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DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 2
INFORMATION IN SUPPORT OF LICENSE RENEWAL
COMMITMENT #13 PROGRAM DESCRIPTION FOR
REACTOR VESSEL INTERNALS INSPECTIONS

Commitment Item No. 13 of NUREG 1838, the Safety Evaluation Report (SER) for the renewed operating license for Millstone Power Station Unit 2, requires the submission of a revised aging management program description for enhanced inspections of the reactor vessel internals. In accordance with Commitment Item No. 13, the inspection program for the reactor vessel internals has been updated to include the ten elements of NUREG-1801 and industry guidance from Electric Power Research Institute (EPRI) guidance document EPRI Report 1022863, "Material Reliability Program: Pressurized Water Reactor Inspection and Evaluation Guidelines (MRP-227-A)". The aging management program description is provided as Enclosure 1.

If you have any questions or require additional information, please contact Mr. William D. Bartron at (860) 444-4301.

Sincerely,

Dan Stoddard

Daniel G. Stoddard
Senior Vice President – Nuclear Operations

Enclosure:
1. Aging Management Program Description: Inservice Inspection – Reactor Vessel Internals

Commitments made in this letter: None

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NRR

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ENCLOSURE 1

AGING MANAGEMENT PROGRAM DESCRIPTION
INSERVICE INSPECTION: REACTOR VESSEL INTERNALS

Millstone Power Station Unit 2
Dominion Nuclear Connecticut, Inc.

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1.0 BACKGROUND

The reactor internals are passive structures within the reactor vessel that provide support for the reactor core as well as guiding the cooling water flow and providing guidance for control rod insertion. The materials are mainly stainless steel and the environment is the reactor coolant system primary water plus irradiation. The aging effects of concern are loss of material, cracking, loss of pre-load, change in dimensions due to void swelling, and loss of fracture toughness. The consequence of these aging effects as well as the potential mechanisms that may cause them have been the subject of study of an industry program, the materials reliability program (MRP), conducted by the Electric Power Research Institute (EPRI) with the participation of utilities and vendors. The result of the study is the inspection guideline MRP-227-A (reference 6.12), which has been adopted by the nuclear industry as the common program for managing the anticipated aging effects of reactor internals.

Until the beginning of the period of extended operation (PEO), periodic inservice inspections (ISI) of the internals are performed in accordance with ASME Section XI (reference 6.1), an existing program that ensures the integrity of the components comprising the reactor vessel internals. In many instances this program remains adequate for management of aging effects and is therefore referenced by this program. The additional examinations specified by the reactor internals program are performed in addition to the requirements of ASME Section XI and are governed by this program. The technical requirements for such augmented examinations are further specified in an EPRI MRP inspection standard, MRP-228 (reference 6.13).

In addition to its utility as a means of satisfying license renewal commitments, MRP-227-A has been issued as a guideline with needed and mandatory requirements under the NEI 03-08 protocol.

The NRC reviewed Revision 0 of MRP-227 and in its safety evaluation (SE) agreed that it formed the basis for an acceptable program for aging management of reactor internals, provided that certain changes to requirements be incorporated in a revised version of the document and each licensee address certain licensee action items (LAI) (reference 6.15). MRP-227-A thus incorporates the changes required by the NRC and also contains a copy of the SE. The LAIs required by the SE are not a requirement of MRP-227-A, but must be addressed in the plant-specific inspection plan which is required to be submitted to the NRC for review and approval.

2.0 DISCUSSION

The reactor vessel internals (RVI) inspection program (The Program) is a plant-specific program that implements aging management activities for the reactor internals as recommended by the relevant industry guideline, MRP-227-A. The Program is supplemental to, and coordinated with, the ASME XI inservice inspection program (reference 6.2). The current inspection plan under MRP-227-A for Millstone Power Station Unit 2 (MPS2) is contained in the tables beginning on page 18. Any changes to The Program or its implementation that constitute a deviation to the MRP-227-A mandatory or needed requirements will be processed in accordance with corporate procedures for implementing NEI 03-08 requirements (references 6.17 - 6.19) and will accordingly be reported to the NRC.

The scope of this aging management program (AMP) includes those passive reactor vessel internal subcomponents that support or by failure could challenge the safety related functions of the reactor, and is fully consistent with MRP-227-A. The subject materials include reactor vessel internals composed of austenitic stainless steels, nickel-based alloys and cast austenitic stainless steels (CASS). The materials are subject to the borated primary water environment. The Program manages the aging effects of:

- Stress corrosion cracking (SCC), primary water stress corrosion cracking (PWSCC), irradiation-assisted stress corrosion cracking (IASCC), and fatigue;
- Loss of material due to wear;
- Loss of fracture toughness due to neutron irradiation embrittlement and thermal aging embrittlement;
- Changes in dimension due to void swelling; and
- Loss of preload due to thermal and irradiation-enhanced stress relaxation or creep.

The Program uses a leading indicator and sampling approach for inspections, in which certain components considered to have a higher susceptibility or greater safety consequence are chosen for inspection as "Primary" inspection items, while certain others are subject to "Expansion" inspection if the Primary inspections detect significant issues. Some items are considered to be adequately managed through "Existing Programs" (including ASME Section XI B-N-3, see below). The remaining reactor internals component items have no specified inspections under The Program but could become subject to augmented inspection if so indicated by operating experience.

ASME XI ISI inspections are performed to demonstrate the long-term integrity and continued functionality of the reactor vessel internals. The Millstone Power Station (MPS) ISI program is broadly described in an administrative procedure, *Dominion Inservice Inspection Program* (reference 6.2). Specific requirements for inservice inspections are identified in the MPS2 ISI program manual[1] (reference 6.3).

For the reactor internals, these include:

- Inservice inspections performed in accordance with Examination Categories B-N-3 for the core support structures made accessible by removal of the reactor internals, and
- Augmented examinations not required by ASME Section XI.

The inspection plan for the reactor internals components governed by The Program is summarized in the attached tables (Tables 1, 2, 3, and 4). The tables are based on and consist of a selection of the table line items from MRP-227-A, Sections 4 and 5. The selection includes the line items applicable to MPS2. There have been no modifications of the MRP-227-A individual line items except as specifically noted.

In addition to the ISI program, the monitoring and control of reactor coolant water chemistry ensures the long-term integrity and continued functionality of the reactor vessel internals. The monitoring and control of primary chemistry is discussed in AMP, *Chemistry Control for Primary Systems* (reference 6.4).

2.1 APPLICABILITY OF MRP-227-A

MRP-227-A, Section 2.4 provides a list of assumptions used in developing MRP-227-A. A brief reconciliation of MPS2 design, operation, and modifications, with respect to the assumptions, is provided in the table below.

Table 2.1-1 MPS2 Reconciliation with MRP-227-A Applicability Assumptions

Assumption	MPS2 Specifics	Reconciliation
30 years of operation with high leakage core loading patterns. Then remaining 30 years of operation with a low-leakage fuel management strategy.	MPS2 transitioned from high leakage to low leakage fuel loading patterns at the beginning of Cycle 10, 11/4/1990, at which time the effective full power years (EFPY) was approximately 8.9 years. This is based on fuel load report ANF-89-019(P) dated 2/1989.	Assumption satisfied
Base load operation	MPS2 has always operated as a base load unit.	Assumption satisfied

[1] MPS ISI Inspections are currently performed to the 2004 Edition of ASME Section XI.

Table 2.1-1 MPS2 Reconciliation with MRP-227-A Applicability Assumptions		
Assumption	MPS2 Specifics	Reconciliation
No design changes affecting Reactor Vessel Internals components beyond those identified in general industry guidance or recommended by the original vendors.	MPS2 is a Combustion Engineering (CE) design. Significant design changes to Millstone Unit 2 include • Stretch Power uprate early in plant operation • Removal of thermal shield • For core barrel, leaving fatigue cracks in place, bounded by crack stop holes	Combustion Engineering and/or Westinghouse have developed or evaluated these design changes. Assumption satisfied.

The limited cracks recorded in the core barrel for MPS2 are not explicitly evaluated by the guideline. The cracks were caused by localized high cycle fatigue loading at thermal shield support brackets welded to the core barrel; the fatigue loading was caused by flow-induced vibration of the thermal shield. The cracks were discovered in 1983, and as a corrective action, the entire thermal shield was removed. 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 (PDCR 2-96-83). The removed material was not credited in the structural evaluation. As a conservative measure, inspection of the cracks was included as an augmented inspection under the ASME XI 10 year ISI program inspections. These were performed during the 1994 and 2008 refueling outages (2R12) and (2R18), respectively. No crack growth or new indications were identified. As an extension of this conservative measure, these same locations will be subjected to a one-time reinspection by enhanced visual examination (EVT-1) under The Program. If new indications are detected, they will be evaluated for flaw growth and tolerance in accordance with the principles of WCAP-17096 (reference 6.14), and reinspected accordingly. 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 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. This augmentation of the MRP-227-A program is included as a primary component line item in Table 1.

2.2 ITEMS SELECTED FOR INSPECTION

Under ASME XI B-N-3, "Core Support" structures are specified for inspection, and unless a more specific delineation of items listed in MRP-227-A applies, this Code scope governs. As provided by MRP-227-A, the attached "Existing Programs" Table 3 provides additional specificity of examination for certain items within the scope of the ASME XI examination scope. Consistent with MRP-227-A, the items selected for inspection are listed in the attached tables for "Primary" and “Expansion” items. The selected table entries and associated notes are identical to the corresponding MRP-227-A table. In addition, the area of the core barrel associated with thermal shield support brackets has been selected for augmented inspection as noted above. The sampling and extent of examination for each item is listed in the “Examination Coverage” column. Inspection of “Expansion” items is required only when criteria listed in the “Examination Acceptance and Expansion Criteria" invoke them.

