UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
2100 RENAISSANCE BLVD., SUITE 100
KING OF PRUSSIA, PA 19406-2713

November 13, 2014

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

SUBJECT: THREE MILE ISLAND STATION, UNIT 1 – NRC INTEGRATED INSPECTION REPORT 05000289/2014004

Dear Mr. Pacilio:

On September 30, 2014, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Three Mile Island, Unit 1 (TMI) facility. The enclosed inspection report documents the inspection results, which were discussed on October 17, 2014, with Mr. Rick Libra, TMI Site Vice President, and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents one violation of NRC requirements, which was of very low safety significance (Green). This finding was determined to involve a violation of NRC requirements. However, because of the very low safety significance, and because it was entered into your corrective action program, the NRC is treating this finding as a non-cited violation, consistent with Section 2.3.2a of the NRC Enforcement Policy. If you contest the non-cited violation in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Three Mile Island. In addition, if you disagree with the cross-cutting aspect assigned to this finding, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region I, and the NRC Resident Inspector at Three Mile Island.

In accordance with 10 CFR 2.390 of the NRCs "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the

NRC's document system (ADAMS). ADAMS is accessible from the NRC website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Silas R. Kennedy, Chief
Reactor Projects Branch 6
Division of Reactor Projects

Docket Nos.: 50-289
License Nos.: DPR-50

Enclosure: Inspection Report 05000289/2014004
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-289
License No:	DPR-50
Report No:	05000289/2014004
Licensee:	Exelon Generation Company
Facility:	Three Mile Island Station, Unit 1
Location:	Middletown, PA 17057
Dates:	July 1 through September 30, 2014
Inspectors:	D. Werkheiser, Senior Resident Inspector, Division of Reactor Projects (DRP) J. Heinly, Resident Inspector, DRP S. Barr, Senior EP Inspector, Division of Reactor Safety (DRS) M. Modes, Senior Reactor Inspector, DRS H. Gray, Senior Reactor Inspector, DRS T. Burns, Reactor Inspector, DRS
Approved by:	S. Kennedy, Chief Projects Branch 6 Division of Reactor Projects (DRP)

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SUMMARY

IR 05000289/2014004, 07/01/2014 - 09/30/2014; Three Mile Island, Unit 1, Maintaining Emergency Preparedness.

This report covered a three-month period of inspection by resident inspectors and announced inspections performed by regional inspectors. Inspectors identified one finding of very low safety significance (Green), which was a non-cited violation (NCV). The significance of most findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, “Significance Determination Process” (SDP), dated June 2, 2011. Cross-cutting aspects are determined using IMC 0310, “Aspects Within Cross-Cutting Areas,” dated December 19, 2013. All violations of NRC requirements are dispositioned in accordance with the NRC’s Enforcement Policy, dated July 9, 2013. The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 5.

Cornerstone: Emergency Preparedness

- Green. The inspectors identified an NCV of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.54(q)(2), 10 CFR 50.47(b)(10), and 10 CFR 50, Appendix E, Section IV.4, for failing to maintain the effectiveness of the Three Mile Island Nuclear Station (TMI) emergency plan as a result of failing to provide the station evacuation time estimate (ETE) to the responsible offsite response organizations (OROs) by the required date. Upon identification, Exelon entered this issue into its corrective action program (CAP) as issue reports (IRs) 1525923 and 1578649. Exelon submitted a third ETE for TMI on April 4, 2014, and the NRC’s review of that ETE is documented in section 1EP4 of this report.

  The finding is more than minor because it is associated with the Emergency Preparedness cornerstone attribute of procedure quality and adversely affected the cornerstone objective of ensuring that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. The ETE is an input into the development of protective action strategies prior to an accident and to the protective action recommendation decision making process during an accident. Inadequate ETEs had the potential to reduce the effectiveness of public protective actions implemented by the OROs. The finding is determined to be of very low safety significance (Green) because it is a failure to comply with a non-risk significant portion of 10 CFR 50.47(b)(10). The cause of the finding is related to cross-cutting aspect of Human Performance, Documentation, because Exelon did not appropriately create and maintain complete, accurate and, up-to-date documentation [H.7]. (Section 1EP5)

REPORT DETAILS

Summary of Plant Status

Unit 1 began the inspection period at 100 percent power. On September 6, 2014, operators reduced power to approximately 89 percent to perform planned turbine valve testing and control rod drive exercises. Operators returned the unit to 100 percent on September 7, 2014, and remained at 100 percent for the remainder of the inspection period.

