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

December 19, 2014

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1 – STAFF ASSESSMENT OF THE REACTOR VESSEL INTERNALS INSPECTION PLAN (TAC NO. MF1459)

Dear Mr. Pacilio:

By letter dated April 16, 2012,[1] as supplemented by letters dated April 17, 2013,[2] and November 6, 2013,[3] Exelon Generation Company, LLC (the licensee), pursuant to license renewal (LR) Activity No. 36 in Appendix A of NUREG-1928, "Safety Evaluation Report Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1 [TMI-1],"[4] submitted a Reactor Vessel Internals (RVI) Inspection Plan. The RVI Inspection Plan is based on MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)."[5]

The U.S. Nuclear Regulatory Commission (NRC) staff's review of the licensee's RVI Inspection Plan is provided in the enclosed staff assessment (SA). Proprietary information in the SA is denoted by red text in double bold brackets **[[ ]]**.

The NRC staff concludes that the licensee's RVI Inspection Plan is acceptable because it is consistent with the inspection and evaluation guidelines of MRP-227-A, the licensee has addressed six of the eight licensee action items specified in MRP-227-A appropriately, and has made regulatory commitments to submit analyses fulfilling the other two action items on an acceptable schedule.

Enclosure 2 herewith contains sensitive unclassified non-safeguards information. When separated from Enclosure 2, this document is decontrolled.

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[1] Agencywide Documents Access and Management System (ADAMS) Accession No. ML12108A029.
[2] ADAMS Accession No. ML13108A004.
[3] ADAMS Accession No. ML13317A931.
[4] ADAMS Accession No. ML092950458.
[5] ADAMS Accession No. ML120170453.

Consequently, LR Commitment No. 36 from Appendix A of the NRC's SE of the TMI-1 LR Application is considered fulfilled. The NRC staff will review the information provided by the licensee in accordance with the two new regulatory commitments documented in the licensee's letter dated November 6, 2013, for adequacy when received. The NRC staff's approval of the TMI-1 RVI Inspection Plan does not reduce, alter, or otherwise affect current American Society of Mechanical Engineers Code, Section XI Inservice Inspection (ISI) requirements, or any TMI-1 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.

If you have any questions concerning this matter, please contact the TMI-1 Project Manager, Mr. John Lamb, at (301) 415-3100 or via e-mail at John.Lamb@nrc.gov.

Sincerely,

Meena K. Khanna, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosures:
1. Non-Proprietary Staff Assessment
2. Proprietary Staff Assessment

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STAFF ASSESSMENT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO REACTOR VESSEL INTERNALS INSPECTION PLAN

EXELON GENERATION COMPANY, LLC

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-289

Proprietary information pursuant to
Title 10 of the *Code of Federal Regulations* (10 CFR), Section 2.390
has been redacted from this document. Redacted information is identified by blank space enclosed within double brackets as shown here [[ ]].

1.0 INTRODUCTION

By letter dated April 16, 2012 (Ref. 1), as supplemented by letters dated April 17, 2013 (Ref. 2), and November 6, 2013 (Ref. 3, 4), Exelon Generation Company, LLC (the licensee), pursuant to license renewal (LR) Activity No. 36 in Appendix A of NUREG-1928, "Safety Evaluation Report Related to the License Renewal of Three Mile Island Nuclear Station, Unit 1 (TMI-1)," (Ref. 5) submitted a Reactor Vessel Internals (RVI) Inspection Plan. The RVI Inspection Plan is based on MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," (Ref. 6).

Appendix A, "Commitments for License Renewal of TMI-1," of the NUREG-1928, (Ref. 5) contains commitments associated with the license renewal (LR) of TMI-1. Commitment No. 36 states:

> The PWR [Pressurized Water Reactor] Vessel Internals Program will commit to the following activities: 1. Participate in the industry programs for investigating and managing aging effects on reactor internals. 2. Evaluate and implement the results of the industry programs as applicable to the reactor internals. 3. Upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals to the NRC for review and approval.

The period of extended operation (PEO) for TMI-1 began April 20, 2014. Accordingly, the licensee submitted its RVI Inspection Plan prior to 24 months before entering the PEO. The licensee stated that the "Inspection Plan for the Three Mile Island Unit 1 Reactor Vessel Internals" is based upon the LR Commitment No. 36 and MRP-227-A. The licensee also stated that its RVI Inspection Plan is complete, but changes to the plan may result from ongoing

Enclosure 1

discussions between the Materials Reliability Program (MRP) and the NRC in resolving final MRP-227-A inspection and evaluation (I&E) methodologies. The licensee stated they were currently working with the Pressurized Water Reactor Owners Group (PWROG) to address the action items. In its submittal dated April 16, 2012, the licensee committed to provide an update of the PWROG progress in evaluating licensee action items 2, 6, and 7, by April 19, 2013. In Appendix D to its RVI Inspection Plan, the licensee provided three new commitments that correspond to the open action items described above. The licensee stated that it considers that LR Commitment No. 36 was satisfied via the submittal of the RVI Inspection Plan and the new commitments made in Appendix D. In its letter dated April 17, 2013, the licensee provided the update on action items 2, 6 and 7. For licensee action item 2, the licensee provided the resolution of the item. For licensee action items 6 and 7, the licensee committed to provide the analyses resolving these items by certain dates.

The licensee also stated in its submittal dated April 16, 2012, that the impact of any future power uprates on this RVI Inspection Plan was not addressed, but that the impact of future power uprates at TMI-1 will be addressed, as appropriate, in the license amendment request to implement the proposed uprate.

In its submittal dated April 16, 2012, (Ref. 1), the licensee indicated that all initial MRP-227-A inspections were planned for the fall 2015 refueling outage. However, in its letter dated November 6, 2013 (Ref. 3), the licensee stated that examination of RVI components that are accessible only while the Core Support Assembly (CSA) is removed will be performed during the 2015 refueling outage (RFO), and that other RVI components will be examined during the 2017 RFO.

2.0 REGULATORY EVALUATION

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

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

On January 12, 2009, the Electric Power Research Institute (EPRI) submitted for NRC staff review and approval Revision 0 of MRP Report MRP-227 (Ref. 7), which was intended as guidance for applicants in developing their plant-specific AMP for RVI components. The scope of components considered for inspection under the guidance of MRP-227, Revision 0, includes core support structures, which are typically denoted as Examination Category B-N-3 by Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), and those RVI components that serve an intended safety function consistent with the criteria in 10 CFR 54.4(a)(1). The scope of the program does not include consumable items such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation because these components are not subject to an AMR, as defined in 10 CFR 54.21(a)(1).

Revision 1 to the final safety evaluation (SE) regarding MRP-227, Revision 0, was issued on December 16, 2011 (Ref. 8). This SE contains specific conditions on the use of the topical report and applicant/licensee action items that must be addressed by those utilizing the topical report as the basis for a submittal to the NRC. On January 9, 2012, EPRI published the NRC approved version of topical report MRP-227-A (Ref. 6). MRP-227-A contains a discussion of the technical basis for the development of plant-specific AMPs for RVI components in PWR vessels and also provides inspection and evaluation guidelines for PWR applicants to use in their plant-specific AMPs. MRP-227-A provides the basis for renewed license holders to develop plant-specific inspection plans to manage aging effects on RVI components, as described by their LR commitment.

Subsequent to the submittal of MRP-227 and prior to the issuance of the SE for MRP-227, NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," Final Report (Ref. 9), was issued, providing new AMR line items and aging management guidance in AMP XI.M16A, "PWR Vessel Internals." This AMP was based on NRC staff expectations for the guidance to be provided in MRP-227-A. License Renewal Interim Staff Guidance LR-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors," (Ref. 10) was published on May 28, 2013, to update AMP XI.M16A for consistency with MRP-227-A.

TMI-1 License Condition 2.c.(21) states the following:

> The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), describes certain future activities to be completed prior to and/or during the period of extended operation. The licensee shall complete these activities in accordance with Appendix A of NUREG-1928, "Safety Evaluation Report Related to the License Renewal of Three Mile Island, Unit 1," dated, October, 2009. The licensee shall notify the NRC in writing when activities to be completed prior to the period of extended operation are complete and can be verified by NRC inspection.

