September 5, 2013

MEMORANDUM TO: Veronica M. Rodriguez, Acting Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

FROM: Richard B. Ennis, Senior Project Manager ***/ra/***
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

SUBJECT: THREE MILE ISLAND NUCLEAR STATION, UNIT 1, DRAFT REQUEST FOR ADDITIONAL INFORMATION (TAC NO. MF1459)

The attached draft request for additional information (RAI) was transmitted on September 5, 2013, to Mr. Thomas Loomis of Exelon Generation Company, LLC (Exelon, the licensee). This information was transmitted to facilitate an upcoming conference call in order to clarify the licensee's reactor vessel internals inspection plan for Three Mile Island Nuclear Station, Unit 1, which was submitted via Exelon's letter dated April 16, 2012, as supplemented by letter dated April 17, 2013.

The draft RAI was sent to Exelon to ensure that the questions are understandable, the regulatory basis for the questions is clear, and to determine if the information was previously docketed. This memorandum and the attachment do not convey or represent an NRC staff position regarding the licensee's submittal.

Docket No. 50-289

Attachment: Draft RAI

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Attachment: Draft RAI

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DRAFT REQUEST FOR ADDITIONAL INFORMATION

REACTOR VESSEL INTERNALS INSPECTION PLAN

EXELON GENERATION COMPANY, LLC

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

DOCKET NO. 50-289

By letter dated April 16, 2012, as supplemented by letter dated April 17, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML12108A029 and ML13108A004, respectively), Exelon Generation Company, LLC (Exelon, the licensee) submitted an inspection plan for the Three Mile Island Nuclear Station, Unit 1 (TMI-1) reactor vessel internals (RVI). The RVI inspection plan was submitted for Nuclear Regulatory Commission (NRC) staff review and approval in accordance with a license renewal commitment. The inspection plan, Attachment 1 to the licensee's letter dated April 16, 2012, is based on Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) report MRP-227-A, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines" (Reference 1).

The NRC staff is reviewing your submittal and has determined that additional information is needed to complete its review. The specific request for additional information (RAI) is addressed below.

RAI 1

Please provide the following information:

1. Identify all TMI-1 RVI components that are defined in the current licensing basis (CLB) as American Society of Mechanical Engineers (ASME), *Boiler and Pressure Vessel Code* (Code), Section XI, Examination Category B-N-3 core support structure components. For these components, identify which of the four inspection categories in MRP-227-A (Reference 1) is applicable to the component.

2. For those components listed in the response to item 1 of this RAI question, in the Primary or Expansion inspection categories, if the MRP-227-A inspections will be credited with fulfilling the ASME Code, Section XI Examination Category B-N-3 inspection requirements, clarify how differences in inspection requirements between the ASME Code and MRP-227-A will be reconciled.

RAI 2

In Section 4.1.6 of the RVI inspection plan (Reference 2), the licensee indicated that it has performed analyses to determine the tolerance of the RVI to broken or degraded upper core barrel (UCB) and lower core barrel (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 less. Section 5.6.2 of the RVI inspection plan states that the component degradation that exceeds the examination acceptance criteria will be evaluated per MRP-227-A, but will also be evaluated considering the supplemental guidance in WCAP-17096, including any additional guidance resulting from the ongoing NRC review of that document.

Please provide the following information:

1. Are the UCB and LCB bolt analyses consistent with the recommended methodology and acceptance criteria of WCAP-17096-NP, Revision 2 (Reference 3), as modified by the NRC staff's draft safety evaluation of that report?

2. If not, will the analyses be revised to be consistent with the approved version of WCAP-17096-NP if degradation is found in the bolts?

RAI 3

In Section 4.1.7 of the RVI inspection plan, the licensee described RVI inspections that have been performed in the past at TMI-1. These include vent valve inspections, UCB bolt ultrasonic examinations (UT), and core clamping measurements. One hundred percent of the UCB bolts were examined in 2009 via UT with no recordable indications found. One hundred percent UT examination of the UCB bolts was also performed in 1991 with no recordable indications. MRP-227-A specifies UT examination of the UCB bolts within two refueling outages of January 1, 2006, or the next ten-year inservice inspection (ISI) interval, whichever comes first.

Please provide the following information:

1. Confirm that the UCB bolt examinations performed in 2009 constitute the initial MRP-227-A inspection requirement for the UCB bolts.

2. Will the UCB bolts be reinspected during the Fall 2015 refueling outage, when MRP-227-A inspections are scheduled in conjunction with the ASME Section XI inspections of the RVI?

RAI 4

In Attachment 2 to Exelon's submittal dated April 16, 2012, the licensee committed to perform an evaluation of the RVI vent valve locking devices, which are a TMI-1-specific component not covered by the generic MRP-227-A aging management recommendations. The licensee provided the results of the review of the vent valve locking device in its letter dated April 17, 2013 (Reference 4). In Reference 4, the licensee stated that the Pressurized Water Reactor Owner's Group (PWROG) proposes to accommodate the vent valve locking devices as an existing program within Table 4-7, "B&W [Babcock & Wilcox] plants Existing Programs components," of MRP-227-A. The licensee stated that the vent valve locking devices shall be addressed by ASME Code, Section XI Examination Category B-N-3, per BAW-2248-A (Reference 5), 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 1 will examine the vent valve locking devices under the ASME Section XI ISI program. The licensee finally stated that this commitment is complete.