2.3 INSPECTION SCHEDULE

ASME XI visual ISI inspections are implemented in accordance with the schedule required for Category B-N-3, Removable Core Support Structures, of ASME Section XI, Subsection IWB. The inspection schedule of MRP-227-A items is in accordance with MRP-227-A as shown in the attached tables. The tables refer to the start of the license renewal period, also known as the period of extended operation (PEO). Per the operating license, the PEO for MPS2 begins at midnight, July 31, 2015. The second refueling outage beyond this date, at which time the MRP-227-A inspections must be complete, is anticipated to occur in the spring of 2017.

2.4 INSPECTION STANDARDS AND EXAMINATION ACCEPTANCE STANDARDS

ASME XI visual ISI Inspections are implemented in accordance with Category B-N-3, Removable Core Support Structures, of ASME Section XI, Subsection IWB. The examination acceptance standards for the visual examinations (VT-3) of Category B-N-3 are summarized in IWB-3520.2, Visual Examination, VT-3 (reference 6.1).

The visual examinations (VT-3, VT-1 and EVT-1) of items listed in the attached tables are performed in compliance with the industry standards established in MRP-228 (reference 6.13). Since there are no surface or volumetric examinations specified by MRP-227-A for CE reactor designs, only the visual examination requirements of MRP-228 are relevant to The Program.

The examination acceptance standards for items inspected in accordance with MRP-227-A are described in MRP-227-A Section 5. Standards specific to Millstone are excerpted in TABLE 4 - MRP-227-A TABLE 5-2 CE PLANTS EXAMINATION ACCEPTANCE AND EXPANSION CRITERIA, which is attached to this document. The table has been enhanced to include a line corresponding to the inspection of repaired core barrel area noted above in Section 2.2.

Relevant conditions and relevant indications identified in both the ASME XI ISI program and the MRP-227-A inspections are entered into the corrective action program (CAP).

2.5 DISPOSITION OF INSPECTION RESULTS

Adverse inspection results are entered into the CAP for disposition. Engineering evaluation methodologies used to disposition conditions identified by MRP-227-A inspections are in accordance with WCAP-17096, including conditions imposed by the NRC (reference 6.14). Evaluation methodologies for conditions identified by the ASME XI ISI program are in accordance with ASME XI requirements or approved alternatives. Any repair/replacement activities required as a result of disposition will be in accordance with ASME XI requirements or approved alternatives.

2.6 REPORTING OF INSPECTION RESULTS

The inspection results and any required dispositions are reported to the MRP in accordance with MRP-227-A Section 7.6. The results of the ASME XI inspections are reported to the NRC in accordance with ASME XI requirements.

2.7 DISCUSSION OF GENERIC ISSUES

Generic issues that are directly addressed by the EPRI MRP-227-A program include void swelling, IASCC, and thermal and irradiation embrittlement of CASS. These effects are adequately managed by the requirements of MRP-227-A together with the associated SER licensee action items. Detailed discussion of these issues is, therefore, removed from this section of The Program.

2.8 ONGOING ACTIVITIES

Ongoing activities include participation in, and maintaining awareness of, continuing industry activities that could affect the future requirements of The Program. Activities also include an assessment of related industry experience. There are no ongoing activities that are required to be resolved prior to implementation of The Program.

For the MPS2 CE design, evaluations were required by MRP-227-A to determine the actual inspection requirement for some components. 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 (reference 6.23). Therefore, no inspection for fatigue cracking of these locations is required and a notation to this effect has been added to Table 1.

Industry operating experience has been assessed and incorporated in MRP-227-A as appropriate. The EPRI MRP has an ongoing program to gather and assess industry operating experience, including available experience from non-domestic reactors. A summary of industry experience is included in Section 3.10 of The Program. Industry experience through spring 2013 has been consistent with the expectations of MRP-227-A, with one exception. The exception is the failure of a number of baffle-former bolts at DC Cook Unit 2 in a clustered pattern, and only on one wide baffle plate. Since MPS2 has a welded shroud and does not utilize baffle bolting, this OE is not directly applicable.

At MPS2 during spring 2011 refueling outage (2R20), a nuclear instrumentation flux thimble at location G18 was found to be failed in the fluted region that, during operation, resides within the fuel assembly guide tube. The cause of the failure was determined to be wear at the location corresponding to the lower end of the fuel assembly guide tube wear sleeve. Wear of the flux thimble at the bottom of the wear sleeve created a step discontinuity in the flux thimble. During reactor disassembly, the discontinuity caused the thimble to catch on the lower end of the wear sleeve as the upper internals were lifted. During the prior cycle operation, there was no adverse effect on the G18 instrument readings. The flux thimbles guide the flexible instrumentation cables into the fuel assembly, and are not a structural member or pressure boundary. Upon evaluation, it was determined that there is no potential for loss of redundant safety function for such wear, and such wear should be trended and managed under the CAP.

3.0 EVALUATION USING NUREG-1801, GENERIC AGING LESSONS LEARNED (GALL) REPORT ELEMENTS

Note: The existing license renewal basis for MPS2 is GALL Rev. 0 (reference 6.11). This version of the GALL was referenced in previous versions of this AMP. However, the NRC SER (reference 6.15) on MRP-227-A requested that comparisons of the AMP be made with respect to Section XI.M16A, PWR Vessel Internals, of GALL Rev. 2 (reference 6.16). The GALL, Rev. 2 comparison is contained in Section 4. However, this reference to the later version of the GALL is not a change to the licensing basis for MPS2.

3.1 SCOPE

The scope of The Program includes the reactor internals components of the MPS2 reactor, which is a Combustion Engineering NSSS design. The scope of The Program applies the methodology and guidance provided in the most recently NRC-endorsed version of MRP-227-A, which provides augmented inspection and flaw evaluation methodology for assuring the functional integrity of safety-related internals in commercial operating U.S. pressurize water reactor (PWR) nuclear power plants designed by Babcock & Wilcox (B&W), CE, and Westinghouse. The scope of components considered for inspection under MRP-227-A guidance includes core support structures (typically denoted as examination category B-N-3 by the ASME Code, Section XI), those reactor internals components that serve an intended license renewal safety function pursuant to criteria in 10 CFR 54.4(a)(1), and other RVI components whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii).

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 typically within the scope of the components that are required to be subject to an aging management review (AMR), as defined by the criteria set in 10 CFR 54.21(a)(1). The scope of The Program also does not include welded attachments to the internal surface of the reactor vessel because these components are considered to be ASME Code Class 1 appurtenances to the reactor vessel and are adequately managed in accordance with the ISI Program: Systems, Components, and Supports (reference 6.7).

The scope of The Program includes the response bases for applicable license renewal applicant action items (LRAAI's) on the MRP-227 methodology, and any additional programs, actions, or activities that are discussed in these LRAAI responses and credited for aging management of the applicant's RVI components. The LRAAI's are identified in the staff's SER for MRP-227. They include applicable action items for meeting the assumptions that formed the basis of the MRP's augmented inspection and flaw evaluation methodology (as discussed in Section 2.4 of MRP-227), and NSSS vendor-specific or plant-specific LRAAI's as well. The responses to the LRAAI's on MRP-227 are provided in Attachment 2 of this AMP.

A review for applicability of MRP-227-A has been performed in accordance with Section 2.4 of the guideline, and it has been determined that the baseline assumptions of MRP-227-A are satisfied by the reactor internals inspection program for MPS2.

3.2 PREVENTIVE ACTIONS

The Program is an inspection program and is designated condition monitoring. The Program does not include preventive actions.

The guidance in MRP-227-A relies on PWR water chemistry control to prevent or mitigate aging effects that can be induced by corrosive aging mechanisms (e.g., loss of material induced by general corrosion, pitting corrosion, crevice corrosion, or stress corrosion cracking [including its various forms of SCC, PWSCC, and IASCC]). Reactor coolant water chemistry is monitored and maintained in accordance with the chemistry control for primary systems aging management program (reference 6.4).

3.3 PARAMETERS MONITORED OR INSPECTED

The Program manages the following age-related degradation effects and mechanisms that are applicable in general to the RVI components at the facility:

a. Cracking induced by SCC, PWSCC, IASCC, or fatigue/cyclical loading;
b. Loss of material induced by wear;
c. Loss of fracture toughness induced by either thermal aging or neutron irradiation embrittlement;
d. Changes in dimension due to void swelling and irradiation growth, distortion, or deflection; and
e. Loss of preload caused by thermal and irradiation-enhanced stress relaxation or creep.

For the management of cracking, The Program monitors for evidence of surface breaking linear discontinuities for the visual inspection technique that is used as the non-destruction examination (NDE) method. For the management of loss of material, The Program monitors for gross or abnormal surface conditions that may be indicative of loss of material occurring in the components. The Program does not directly monitor for loss of fracture toughness that is induced by thermal aging or neutron irradiation embrittlement, or by void swelling and irradiation growth. Instead, the impact of loss of fracture toughness on component integrity is indirectly managed by:

1) Using visual examination techniques to monitor for cracking in components, and
2) Applying applicable reduced fracture toughness properties in the flaw evaluation if cracking is detected in the component and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation under the MRP-227-A guidance or ASME Code, Section XI requirements.

The Program uses physical measurements to monitor for any dimensional changes due to void swelling, irradiation growth, distortion, or deflection.