2.1 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: (1) stress-corrosion cracking (SCC); (2) irradiation-assisted stress-corrosion cracking (IASCC); (3) wear; (4) fatigue; (5) thermal aging embrittlement (TE); (6) irradiation embrittlement (IE); (7) irradiation-enhanced stress relaxation and creep; and (8) 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 category 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 (RFOs) 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[1] for the inspection. The inspection strategy generally employs VT-3 level visual examinations to evaluate general component condition, EVT-1 level visual examinations to identify surface

[1] For Babcock & Wilcox (B&W) - design RVI such as TMI-1, inspection schedules, methods, and inspection coverage are defined in MRP-227-A, Table 4-1 for Primary category components and Table 4-4 for Expansion category components.

breaking flaws, and VT-1 level visual examination to identify surface discontinuities such as gaps. Cracking in baffle-former bolts is monitored with ultrasonic (UT) techniques.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the RVI Inspection Plan 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 limitations and applicant/licensee action items 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 Inspection Plan focused on determining whether the plan is consistent with the recommendations of MRP-227-A, and that it addresses the plant-specific applicant/licensee action items.

3.1 Industry and TMI-1 Programs and Activities (Section 4 of the TMI-1 Inspection Plan)

Section 4 of the TMI-1 inspection plan contains pertinent TMI-1 and industry programs and activities used for the development and implementation of MRP-227-A. It also contains discussion of specific technical issues which affect the TMI-1 inspection plan based on MRP-227-A. The NRC staff's evaluation of these issues follows.

3.1.1 ASME Code Section XI Inservice Inspection Program

In this subsection, the licensee described its ASME Code, Section XI Inservice Inspection (ISI) Program which is credited as an existing program implementing ASME Code Section XI, IWB-2500, Inspection Category B-N-3 inspections which consist of visual VT-3 examinations of core support structures (CSS), once every 10-year inspection interval. The licensee also stated the next TMI-1 ASME Code, Section XI 10-year ISI inspection requiring core support assembly (CSA) removal is currently scheduled for the fall 2015 RFO. As noted in Section 1.0 of this staff assessment (SA), the MRP-227-A initial inspections requiring CSA removal are also scheduled for the fall 2015 RFO.

3.1.2 Time-Limited Aging Analyses

The licensee stated that the TMI-1 RVI AMP includes three TLAAs that were evaluated and dispositioned in the LR Application (Ref. 11), and that the three TLAAs for the RVI components include low-cycle fatigue, high-cycle fatigue, and neutron embrittlement. The licensee further stated that low-cycle fatigue is managed by the Fatigue Monitoring Program (LRA Section 4.3.4) and that high-cycle fatigue (flow induced vibration) is projected through the PEO (LRA Section 4.3.5).

Finally, the licensee stated that the LRA committed to manage neutron embrittlement by participating in, evaluating, and implementing the results of RVI industry programs that were then in development. The licensee stated that those programs (i.e., MRP-227-A) are now approved and are implemented via this inspection plan as part of the TMI-1 RVI AMP and that neutron embrittlement is one of the aging effects considered in the development of MRP-227-A (see Section 3.2.6 of MRP-227-A).

The NRC staff finds that the licensee's use of its RVI Inspection Program in accordance with MRP-227-A is acceptable and fulfills its commitment to manage neutron embrittlement via implementation of the industry program. No further action is necessary.

The licensee also stated that the responses to applicant/licensee action items 6 and 7 from the NRC SE for MRP-227-A may result in additional analyses justifying the use of the TMI-1 RVI and that as these are developed, any time-dependent aspects of these analyses will be identified and addressed through the PEO.

3.1.3 Upper Core Barrel (UCB) Bolt and Lower Core Barrel (LCB) Bolt Analyses

In Section 4.1.6 of the RVI Inspection Plan, the licensee indicated that they have performed analyses to determine the tolerance of the RVI to broken or degraded UCB and LCB bolts. The analyses used the stress limits of the ASME Code, Section III, Subsection NG for threaded fasteners. Five different hypothetical combinations of degraded bolts were analyzed, which showed that large numbers of degraded bolts could be tolerated provided the degraded bolts are not adjacent. If the degraded bolts are adjacent, the number of degraded bolts that can be tolerated is fewer.

In Request for Additional Information (RAI) 1, the NRC staff asked the licensee whether the UCB and LCB bolt analyses are consistent with the recommended methodology of WCAP-17096-NP, Revision 2 (Ref. 12) for these bolts, and if not, whether the analyses would be revised if degradation is found in the bolts. In its November 6, 2013, non-proprietary response to RAI 1 (Ref. 3, Attachment 4), the licensee stated that the analyses would be consistent with the methodology of WCAP-17096-NP, Revison 2, as modified by the NRC staff's draft SE of the WCAP. The NRC staff finds this response acceptable since all industry comments have been resolved for the B&W component analyses, thus the methodology used for the UCB and LCB bolt analyses will be consistent with an NRC-approved methodology. Therefore, RAI 1 is resolved.

3.1.4 Past RVI Inspections

In Section 4.1.7 of the inspection plan, the licensee described RVI inspections that have been performed in the past at TMI-1. These include vent valve inspections, UCB bolt UT examinations, and core clamping measurements. All of the UCB bolts were examined in 2009 via UT with no recordable indications found. UT examination of the UCB bolts was also performed in 1991 with no recordable indications. The core clamping measurements were conducted in 2010 to fulfill the examination requirements for a one-time physical measurement of the differential height of top of the plenum rib pads to the reactor vessel seating surface. This measurement was taken with the plenum cover weldment rib pads, plenum cover support flanges, and CSS top flange inside the reactor vessel, but with the fuel assemblies removed per Section 4.3.1 of MRP-227-A. The licensee stated that the core clamping summary document concluded that there was no evidence of wear occurring during the service period and the measurements were acceptable. Therefore, the core clamping measurements at TMI-1 described above meet the one-time physical measurement requirement of MRP-227-A.
In its November 6, 2013, response to RAI 2 (Ref. 3), the licensee confirmed that the 2009 UCB bolt examinations constitute the initial MRP-227-A required inspection of the UCB bolts for TMI-1, which were required within two RFOs of January 1, 2006. The licensee also noted that MRP-227-A and the Final Report for MRP-228, "Materials Reliability Program: Inspection Standard for PWR Internals," (Ref. 17) were not completed in time to be fully implemented at

TMI-1 prior to this inspection; thus, the additional requirements for training and equipment as designated in MRP-228 were not fully applied. The licensee also stated that the examination technique used in 2009 was demonstrated at EPRI in Charlotte, North Carolina. The licensee additionally stated that a technical justification was completed for the examination process that was in accordance with MRP-228, Revision 0, and ASME Section V, Article 14, 2004 Edition, and that the examination technique was determined to be capable of detecting flaws greater than 28 percent cross sectional area with no false calls. The licensee also stated in its response to RAI 2 that the UCB bolts would be re-inspected in 2015 when the CSA is removed for the 10-year ISI inspections, and that these inspections will be fully compliant with MRP-227-A and MRP-228.

The NRC staff finds the response to RAI 2 acceptable because the licensee performed an inspection that was compliant with the guidance available at the time, followed the requirements of MRP-228 to the extent possible by preparing a technical justification, and will re-inspect the bolts at the next opportunity in full compliance with MRP-227-A requirements. Therefore, RAI 2 is resolved.

3.1.5 Consistency of the TMI-1 RVI Inspection Plan with MRP-227-A

In addition to its review of the licensee action items, the NRC staff reviewed the information provided in TMI's RVI Inspection Plan for general consistency with MRP-227-A. Section 1.0 of MRP-227-A is the executive summary and is therefore not specifically addressed below.

3.1.5.1 Background (Section 2.0 of MRP-227-A)

Section 2.0 of MRP-227-A contains general background, a discussion of the scope, and applicability information.

The NRC staff verified the scope of the applicant's RVI program is consistent with that described in MRP-227-A, Section 2.0 (i.e., is limited to the RVI structural components). The applicability information of Section 2.4 of MRP-227-A is addressed in detail in Section 3.3.1 of this SE under the discussion of licensee action item 1.

3.1.5.2 Component Categorization and Aging Management Strategy Development (Section 3.0 of MRP-227-A)

This section of MRP-227-A provides information on the RVI design characteristics for the three different nuclear steam supply system (NSSS) designs, an overview of the applicable degradation mechanisms and effects, and a description of the process used to develop the aging management strategy recommendations of MRP-227-A, including screening for degradation mechanisms, categorization, and FMECA. Since this information is generic, it does not have a plant-specific parallel.

