, MRP-227-A does not specify an expansion to the main portion of the column. Therefore, in RAI 2-1, the NRC staff requested the licensee clarify what actions would be taken if evidence of service-induced cracking was detected during the "Primary" inspection of the core support column welds. In its response to RAI 2-1, dated January 26, 2015, the licensee indicated that discovery of service-induced cracking in the core support column welds would be entered into the licensee's corrective action program, which would initiate an operability determination involving an engineering evaluation. The licensee's response further indicated that the operability determination process may require additional characterization of the cracking conditions and additional activities to establish the extent of condition. Finally the licensee's response indicates that core reload would be contingent on achieving satisfactory conclusions regarding structural integrity of the core support system and the capability of the core support system to perform its safety-related functions. The licensee also noted in the response to RAI 2-1 that its RVI Program indicates that the methodology of WCAP-17096-NP, Rev. 2 (Reference 24), as approved by the NRC staff, will be used for engineering evaluations of conditions found that do not meet the MRP-227-A acceptance criteria. The licensee further noted that the NRC staff has proposed conditions on the use of WCAP-17096-NP that would require licensees to submit such engineering evaluations within one year of discovery of the unacceptable conditions. The NRC staff finds the licensee's response to RAI 2-1 acceptable because the process that will be used to evaluate any cracking found via the inspections of the core support column welds is adequate to provide reasonable assurance of continued operability of the lower core support system. Further, once WCAP-17096 is approved by the NRC staff, the licensee will apply this topical report for any engineering evaluations of unacceptable degradation in RVI, including abiding by conditions requiring timely submittal of such evaluations to the NRC staff. The NRC staff's concern detailed in RAI 2-1 has been resolved.

The NRC staff finds the licensee has demonstrated that it is unlikely any columns will have significant loss of fracture toughness due to TE. The NRC staff also finds that the "Primary" inspection of the core support column welds provides a leading indicator of cracking due to IASCC, SCC, or fatigue of the CASS core support column material. Finally, the NRC staff finds the licensee has described an appropriate process to determine the significance of any cracking found in the core support column welds, including implications for the core support system as a whole; and to determine the need for inspections of the core support columns if cracking is found in the welds. Based on the licensee's screening, primary inspections of the core support column welds, and process if degradation is found in these welds, the NRC staff concludes that the licensee has provided reasonable assurance that the CASS core support columns will remain functional for the duration of the PEO. Therefore, the NRC staff finds that the licensee has adequately addressed A/LAI 7.

3.3.8 A/LAI 8 – Submittal of Information for NRC staff Review and Approval

This action item requires licensees to make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SE, as an AMP for the RVI components at their facility. This submittal shall include the information identified in Section 3.5.1 of NRC staff's final SE of MRP-227, Revision 0.

Section 3.5.1 of the NRC staff's final SE of MRP-227, Rev. 0 (Reference 7), states that in addition to the implementation of MRP-227, Revision 0, in accordance with NEI 03-08, applicants/licensees whose licensing basis contains a commitment to submit a PWR RVI AMP and/or inspection program shall also make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended [by the NRC staff's final SE]. Section 3.5.1 of Reference 7 further states that an licensee's application to implement MRP-227, as amended by this SE, shall include the following items (1) and (2):

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

2. To ensure the MRP-227, Revision 0, program and the plant-specific action items will be carried out by applicants/licensees, applicants/licensees are to submit an inspection plan which addresses the identified plant-specific action items for NRC staff review and approval consistent with the licensing basis for the plant. If an applicant/licensee plans to implement an AMP which deviates from the guidance provided in MRP-227, as approved by the NRC, the applicant/applicant shall identify where their program deviates from the recommendations of MRP-227, as approved by the NRC, and shall provide a

justification for any deviation which includes a consideration of how the deviation affects both “Primary” and “Expansion” inspection category components.

Applicants that submitted applications for LR after the issuance of the MRP-227, Rev. 0, final SE are required to submit additional information items. The NRC staff notes that since the MPS2 LRA was submitted prior to the issuance of the NRC staff's final SE related to MRP-227, the licensee is only required to submit the above two information items.

Licensee Evaluation

The licensee provided an AMP addressing the ten elements of NUREG-1801, Rev. 2, AMP XI.M16A, contained in Section 3.0 of Reference 1.

The licensee submitted an inspection plan in its submittal (Reference 1) that addresses the A/LAI's that are applicable to CE-design RVI.

NRC Staff Evaluation

Since the licensee's submittal (Reference 1) contains both an AMP addressing the required ten elements, and an RVI Inspection Plan addressing the applicable A/LAI's, the NRC staff finds that the licensee has adequately addressed A/LAI 8.

### 3.3.9 Conclusion of A/LAI Evaluation

The NRC staff finds the licensee has adequately addressed those A/LAIs applicable to the design of MPS2; specifically, A/LAIs 1, 2, 3, 5, 7, and 8.

## 4.0 CONCLUSION

The NRC staff has reviewed MPS2 RVI Program description, and concluded that there is reasonable assurance that the MPS2 RVI AMP will manage aging of the RVI components at MPS2. The basis for the NRC staff's conclusion is that the RVI Inspection Plan is consistent with the I&E guidelines of MRP-227-A, and because all A/LAIs specified in MRP-227-A applicable to the design of MPS2 have been addressed in a manner acceptable to the NRC staff. Therefore, the NRC staff approves the MPS2 RVI AMP and RVI Inspection Plan.

Consequently, LR Commitment No. 13 from Appendix A of the NRC's SE of the MPS2 LRA is considered fulfilled. The NRC staff's approval of the MPS2 RVI Inspection Plan does not reduce, alter, or otherwise affect current ASME Code, Section XI, ISI requirements or any MPS2 specific licensing requirements related to ISI. The licensee must follow the implementation requirements as defined in Section 7.0 of MRP-227-A, which require that the NRC be notified of any deviations from the “needed” requirements.

In addition, the NRC staff finds that fatigue evaluations of the CSP and flexure weld performed in lieu of MRP-227-A inspections are now part of the MPS2 CLB, thus would need to be evaluated as TLAAs as part of an application for subsequent license renewal.

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Date: May 29, 2015

May 29, 2015

Mr. David A. Heacock
President and Chief Nuclear Officer
Dominion Nuclear Connecticut, Inc.
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 2 – REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM AND INSPECTION PLAN FOR LICENSE RENEWAL COMMITMENT NO. 13. (TAC NO. MF3402)

Dear Mr. Heacock:

By letter dated July 31, 2013, as supplemented by letters dated July 21, 2014, December 19, 2014, and January 26, 2015, Dominion Nuclear Connecticut, Inc. (the licensee), submitted a document entitled, "Aging Management Program Description: Inservice Inspection – Reactor Vessel Internals," in support of License Renewal (LR) Commitment No. 13 for Millstone Power Station, Unit 2. Specifically, the licensee submitted an updated Reactor Vessel Internals (RVI) Aging Management Program (AMP) and RVI Inspection Plan in accordance with topical report, "Material Reliability Program: Pressurized Water Reactor Inspection and Evaluation Guidelines (MRP-227-A).

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's RVI AMP and RVI Inspection Plan. Based on that review, the NRC staff concludes that the licensee's RVI AMP and RVI Inspection Plan are acceptable and the licensee has fulfilled LR Commitment No. 13.

If you have any questions, please contact me at (301) 415-1030 or Richard.Guzman@nrc.gov.

Sincerely,
***/RA/***
Richard V. Guzman, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-336

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