Specifically, The Program implements the parameters monitored/inspected criteria for CE designed primary components in Table 4-2 of MRP-227-A. Additionally, The Program implements the parameters monitored/inspection criteria for CE designed expansion components in Table 4-5 of MRP-227-A. The parameters monitored/inspected for existing program components follow the bases for referenced existing programs, such as the requirements for ASME Code Class RVI components in ASME Code, Section XI, Table IWB-2500-1, Examination Categories B-N-3, as implemented through the ASME Code, Section XI program. No inspections, except for those specified in ASME Code, Section XI, are required for components that are identified as requiring "No Additional Measures," in accordance with the analyses reported in MRP-227-A. The specific NDE examinations to be utilized in this program are included in the attached tables. Since no bolting is inspected for cracking under this program, no volumetric inspections are anticipated, except potentially as supplemental examinations to further characterize an indication found by visual methods.

3.4 DETECTION OF AGING EFFECTS

The detection of aging effects is covered in two places:

1) The guidance in Section 4 of MRP-227-A provides an introductory discussion and justification of the examination methods selected for detecting the aging effects of interest; and
2) Standards for examination methods, procedures, and personnel are provided in a companion document, MRP-228.

In each case, well-established methods are selected. These methods include physical measurements for detecting changes in dimension, and various visual examinations (VT-3, VT-1, and EVT-1) for detecting effects ranging from general conditions to detection and sizing of surface-breaking discontinuities. The inspection methods required by MRP-227-A for CE reactor designs are implemented in full by The Program.

Cracking caused by SCC, IASCC, and fatigue is monitored/inspected in The Program by either VT-1 or EVT-1 visual examinations. The VT-3 visual methods are utilized to determine the general mechanical and structural condition of components by detecting discontinuities and imperfections, such as loss of integrity at welded connections, loose or missing parts, debris, corrosion, wear, or erosion; and by identifying conditions that could affect operational or functional adequacy of components. In addition, VT-3 examinations are used to monitor/inspect for loss of material induced by wear and for general aging conditions, such as gross distortion caused by void swelling and irradiation growth or by gross effects of loss of preload caused by thermal and irradiation-enhanced stress relaxation and creep.

In addition, The Program adopts the recommended guidance in MRP-227-A for defining the expansion criteria required to be applied to inspections of primary components and existing requirement components, and for expanding the examinations to include additional expansion components. As a result, inspections performed on the RVI components are performed consistent with the inspection frequency and sampling bases for primary components, existing requirement components, and expansion components in MRP-227-A, which have been demonstrated to be in conformance with the inspection criteria, sampling basis criteria, and sample expansion criteria in Section A.1.2.3.4 of NRC Branch Position RLSB-1.

Specifically, for MPS2 The Program implements the parameters monitored/inspected criteria and bases for inspecting the relevant parameter conditions for CE designed primary components in Table 4-2 of MRP-227-A and for expansion components in Table 4-5 of MRP-227-A. As an enhancement, the previously discovered and repaired cracking at thermal shield bracket locations (230 and 270 degrees) will be subject to a one-time reinspection as a Primary item by enhanced visual examination EVT-1 to identify any evidence of fatigue crack growth or new fatigue cracks emanating from the crack stop holes that were bored in 1983. The one-time additional reinspection is sufficient because the fatigue cracking mechanism driving force (thermal shield) has been eliminated, and two reinspections to date (1994 and 2008) have not identified any changes for the original pattern of cracking. Due to the unique nature of this historical mechanism, there is no expansion inspection item related to this additional primary inspection.

In addition, in some cases (as defined in MRP-227-A), physical measurements are used as supplemental techniques to manage for the gross effects of wear, loss of preload due to stress relaxation, or for changes in dimension due to void swelling, deflection or distortion. The only physical measurements anticipated in this program would be for the gap, if observed, between the upper and lower portions of the core shroud assembly. An acceptance limit for the potential gap has been determined and is described in Attachment 2 of The Program.

3.5 MONITORING AND TRENDING

The methods for monitoring, recording, evaluating, and trending the data that result from The Program's inspections are given in Section 6 of MRP-227-A and its subsections. The evaluation methods include recommendations for flaw depth sizing and for crack growth determinations as well as for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications. The examinations and reexaminations required by the MRP-227-A guidance, together with the requirements specified in MRP-228 for inspection methodologies, inspection procedures, and inspection personnel, provide timely detection, reporting, and corrective actions with respect to the effects of the age-related degradation mechanisms within the scope of The Program. The extent of the examinations, beginning with the sample of susceptible CE PWR internals component locations identified as primary component locations, with the potential for inclusion of expansion component locations if the inspection findings exceed the acceptance criteria, plus the continuation of the existing programs activities, such as the ASME Code, Section XI, Examination Category B-N-3 examinations for core support structures, provides a high degree of confidence in the total program.

3.6 ACCEPTANCE CRITERIA

Section 5 of MRP-227-A provides specific examination acceptance criteria for the primary and expansion component examinations. For components addressed by examinations referenced to ASME Code, Section XI, the IWB-3500 acceptance criteria apply. For other components covered by existing programs, the examination acceptance criteria are described within the existing program reference document.

The guidance in MRP-227-A contains three types of examination acceptance criteria:

- For visual examination (and surface examination as an alternative to visual examination), the examination acceptance criterion is the absence of any of the specific, descriptive relevant conditions. In addition, there are requirements to record and disposition surface breaking indications that are detected and sized for length by VT- 1/EVT-1 visual examinations.
- For volumetric examination (not applicable for MPS2), the examination acceptance criterion is the capability for reliable detection of indications in bolting, as demonstrated in the examination technical justification. In addition, there are requirements for system-level assessment of bolted or pinned assemblies with unacceptable volumetric (UT) examination indications that exceed specified limits; and
- For physical measurements of the core barrel holddown spring, when required by MRP-227-A. For the CE design of MPS2, no measurements of the hold-down spring are required by MRP-227-A.

MPS2 Supplemental Information

Upon detection of gaps in the MPS2 core shroud mid-height joint, physical measurements will be obtained for evaluation within the corrective actions element of The Program.

3.7 CORRECTIVE ACTIONS

Corrective actions following the detection of unacceptable conditions are fundamentally provided for in MPS's CAP. Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the plant corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection. The disposition will ensure that design basis functions of the RVI components will continue to be fulfilled for the licensing basis loads and events. Examples of methodologies that can be used to analytically disposition unacceptable conditions are found in the ASME Code, Section XI and in Section 6 of MRP-227-A. Section 6 of MRP-227-A describes the options that are available for disposition of detected conditions that exceed the examination acceptance criteria of Section 5 of the report. These include engineering evaluation methods, as well as supplementary examinations to further characterize the detected condition, or the alternative of component repair and replacement procedures. The latter are subject to the requirements of the ASME Code, Section XI (reference 6.10). The implementation of the guidance in MRP-227-A, plus the implementation of any ASME Code requirements, provides an acceptable level of aging management of safety-related components addressed in accordance with the corrective actions of 10 CFR Part 50, Appendix B or its equivalent, as applicable.

Other alternative corrective action may be used to disposition relevant conditions if they have been previously approved or endorsed by the NRC. Examples of previously NRC-endorsed alternative corrective actions include those corrective actions for Westinghouse-design RVI components that are defined in Tables 4-1, 4-2, 4-3, 4-4, 4-5, 4-6, 4-7 and 4-8 of Westinghouse Report No. WCAP-14577-Rev. 1-A. Westinghouse Report No. WCAP-14577-Rev. 1-A was endorsed for use in an NRC SER to the Westinghouse Owners Group, dated February 10, 2001. A more current acceptance criteria methodology document, WCAP-17096 Rev. 2, has been proposed, and to the extent it is subsequently approved for use, it will be the preferred methodology for determining engineering acceptance or corrective action for items requiring disposition as a result of MRP-227-A inspections. MPS2 will comply with the requirements and limitations of the WCAP document as stipulated in the NRC SER.

3.8 CONFIRMATION PROCESS

Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B, or their equivalent, as applicable. It is expected that the implementation of the guidance in MRP-227-A will provide an acceptable level of quality for inspection, flaw evaluation, and other elements of aging management of the PWR internals that are addressed in accordance with the 10 CFR Part 50, Appendix B, or their equivalent (as applicable), confirmation process, and administrative controls.

3.9 ADMINISTRATIVE CONTROLS

The administrative controls (references 6.2, 6.17 – 6.19) for The Program, including their implementing procedures and review and approval processes, are under existing site 10 CFR 50 Appendix B QA Programs, or their equivalent, as applicable.

3.10 OPERATING EXPERIENCE

MPS benefits from corporate procedures to identify and review relevant operating experience for reactor internals (reference 6.18). The procedures have provisions to modify The Program as required to consider future operating experience events. The following summarizes both MPS-specific and external operating experience.

MPS2 is currently in its fourth 10-year ASME XI inspection program interval.

MPS2 had an event in 1983 requiring removal of the thermal shield. As described in Section 2.1, cracks in the core barrel were identified, removed to the extent possible, and crack stop holes were bored at the ends of the remaining cracks to prevent further growth. The cracks cannot extend because of the crack stop holes, and the flow induced vibration driving force that initiated the cracking has been eliminated. Two subsequent reinspections have not detected any crack extension or initiation relative to the as-left condition of 1983. The condition of these mitigated flaws will be reexamined by enhanced visual examination EVT-1 one additional time as part of The Program.

Prior examinations of the RVIs, completed as part of the second inspection interval, identified a misalignment of one of four core barrel alignment keys. The issue was documented in an Unresolved Indication Report (UIR). CE was consulted, and it was determined that the alignment key would still be able to properly perform its aligning function during installation and operation without repositioning. The core barrel alignment keys continue to be specifically monitored under the ASME XI ISI program.