Please provide the following information:

Provide a summary of the evaluation of the RVI vent valve locking devices, including the material type, screening parameters (temperature, neutron fluence, stress) results of the screening for degradation mechanisms, Failure Modes, Effects and Consequence Analysis (FMECA) results, initial categorization, and how the final inspection category of the locking devices was determined (Existing Programs).

RAI 5

NUREG-1801, Revision 2, “Generic Aging Lessons Learned Report,” (GALL Report, Revision 2), Chapter XI.M16A, “PWR [pressurized water reactor] Vessel Internals Program,” states, under “Detection of Aging,” 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. 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 six Expansion components for which visual VT-3 examination is specified for detection of cracking (consistent with
MRP-227-A).

License Renewal Interim Staff Guidance LR-ISG-2011-04, “Updated Aging Management Criteria for PWR Reactor Vessel Internal Components,” (Reference 6) modified the statement regarding VT-3 examination such that a flaw tolerance evaluation would only be required for non-redundant components. Reference 1 also stated that VT-3 visual methods are acceptable for the detection of cracking in redundant RVI components (e.g., redundant bolts or pins used to secure a fastened RVI assembly).

Please provide the following information:

1. Which TMI-1 RVI components require flaw evaluations to justify VT-3 examination?

2. Are any of the components for which VT-3 examination is specified redundant components that do not require a flaw evaluation? If so, justify considering the components redundant.

3. How will the flaw tolerance evaluations of the TMI-1 Primary and Expansion components be documented and communicated to the NRC to support the NRC staff's review?

4. Confirm that these flaw tolerance evaluations will be completed prior to the initial inspections of the TMI-1 Primary components, scheduled for the fall 2015 refueling outage.

5. Will the flaw tolerance evaluations of the control rod guide tube spacer castings and incore monitoring instrumentation guide tube spiders be part of the plant-specific evaluation of cast austenitic stainless steel components that the licensee committed to submit to the NRC by fall 2015?

RAI 6

Appendix A of MRP-227-A discusses operating experience under various degradation mechanisms for the RVI components of all nuclear steam supply system (NSSS) designs, including B&W plants.

Please provide the following information:

1. Identify the MRP-227-A, Appendix A experience that occurred at TMI-1.

2. Describe any TMI-1 experience with RVI component degradation that is not discussed in Appendix A of MRP-227-A, and is not discussed in Section 4.1.7 of the RVI inspection plan.

3. Describe any changes to the RVI inspection plan made as a result of the operating experience described in the response to item 2 of this RAI question.

RAI 7

Appendix D to the RVI inspection plan summarizes the TMI-1 response to the Applicant/Licensee Action Items from MRP-227-A. With respect to Action Item 2, the licensee stated that the only TMI-1 component not addressed by MRP-227-A was the vent valve locking devices. However, the discussion of Applicant/Licensee Action Item 2 does not address whether any components were fabricated from materials not consistent with "Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items (MRP-189-Revision 1)" (Reference 7).

Confirm that the materials of the TMI-1 RVI components are consistent with those listed in MRP-189. If any materials are different, discuss whether additional aging mechanisms screened in for the components. Provide plant-specific aging management recommendations for these components as necessary to address the additional aging mechanisms.

RAI 8

The following materials used in the PWR RVI components are known to be susceptible to some of the aging degradation mechanisms that are identified in MRP-227-A, but may not have been identified in the evaluation of the generic B&W plant design in MRP-189.

- Nickel base alloys – Alloy 600; Weld Metals - Alloy 82 and 182 and Alloy X-750
- Stainless steel type 347 material (excluding baffle-former bolts)
- Precipitation hardened (PH) stainless steel materials - 17-4 and 15-5
- Type 431 stainless steel material.

Please provide the following information:

1. Identify any such materials currently used in the RVI components at TMI-1, unless already identified in MRP-189.

2. If any of the materials listed above were used in the RVI components at TMI-1, and not consistent with the material for that component identified in MRP-189, provide a plant-specific aging management recommendation for the components.

References

1. EPRI MRP letter to NRC dated January 9, 2012, “Transmittal: PWR Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)” (ADAMS Accession No. ML120170453).

2. Exelon letter TMI-12-069 to the NRC dated April 16, 2012 (ADAMS Accession No. ML12108A029), Attachment 1, “Inspection Plan for the Three Mile Island Unit 1 Reactor Vessel Internals,” Areva Document 77-2952-001, ANP-2952, Revision 001, dated March 2012.

3. Westinghouse report WCAP-17096-NP, Revision 2, dated December 2009, “Reactor Internals Acceptance Criteria Methodology and Data Requirements” (ADAMS Accession No. ML101460157).

4. Exelon letter TMI-13-069 to NRC dated April 17, 2013, “Submittal of Inspection Plan for Reactor Internals” (ADAMS Accession No. ML13108A004).

5. Babcock and Wilcox Owners Group (B&WOG) letter to the NRC dated April 21, 2000, “B&WOG License Renewal Task Force Topical Report BAW-2248A, “Demonstration of the Management of Aging Effects for the Reactor Vessel Internals,” (ADAMS Accession No. ML003708443).

6. NRC letter to EPRI dated May 28, 2013, “License Renewal Interim Staff Guidance LR-ISG-2011-04: Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors” (ADAMS Accession No. ML12270A464).

7. EPRI report MRP-189, Revision. 1, “Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items,” dated March 31, 2009 (non-publicly available).