3.1.5.3 Aging Management Requirements (Section 4.0 of MRP-227-A)

This section of MRP-227-A provides the tables with the recommended inspections (technique, schedule of initial and subsequent inspections, and inspection coverage) for RVI components in the Primary, Expansion, and Existing Programs categories. The NRC staff checked the corresponding information for TMI-1 contained in Appendixes A and B of the RVI Inspection Plan, and found the information to be consistent with MRP-227-A, Table 4-1 (B&W Plant

Primary Components") and 4-4 ("B&W Plants Expansion Components"). There are no Existing Programs components for B&W-design RVI.

3.1.5.4 Examination Acceptance Criteria and Expansion Requirements

Table 5-1 of MRP-227-A provides the acceptance criteria for the Primary and Expansion category components of B&W-design RVI. The expansion criteria define whether the inspection results for the Primary components trigger inspections of the expansion components. The NRC staff checked the corresponding information in Appendix C of the RVI Inspection Plan and finds it is consistent with the information of Table 5-1 of MRP-227-A.

3.1.5.5 Evaluation Methodologies (Section 6.0 of MRP-227-A)

This section of MRP-227-A provides recommended flaw evaluation methods to be used when the examinations recommended in Section 4.0 reveal relevant conditions. See Section 3.2 for the NRC staff's evaluation of the licensee's proposed evaluation methodologies.

3.1.5.6 Implementation Requirements (Section 7.0 of MRP-227-A)

Section 7 of MRP-227-A provides a summary of the implementation requirements. The implementation requirements are defined by the latest edition of Nuclear Energy Institute (NEI) Implementation Protocol NEI 03-08, "Guideline for the Management of Materials Issues," (Ref. 19) which includes implementation categories used in MRP-227-A including: (a) mandatory, which requires implementation of the guidelines at all plants; (b) needed, which provides an option for implementing the guidelines wherever possible or implementing alternative approaches; or (c) good practice, which recommends implementation of the guidelines as an option whereby significant operational and reliability benefits can be achieved at a given plant. Failure to meet a "needed" or a "mandatory" requirement is a deviation from the guidelines and a written justification for deviation must be prepared and approved as described in Appendix B to NEI 03-08. A copy of the deviation is sent to the MRP so that, if needed, improvements to the guidelines can be developed. A copy of the deviation is also sent, for information, to the NRC.

Based on the above, the NRC staff finds that the RVI Inspection Plan for TMI-1 is consistent with MRP-227-A, with respect to scope, examination inspection requirements and expansion criteria, acceptance criteria, evaluation methodologies, and implementation requirements.

3.2 TMI-1 RVI AMP Evaluation

The licensee evaluated the ten attributes of its AMP for RVI against the corresponding attributes from the GALL Report, Revision 2, Chapter XI.M16A, "PWR Vessel Internals Program Description," in Section 5.0 of the RVI Inspection Program. The attributes are: program scope, preventive actions, parameters monitored/inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience. The licensee concluded that the TMI-1 Reactor Internals AMP is consistent with the NUREG-1801 XI.MI6A program with no exceptions and no enhancements.

The NRC staff reviewed the licensee's evaluation of the ten AMP attributes described above against the criteria of the GALL Report, Revision 2, Chapter XI.M16A. The GALL Report, Revision 2, Chapter XI.M16A, states, under "Detection of Aging Effects," that the VT-3 visual methods may be applied for the detection of cracking only when the flaw tolerance of the component or affected assembly, as evaluated for reduced fracture toughness properties, is known and has been shown to be tolerant of easily detected large flaws, even under reduced fracture toughness conditions. LR Interim Staff Guidance LR-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors," (Ref. 10) modified the statement regarding VT-3 examination such that a flaw tolerance evaluation would only be required for non-redundant components. In the licensee's evaluation of consistency with the detection of aging effects attribute of GALL, Revision 2, Chapter XI.M16A, the licensee stated that VT-3 examinations will be used to detect cracking only after evaluation of the flaw tolerance of the component or affected assembly, under reduced fracture toughness conditions, has been shown to be tolerant of easily detected flaws. The NRC staff notes that there are six TMI-1 Primary components and three Expansion components for which visual VT-3 examination is specified for detection of cracking (consistent with MRP-227-A). Since these flaw evaluations were not included in the RVI Inspection Plan (there is no requirement to do so), in RAI 4 the NRC staff requested the following of the licensee: (1) identify which components to be inspected with VT-3 will require flaw evaluations; (2) justify considering any of the components as redundant, and thus not performing a flaw evaluation; (3) describe how the flaw tolerance evaluations of the TMI-1 Primary and Expansion components will be documented and communicated to the NRC staff to support the staff's review; (4) confirm that these flaw tolerance evaluations will be completed prior to the initial inspections of the TMI-1 Primary components; and (5) discuss whether the flaw tolerance evaluations of the control rod guide tube (CRGT) spacer castings and incore monitoring instrumentation (IMI) guide tube spiders will be part of the plant-specific evaluation of cast austenitic stainless steel (CASS) components. The licensee committed to submit these flaw tolerance evaluations to the NRC by fall 2015.

In its November 6, 2013, response to RAI 4 (Ref. 3, 4), the licensee identified the 6 Primary and 3 Expansion components for which MRP-227-A specifies VT-3 inspections to detect cracking. In response to Item 1 of RAI 4, Table 2-6 of the licensee's response indicates that only one of the 6 Primary components would require a flaw evaluation. In response to Item 2 of RAI 4, the licensee provided a discussion justifying the redundancy of each component determined to be exempt from a flaw evaluation. The licensee generally determined the components to be redundant based on the fact that there is a population of like components supporting the same function, or because the design function of the component is relatively minor (for example, the Alloy X-750 lower grid dowel welds – the dowels cannot fall out of the lower grid even if the weld fails due to the orientation of the dowels). In addition, certain welds can perform their function if they are partially cracked. Also, in some cases the components provide an alternate method of attachment; for example, the lower grid pad-to-rib section welds, which are a backup to the Alloy X-750 dowels and stainless steel capscrews. Some components can also crack and still perform their design function. For example, each IMI spider casting has four arms, one of which could completely fracture and still allow the casting to perform its design function. The NRC staff compared the licensee's redundancy arguments with information on the affected components in MRP-227-A and Revision 1 of MRP-231, "Materials Reliability Program: Aging Management Strategies for B&W PWR Internals," (Ref 20) and finds that the arguments presented are consistent with other information on design, expected aging mechanisms, and the functionality analysis results for these components, and are therefore acceptable.

In response to RAI 4 Items 3 and 4, the licensee stated that for components requiring flaw tolerance evaluations, the evaluations will be completed and available onsite prior to the outage during which the inspections of the affected components will be performed, but that per MRP-227-A and MRP-228 there is no requirement that these analyses be submitted to the NRC. The only component identified as requiring a flaw tolerance evaluation is the core barrel assembly: baffle plates, which will be inspected in 2017 per Table 4-1 of the RAI 4 response. The licensee also indicated in response to Item 5 of RAI 4 that the flaw tolerance evaluations of the CASS components subject to VT-3 would be separate evaluations and not part of the plant-specific evaluation of CASS the licensee committed to provide in its response to action item 7. However, in response to action item 2 of RAI 4, both CASS components were determined to be redundant. Therefore, no flaw tolerance evaluations of CASS components will be necessary. The NRC staff finds the licensee's response to RAI 4 acceptable because it identifies the components requiring a flaw evaluation, justifies the redundancy of components subject to VT-3 inspection for cracking, and confirmed that required flaw tolerance evaluations will be completed prior to the affected component inspections. Therefore, RAI 4 is resolved.

Also, in its description of the acceptance criteria attribute the licensee stated that components with degradation exceeding the examination acceptance criteria will be evaluated per MRP-227-A, but will also use the supplemental guidance of WCAP-17096, including any guidance resulting from the ongoing NRC review of that report. The NRC staff finds the licensee's proposed evaluation methodologies to be acceptable, provided the licensee uses MRP-227-A recommended evaluation methodologies and/or the guidance of the NRC-approved version of WCAP-17096 when available.