1. REACTOR SAFETY [R]

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 – 1 sample)

External Flooding

a. Inspection Scope

During the week of July 21, 2014, the inspectors performed an inspection of the external flood protection measures for TMI. The inspectors reviewed technical specifications; procedures; design documents; and the Updated Final Safety Analysis Report (UFSAR), Chapter 2.4.2.4, which depicted the design flood levels and protection areas containing safety-related equipment, to identify areas that may be affected by an external flooding event. The inspectors conducted a general site walkdown of all external areas of the plant, including a detailed review of the control building, and dike to ensure that Exelon erected and maintained flood protection measures in accordance with design specifications. The inspectors also reviewed operating procedures for mitigating external flooding during severe weather to determine if Exelon planned or established adequate measures to protect against external flooding events. Documents reviewed for each section of this inspection report are listed in the Attachment.

b. Findings

No findings were identified.

1R04 Equipment Alignment

.1 Partial System Walkdowns (71111.04Q – 3 samples)

a. Inspection Scope

The inspectors performed partial walkdowns of the following systems:

- 'B' make-up system alignment during 'A' unplanned system outage on July 14, 2014
- Emergency feedwater systems during 'B' emergency feedwater system outage on September 10 – 12, 2014
- Nuclear river alignment during seasonal clamicide and heat exchanger backwashes, on September 25 - 26, 2014

The inspectors selected these systems based on their risk-significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors reviewed applicable operating procedures, system diagrams, the UFSAR, technical specifications, work orders, issue reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have impacted system performance of their intended safety functions. The inspectors also performed field walkdowns of accessible portions of the systems to verify system components and support equipment were aligned correctly and were operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no deficiencies. The inspectors also reviewed whether Exelon staff had properly identified equipment issues and entered them into the corrective action program for resolution with the appropriate significance characterization.

b. Findings

No findings were identified.

.2 Full System Walkdown (71111.04S – 1 sample)

a. Inspection Scope

On September 26, 2014, the inspectors performed a complete system walkdown of accessible portions of the instrument air system to verify the existing equipment lineup was correct after a system outage of the main air compressor (IA-P-4) and upgrade to the air dryer (IA-Q-2) controller. The inspectors reviewed operating procedures, functional tests, drawings, equipment line-up check-off lists, and the UFSAR to verify the system was aligned to perform its required important-to-safety functions. The inspectors also reviewed electrical power availability, component lubrication and equipment cooling, hangar and support functionality, and functionality of support systems. The inspectors performed field walkdowns of accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no deficiencies. Additionally, the inspectors reviewed a sample of related issue reports and work orders to ensure Exelon appropriately evaluated and resolved any deficiencies.

b. Findings

No findings were identified.

1R05 Fire Protection

.1 Resident Inspector Quarterly Walkdowns (71111.05Q – 5 samples)

a. Inspection Scope

The inspectors conducted tours of the areas listed below to assess the material condition and operational status of fire protection features. The inspectors verified that Exelon controlled combustible materials and ignition sources in accordance with administrative procedures. The inspectors verified that fire protection and suppression equipment was available for use as specified in the area pre-fire plan, and passive fire barriers were maintained in good material condition. The inspectors also verified that station personnel implemented compensatory measures for out of service, degraded, or inoperable fire protection equipment, as applicable, in accordance with procedures.

- Fuel handling building 380' elevation (FH-FZ-5) on July 8, 2014
- Diesel Room 'B' and Control Panel (DG-FA-2) on July 21, 2014
- Control Building "A' Inverter Room (CB-FA-2D) on August 7, 2014
- Fuel handling building 355'/365' elevations (FH-FZ-5) on August 25, 2014
- South Heating and Ventilation Equipment Room (CB-FZ-5A) on September 4, 2014

b. Findings

No findings were identified.

.2 Fire Protection – Drill Observation (71111.05A – 1 sample)

a. Inspection Scope

The inspectors observed an unannounced fire brigade drill conducted on July 31, 2014, that simulated a fire in the radiological controlled area of the 306 foot elevation of the TMI-1 fuel handling building, specifically at air-handling unit AH-E-26. The inspectors evaluated the readiness of the plant fire brigade to fight fires. The inspectors verified that Exelon personnel identified deficiencies, openly discussed them in a self-critical manner at the drill debrief, and took appropriate corrective actions as required. The inspectors evaluated specific attributes as follows:

- Proper wearing of turnout gear and self-contained breathing apparatus
- Proper use and layout of fire hoses
- Employment of appropriate fire-fighting techniques
- Sufficient fire-fighting equipment brought to the scene
- Effectiveness of command and control
- Search for victims and propagation of the fire into other plant areas
- Smoke removal operations
- Utilization of pre-planned strategies
- Adherence to the pre-planned drill scenario
- Drill objectives met

The inspectors also evaluated the fire brigade's actions to determine whether these actions were in accordance with Exelon's fire-fighting strategies.