Most recently, the MPS2 nuclear instrumentation flux thimbles were replaced in 2009 due to irradiation induced elongation of the zircaloy thimbles. In 2011, one of the replaced thimbles was found to be severed at a location within the fuel assembly. The fuel assembly design has short wear sleeves inserted into the guide tubes at the top of the assembly in order to reduce wear of control element rods. An evaluation of the event determined that fluid induced vibration wear of the flux thimble against the lower edge of the wear sleeves created a wear scar on the flux thimble which impeded the removal of the flux thimble upon reactor disassembly during refueling. The high loads resulting from this restriction caused severance of the flux thimble. The MPS2 CAP is considering fuel design modifications to prevent recurrence of this issue. Because the flux thimble is not a pressure boundary and its function of guiding the nuclear instrumentation cable is not affected by the wear, no supplement to the reactor internals aging management program is required.

Industry Operating Experience (OE)

The MRP-227-A, Appendix A, listing of PWR operating experience was reviewed for experience potentially applicable to MPS2. The result is that the experience has also occurred for MPS2, as noted above, or is not applicable to MPS2 based on its design.

MRP-227-A also requires reporting of reactor internals inspection results. The compiled results are periodically published in EPRI document MRP-219 (reference 6.22). The latest available version of this document, Revision 8, has been reviewed for OE applicable to MPS2. This document does not contain any reports of MRP-227-A inspections of CE design reactor internals. The Westinghouse design OE was also reviewed and no adverse events relevant to the CE design were identified.

The MPS2 shroud assembly is welded, so baffle bolting degradation as seen at other reactors is not applicable.

License Renewal Interim Staff Guidance LR-ISG-2011-04: Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors (reference 6.20) has been reviewed for relevant operating experience. There were no changes to this AMP required as a result of this review.

4.0 AGING MANAGEMENT PROGRAM COMPARISON: MILLSTONE PROGRAM AND NUREG-1801 (GALL REPORT)

The license renewal application and subsequent approval for MPS2 and MPS3 were developed with reference to Chapter XI of Revision 0 of the GALL Report (reference 6.11). In the intervening years, Revision 1 of the report was issued in

2005, and Revision 2 of the report (reference 6.16) was issued to accommodate the inspection strategies developed in MRP-227-A, which incorporated changes identified in the NRC SER (reference 6.15). The SER requires submittal of aging management programs, in accordance with commitments for plants that have received renewed licenses, with reference to the GALL Report, Revision 2. The later revision resolves many of the issues that were outstanding in Revision 0.

In consideration of the SER (Reference 6.15), requirement A/LAI 8, Section 3 of The Program was revised to be compatible with Revision 2 of the GALL Report. However, the licensing basis and commitments for MPS are still based on GALL, Rev. 0 (reference 6.11).

The MPS ISI Program: Reactor Vessel Internals complies without deviation to MRP-227-A (reference 6.12) and is compatible with the aging management programs described in Chapter XI of Revision 2 of the GALL Report. The specific GALL sections and corresponding titles are as follows:

- Section XI.M16A, PWR Vessel Internals

Exceptions to the GALL (Section XI.M16A) – None.

Note: The published version of MRP-227-A, Table 4-2, contained an editorial omission. For the core support barrel assembly/upper (core support barrel) flange weld, the expansion column components should have included the lower core barrel flange weld. This missing component reference has been included in primary component inspection tables attached to this report.

Enhancements to GALL

Augmented examination of mitigated flaws related to the 1983 thermal shield damage and removal on MPS2 Core Barrel will be performed as a one-time augmented examination using enhanced visual examination EVT-1. This examination will repeat the scope of re-examinations for that component that was performed in the last two 10-year ISI examinations. The examination will be performed one time during the period of extended operation, and no further examinations are planned provided the results identify no new flaws or extensions of mitigated flaws related to the 1983 event. This enhanced requirement is included as a primary inspection item consistent with MRP-227-A format in Table 1, attached. A corresponding examination acceptance criterion has been added to Table 4, attached. The basis for having only the one additional examination is the fact that upon completing that examination, there will be a total of three re-examinations with no further degradation observed, and the fact that the major cause of the cracking, which was the highly localized loading of the thermal shield support bracket subject to thermal shield vibrations, has been corrected by removal of the thermal shield.

Program Elements Affected -

Scope of Program - This program element identifies the specific components subject to aging management for license renewal. Augmented examinations will be performed for the MPS2 core barrel in the area affected by the 1983 thermal shield damage.

Detection of Aging Effects - This program element identifies methods or techniques to ensure timely detection of aging effects. The MPS ISI program for MPS2 will incorporate enhanced visual examination EVT-1 techniques to detect cracking as an aging effect for the MPS2 core barrel in the area affected by the 1983 thermal shield damage.

5.0 SUMMARY

Inservice Inspection: The Program ensures that the effects of aging associated with the in-scope components will be adequately managed so that there is reasonable assurance that their intended functions will be maintained consistently with the current licensing basis through the period of extended operation.

6.0 REFERENCES

6.1. *Rules for Inservice Inspection of Nuclear Power Plant Components*, Section XI, American Society of Mechanical Engineers, New York, NY.

6.2. ER-AA-ISI-100, Dominion Inservice Inspection Program, Rev. 3, Dominion.

6.3. Millstone Unit 2 Inservice Inspection Program Manual, Fourth Ten-Year Interval, Revision 3, Program Manual, Millstone Unit 2, Dominion.

6.4. ETE-MP-2013-1041, Rev. 0, *Chemistry Control for Primary Systems*, License Renewal Aging Management Program (MP-LR-3702/MP-LR-4702), Dominion Engineering Technical Evaluation, 2013.

6.5. MP-LR-3503, *Reactor Vessel Internals*, Technical Report, License Renewal Project, Dominion Nuclear Connecticut.

6.6. MP-LR-3501, *Reactor Vessel*, Technical Report, License Renewal Project, Dominion Nuclear Connecticut.

6.7. ETE-MP-2013-1040, Rev.0. *Inservice Inspection Program: Systems, Components, and Supports*; License Renewal Aging Management Program (MP-LR-3701/MP-LR-4701), Dominion Engineering technical Evaluation.

6.8. MP-LR-3724/MP-LR-4724, *Reactor Vessel Surveillance*, Aging Management Program Report, License Renewal Project, Dominion Nuclear Connecticut.

6.9. PI-AA-200, Rev. 21, Corrective Action, Nuclear Fleet Administrative Procedure, Dominion Nuclear Connecticut.

6.10. ER-AA-RRM-100, ASME Section XI Repair/Replacement Program Fleet Implementation Requirements, Dominion.

6.11. NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, US Nuclear Regulatory Commission, July 2001.

6.12. EPRI Report 1022863, Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A), Electric Power Research Institute, Palo Alto, CA dated December 2011.

6.13. EPRI Report 1025147, Materials Reliability Program: Inspection Standard for PWR Internals – 2012 update (MRP-228 Rev. 1), 2012, Electric Power Research Institute, Palo Alto, CA.

6.14. Westinghouse WCAP-17096 Rev. 2, *Reactor Internals Acceptance Criteria Methodology and Data Requirements,* December 2009, Westinghouse Electric Company LLC, Pittsburgh, PA.

6.15. NRC SER: Revision 1 to The Final Safety Evaluation Of Electric Power Research Institute (EPRI) Report Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" (TAC No. ME0680).

6.16. NUREG-1801, Revision 2, *Generic Aging Lessons Learned (GALL) Report*, US Nuclear Regulatory Commission, December 2010.

6.17. ER-AA-RII-10, Fleet Reactor Internals Inspection Program Description, Rev. 2, Dominion.

6.18. ER-AA-RII-101, Fleet Reactor Internals Inspection Program, Rev. 2, Dominion.

6.19. ER-AA-MAT-10, Reactor Coolant System Materials Degradation Management Program, Rev. 5, Dominion.

6.20. LR-ISG-2011-04, License Renewal Interim Staff Guidance LR-ISG-2011-04: Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors, May 28, 2013 (Interim Staff Guidance to NUREG-1801 Rev. 2), U.S. Nuclear Regulatory Commission.

6.21. Westinghouse Letter CMIL-13-10 Rev. 1, "Transmittal of Final Task 1 Summary Letter for Millstone Power Station Unit 2 Applicant/Licensee Action Items 1, 2, 5 and 7", dated 6/7/2013.

6.22. Materials Reliability Program: Inspection Data Survey Results (MRP-219, Revision 8). Electric Power Research Institute, Palo Alto, CA: 2012. 1025151.

6.23. Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and threshold Values (MRP-175, Revision 0), Electric Power Research Institute, Palo Alto, CA, 2005, 1012081.

ATTACHMENT 1 – MPS2 MRP-227-A TABLES

MILLSTONE UNIT 2 (MPS2) REACTOR INTERNALS INSPECTION PLAN IN ACCORDANCE WITH MRP-227-A

The following tables are applicable to MPS2.

TABLE 1 – MRP-227- A TABLE 4-2 CE PLANT PRIMARY COMPONENTS

Inspection of the listed components is required in accordance with MRP-227-A and MRP-228

TABLE 2 – MRP-227- A TABLE 4-2 CE PLANT EXPANSION COMPONENTS

Inspection of listed components is required in accordance with MRP-227-A and MRP-228 when indicated by inspection results of primary components

TABLE 3 - MRP-227-A TABLE 4-8 CE PLANTS EXISTING PROGRAMS COMPONENTS

Inspection of the listed components is required in accordance with the referenced existing program. For references to the ASME Section XI ISI program, that program governs and this program is for reference only.

TABLE 4 - MRP-227-A TABLE 5-2 CE PLANTS EXAMINATION ACCEPTANCE AND EXPANSION CRITERIA

Contains the examination expansion criteria for the results of the Primary inspection components, and examination acceptance standards for primary and expansion components.