The GALL Report, Revision 2, Chapter XI.M16A, under "Operating Experience," states that a summary of observations of RVI aging degradation to date is provided in Appendix A of MRP-227-A. The applicant is expected to review subsequent operating experience (OE) for impact on its program or to participate in industry initiatives that perform this function. In its discussion of the OE element, the licensee identified B&W RVI-specific OE, but not TMI-1-specific OE. Therefore, in RAI 5, the NRC staff requested that the licensee identify any TMI-specific OE, the details of the OE if not contained in MRP-227-A, Appendix A, and any changes to the TMI-1 RVI Inspection Plan made as a result of the OE. The licensee's response to RAI 5 dated November 6, 2013, indicated that there was no TMI-1 degradation described in MRP-227-A or that occurred subsequent to the publication of MRP-227-A. The response noted the presence of upset metal on several components attributed to the plenum assembly being out of level during removal or installation. The licensee does not consider this aging degradation. The response also described degradation of two in-vessel components that are outside the scope of MRP-227-A. The NRC staff reviewed this information and finds that the licensee adequately addressed plant-specific OE. Therefore, RAI 5 is resolved.

Based on its review, as supplemented by the responses to RAI 4 and 5, the NRC staff found the attributes of the licensee's program are consistent with the GALL Report, Revision 2, with no exceptions and no enhancements. The NRC staff also notes that the licensee included implementation of a low-leakage core design as a preventive action in addition to the Primary Water Chemistry Program. With regard to components exhibiting degradation exceeding the examination acceptance criteria of MRP-227-A, the licensee will use the supplemental guidance of WCAP-17096, including any guidance resulting from the ongoing NRC review of that report.

3.3 Licensee Action Items from SE of MRP-227, Revision 0

The NRC staff's final SE of MRP-227, Revision 0 (Ref. 8), contained 8 plant-specific applicant/licensee action items. The NRC staff determined that licensee action items 1, 2, 4, 6, 7, and 8 are applicable to TMI-1 and licensee action items 3 and 5 are not applicable to TMI-1.

3.3.1 Licensee Action Item 1

Per Section 4.2.1 of Reference 8, each licensee is responsible for assessing its plant's design and operating history and demonstrating that MRP-227-A is applicable to the facility. Each licensee shall refer, 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, Combustion Engineering (CE), or B&W) which support MRP-227-A. The licensee shall also describe the process used for determining plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection categories. Finally, the licensee shall submit this evaluation for NRC review and approval as part of its application to implement MRP-227-A.

3.3.1.1 Licensee Evaluation

In Section 4.2.2.1 of the RVI Inspection Plan, the licensee listed the assumptions found in Section 2.4 of MRP-227-A, followed by their applicability to TMI-1 as follows:

- 30 years of operation with high leakage core loading patterns (fresh fuel assemblies loaded in peripheral locations) followed by implementation of a low-leakage fuel management strategy for the remaining 30 years of operation.

  The fuel management program for TMI-1 changed from a high to a low leakage core loading pattern prior to 30 years of plant operation. This change was started in TMI-1 Cycle 6 (1987) and has been continually implemented through the most recent fuel cycle, TMI-1 Cycle 19 (2011). This change is considered to be a preventative action to lessen the effects of aging on the TMI-1 RV Internals. TMI-1 will continue to use low-leakage core loading pattern.

- Base load operation, i.e., typically operates at fixed power levels and does not usually vary power on a calendar or load demand schedule.

  TMI-1 operates as a base load unit.

- No design changes beyond those identified in general industry guidance or recommended by the original vendors.

  MRP-227-A states that the requirements are applicable to all U. S. PWR operating plants as of May 2007 for the three designs (i.e., B&W, Westinghouse, and CE) considered. No modifications have been made to the TMI-1 RV Internals since May 2007.

The licensee also stated that Section 4 of MRP-190, "Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals (MRP-190)," (Ref. 13) (the FMECA) contains 6 assumptions and observations and that, these assumptions are either bounding or methodological, and do not require plant-specific verification for each of the B&W-designed operating units. The licensee also stated that TMI-1 intends to apply for a measurement uncertainty recapture (MUR) power uprate in the future, and that the impact of that power uprate or any other power uprates will be formally reviewed and reported as part of the license amendment request for the power uprate.

3.3.1.2 NRC Staff Assessment

In its response to action item 1, the licensee addresses the three general assumptions made in the analyses used to develop the MRP-227-A inspection. The licensee's response confirms that TMI-1 has operated consistently with the 3 general assumptions stated in MRP-227-A Section 2.4: (1) switch to low-leakage core at or before 30 calendar years of operation; (2) operation as a base load unit; and (3) no design modifications other than those recommended by the NSSS vendor or general industry guidance. The licensee further stated that it would continue to implement a low-leakage core design, and will evaluate the effects of any future power uprates on the RVI Inspection Program.

However, MRP-227-A, Section 2.4 also states:

> The guidelines are based on a broad set of analyses about plant operation, which encompass the range of current plant conditions for the U.S. domestic fleet of PWRs. The functionality assessments and supporting aging management strategies in MRP-231 and MRP-232 provide the basis for these guidelines. These evaluations were based on representative configurations and operational histories, which were generally conservative, but not necessarily bounding in every parameter.

In its review of MRP-227, Revision 0, the NRC staff did not endorse the three basic assumptions above as sufficient to verify plant-specific applicability of the guidelines. Section 3.2.5.1 of the NRC staff's final SE of MRP-227-A provides additional background on the concerns regarding plant-specific applicability verification. The NRC staff performed an independent assessment to determine if plant-specific information could potentially affect the applicability of MRP-227-A to TMI-1.

As described in the report summary for MRP-227-A, the key steps in development of the guidelines are:

1. development of screening criteria, with susceptibility levels for the eight postulated aging mechanisms relevant to reactor internals and their effects;

2. initial component screening and categorization, using susceptibility levels and FMECA (failure modes, effects, and criticality analysis) to identify the relative ranking of components;

3. functionality assessment of degradation for components and assemblies of components; and

4. aging management strategy development combining results of the functionality assessment with component accessibility, operating experience, 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.

For B&W plants, the screening process is documented in MRP-189, Rev. 1, "Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items (MRP-189-Rev. 1)," (Ref. 15); FMECAs are documented in MRP-190 (Ref. 13); functionality analyses are described in MRP-229, Rev. 3, "Materials Reliability Program: Functionality Analysis for Babcock & Wilcox Representative PWR Internals (MRP-229-Rev. 3)," (Ref. 14); and generic aging management strategies are documented in MRP-231, Rev. 2.

Section 2.2 of the NRC staff's SE of MRP-227, Rev. 0 (Ref. 8), notes that the FMECA was a qualitative process that included expert elicitation by technical experts, and that expert elicitation was used for developing the technical basis for categorization of various RVI components under different categories based on the combination of the likelihood of component degradation due to one or more of the eight degradation mechanisms, and the severity of safety consequences. Based on this, the NRC staff concludes that the FMECA results are unlikely to be affected by plant-specific variations in parameters such as neutron fluence, unless additional aging mechanisms could screen in as a result of the variation, or aging mechanisms becoming significantly more severe.

According to MRP-227-A, Section 3.3.1, the most affected PWR internals were placed into Category C based upon the FMECA results, while the components that are only moderately affected were placed into Category B. In addition, the FMECA process determined that some components not initially in Category A were sufficiently unaffected by consequences to be subsequently placed into Category A. In addition to this categorization using FMECA, a more refined assessment involved a functionality assessment of some of the components other than Category A components with the intent to determine the tolerance of components and systems of components to aging degradation effects. When the functionality assessments were completed, all PWR internals were placed into the four functional groups (Primary, Expansion, Existing Programs, or No Additional Measures).

Since the final categorization of RVI components largely depended on the results of the FMECA and functionality analyses, the NRC staff's evaluation of plant-specific applicability of MRP-227-A to TMI-1 focused on identification of the plant-specific information which could affect the conclusions of the FMECAs and generic functionality analyses supporting MRP-227-A.

Any RVI component item designated as Category C, or later as Primary, in accordance with the new categorization need not be reviewed because it is already in the highest risk category. Therefore, the NRC staff's review focused on examination of the RVI component items in Table 3-1 of MRP-227-A to determine whether the RVI component items designated as Category B could be reclassified as Category C based on plant-specific variations in parameters such as neutron fluence, temperature, and stress, instead of the generic assumptions used in the screening, FMECA, and functionality analyses.