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06 – 1 sample)

Internal Flooding Review

a. Inspection Scope

The inspectors reviewed the UFSAR, the site flooding analysis, and plant procedures to assess susceptibilities involving internal flooding. The inspectors also reviewed the corrective action program to determine if Exelon identified and corrected flooding problems and whether operator actions for coping with flooding were adequate. The inspectors focused on building spray and decay heat pump vaults to verify the adequacy of equipment seals located below the flood-line, floor and water penetration seals, watertight door seals, common drain lines and sumps, sump pumps, level alarms, control circuits, and temporary or removable flood barriers.

b. Findings

No findings were identified.

1R08 In-service Inspection (71111.08 - 1 sample)

a. Inspection Scope

On September 18, 2014, the inspectors concluded an inspection of Exelon staff's evaluation of the tube wear indications of the two replacement once-through-steam generators which were discovered during the eddy current examination of the generators during the TMI Unit 1 20th refueling outage (T1R20).

Steam Generator (SG) Tube Inspection Activities (IMC Section 02.04)

The inspectors previously noted, in report 05000289/2013005, three SG tubes (48-118, 49-118, and 49-119) in SG B containing wear indications that increased from 0% through-wall-thickness to greater than the administrative plugging limit of 40% through-wall-thickness in one operating cycle. These tube to tube-support plate wear indications required additional review and evaluation by Exelon staff. Tube 49-119 with a wear indication of 62%, at tube support plate (TSP) 13, intersection 13S, was subject to the most limiting in-situ pressure testing criteria prior to plugging, and met the required pressure limits, without leaking, of 1.4 times the main-steam-line-break pressure (3605 psid), and 3 times the operating tube pressure differential (3ΔP of 3830 psid). The inspectors noted that tube 49-119 was held at 4800 psig for two minutes without structural failure or leakage. The inspectors further noted that tube 49-119 also had 58% wear at 10S, 48% wear at 11S, and 40% wear at 12S.

The inspectors noted that the number of new tube-support wear indications identified in T1R19 was 952 in SG A and 1232 in SG B. This increased during T1R20 to 2668 indications in SG A and 3600 indications in SG B. As a consequence of the 40% through wall wear limit Exelon staff plugged a total of 31 tubes in SG B with one tube preventatively plugged in SG A.

The inspectors reviewed the Exelon's probable root-cause report referenced in the Attachment to this inspection report. The inspectors verified Exelon's conclusion that the most probable cause was TSP partial locking to the shroud at the contact areas (wedges and alignment keys), by evaluating the basis for the identified contributing causes:

1. Reduced preload of the tubes may have increased TSP wear rates.

2. Manufacturing deviation from design specifications for shroud ovality during construction of the B SG may have increased the probability of TSP locking.

3. Higher steam flow exists at the location of the locked TSPs toward the periphery of the upper bundle in the B enhanced once-through-steam-generator.

The inspectors compared the results of Exelon's probable root cause with Exelon's Condition Monitoring and Operational Assessment (CMOA) report completed after T1R19 (listed in the Attachment to this inspection report). In the condition monitoring report, Exelon staff compared the as-found wear examination results from T1R19 for SG A and B with respect to criteria for structural integrity and accident leakage limits. The Operational Assessment section provided a "forward looking" evaluation of the A and B SG conditions to ensure structural integrity and accident leakage performance criteria identified in the TMI Unit 1 technical specifications will not be exceeded during the current operating cycle.

The condition monitoring report documented Exelon's conclusions that structural integrity and accident leakage limits were met during the previous operating cycle. The operational assessment report documented Exelon's conclusions that the structural integrity and leakage performance criteria were predicted to be met over the next period of operation until the next planned SG inspection.