TABLE 1 - MRP-227-A TABLE 4-2 CE PLANTS PRIMARY COMPONENTS

Table 4-2 CE Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Shroud Assembly (Welded) Core shroud plate-former plate weld	Plant designs with core shrouds assembled in two vertical sections	Cracking (IASCC) Aging Management (IE) (Note 2)	Remaining axial welds	Enhanced visual examination (EVT-1) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.	Axial and horizontal weld seams at the core shroud re-entrant corners as visible from the core side of the shroud, within six inches of central flange and horizontal stiffeners. See Figures 4-12 and 4-14.

TABLE 1 - MRP-227-A TABLE 4-2 CE PLANTS PRIMARY COMPONENTS

Table 4-2 CE Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Shroud Assembly (Welded) Assembly	Plant designs with core shrouds assembled in two vertical sections	Distortion (Void Swelling), as evidenced by separation between the upper and lower core shroud segments Aging Management (IE)	None	Visual examination (VT-1) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	If a gap exists, make three to five measurements of gap opening from the core side at the core shroud re-entrant corners. Then, evaluate the swelling on a plant-specific basis to determine frequency and method for additional examinations. See Figures 4-12 and 4-14.
Core Support Barrel Assembly Upper (core support barrel) flange weld	All plants	Cracking (SCC)	Lower core support beams Core support barrel assembly upper cylinder Upper core barrel flange Lower core barrel flange weld	Enhanced visual examination (EVT-1) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible surfaces of the upper flange weld (Note 4). See Figure 4-15.

TABLE 1 - MRP-227-A TABLE 4-2 CE PLANTS PRIMARY COMPONENTS

Table 4-2 CE Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Support Barrel Assembly Crack stop holes at areas of prior fatigue cracking near thermal shield support bracket attachments	MPS2 only (This plant-specific item is not specified by MRP-227-A Table 4-2)	Cracking(fatigue)	None	One time enhanced visual examination (EVT-1) no later than 2 refueling outages from the beginning of the license renewal period. If flaw initiation detected, subsequent examination on a schedule determined by flaw tolerance evaluation.	100% of crack stop holes as shown on MPS2 drawings 25203-29141-00097A and -00097B.
Core Support Barrel Assembly Lower cylinder girth welds	All plants	Cracking (SCC, IASCC) Aging Management (IE)	Lower cylinder axial welds	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible surfaces of the lower cylinder welds (Note 4). See Figure 4-15
Lower Support Structure Core support column welds	All plants	Cracking (SCC, IASCC, Fatigue including damaged or fractured material) Aging Management (IE, TE)	None	Visual examination (VT-3) no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible surfaces of the core support column welds (Note 5). See Figures 4-16 and 4-31

TABLE 1 - MRP-227-A TABLE 4-2 CE PLANTS PRIMARY COMPONENTS

Table 4-2 CE Plants Primary Components					
Item	**Applicability**	**Effect (Mechanism)**	**Expansion Link (Note 1)**	**Examination Method/Frequency (Note 1)**	**Examination Coverage**
Core Support Barrel Assembly Lower flange weld	All plants	Cracking (Fatigue)	None	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), enhanced visual examination (EVT-1), no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval. Based on fatigue life demonstration, no EVT-1 examination required. See Note 6	Examination coverage to be defined by evaluation to determine the potential location and extent of fatigue cracking. See Figures 4-15 and 4-16.

TABLE 1 - MRP-227-A TABLE 4-2 CE PLANTS PRIMARY COMPONENTS

Table 4-2 CE Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Lower Support Structure Core support plate	All plants with a core support plate	Cracking (Fatigue) Aging Management (IE)	None	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), enhanced visual examination (EVT-1), no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval. Based on fatigue life demonstration, no EVT-1 examination required. See Note 6	Examination coverage to be defined by evaluation to determine the potential location and extent of fatigue cracking. See Figure 4-16.

TABLE 1 - MRP-227-A TABLE 4-2 CE PLANTS PRIMARY COMPONENTS

Table 4-2 CE Plants Primary Components					
Item	**Applicability**	**Effect (Mechanism)**	**Expansion Link (Note 1)**	**Examination Method/Frequency (Note 1)**	**Examination Coverage**
Control Element Assembly Instrument guide tubes	All plants with instrument guide tubes in the control element assembly (CEA) shroud assembly	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports	Remaining instrument guide tubes within the CEA shroud assemblies	Visual examination (VT-3), no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval. Plant-specific component integrity assessments may be required if degradation is detected and remedial action is needed.	100% of tubes in peripheral CEA shroud assemblies (i.e., those adjacent to the perimeter of the fuel alignment plate). See Figure 4-18.

Notes to Table 4-2:

1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-2.
2. Void swelling effects on this component is managed through management of void swelling on the entire core shroud assembly.
3. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-2, must be examined for inspection credit.
4. A minimum of 75% of the total weld length (examined + unexamined), including coverage consistent with the expansion criteria in Table 5-2, must be examined from either the inner or outer diameter for inspection credit.
5. A minimum of 75% of the total population of core support column welds.
6. 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 listed enhanced visual examination EVT-1 is not required and this item is subject to the normal ASME Section XI B-N-3 inspection requirements.

TABLE 2 - MRP-227-A TABLE 4-5 CE PLANTS EXPANSION COMPONENTS

Table 4-5 CE Plants Expansion Components

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Support Barrel Assembly Lower core barrel flange weld	All plants	Cracking (SCC, Fatigue)	Upper (core support barrel) flange weld	Enhanced visual examination (EVT-1). Re-inspection every 10 years following initial inspection.	100% of accessible welds and adjacent base metal (Note 2). See Figure 4-15.
Core Support Barrel Assembly Upper cylinder (including welds)	All plants	Cracking (SCC) Aging Management (IE)	Upper (core support barrel) flange weld	Enhanced visual examination (EVT-1). Re-inspection every 10 years following initial inspection.	100% of accessible surfaces of the welds and adjacent base metal (Note 2). See Figure 4-15.
Core Support Barrel Assembly Upper core barrel flange	All plants	Cracking (SCC)	Upper (core support barrel) flange weld	Enhanced visual examination (EVT-1). Re-inspection every 10 years following initial inspection.	100% of accessible bottom surface of the flange (Note 2). See Figure 4-15.

TABLE 2 - MRP-227-A TABLE 4-5 CE PLANTS EXPANSION COMPONENTS

Table 4-5 CE Plants Expansion Components

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Lower Support Structure Lower core support beams	All plants except those with core shrouds assembled with full-height shroud plates	Cracking (SCC, Fatigue) including damaged or fractured material	Upper (core support barrel) flange weld	Enhanced visual examination (EVT-1). Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2). See Figures 4-16 and 4-31.
Core Shroud Assembly (Welded) Remaining axial welds	Plant designs with core shrouds assembled in two vertical sections	Cracking (IASCC) Aging Management (IE)	Core shroud plate-former plate weld	Enhanced visual examination (EVT-1). Re-inspection every 10 years following initial inspection.	Axial weld seams other than the core shroud re-entrant corner welds at the core mid-plane. See Figure 4-12.
Control Element Assembly Remaining instrument guide tubes	All plants with instrument guide tubes in the CEA shroud assembly	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports.	Peripheral instrument guide tubes within the CEA shroud assemblies	Visual examination (VT-3). Re-inspection every 10 years following initial inspection.	100% of tubes in CEA shroud assemblies (Note 2). See Figure 4-18.

Notes to Table 4-5:

1. Examination acceptance criteria and expansion criteria for the CE components are in Table 5-2.
2. A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).

TABLE 3 - MRP-227-A TABLE 4-8 CE PLANTS EXISTING PROGRAMS COMPONENTS

Table 4-8 CE Plants Existing Programs Components

Item	Applicability	Effect (Mechanism)	Reference	Examination Method	Examination Coverage
Core Shroud Assembly Guide lugs Guide lug inserts and bolts	All plants	Loss of material (Wear) Aging Management (ISR)	ASME Code Section XI	Visual examination (VT-3), general condition examination for detection of excessive or asymmetrical wear.	First 10-year ISI after 40 years of operation, and at each subsequent inspection interval. Accessible surfaces at specified frequency.
Lower Support Structure Fuel alignment pins	All plants with core shrouds assembled in two vertical sections	Loss of material (Wear) Aging Management (IE and ISR)	ASME Code Section XI	Visual (VT-3) examination.	Accessible surfaces at specified frequency.
Core Barrel Assembly Upper flange	All plants	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	Area of the upper flange potentially susceptible to wear.

TABLE 4 - MRP-227-A TABLE 5-2 CE PLANTS EXAMINATION ACCEPTANCE AND EXPANSION CRITERIA

Table 5-2 CE Plants Examination Acceptance and Expansion Criteria

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Shroud Assembly (Welded) Core shroud plate-former plate weld	Plant designs with core shrouds assembled in two vertical sections	Visual examination (EVT-1). The specific relevant condition is a detectable crack-like surface indication.	Remaining axial welds	Confirmation that a surface-breaking indication > 2 inches in length has been detected and sized in the core shroud plate-former plate weld at the core shroud re-entrant corners (as visible from the core side of the shroud), within 6 inches of the central flange and horizontal stiffeners, shall require EVT-1 examination of all remaining axial welds by the completion of the next refueling outage.	The specific relevant condition is a detectable crack-like surface indication.