According to this principle, the following 12 RVI component items designated as Category B in Table 3-1 needed to be reviewed (except for those that do not apply to TMI-1):

- CRGT Assembly
    - CRGT Spacer Castings
- Core Barrel Assembly
    - Core Barrel Cylinder (including the vertical and circumferential seam welds)
    - Former plates
    - External BB bolts
    - Inaccessible Locking Device and Locking Weld for Core Barrel-to-Former (CB) Bolts and External BB Bolts
    - Upper Thermal Shield (UTS) Bolts
    - Surveillance Specimen Holder Tube (SSHT) Bolts (Crystal River Unit 3 (CR-3), Davis-Besse (DB))
- Upper Grid Assembly
    - Upper Grid Fuel Assembly Support Pads: Alloy X-750 Dowel Locking Weld (except DB)
- Lower Grid Assembly
    - Lower Grid Fuel Assembly Support Pad Component items: Pad, Pad-to-Rib Section Weld,
        - Alloy X-750 Dowel, Cap Screw, Their Locking Welds
    - X-750 Bolts for Lower Grid Shock Pads (TMI-1 only)
    - Lower Thermal Shield (LTS) Bolts

For the eleven components listed above applicable to TMI-1, the NRC staff used the following process to evaluate whether plant-specific information could change the aging management recommendations:

- Evaluate whether a fluence increase of 50 percent from the assumed fluence screening value in MRP-189 would result in additional aging mechanisms screening in, or whether already screened-in aging mechanisms would increase in severity.
- Review information in FMECA and functionality analysis to determine if any detailed quantitative analyses were performed that assumed certain numerical values of

parameters, such as the neutron fluence and stress analyses performed for certain components in MRP-229. If generic quantitative analyses were performed, evaluate whether a change in the analysis conclusions would result from a fluence increase of 50 percent from the values assumed in the analysis.

- If additional aging mechanisms or an increase in severity could occur, evaluate the effect on component categorization, (i.e., should Category B be elevated to Category C).

Only variations in neutron fluence, not temperature were considered because the variation in core inlet and outlet temperatures across the B&W fleet is less than 4°F; however, reactor pressure vessel (RPV) inner diameter fluences vary almost 50 percent from the highest to lowest plant in the fleet.

3.3.1.2.1 Evaluations of Specific Components

CRGT Spacer Castings

[[ ]] this component was elevated to "Primary" in MRP-227-A. Therefore, the only potential impact from a change in screening assumptions would be the screening in of additional degradation mechanisms which might affect the type of inspection performed. Although the castings were [[ ]] the castings are already being inspected for cracking since TE is only a problem if cracking is present.

[[ ]] Therefore, the NRC staff concludes that determination of categorization of CRGT spacer castings with respect to various degradation mechanisms remains qualitative, and this determination was not based on any generic quantitative analysis from which certain parameters can be revised using TMI-1 plant-specific information. [[ ]]

[[ ]] Even a doubling of this fluence value would not result in screening in the CRGT spacer castings for additional fluence-dependent aging mechanisms, for instance IE. Further, although TE is dependent on operating temperature and time, the variation in coolant temperature across the B&W fleet is too small for significant differences in the rate of TE or the final fracture toughness. Since no additional aging mechanisms would screen in based on a 50 percent fluence increase, the NRC staff concludes that the final categorization for the CRGT spacer castings is unlikely to be affected by any TMI-1 plant-specific information.

Core Barrel Cylinder (including the vertical and circumferential seam welds)

[[

]] Based on this information, the NRC staff determined an increase in the fluence of 50 percent would not result in the core barrel cylinder exceeding the screening criteria for additional fluence-related aging mechanisms (IASCC and void swelling), and the relative susceptibility of the baffle plates and core barrel cylinder would remain unchanged, [[

]] Since the applicable aging mechanisms and the relative susceptibility to these mechanisms would not change with a 50 percent increase in fluence, the NRC staff finds final categorization of the baffle plates as Primary and core barrel cylinder as Expansion is appropriate, and that this categorization would be unlikely to change based on TMI-1 specific information.

Former Plates

[[

]]

[[

]] Therefore, the NRC staff concludes that revision of categorization of IASCC for former plates from Category C to Category A is not sensitive to plant-specific information.

Further, since [[

]] the NRC staff determined that it is unlikely that a TMI-1 plant-specific void swelling analysis for former plates would show an increase of 50 percent void swelling and elevate its categorization from Category A to Category C. Therefore, the NRC staff concludes that revision of categorization of void swelling for former plates from Category C to Category A is not sensitive to plant-specific information.

It should be noted that determination of categorization of various degradation mechanisms other than IASCC and void swelling for former plates remains qualitative, and this determination was not based on any generic quantitative analysis from which certain parameters can be revised using TMI-1 plant-specific information. In summary, the NRC staff concludes that the re-categorization of the former plates and the underlying generic analyses is not sensitive to plant-specific information.

The former plates were re-defined as **[[** **]]** Since the limited additional information provided in this section regarding the former plates is very general, the NRC staff concludes that the final categorization for former plates is unlikely to be affected by any TMI-1 plant-specific information.

External Baffle-to-Baffle (BB) Bolts

[[**]]** The NRC staff reviewed the calculated values of the critical parameters in the two generic evaluations/analyses and determined that in each case, the margin in the results is not sufficient to tolerate possible changes caused by TMI-1 plant-specific information while still supporting Category B categorization.

Nevertheless, the NRC staff concludes that even if the categorization of external BB bolts was changed from Category B to Category C after TMI-1 plant-specific information was considered, it would have no impact on the applicability of MRP-227-A to TMI-1. This is because inaccessibility of the external BB bolts makes their final categorization as Primary not practical. Therefore, redefining them as Expansion **[[** **]]** is appropriate. Further, due to the relatively low neutron fluence level, the external BB bolts are less susceptible to degradation than the internal BB bolts and baffle-to-former (FB) bolts **[[** **]]**, and the condition of the external BB bolts can be inferred from the inspection results of internal BB bolts and FB bolts. The functionality of the external BB bolts will be evaluated as part of the licensee's analysis fulfilling applicant/licensee action item 6 (see SA Section 3.3.6). Based on the above discussion and the rather general additional information provided in **[[** **]]** regarding the external BB bolts, the NRC staff concludes that the final categorization for the external BB bolts is unlikely to be affected by any TMI-1 plant-specific information.

Internal BB Bolts

[[**]]** Therefore, the internal BB bolts will remain bounded by the external BB bolts with respect to IASCC susceptibility. This relationship is unlikely to change due to plant specific information since the fluence on the internal BB bolts will always be greater than the fluence on the external BB bolts. Therefore, the internal BB bolts will always have greater stress relaxation, which results in lower stresses on the bolts, and thus, decreases the IASCC susceptibility.

<u>Inaccessible Locking Device and Locking Weld (CB Bolts and External BB Bolts)</u>

The staff reviewed the **[[** **]]** for these devices, and determined that a 50% increase in fluence would not cause an additional mechanism to screen in, since all fluence-related aging mechanisms already screen in for the locking devices. Variations in temperature and fluence would not cause these devices to screen in for fatigue, wear, or stress relaxation. The NRC staff reviewed **[[** **]]** and found no generic evaluation supporting the categorization of this item as Category B. Therefore, the NRC staff concludes that determination of categorization of inaccessible locking device and locking weld (CB bolts and external BB bolts) with respect to various degradation mechanisms remains qualitative, and this determination was not based on any generic quantitative analysis from which certain parameters can be revised using TMI-1 plant-specific information.

Further, inaccessibility of this item makes its classification as Expansion **[[** **]]** appropriate. Due to the lower neutron fluence, the locking device and locking weld of CB bolts and external BB bolts are less susceptible to degradation than the locking device and locking weld of FB bolts and internal BB Bolts **[[** **]]**, and the condition of this RVI component item can be inferred from the inspection results of the accessible locking device and locking weld of FB bolts and internal BB Bolts.

Based on the above discussion and the rather general additional information provided in **[[** **]]** regarding the inaccessible locking device and locking weld, the staff concludes that the final categorization for the inaccessible locking device and locking weld is unlikely to be affected by any TMI-1 plant-specific information.