The inspectors reviewed the CMOA and requested additional clarifying information from Exelon staff (ADAMS ML14301A200). These information requests and Exelon's responses are contained in the Special Attachment to this report. The inspectors subsequently completed onsite reviews of the CMOA and additional supporting analysis. The operational assessment and supporting analyses was compared against Nuclear Energy Institute (NEI) document 97-06, "Steam Generator Program Guidelines," Revision 3 (ADAMS ML111310712), and Electric Power Research Institute's "Steam Generator Integrity Assessment Guideline," Rev. 3 (ADAMS ML100480264). The inspectors noted the probability that the tube bundle will meet the minimum 3 delta P is predicated upon the probability of survival of each indication. The inspectors observed that Exelon staff computed over $1x10^6$ Monte Carlo simulations in order to conservatively assess the probability-of-survival for repeat indications returned to service which resulted in a probability of 0.978. The inspectors also noted that Exelon calculated the maximum depth indication for the next outage based on 1×10^4 Monte Carlo simulations and a conservatively increased population of 3000 indications. This resulted in an expectation of a median deep indication of 69% through wall with an upper 95^{th} value of 77% through wall. The probability of a structural flaw that exceeds the structural integrity limit is thus small.

b. Findings and Observations

No findings were identified.

The inspectors concluded:

1. The sensitivity analysis performed provides credibility to the conclusion that the performance criteria will be met at the next inspection (i.e., tube integrity will be maintained until the next refueling outage).

2. The analysis performed supporting operation until the next refueling outage is at the limits of the methodology/technology since it is based on a benchmark that is highly uncertain.

3. Our initial concern that the operational assessment was inconsistent with past inspection results was supported by the sensitivity study (namely, the Bernard probability function analysis).

1R11 Licensed Operator Requalification Program (71111.11Q – 2 samples)

.1 Quarterly Review of Licensed Operator Requalification Testing and Training

a. Inspection Scope

The inspectors observed licensed operator simulator training on September 9, 2014, which included a steam leak coincident with the failure of all alternating current power. The inspectors evaluated operator performance during the simulated event and verified completion of risk significant operator actions, including the use of abnormal and emergency operating procedures. The inspectors assessed the clarity and effectiveness of communications, implementation of actions in response to alarms and degrading plant conditions, and the oversight and direction provided by the control room supervisor. The inspectors verified the accuracy and timeliness of the emergency classification made by the shift manager and the technical specification action statements entered by the shift technical advisor. Additionally, the inspectors assessed the ability of the crew and training staff to identify and document crew performance problems.

b. Findings

No findings were identified.

.2 Quarterly Review of Licensed Operator Performance in the Main Control Room

a. Inspection Scope

The inspectors observed crew ‘A’ control room operations on September 10 and 11, 2014. Main activities were line-ups to support emergency feedwater partial system outage and control building ventilation changes for routine plant operations. The inspectors observed licensed operators performance to verify that procedure use, crew communications, and coordination of activities between work groups met the criteria specified in Exelon’s OP-AA-1, “Conduct of Operations,” Revision 0. In addition, the inspectors verified that licensee supervision and management were adequately engaged in plant operations oversight and appropriately assessed control room operator performance and similarly met established expectations and standards.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12Q – 1 sample)

a. Inspection Scope

The inspectors reviewed the sample listed below to assess the effectiveness of maintenance activities on structure, system, or component (SSC) performance and reliability. The inspectors reviewed system health reports, corrective action program documents, maintenance work orders, and maintenance rule basis documents to ensure that Exelon was identifying and properly evaluating performance problems within the scope of the maintenance rule. For the sample selected, the inspectors verified that the SSC was properly scoped into the maintenance rule in accordance with 10 CFR 50.65 and verified that the (a)(2) performance criteria established by Exelon staff was reasonable. Additionally, the inspectors ensured that Exelon staff was identifying and addressing common cause failures that occurred within and across maintenance rule system boundaries.

- North Bridge Structural Repairs on August 18, 2014

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 – 4 samples)

a. Inspection Scope

The inspectors reviewed station evaluation and management of plant risk for the maintenance and emergent work activities listed below to verify that Exelon performed the appropriate risk assessments prior to removing equipment for work. The inspectors selected these activities based on potential risk significance relative to the reactor safety cornerstones. As applicable for each activity, the inspectors verified that Exelon personnel performed risk assessments as required by 10 CFR 50.65(a)(4) and that the assessments were accurate and complete. When Exelon performed emergent work, the inspectors verified that operations personnel promptly assessed and managed plant risk. The inspectors reviewed the scope of maintenance work and discussed the results of the assessment with the station's probabilistic risk analyst to verify plant conditions were consistent with the risk assessment. The inspectors also reviewed the technical specification requirements and inspected portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met.