TABLE 4 - MRP-227-A TABLE 5-2 CE PLANTS EXAMINATION ACCEPTANCE AND EXPANSION CRITERIA

Table 5-2 CE Plants Examination Acceptance and Expansion Criteria

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Shroud Assembly (Welded) Assembly	Plant designs with core shrouds assembled in two vertical sections	Visual examination (VT-1). The specific relevant condition is evidence of physical separation between the upper and lower core shroud sections.	None	N/A	N/A
Core Support Barrel Assembly Upper (core support barrel) flange weld	All plants	Visual examination (EVT-1). The specific relevant condition is a detectable crack-like surface indication.	Lower core support beams Upper core barrel cylinder (including welds) Upper core barrel flange	Confirmation that a surface-breaking indication >2 inches in length has been detected and sized in the upper flange weld shall require that an EVT-1 examination of the lower core support beams, upper core barrel cylinder and upper core barrel flange be performed by the completion of the next refueling outage.	The specific relevant condition is a detectable crack-like surface indication.

TABLE 4 - MRP-227-A TABLE 5-2 CE PLANTS EXAMINATION ACCEPTANCE AND EXPANSION CRITERIA

Table 5-2 CE Plants Examination Acceptance and Expansion Criteria					
Item	**Applicability**	**Examination Acceptance Criteria (Note 1)**	**Expansion Link(s)**	**Expansion Criteria**	**Additional Examination Acceptance Criteria**
Core Support Barrel Assembly Lower cylinder girth welds	All plants	Visual examination (EVT-1). The specific relevant condition is a detectable crack-like surface indication.	Lower cylinder axial welds	Confirmation that a surface-breaking indication >2 inches in length has been detected and sized in the lower cylinder girth weld shall require an EVT-1 examination of all accessible lower cylinder axial welds by the completion of the next refueling outage.	The specific relevant condition for the expansion lower cylinder axial welds is a detectable crack-like surface indication.
Core Support Barrel Assembly Crack stop holes at areas of prior fatigue cracking near thermal shield support bracket attachments	MPS2 only (This plant-specific item is not specified by MRP-227-A Table 4-2)	Visual examination (EVT-1). The specific relevant condition is a detectable crack-like indication emanating from crack stop holes, or extension of mitigated flaws	None	NA	N/A

TABLE 4 - MRP-227-A TABLE 5-2 CE PLANTS EXAMINATION ACCEPTANCE AND EXPANSION CRITERIA

Table 5-2 CE Plants Examination Acceptance and Expansion Criteria

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Lower Support Structure Core support column welds	All plants	Visual examination (VT-3). The specific relevant condition is missing or separated welds.	None	None	
Core Support Barrel Assembly Lower flange weld	All plants	Visual examination (EVT-1). The specific relevant condition is a detectable crack-like indication.	None	N/A	N/A
Lower Support Structure Core support plate	All plants with a core support plate	Visual examination (EVT-1). The specific relevant condition is a detectable crack-like surface indication.	None	N/A	N/A

TABLE 4 - MRP-227-A TABLE 5-2 CE PLANTS EXAMINATION ACCEPTANCE AND EXPANSION CRITERIA

Table 5-2 CE Plants Examination Acceptance and Expansion Criteria					
Item	**Applicability**	**Examination Acceptance Criteria (Note 1)**	**Expansion Link(s)**	**Expansion Criteria**	**Additional Examination Acceptance Criteria**
Control Element Assembly Instrument guide tubes	All plants with instruments tubes in the CEA shroud assembly	Visual examination (VT-3). The specific relevant conditions are missing supports and separation at the welded joint between the tubes and the supports.	Remaining instrument tubes within the CEA shroud assemblies	Confirmed evidence of missing supports or separation at the welded joint between the tubes and supports shall require the visual examination (VT-3) to be expanded to the remaining instrument tubes within the CEA shroud assemblies by completion of the next refueling outage.	The specific relevant conditions are missing supports and separation at the welded joint between the tubes and the supports.

Notes to Table 5-2:

1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s).

ATTACHMENT 2 – LICENSEE ACTION ITEMS

Millstone Power Station Unit 2 (MPS2) Applicant/Licensee Action Items (A/LAIs) 1, 2, 3, 5, and 7

1.0 *Background and Purpose*

The current MPS2 aging management program (AMP) for the reactor internals is based on the Materials Reliability Program (MRP) specifications contained in MRP-227-A. The applicable NRC safety evaluation (SE) (reference 7.15 of the AMP) specifies that additional information be submitted in support of the AMP. Dominion Nuclear Connecticut (DNC) has contracted Westinghouse to address these new requirements for A/LAIs 1, 2, 5 and 7. The response to A/LAI 3 was prepared by DNC. A/LAIs 4 and 6 are applicable to Babcock & Wilcox reactor designs and thus are not applicable to MPS2.

References for this attachment are contained in Section 7.0 herein.

2.0 *SE A/LAI 1: Applicability of Failure Mode, Effects, and Criticality Analysis (FMECA) and Functionality Analysis Assumptions*

The action item text from the SE contained in MRP-227-A (reference 7.3) states:

"As addressed in Section 3.2.5.1 of this SE, each applicant/licensee is responsible for assessing its plant's design and operating history and demonstrating that the approved version of MRP-227 is applicable to the facility. Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE, or B&W) which support MRP-227 and describe the process used for determining plant-specific differences in the design of their RVI [reactor vessel internal] *components or plant operating conditions, which result in different component inspection categories. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227. This is Applicant/Licensee Action Item 1."*

Millstone Unit 2 Compliance

The process used to provide assurance that MPS2 is reasonably represented by the generic industry program assumptions (with regard to neutron fluence, temperature, stress values, and materials used in the development of MRP-227-A (reference 7.3)) is:

1. Identification of typical Combustion Engineering (CE)-designed pressurized water reactor (PWR) RVI components in Table 4-5 of MRP-191 (reference 7.1).
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 (reference 7.1) 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 (reference 7.3) 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 (reference 7.3).

The 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 (reference 7.1) and in the MRP-232 functionality analysis (reference 7.5) based on the following:

1. MPS2 operating history is consistent with the assumptions in MRP-227-A (reference 7.3) with regard to neutron fluence.
   a. The FMECA and functionality analysis for MRP-227-A (reference 7.3) 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 (reference 7.6), 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 (reference 7.1) and meets the requirements for application of MRP-227-A (reference 7.3).
   b. MPS2 has operated under base load conditions over the life of the plant, as stated in the MPS2 AMP (reference 7.6). Therefore, MPS2 satisfies the assumptions in MRP-227-A (reference 7.3) regarding base load operation.

2. The MPS2 RVI components operate between T_{hot} and T_{cold} (reference 7.9), 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 (reference 7.9). 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 [7.3] 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 (reference 7.1), as summarized in (reference 7.9).
   a. No additional components are identified for MPS2 by the comparison of the listing of Millstone Unit 2 RVI components listed in NUREG-1838 (reference 7.8) to the listing in Table 4-5 of MRP-191 (reference 7.1).
   b. MPS2 RVI component materials are consistent with, or nearly equivalent to, those materials identified in Table 4-5 of MRP-191 (reference 7.1) for CE-

designed plants. Where differences exist, there is no impact on the MPS2 RVI (reference 7.9).

c. Design and fabrication of Millstone Unit 2 RVI components are the same as, or equivalent to, the typical CE-designed PWR RVI components (reference 7.9).

4. As stated in the AMP (reference 7.6), 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 (reference 7.3). The components and materials are equivalent to those considered in MRP-191 (reference 7.1). Therefore, the MPS2 RVI components are represented by the assumptions in MRP-191 (reference 7.1), MRP-227-A (reference 7.3), and MRP-232 (reference 7.5), confirming the applicability of the generic FMECA.

Conclusion

MPS2 complies with A/LAI 1 of the SE on MRP-227, Revision 0 (reference 7.2). Therefore, MPS2 meets the requirement for application of MRP-227-A (reference 7.3) as a strategy for managing age-related material degradation in RVI components.

3.0 *SE A/LAI 2: PWR Vessel Internals Components within the Scope of License Renewal MPS2 AMP Text*

The action item text from the SE contained in MRP-227-A (reference 7.3) states:

"As discussed in Section 3.2.5.2 of this SE, consistent with the requirements addressed in 10 CFR 54.4, each applicant/licensee is responsible for identifying which RVI components are within the scope of LR for its facility. Applicants/licensees shall review the information in Tables 4-1 and 4-2 in MRP-189, Revision 1, and Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RVI components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. 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 period of extended operation. This issue is Applicant/Licensee Action Item 2."

MPS2 Compliance

This action item requires comparison of the MPS2 RVI components that are within the scope of license renewal for MPS2 to those components contained in Table 4-5 of MRP-191 (reference 7.1). The MPS2 RVI components within the scope of license renewal from the program AMP (reference 7.6) and NUREG-1838 (reference 7.8) compared favorably to the typical CE-designed PWR RVI components in MRP-191 (reference 7.1). Several components had a different material than that specified in MRP-191 (reference 7.1), but those differences have no effect on the recommended MRP aging strategy. Therefore, no modifications to the program details in MRP-227-A (reference 7.3) need to be proposed (reference 7.9). This supports the requirement that the AMP (reference 7.6) 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 (reference 7.1), will be managed for the period of extended operation.

The generic scoping and screening of the RVI components, as summarized in MRP-191 (reference 7.1) and MRP-232 (reference 7.5), to support the inspection sampling approach for aging management of the RVI components specified in MRP-227-A (reference 7.3), are applicable to MPS2 with no modifications.

Conclusion

MPS2 complies with A/LAI 2 of the SE on MRP-227, Revision 0 (reference 7.2). Therefore, MPS2 meets the requirement for application of MRP-227-A (reference 7.3) as a strategy for managing age-related material degradation in RVI components.