<u>UTS and LTS Bolts</u>

[[**]]** An increase in the fluence of 50% would not cause the UTS bolts to screen in for any fluence-dependent aging mechanisms. These bolts were not within the scope of the functionality analysis. Therefore, the NRC staff concludes that determination of categorization of various degradation mechanisms for UTS and LTS bolts is qualitative, and this determination was not based on any generic quantitative analysis from which certain parameters can be revised using TMI-1 plant-specific information. Based on the lower susceptibility to SCC and less severe failure consequences compared to the UCB and LCB bolts, the UTS and LTS bolts were categorized as **[[** **]]** Therefore, any plant-specific variations in material would not affect the categorization of these bolts.

<u>SSHT Bolts (CR-3, DB)</u>

Not applicable to TMI-1.

Upper Grid Fuel Assembly Support Pads: Alloy X-750 Dowel Locking Weld (except DB)

[[

]] The NRC staff reviewed [[
]] and found no generic evaluation supporting the categorization of this item as Category B. Therefore, the NRC staff concludes that determination of categorization of upper grid fuel assembly support pads (Alloy X-750 dowel locking weld) with respect to various degradation mechanisms remains qualitative, and this determination was not based on any generic quantitative analysis from which certain parameters can be revised using TMI-1 plant-specific information.

[[

]] Since no additional information was provided in this section regarding this RVI component item, the NRC staff concludes that the final categorization for the upper grid fuel assembly support pads (Alloy X-750 dowel locking weld) is unlikely to be affected by any TMI-1 plant-specific information.

Lower Grid Fuel Assembly Support Pad Component Items: Pad, Pad-to-Rib Section Weld, Alloy X-750 Dowel, Cap Screw, Their Locking Welds

[[

]] The NRC staff reviewed [[
]] and found no generic evaluation supporting the categorization of this item as Category B. Therefore, the NRC staff concludes that determination of categorization of this RVI item with respect to various degradation mechanisms remains qualitative, and this determination was not based on any generic quantitative analysis from which certain parameters can be revised using TMI-1 plant-specific information.

The lower grid fuel assembly support pad component items (pad, pad-to-rib section weld, Alloy X-750 dowel, cap screw, and their locking welds) were categorized as [[

]] Since the relationship of these items to the linked Primary component (the IMI guide tube spiders) is thus unlikely to be affected by any TMI-1 plant-specific information, the NRC staff concludes that the final categorization for this RVI item as Expansion is unlikely to be affected by any TMI-1 plant-specific information.

X-750 Bolts for Lower Grid Shock Pads (TMI-1 only)

[[

]] Since these bolts were not within the scope of a quantitative functionality analysis, it is unlikely any plant-specific information from TMI-1 would affect the categorization of these bolts.

FB Bolts

[[

]]

Therefore, the categorization of FB bolts as Expansion in MRP-227-A is unlikely to be affected by the TMI-1 plant-specific information.

The FB bolts were categorized as [[

]] the staff concludes that the final categorization for the FB bolts is unlikely to be affected by any TMI-1 plant-specific information.

3.3.1.2.2 Licensee Action Item 1 - Conclusion

Based on its independent evaluation following the process described above for the eleven components applicable to TMI-1, the NRC staff determined that the component aging management recommendations in MRP-227-A for generic B&W-designed RVI components will not be affected by TMI-1 plant-specific information. Therefore, the NRC staff concluded that the licensee's response to licensee action item 1, which confirmed that TMI-1 meets the three basic assumptions outlined in MRP-227-A, Section 2.4, stated that TMI-1 will continue to use a low-leakage core in the future, and will evaluate the effect of future power uprates on the RVI Inspection Plan, is sufficient to resolve licensee action item 1 for TMI-1.

3.3.2 Licensee Action Item 2

Per Section 4.2.2 of Reference 8, this action item requires that, consistent with the requirements addressed in 10 CFR 54.4, each applicant/licensee is responsible for identifying the RVI components that are within the scope of LR for its facility. Applicants/licensees shall review the information in Tables 4-1 and 4-2 in MRP-189, Revision 1, and Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RVI components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. (Note: Table 4-4 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.

3.3.2.1 Licensee Evaluation

In Appendix D of its submittal dated April 16, 2012, (Ref. 1) the licensee stated that a review of the TMI-1 LR Project documentation shows that all RVI sub-components in the scope of LR were included in LRA Table 3.1.2-3; (i.e., no components were screened out during the AMR). The licensee further stated that Appendix E of the inspection plan provides a listing of components in the TMI-1 LRA Table 3.1.2-3 and where those components are addressed in the MRP-227-A supporting documents (MRP-189 and MRP-231).

Finally, the licensee stated that the only component identified that is not included in the MRP-227-A development is the vent valve locking device. A review of this component using the MRP-227-A development methodology will be performed and will be submitted to the NRC upon completion. In the cover letter of Reference 1, the licensee indicated it was working with the PWROG to evaluate the vent valve locking device and committed to submitting an update on progress on evaluating this item by April 19, 2013.

The licensee provided the results of the review of the vent valve locking device in its letter dated April 17, 2013 (Ref. 2). In Reference 2, the licensee stated that the PWROG proposes to accommodate the vent valve locking devices as an existing program within Table 4-7, "B&W Plants Existing Programs Components," of MRP-227-A. The licensee stated that the vent valve locking devices shall be addressed by ASME Section XI Examination Category B-N-3, per BAW-2248-A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," (Ref. 18), and that these inspections shall require a VT-3 examination of 100% of accessible surfaces of the vent valve locking devices during each 10-year ISI interval. The licensee further stated that TMI-1 will examine the vent valve locking devices under the ASME Section XI ISI program. The licensee finally stated that this commitment is complete.

3.3.2.2 NRC Staff Assessment

The NRC staff reviewed the information provided by the licensee in Appendix E of the inspection plan against the components subject to AMR in the TMI-1 LRA and the components evaluated in MRP-189 for B&W plants. The NRC staff finds that MRP-227-A (through MRP-189) addresses all the TMI-1 components subject to AMR, with the exception of the vent valve locking devices. However, Appendix E does not identify the material of the TMI-1 components, and does not address whether there were any material differences between the TMI-1 components and the generic components in MRP-189. Therefore, in RAI 6, the NRC staff requested that the licensee confirm that the TMI-1-specific materials are consistent with those listed in MRP-189, to discuss whether additional aging mechanisms are applicable to the TMI-1 materials, and provide plant-specific aging management recommendations as necessary to address the additional mechanisms. In its November 6, 2013, response to RAI 6, the licensee identified a few minor differences in material specifications; for example, the TMI-1 lower grid assembly shock pads are fabricated from ASTM A-240 Type 304 plate versus ASTM A-276 Type 304 bar as described in MRP-189. In all cases, the licensee indicated the materials were equivalent and would have no additional aging effects. The NRC staff reviewed the material differences and finds that the materials are equivalent, with respect to the applicable aging mechanisms, thus no TMI-1 specific aging management evaluations are necessary to address the material differences. Therefore, RAI 6 is resolved.

In RAI 3, the NRC staff requested that the licensee provide the details of the evaluation of the vent valve locking devices using the MRP-227-A methodology. In its November 6, 2013, response to RAI 3 (Ref. 3), the licensee provided the details of the screening, FMECA, and final categorization of the vent valve locking devices. (Certain details of the vent valve evaluation are proprietary and are included in the version of the RAI 3 response in Reference 4.) The NRC staff reviewed the information and finds that the methodology used for these processes is consistent with the methodology applied to the generic B&W RVI design components in MRP-189 and MRP-190. The licensee determined the most significant aging mechanism in terms of risk was wear of certain parts of the vent valve locking devices. Wear of these parts occurred early in plant life at some B&W units necessitating the replacement of some vent valves. VT-3 visual examination is effective for detecting wear of components; therefore, the NRC staff agrees with the inspection method chosen for the locking devices. The NRC staff also found the licensee's evaluation supports the final categorization of the vent valve locking devices as Existing Programs components, and that the inspection method, scope and frequency are appropriate. Therefore, RAI 3 is resolved.

3.3.3 Licensee Action Item 3

Not applicable to B&W design units.