- Unplanned orange risk condition for an identified 'A' high pressure injection header piping weld leak and subsequent repair on July 10 - 15, 2014
- 'A' emergency diesel generator during 'A' air starting air compressor (EG-P-1A) unloader issues documented in IR 1686110 on July 29, 2014
- Yellow risk condition for a planned calibration of the '1E' 4 kilovolt degraded-grid relays on September 11, 2014
- Yellow risk condition for a planned station outage window of the main instrument air compressor (IA-P-4) on September 23, 2014

b. Findings

No findings were identified.

1R15 Operability Determinations and Functionality Assessments (71111.15 – 3 samples)

a. Inspection Scope

The inspectors reviewed operability determinations for the following degraded or non-conforming conditions:

- TMI review of Westinghouse nuclear safety advisory letter #14-1 regarding performance of reactor coolant pump seals during a postulated loss of seal cooling, documented in IRs 1625829 and 1624353, on July 3, 2014
- Identification of as-found zero-shift of the 'B' high pressure injection flow transmitter (MU-FT-1127) following recovery of the 'A' high pressure injection header as documented in IRs 1682046 and 1682189, on July 15, 2014
- TMI review of potential defects regarding 3 inch and 4 inch valve diaphragms reported by ITT Engineered Valves LLC under 10 CFR 21 (ENs 48976 and 50285), and documented in IRs 1508556 and 1527123, on July 31, 2014

The inspectors selected these issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the operability determinations to assess whether technical specification operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the technical specifications and UFSAR to Exelon's evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled by Exelon. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18 – 2 samples)

Permanent Modifications

a. Inspection Scope

The inspectors evaluated the following permanent modifications:

- Engineered Safeguards Actuation System (ESAS) relay replacement per engineering change request (ECR) 2013-00503
- Instrument Air Compressor Dryer (IA-Q-2) controller upgrade implemented by ECR 2014-00310

The inspectors verified that the design bases, licensing bases, and performance capability of the affected systems were not degraded by the modification. In addition, the inspectors reviewed modification documents associated with the upgrade and design change, including observing the physical replacement of the IA-Q-2 controller and the installation of the new ESAS cutler hammer relays. The inspectors also reviewed revisions to operating and test procedures.

a. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19 – 5 samples)

a. Inspection Scope

The inspectors reviewed the post-maintenance tests for the maintenance activities listed below to verify that procedures and test activities ensured system operability and functional capability. The inspectors reviewed the test procedure to verify that the procedure adequately tested the safety functions that may have been affected by the maintenance activity, that the acceptance criteria in the procedure was consistent with the information in the applicable licensing basis and/or design basis documents, and that the procedure had been properly reviewed and approved. The inspectors also witnessed the test or reviewed test data to verify that the test results adequately demonstrated restoration of the affected safety functions.

- ‘A’ high pressure injection header piping weld (near MU-V-1034) repair on July 15, 2014
- 1A containment purge valve diagnostic testing and repairs on August 6, 2014
- ‘B’ ESAS relays (63X/RC8B & 43/RC4B) replacement on July 17, 2014
- ‘B’ emergency feedwater surveillance testing following system outage that included a rebuild of the pump discharge check valve (EF-V-11B), and ‘B’ flow control valve controller (EF-V-30B) on September 9 – 12, 2014
- Rebuild of the ‘A’ condensate storage tank cross-connect valve actuator (EF-V-1A) on September 22, 2014

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22 – 5 samples)

a. Inspection Scope

The inspectors observed performance of surveillance tests and/or reviewed test data of selected risk-significant SSCs to assess whether test results satisfied technical specifications, the UFSAR, and Exelon procedure requirements. The inspectors verified that test acceptance criteria were clear, tests demonstrated operational readiness and were consistent with design documentation, test instrumentation had current calibrations and the range and accuracy for the application, tests were performed as written, and applicable test prerequisites were satisfied. Upon test completion, the inspectors considered whether the test results supported that equipment was capable of performing the required safety functions. The inspectors reviewed the following surveillance tests:

- 1302-5.18E, Calibration of 'A' Decay Heat Flow Transmitter DH-DPT-802, on July 1, 2014
- 1300-6F, Spent Fuel Pool Leakage Exam for IST, on July 21, 2014 (in-service test)
- OP-TM-541-203, IST of NS-V52A/B/C and NS-V-53A/B/C, on July 30, 2014 (in-service test)
- 1300-3Q.5, Quarterly In-service Testing of CM-V-1/2/3/4 Valves During Normal Plant Operations, on August 11, 2014 (containment isolation valve)
- OP-TM-220-252, Primary to Secondary Leak Rate Determination, on August 27, 2014 (leak rate test)

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness [EP]

1EP4 Emergency Action Level and Emergency Plan Changes (IP 71114.04)

a. Inspection Scope

The staff from the office of Nuclear Security and Incident Response (NSIR) performed an in-office review of the latest revision to "Evacuation Time Estimate Analysis for Three Mile Island Nuclear Station," located under ADAMS Accession Number ML14101A164, as listed in the Attachment.

The staff performed a review using the guidance provided in NUREG/CR-7002, "Criteria for Development of Evacuation Time Estimate Studies." The Updated Evacuation Time Estimate was found to be complete in accordance with 10 CFR Part 50, Appendix E.IV.3. The NRC review was only intended to verify consistent application of the ETE guidance contained in NUREG/CR-7002; and therefore remains subject to future NRC inspection in its entirety.

b. Findings

No findings were identified.

1EP5 Maintaining Emergency Preparedness (IP 71114.05)

Inspection Scope

NRC Emergency Preparedness (EP) rulemaking, which became effective on December 23, 2011, added a new regulation which required a licensee to develop an ETE analysis and submit it to the NRC by December 23, 2012. This inspection was a follow-up of issues identified by the NSIR staff during its review of the Exelon submittal of the ETEs for the ten sites that it operated at the time. The NSIR staff related those issues to Exelon, which provided responses through 2013 and into 2014. During this inspection period, regional EP inspectors reviewed applicable licensee documents, conducted discussions with licensee personnel, and provided assessment of the Exelon responses.

Findings

Introduction: The inspectors identified a Green NCV of 10 CFR 50.54(q)(2) for failing to maintain the effectiveness of the TMI emergency plan. Specifically, Exelon failed to provide the station ETE to responsible OROs and failed to update its site-specific protective action strategies as necessary as outlined in the requirements listed in 10 CFR 50.47(b)(10), and Section IV, Paragraph 4, of Appendix E to 10 CFR Part 50.

Description: On November 23, 2011, the NRC issued final new and amended emergency preparedness regulations (76 *Federal Register* 72560) that required all licensees to update the ETE on a periodic basis. This rulemaking became effective on December 23, 2011. The rulemaking also added a new regulation 10 CFR Part 50, Appendix E, Section IV.4, which required licensees to develop an ETE analysis using the most recent decennial census data and submit it to the NRC within 365 days of December 23, 2011. Concurrently, with the issuance of the rulemaking, the NRC published a new report entitled "Criteria for Development of Evacuation Time Estimate Studies," NUREG/CR-7002. The Statements of Consideration for the rulemaking (76 Federal Register 72580) identified that the NRC staff would review the submitted ETEs for completeness using that document. The Statements also provided that the guidance of NUREG/CR-2002 guidance was an acceptable template to meet the requirements and licensees should use the guidance or an appropriate alternative.

By individual letters dated December 12, 2012, Exelon submitted the ETEs for the sites for which it held the operating licenses, including TMI. By letter dated January 23, 2013, Exelon submitted the NUREG/CR-7002 checklists for the ETEs that identified where a particular criterion was addressed in the ETEs, facilitating the NRC review.

As provided in the Statements of Consideration, the NRC staff performed a completeness review using the checklists and found the ETEs (including the ETEs for TMI) to be incomplete due to common and site-specific deficiencies. The staff discussed its concerns regarding the completeness of the ETEs, in a teleconference with Exelon conducted on June 10, 2013. On September 5, 2013, Exelon resubmitted the ETEs and the associated checklists for its sites. The NRC staff performed another completeness review and again found the ETEs to be incomplete. Examples of information missing from the submittal included: peak and average attendance were not stated (NUREG/CR-7002 Criteria Item 2.1.2.a); the ETE used a value based on campsite and hotel capacity, vice an average value (2.1.2.b); basis for speed and capacity reduction factors due to weather was not provided (3.4.b); snow removal was not addressed (3.4.c); no bus routes or plans were included in the ETE analysis (4.1.2.a); and, no discussion on the means of evacuating ambulatory and non-ambulatory residents was included (4.1.2.b). The staff communicated the various ETE issues to Exelon through several telephone conference calls.