4.0 *SE A/LAI 3: Evaluation of the Adequacy of Plant-Specific Existing Programs*[2]

The action item text from the SE contained in MRP-227-A (reference 7.3) states:

"As addressed in Section 3.2.5.3 in this SE, applicants/licensees of CE and Westinghouse are required to perform plant-specific analysis either to justify the acceptability of an applicant's/licensee's existing programs, or to identify changes to the programs that should be implemented to manage the aging of these components for the period of extended operation. The results of this plant-specific 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 applicant's/licensee's AMP application. The CE and Westinghouse components identified for this type of plant-specific evaluation include: CE thermal shield positioning pins and CE in-core instrumentation thimble tubes (Section 4.3.2 in MRP-227), and Westinghouse guide tube support pins (split pins) (Section 4.3.3 in MRP-227). This is Applicant/Licensee Action Item 3."

MPS2 Compliance

The one component applicable to MPS2 in this LAI is the CE in-core instrumentation thimble tubes. The thermal shield positioning pins are no longer applicable because the thermal shield was removed in 1983 (see program section 2.1).

All in-core instrumentation flux thimble tubes for MPS2 were replaced during refueling outage 2R19 (Fall 2009) after 22.65 (EFPY) of operation. The design change was documented in DM2-01-0044-08. The original zircaloy thimbles had

[2] This section on A/LAI 3 prepared by Dominion Nuclear Connecticut

approximately a 4" gap allowance for zircaloy growth before contacting the lower end of the fuel assembly guide tube fitting. The new replacement thimbles have almost a 14" gap for future growth. 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. The functionality of the incore detector instrumentation is maintained in accordance with Technical Specification 4.3.3.2.

Conclusion

MPS2 complies with A/LAI 3 of the NRC SE on (reference 7.2). Therefore, MPS2 meets the requirement for application of (reference 7.3) as a strategy for managing age-related material degradation in RVI components.

5.0 *SE A/LAI 5: Application of Physical Measurements as Part of the I&E Guidelines for B&W, CE, and Westinghouse Reactor Internals (RI) Components*

The action item text from the SE contained in MRP-227-A (reference 7.3) states:

"As addressed in Section 3.3.5 in this SE, applicants/licensees shall identify plant-specific acceptance criteria to be applied when performing the physical measurements required by the NRC-approved version of MRP-227 for loss of compressibility for Westinghouse hold down springs, and for distortion in the gap between the top and bottom core shroud segments in CE units with core barrel shrouds assembled in two vertical sections. The applicant/licensee shall include its proposed acceptance criteria and an explanation of how the proposed acceptance criteria are consistent with the plants' licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation during the period of extended operation as part of their submittal to apply the approved version of MRP- 227. This is Applicant/Licensee Action Item 5."

MPS2 Compliance

This effort was focused on the core shroud (CS). The CS assembly for MPS2 comprises an upper section and a lower section, as shown in FIGURE 1: CS ASSEMBLY – ELEVATION (REFERENCE 7.9) and FIGURE 2: CS ASSEMBLY – PLAN VIEW (REFERENCE 7.9)

The bottom plate of the upper section sits on the top plate of the lower section. The two sections are attached to one another and to the core support plate (CSP) via eight tie rods. Tapered pins are inserted through the interfacing bottom and top plates to provide lateral restraint and alignment between the two sections.

The elevation of the bottom plate/top plate interface is close to the mid-plane of the core; therefore, it is subjected to high levels of irradiation. The irradiation dose contour plots in these plates indicate that the dose [in displacements per atom (dpa)] is highest at the innermost corners, which are closest to the center of the core. This irradiation produces both gamma heating and potential void swelling in the plates. The gamma heating will produce a temperature gradient between the center of the bottom plate/top plate combination (at the interfacing surfaces) and at the outer surfaces of the bottom plate/top plate combination. These temperature gradients, like the irradiation dose, will be greatest at the innermost corners of the interfacing plates. The void swelling increases with temperature as well as dose. Therefore, it is also greatest at these innermost corner locations.

Per (reference 7.9), this gamma heating and void swelling results in a deflection of the bottom plate relative to the top plate, so that gaps form at the inner and outer peripheries of the bottom plate/top plate interface (see FIGURE 3). Gamma heating and void swelling also cause local increases in plate thickness. Both of these effects are greatest at the innermost corners of the interfacing plates. The functions of the tie rods and tapered pins are not adversely affected by gamma heating and void swelling of the interfacing plates. The tie rods would continue to clamp the upper and lower CS sections together, and the tapered pins would continue to prevent lateral translation of one plate relative to the other. Therefore, plate-to-plate contact would be maintained at the locations of maximum plate thickness [i.e., at the innermost corners, between the inner and outer peripheries where the gaps occur (see FIGURE 3)]. However, additional gaps could form between the two plates away from the innermost corners, where plate thicknesses are smaller, and these gaps could extend through the bottom plate/top plate interface (refer to FIGURE 4).

Based on the preceding discussion, there could be two types of gap between the interfacing plates of the CS upper and lower sections. The gaps at the innermost corners would not extend through the bottom plate/top plate interface; the gaps away from the innermost corners could extend through this interface. Both types of gap would have a thermal contribution and a void swelling contribution. The thermal contribution would only be present during power operation. The void swelling contribution would be present under all conditions, including plant shutdown, during which physical examinations of the CS will be performed.

The maximum total gaps between the interfacing horizontal plates of the upper and lower sections of the MPS2 CS, which occur during plant operation, must be acceptable from both structural and functional standpoints.

CS gaps are calculated in (reference 7.9). The maximum gap during plant shutdown, constituting the acceptance criterion for physical examination of gaps in the MPS2 CS, must be within the range that can be detected by visual examination VT-1. Per MRP-228 (reference 7.4, paragraph 2.3.6.3,b.1), visual examination VT-1 processes shall be demonstrated as capable of resolving lower case characters with heights no greater than 0.044 inches at the maximum examination distance. Based on this requirement, it is conservative to conclude that gaps of 0.044 inches or greater can be detected by visual examination VT-1.

The maximum value for the gap during operation at the end of 60 years between the interfacing plates of the CS upper and lower sections is bounded at 0.212 inches. This maximum gap, which occurs at the innermost corners of these interfacing plates, reflects both differential thermal expansion (from gamma heating) and irradiation-induced void swelling, and also includes the permissible as-fabricated gap. The maximum gap away from the innermost corners is 0.090 inches. The structural and functional effects associated with the presence of these gaps have been evaluated in (reference 7.9), and are acceptable.

Those portions of these total gaps due to differential thermal expansion would only be present during power operation. Those portions of the total gaps due to irradiation-induced void swelling would be present under all conditions, including plant shutdown, during which physical examinations of the CS will be performed.

During plant shutdown, the maximum value for the gap between the interfacing plates of the CS upper and lower sections, reflecting irradiation-induced void swelling, is 0.125 inches ($1/8^{th}$ of an inch). Again, the maximum gap would occur at the innermost corners. The maximum gap away from the innermost corners is 0.063 inches ($1/16^{th}$ of an inch).

Based on these results, a maximum, bounding gap between the interfacing plates of the CS upper and lower subassemblies, as could be present during plant shutdown, is set at $1/8^{th}$ of an inch. This gap is used as an acceptance criterion for the physical examination of gaps in the MPS2 CS.

Conclusion

MPS2 complies with A/LAI 5 of the SE on MRP-227, Revision 0 (reference 7.2). Therefore, MPS2 meets the requirement for application of MRP-227-A [7.3] as a strategy for managing age-related material degradation in RVI components.

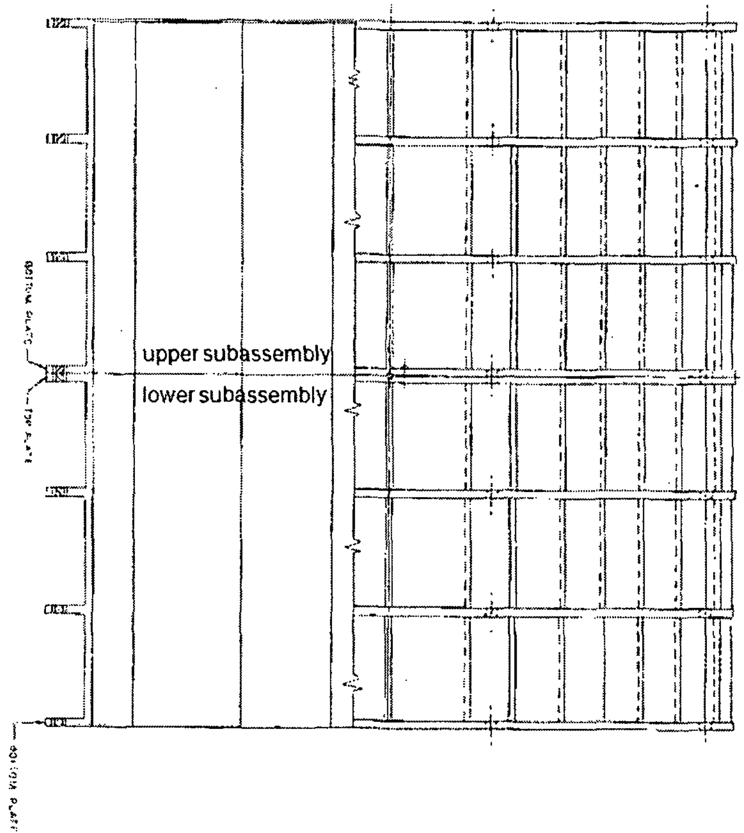


FIGURE 1: CS ASSEMBLY – ELEVATION (REFERENCE 7.9)