3.3.4 Licensee Action Item 4

This action item requires that the B&W applicants/licensees confirm that the CSS upper flange weld was stress relieved during the original fabrication of the RPV in order to confirm the applicability of MRP-227, as approved by the NRC, to their facility. If the upper flange weld has not been stress relieved, then this component shall be inspected as a Primary inspection category component. If necessary, the examination methods and frequency for non-stress relieved B&W CSS upper flange welds shall be consistent with the recommendations in MRP-227, as approved by the NRC, for the Westinghouse and CE upper core support barrel welds. The examination coverage for this B&W flange weld shall conform to the NRC staff's imposed criteria as described in Sections 3.3.1 and 4.3.1 of Reference 8. The applicant's/licensee's resolution of this plant-specific action item shall be submitted to the NRC for review and approval.

3.3.4.1 Licensee Evaluation

In Appendix D of its submittal (Ref. 1), in response to applicant/licensee action item 4, the licensee stated original fabrication records have confirmed that the TMI-1 CSS upper flange weld was stress relieved and referred to Section 4.1.5 of Attachment 1 to the submittal. The licensee therefore concluded that the CSS upper flange weld does not need to be inspected as a Primary component.

3.3.4.2 NRC Staff Assessment

Since the licensee confirmed via examination of fabrication records that the CSS upper flange weld is stress relieved, the NRC staff finds the licensee's response to licensee action item 4 acceptable.

3.3.5 Licensee Action Item 5

Not applicable to B&W design units.

3.3.6 Licensee Action Item 6

Licensees shall justify the acceptability of these components [[

]] for continued operation through the PEO by performing an evaluation, or by proposing a scheduled replacement of the components. As part of their application to implement MRP-227, applicants/licensees shall provide their justification for the continued operability of each of the inaccessible components and, if necessary, provide their plan for the replacement of the components for NRC review and approval.

3.3.6.1 Licensee Evaluation

In Appendix D of the inspection plan, the licensee stated that they are working with the PWROG to evaluate the acceptability of the subject components for continued service without inspection. The licensee further stated that the PWROG is currently concentrating analyses tasks for Primary components and is defining future analyses to support continued operation of the subject components. The licensee stated that by April 19, 2013, Exelon will provide an update of the PWROG progress including a schedule showing when Exelon will submit for NRC review and approval either: (1) an analysis of the acceptability of the subject components for continued service without inspection; or (2) a schedule for replacement of the subject components. Finally the licensee stated that the submittal schedule is one year prior to entering the PEO and approximately 2 years prior to the majority of the currently planned TMI-1 MRP-227-A examinations.

In its letter dated November 6, 2013 (Ref. 3), the licensee updated its response to action item 6 via the following regulatory commitment, with a committed date of December 15, 2018:

> The analyses of the inaccessible components identified in Table 4-4 of MRP-227-A are being pursued as TMI, Unit 1 plant-specific analyses. As the inaccessible components are defined as Expansion components under MRP-227-A, their inspection (analysis) is only required if the primary component inspection does not meet MRP-227-A acceptance criteria presented in Table 5-1. TMI Unit 1 will either submit a detailed analysis, a replacement schedule, or a justification for some other alternative process within one year of the initial inspection (Fall 2017) of the linked MRP-227-A primary component items, if the inspection results in indications beyond the threshold for expansion criteria presented in Table 5-1. This schedule is consistent with the current NRC and Industry proposed schedules concerning topical report WCAP-17096-NP.

3.3.6.2 NRC Staff Assessment

Per Section 3.3.6 of the NRC staff's final SE of MRP-227, the components within the scope of action item 6 are the core barrel cylinder (including vertical and circumferential seam welds), the former plates, the external BB bolts and their locking devices, and the CB bolts and their locking devices. These components are all categorized as Expansion items in MRP-227-A. In addition, the NRC staff reviewed the basis for the final categorization of these components in MRP-231, Rev. 2 (Ref. 16). The inaccessible components, in most cases, have lower susceptibility to the relevant degradation mechanisms, such as IASCC and IE, than the associated Primary components. Therefore, the Primary components should be expected to exhibit degradation before it occurs in the linked Expansion components. The licensee's proposal to submit the detailed analysis, a replacement schedule, or a justification for some other alternative process within one year of the initial inspection (fall 2017) of the linked MRP-227-A primary component items, if the inspection results in indications beyond the threshold for expansion criteria presented in Table 5-1, is therefore reasonable, because degradation would not be expected to occur in these components before the linked Primary components experience degradation. This schedule is also consistent with the NRC staff's draft safety evaluation of topical report WCAP-17096-NP, Rev. 2 (Ref. 11), which provides the methodology for evaluating detected degradation in RVI components inspected in accordance with MRP-227-A. In the NRC staff's draft SE of WCAP-17096-NP, Rev. 2, the NRC staff imposed conditions requiring detailed analyses justifying operation with degradation exceeding the acceptance criteria of MRP-227-A, to be submitted to the NRC within one year of the detection of the degradation. The NRC staff therefore finds the licensee's proposed commitment for submittal of the detailed analyses of the expansion components acceptable.

3.3.7 Licensee Action Item 7

This action item 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 IMI guide tube assembly spiders CRGT 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 (PH) 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 thermal embrittlement (TE) and irradiation embrittlement (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 applicants/licensees shall include the plant-specific analysis as part of their submittal to apply the approved version of MRP-227.

3.3.7.1 Licensee Evaluation

In the RVI Inspection Plan, Appendix D, the licensee stated that it will develop either a plant-specific analysis or a PWROG generic analysis that bounds TMI-1, to evaluate the acceptability of the subject components, and any additional RVI components that may be fabricated from CASS, martensitic stainless or precipitation hardened stainless steel materials, for continued service. The licensee further stated that the analysis will consider the possible loss of fracture toughness in these components due to TE and/or IE, as well as limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. Finally, the licensee stated that by April 19, 2013, Exelon will provide an update including a schedule showing when Exelon will submit for NRC review and approval either: (1) analyses of the acceptability of the subject components for the maintenance of functionality during the PEO; or (2) a schedule for replacement of the components, and that the proposed date of April 19, 2013, is one year prior to the PEO and approximately 2-1/2 years prior to the majority of the currently planned TMI-1 MRP-227-A examinations.

In its letter dated November 6, 2013 (Ref. 3), the licensee updated its response to Action Item 7 via the following regulatory commitment:

The analysis and evaluation of the Control Rod Guide Tube (CRGT) Spacer Castings is in progress with the PWROG, with an estimated completion date of December 31, 2013. The Incore Monitoring Instrumentation (IMI) Spider Castings evaluation will be pursued as a TMI, Unit 1 plant specific analysis and evaluation, separate from the CRGT Spacer Casting analyses being performed by the PWROG. TMI will submit the CRGT Spacer Castings and IMI Spider Castings evaluations to the NRC by October 30, 2016, which is approximately one year prior to the inspection outage (2017). Additionally, the vent valve retaining ring analysis will also be submitted by October 30, 2016.

3.3.7.2 NRC Staff Assessment

The NRC staff notes that there are four components listed in Table 3-1 of MRP-227-A as fabricated from CASS material, martensitic stainless steel, or precipitation hardened stainless steel: the CRGT spacer castings, IMI guide tube spiders, and core support shield vent valve top retaining rings, and core support shield bottom retaining rings. The CRGT spacer castings are fabricated from Type CF3M CASS, the IMI spider castings are Type CF8 CASS, and the retaining rings are Type 15-5 PH stainless steel. All four components are Primary components for TE. The IMI guide tube spiders are also Primary for IE. MRP-227-A prescribes VT-3 visual examination for all four components. The licensee committed to submit plant-specific evaluations for all four of these components. Therefore, all CASS, martensitic or PH stainless steel components requiring aging management will be evaluated by the licensee in accordance with applicant/licensee action item 7.

The licensee proposed a commitment to submit the evaluations of the CASS and PH stainless steel components fulfilling applicant/licensee action item 7 by October 30, 2016, approximately one year prior to the initial RVI inspections of components not requiring CSA removal planned for the Fall, 2017 refueling outage. Based on the figures of B&W design RVI in MRP-227-A, all the components in scope of applicant/licensee action item 7 are accessible for inspection without CSA removal. Since all the in-scope components are already in the Primary category, the NRC staff finds the proposed commitment to provide the plant-specific analysis by one year prior to the inspection outage to be acceptable because: (1) it addresses all CASS, martensitic or PH stainless steel components requiring aging management, and because (2) the NRC staff will have time to review the plant-specific analyses prior to the time that the next inspection of the CASS components would occur. Further, since the CRGT spacer castings, IMI spiders, retaining rings are Primary and will be inspected during the initial inspection (2017), the NRC staff's review would not have to be complete prior to the initial inspection. Any necessary adjustments to the re-inspection interval identified during the NRC staff's review could be implemented by the subsequent refueling outage.