Upon identification, Exelon entered this issue into its CAP as IRs 1525923 and 1578649. Exelon submitted a third ETE for TMI on April 4, 2014, and the NRC's review of that ETE is documented in section 1EP4 of this report.

Analysis: The inspectors determined that the failure to submit a complete updated ETE for the TMI by December 23, 2012, is a performance deficiency because Exelon failed to meet a regulatory requirement that was reasonably within its ability to foresee and correct, and should have been prevented, for both the December 12, 2012, and September 5, 2013, submittals.

Using IMC 0612, Appendix B, "Issue Screening," the inspector determined that the performance deficiency is associated with the Emergency Preparedness cornerstone attribute of procedure quality and is more than minor because it adversely affected the cornerstone objective of ensuring that the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. The ETE is an input into the development of protective action strategies prior to an accident and to the protective action recommendation decision making process during an accident. Inadequate ETEs had the potential to reduce the effectiveness of public protective actions implemented by the OROs.

The inspectors utilized IMC 0609, Appendix B, "Emergency Preparedness Significance Determination Process (SDP)," to determine the significance of the performance deficiency. The performance deficiency was associated with planning standard 10 CFR 50.47(b)(10). EP SDP Table 5.10-1, "Significance Examples §50.47(b)(10)," provides two Green significance examples: "ETEs and updates to the ETEs were not provided to responsible OROs," and "The current public protective action strategies documented in emergency preparedness implementing procedures (EPIPs) are not consistent with the current ETE." The inspectors concluded that, because the performance deficiency delayed the NRC's approval of the TMI ETE, the ETE was not provided to the site OROs nor was it used to inform the site EPIPs as required by 10 CFR 50.47(b)(10), and Section IV, Paragraph 4 of Appendix E to 10 CFR Part 50. Therefore, in accordance with EP SDP Table 5.10-1, this was determined to be a finding of very low safety significance (Green).

The cause of the finding has a cross-cutting aspect in the area of Human Performance, Documentation, because Exelon personnel did not create and maintain complete, accurate and, up-to-date documentation. Specifically, the Emergency Preparedness organization did not develop the TMI ETE as required by the new regulation introduced by the NRC's EP Rule [H.7].

Enforcement: 10 CFR 50.54(q)(2) states, in part, that a licensee "shall follow and maintain in effective emergency plans which meet the standards in 10 CFR 50.47(b) and the requirements in Appendix E to this part." 10 CFR 50.47(b)(10), states, in part, that licensees shall develop an evacuation time estimate and update it on a periodic basis. 10 CFR Part 50 Appendix E, Section IV.4, states that within 365 days of December 23, 2011, nuclear power reactor licensees shall develop an ETE analysis and submit it under § 50.4. Contrary to the above, the ETEs submitted by Exelon on December 12, 2012, and on September 5, 2013, for TMI were found to be inadequate. Upon identification, Exelon implemented immediate corrective actions by entering this issue into its CAP as IRs 1525923 and 1578649 and revising the ETE to satisfy NRC requirements. Because this finding is of very low safety significance (Green) and was entered into Exelon's CAP, this issue is being treated as an NCV consistent with Section 2.3.2.a of the Enforcement Policy. **(NCV 05000289/2014004-01, Inadequate Evacuation Time Estimate Submittals)**

4. OTHER ACTIVITIES [OA]

4OA1 Performance Indicator Verification (71151)

.1 Safety System Functional Failures (1 sample)

a. Inspection Scope

The inspectors sampled Exelon's submittals for the Safety System Functional Failures performance indicator for TMI for the period of July 1, 2013 through June 30, 2014. To determine the accuracy of the performance indicator data reported during those periods, inspectors used definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, and NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 10 CFR 50.73." The inspectors reviewed Exelon's operator narrative logs, operability assessments, maintenance rule records, maintenance work orders, issue reports, event reports and NRC integrated inspection reports to validate the accuracy of the submittals.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index (5 samples)

a. Inspection Scope

The inspectors reviewed Exelon's submittal of the Mitigating Systems Performance Index for the following systems for the period of July 1, 2013 through June 30, 2014:

- [MS 06] Emergency AC Power System (Emergency Diesel Generators)
- [MS 07] High Pressure Safety Injection System (Makeup)
- [MS 08] Emergency Feedwater System
- [MS 09] Decay Heat Removal
- [MS 10] Cooling Water Support Systems (Decay Closed, Decay River, Nuclear Closed, Nuclear River)