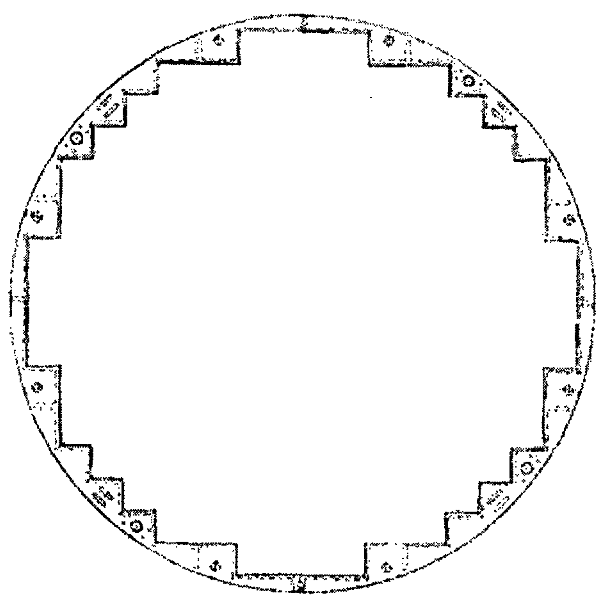


FIGURE 2: CS ASSEMBLY – PLAN VIEW (REFERENCE 7.9)

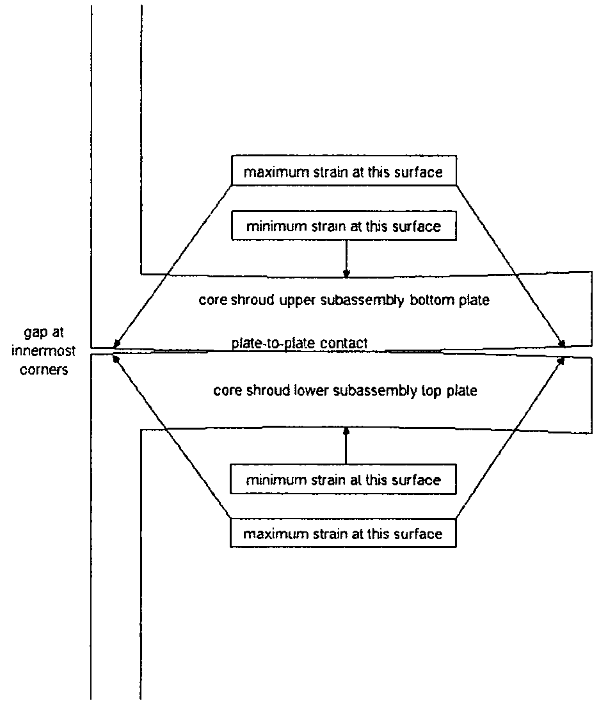
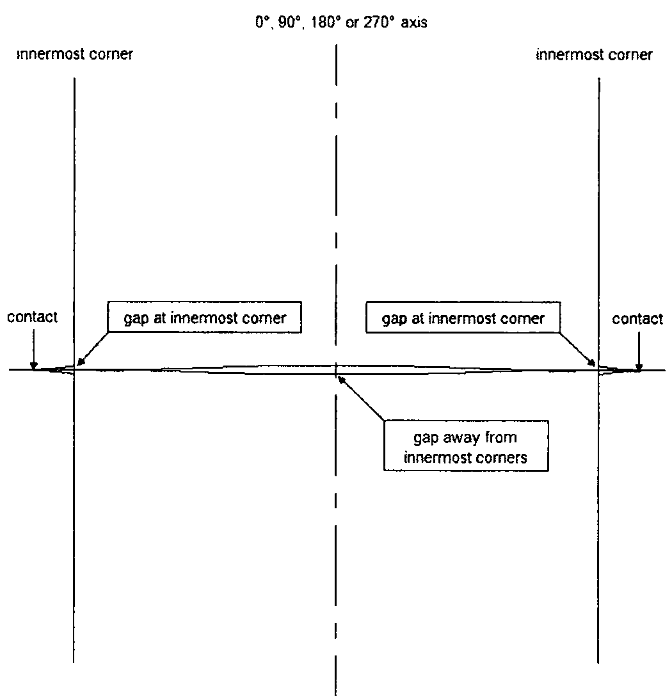


FIGURE 3: GAP BETWEEN CORE SHROUD UPPER AND LOWER SUBASSEMBLIES AT INNERMOST CORNERS



(Viewed from center of core)

FIGURE 4: GAP BETWEEN CORE SHROUD UPPER AND LOWER SUBASSEMBLIES AWAY FROM INNERMOST CORNERS (REFERENCE 7.9)

6.0 *SE A/LAI 7: Plant-specific Evaluation of Cast Austenitic Stainless Steel (CASS) Materials*

The action item text from the SE contained in MRP-227-A [7.3] states:

"As discussed in Section 3.3.7 of this SE, the applicants/licensees of B&W, CE, and Westinghouse reactors are required to develop plant-specific analysis to be applied for their facilities to demonstrate that B&W IMI guide tube assembly spiders and CRGT spacer castings, CE lower support columns, and Westinghouse lower support column bodies will maintain their functionality during the period of extended operation or for additional RVI components that may be fabricated from CASS, martensitic stainless steel or precipitation hardened stainless steel materials. These analyses shall also consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement, and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The requirement may not apply to components that were previously evaluated as not requiring aging management during development of MRP-227. That is, the requirement would apply to components fabricated from susceptible materials for which an individual licensee has determined aging management is required, for example during their review performed in accordance with Applicant/Licensee Action Item 2. 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 applicant/licensee shall include the plant-specific analysis as part of their submittal to apply the approved version of MRP-227. This is Applicant/Licensee Action Item 7."*

MPS2 Compliance

A/LAI 7 from the NRC's final SE on MRP-227, Revision 0 (reference 7.2) states that, for assessment of CASS materials, the licensee or applicant for license renewal may apply the criteria in the License Renewal Issue No. 98-0030 (reference 7.11) as the basis for determining whether the CASS materials are potentially susceptible to the thermal aging mechanism. If the application of the applicable screening criteria for the component material demonstrates that the components are not susceptible to either thermal embrittlement (TE) or irradiation embrittlement (IE), or the synergistic effects of TE and IE combined, then no other evaluation would be necessary.

The MPS2 RVI CASS components and the assessment of their susceptibility to TE are summarized in Table 1. Chemistry for 63 of 68 core support columns was confirmed based on certified material test reports (CMTRs). Based on the criteria of the License Renewal Issue No. 98-0030 (reference 7.11), the 63 columns with CMTR-based chemistry composition are not susceptible to TE. The remaining five columns are potentially susceptible to TE based on the material design specification bounding chemistry as the material confirmation technical basis.

Based on the same criteria from the License Renewal Issue No. 98-0030 (reference 7.11), the MSP2 CASS core support columns are not susceptible to TE. For the core support columns, the MRP-191 assessment for IE remains unchanged, resulting in the core support columns being potentially susceptible to IE. Therefore, the aging management strategies endorsed by MRP-227-A remain applicable.

TABLE 1: SUMMARY OF MILLSTONE UNIT 2 CASS COMPONENTS AND THEIR SUSCEPTIBILITY TO TE (REFERENCE 7.9)

CASS Component	Molybdenum Content (wt %)	Casting	Calculated Ferrite Content(3)	Susceptibility to TE (Based on the NRC Criteria (reference 7.11))
Core Support Columns, Four-Leg	Low, 0.5 Maximum	Static	$\leq$20%	Thirty-five out of 36 columns are not susceptible to TE.[1]
Core Support Columns, Three-Leg	Low, 0.5 Maximum	Static	$\leq$20%	Eight out of eight columns are not susceptible to TE.
Core Support Columns, Two-Leg	Low, 0.5 Maximum	Static	$\leq$20%	Twenty out of 24 columns are not susceptible to TE.[2]

Notes:

1. One column is potentially susceptible to TE based on the material specification bounding chemistry.
2. Four columns are potentially susceptible to TE based on the material specification bounding chemistry.
3. The maximum calculated ferrite content for the core support columns with CMTRs is 9.84%. The calculated ferrite content [for core support columns without CMTRs] is potentially greater than 20% based on the bounding material specification chemistry.

Conclusion

The results of this core support column CASS evaluation do not conflict with the MRP-227-A strategy for aging management of RVI (reference 7.3). 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 (reference 7.3) process. Basis assumptions for the components of MRP-227-A (reference 7.3) methodology are still valid. It is concluded that continued application of the strategy of MRP-227-A (reference 7.3] will meet the requirement for managing age-related degradation of the Millstone Unit 2 CASS core support columns.

7.0 *References*

7.1. Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191). EPRI, Palo Alto, CA: 2006. 1013234.

7.2. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0). EPRI, Palo Alto, CA: 2008. 1016596.

7.3. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A). EPRI, Palo Alto, CA: 2011. 1022863.

7.4. Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228). EPRI, Palo Alto, CA: 2009. 1016609.

7.5. Material Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals (MRP-232). EPRI, Palo Alto, CA: 2008. 1016593.

7.6. DNC Document, MP-LR-3711/MP-LR-4711, Rev. 7, "License Renewal Project Aging Management Program Inservice Inspection: Reactor Vessel Internals Millstone Power Station," June 14, 2012.

7.7. Westinghouse Letter, LTR-AMER-MKG-12-1438, Rev. 1, "Revised Order Acknowledgement and Clarifications to Purchase Order 70245229," August 2, 2012.

7.8. U.S. NRC Document, NUREG-1838, "Safety Evaluation Report Related to the License Renewal of the Millstone Power Station, Units 2 and 3," October 2005.

7.9. Westinghouse Letter, LTR-RIAM-13-51, Rev. 0, "Request for Transmittal of Comment Resolutions for the Millstone Power Station Unit 2 Draft Task 1 Summary Letter CML-13-6 for A/LAIs 1, 2, 5, and 7," May 30, 2013.

7.10. DNC Purchase Order 70245229, dated June 29, 2012.

7.11. NRC Letter, "License Renewal Issue No. 98-0030, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components," May 19, 2000.