3.3.8 Licensee Action Item 8

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

Section 3.5.1 of Reference 8 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 8 further states that an applicant/licensee's application to implement MRP-227, as amended by this SA shall include the following items (1) and (2):

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

2. To ensure the MRP-227 program and the plant-specific action items will be carried out by applicants/licensees, applicants/licensees are to submit an inspection plan which addresses the identified plant-specific action items for staff review and approval consistent with the licensing basis for the plant. If an applicant/licensee plans to implement an AMP which deviates from the guidance provided in MRP-227, as approved by the NRC, the applicant/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 submit 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 TMI-1 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.

3.3.8.1 Licensee Evaluation

Item 1 - The licensee included its ten-element AMP in Section 5.0 of its RVI Inspection Plan.

Item 2 - The licensee stated in Appendix D to the RVI Inspection Plan that the TMI-1 RVI Inspection Plan is consistent with MRP-227-A, contains no deviations, and addresses TMI-1 specific action items.

3.3.8.2 NRC Staff Assessment

Item 1 - The NRC staff's review of the AMP elements is contained in Section 3.2 of this SE. The NRC staff found the AMP was consistent with no exceptions and no enhancements to the GALL Report, Rev. 2, Chapter XI.M16A. Therefore, the NRC staff finds the licensee adequately addressed Item 1 of Applicant/Licensee Action Item 8.

Item 2 - The NRC staff's review of the licensee's resolution of the applicant/licensee action items applicable to TMI-1 is addressed previously in this SA section (i.e., Section 3).

Based on the above, the NRC staff finds the licensee has submitted the information required by applicant/licensee action item 8, thus the action item is resolved.

4.0 CONCLUSION

The NRC staff has reviewed the inspection plan for the TMI-1 RVI components as submitted in the licensee's April 16, 2012, letter (Ref. 1), and supplemented by the licensee's letters dated April 17, 2013 (Ref. 2), and November 6, 2013 (Ref. 3, 4). The NRC staff concludes that the licensee's RVI Inspection Plan is acceptable based on the following: (1) the plan is consistent with the inspection and evaluation guidelines of MRP-227-A, (2) the licensee has addressed six of the eight licensee action items specified in MRP-227-A appropriately, and (3) the licensee has made regulatory commitments to submit analyses fulfilling the other two action items on an acceptable schedule. Consequently, LR Commitment No. 36 from Appendix A of the NRC's SE of the TMI-1 LR Application is considered fulfilled. The NRC staff will review the information provided by the licensee in accordance with the two new regulatory commitments documented in the licensee's letter dated November 6, 2013, for adequacy, when received. The NRC staff's approval of the TMI-1 RVI Inspection Plan does not reduce, alter, or otherwise affect current ASME Code, Section XI ISI requirements, or any TMI-1 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.

5.0 REFERENCES

1. Inspection Plan for the Three Mile Island Reactor Vessel Internals, AREVA Document 77-2952-001, ANP-2952 Revision 001, March 2012, Attachment 1 to Three Mile Island, Unit 1, Submittal of Inspection Plan for Reactor Internals, April 16, 2012 (ADAMS Accession No. ML12108A029).

2. Three Mile Island Nuclear Station, Unit 1, "Submittal of Inspection Plan for Reactor Internals," April 17, 2013 (ADAMS Accession No. ML13108A004).

3. Three Mile Island Unit 1, "Submittal of Inspection Plan for Reactor Internals," November 6, 2013; Attachment 4, "Response to NRC Requests for Additional Information on the Three Mile Island Unit 1 Reactor Vessel Internals Inspection Plan," ANP-2952Q1 NP, Revision 0, Non-Proprietary Version (ADAMS Accession No. ML13317A931).

4. Three Mile Island Unit 1, "Submittal of Inspection Plan for Reactor Internals," November 6, 2013; Attachment 4, "Response to NRC Requests for Additional Information on the Three Mile Island Unit 1 Reactor Vessel Internals Inspection Plan," ANP-2952Q1 P, Revision 0, Proprietary Version (not publicly available, proprietary).

5. NUREG-1928, 1:2 Cover-Chapter 3, Safety Evaluation Report, Related to the LR of Three Mile Island Nuclear Station, Unit 1, Exelon Generation Company, LLC, October, 2009 (ADAMS Accession No. ML092950458).

6. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A), EPRI Final Report 1022863, December, 2011 – Transmitted to NRC by MRP letter MRP-2011-036 dated January 9, 2012 (ADAMS Accession No. ML120170453).

7. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, (MRP-227-Rev. 0), EPRI Final Report 1016596, December, 2008, - Transmitted to NRC by MRP letter number MRP 2009-04 dated January 12, 2009, (ADAMS Accession No. ML090160206).

8. Letter from Robert A. Nelson, NRC, to Neil Wilmshurst, EPRI, dated December 16, 2011; Subject: Revision 1 to the Final Safety Evaluation of EPRI Report, Materials Reliability Program Report 1016596 (MRP-227), Revision 0, "PWR Internals Inspection and Evaluation Guidelines" (TAC NO. ME0680) (ADAMS Accession No. ML11308A770).

9. NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," Final Report, December, 2010 (ADAMS Accession No. ML103490041).

10. Final 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 (ADAMS Accession No. ML12270A436).

11. Three Mile Island, Unit 1, LR Application, Section 3.4, "Aging Management of Steam and Power Conversion System," through Appendix A, January 8, 2008 (ADAMS Accession No. ML080220252).

12. "Reactor Internals Acceptance Criteria Methodology and Data Requirements," (WCAP-17096-NP, Revision 2), December, 2009, (ADAMS Accession No. ML101460157).

13. "Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals (MRP-190)," EPRI Technical Report 1013233, November, 2006, (ADAMS Accession No. ML091910128).

14. "Materials Reliability Program: Functionality Analysis for Babcock & Wilcox Representative PWR Internals (MRP-229-Rev. 3)," EPRI Technical Report 1022402, December, 2010, Proprietary (not publicly available, proprietary).

15. "Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items (MRP-189-Rev. 1)," EPRI Final Report 1018292, March, 2009, Proprietary (not publicly available, proprietary).

16. "Materials Reliability Program: Aging Management Strategies for B&W Pressurized Water Reactor Internals (MRP-231-Rev. 2)," EPRI Final Report 1021028, December, 2010, Proprietary (not publicly available, proprietary).

17. "Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228)," EPRI Final Report 1016609, July, 2009, Non-Proprietary Version (ADAMS Accession No. ML092750569).

18. B&W Owners Group License Renewal Task Force Topical Report BAW-2248A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" March, 2000 (ADAMS Accession No. ML003708443).

19. NEI 03-08, "Guideline for the Management of Materials Issues," Revision 2 (ADAMS Accession No. ML102880028).

20. "Materials Reliability Program: Aging Management Strategies for B&W PWR Internals (MRP-231-Rev. 1)," EPRI Final Report 1019092, July, 2009, Proprietary (not publicly available, proprietary).

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Date: December 19, 2014

Consequently, LR Commitment No. 36 from Appendix A of the NRC's SE of the TMI-1 LR Application is considered fulfilled. The NRC staff will review the information provided by the licensee in accordance with the two new regulatory commitments documented in the licensee's letter dated November 6, 2013, for adequacy when received. The NRC staff's approval of the TMI-1 RVI Inspection Plan does not reduce, alter, or otherwise affect current American Society of Mechanical Engineers Code, Section XI Inservice Inspection (ISI) requirements, or any TMI-1 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.

If you have any questions concerning this matter, please contact the TMI-1 Project Manager, Mr. John Lamb, at (301) 415-3100 or via e-mail at John.Lamb@nrc.gov.

Sincerely,

/RA/

Meena K. Khanna, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosures:
1. Non-Proprietary Staff Assessment
2. Proprietary Staff Assessment

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