To determine the accuracy of the performance indicator data reported during those periods, the inspectors used definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7. The inspectors also reviewed Exelon's operator narrative logs, issue reports, mitigating systems performance index derivation reports, event reports, and NRC integrated inspection reports to validate the accuracy of the submittals.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152 – 2 samples)

.1 Routine Review of Problem Identification and Resolution Activities

a. Inspection Scope

As required by Inspection Procedure 71152, "Problem Identification and Resolution," the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that Exelon entered issues into the CAP at an appropriate threshold, gave adequate attention to timely corrective actions, and identified and addressed adverse trends. In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the CAP and periodically attended issue report screening meetings.

b. Findings

No findings were identified.

.2 Annual Sample: Underground Pipe Leak from the "A" Decay River (DR) System

a. Inspection Scope

The inspectors performed an in-depth review of Exelon staff's identification, evaluation and corrective actions related to a leak in underground pipe from the TMI Unit 1 Intake Screen and Pump House (ISPH) in January 2012. Exelon staff observed indications of a leak on January 7, 2012. Exelon staff manipulated Decay Heat River Water System flows to determine the source of the water leaking to the surface was from "A" train of the Decay Heat River Water (DR-A) buried system piping. Exelon staff completed an operability evaluation and concluded the DR-A remained operable in part because the measured leak rate was well below the allowable leak rate of 160 gallons per minute for the DR system. Exelon staff established leakage limits for the remainder of the operating cycle under an "adverse condition monitoring plan" with action levels. During the refuel outage in November 2013, Exelon staff excavated the area where the surface water had accumulated and identified a circumferential crack in the DR piping. This component is pre-stressed concrete lined cylinder pipe twenty-four inches in diameter. The crack was located 14 foot south of the ISPH wall. Subsequently, an additional crack was discovered in the adjacent Secondary Services River Water (SR) system about one foot north of the DR-A failure. Exelon staff removed these pipe sections and completed testing and analysis to determine the causes of the pipe cracks.

Based on metallurgical analysis at the Exelon Power Labs, Exelon staff concluded cause of the pipe failures was due to pipeline settlement and likely thrust block movement resulting from a substandard installation during original construction. This conclusion was supported by a detailed analysis of the pipe failure and the as-found positions of the pipelines. The failures of the DR-A and SR pipe sections likely occurred over a substantial period of time. Inadequate pipe installation during initial plant construction, as noted during the pipeline excavation, caused pipeline and likely thrust block movement. The settling was not rapid and initially precipitated mortar cracks at the bottom outside diameter of the subject pipelines. The breech in the mortar coating enabled ground water to contact the inner steel liner and initiated a stress amplified corrosion process that eventually resulted in leakage from the pipe.

The DR-A piping was replaced during refuel outage T1R20. The SR pipe crack was repaired with an external seal and internal "Weko" Seals. Exelon staff concluded that since the SR pipe has been repaired, the degradation process has been arrested and structural integrity should be maintained. In addition, Exelon staff utilized "flowable" fill for the repair of the DR-A and SR Piping. The flowable fill provided bedding that does not have voids and cures to form a stable mass similar to a cast in place thrust block.

The inspectors reviewed Exelon staff's problem identification threshold, apparent cause analyses, extent of condition reviews, and timeliness of corrective actions related to this issue. The inspectors reviewed the documents noted in the Attachment to this report and interviewed engineering personnel to assess the effectiveness of the planned, scheduled, and completed corrective actions to arrest the system leakage.

b. Findings and Observations

No findings were identified.

The inspectors reviewed action requests, issue reports, system health reports, drawings, photographs, and procedures and determined the pipe repair was appropriately identified, documented, characterized and entered into Exelon's corrective action process consistent with Exelon CAP guidance and in compliance with 10 CFR Part 50, Appendix B requirements. Exelon Power Labs personnel determined that cracking was caused by corrosion and movement of pipe sections (bending) resulting from inadequate soil support.

The inspectors determined that this issue received appropriate management attention as indicated by the immediate corrective action taken to perform inspection and evaluation of both the "A" and "B" trains and the subsequent replacement of leaking pipe sections. The evaluations were of sufficient technical detail to identify the likely causes of the pipe leaks and to develop corrective actions that will likely be effective. Additionally, the inspectors concluded the extent of condition reviews were sufficient and involved locating and inspecting similar buried pipe sections that did not identify additional leaks. The inspectors determined that the corrective actions taken and planned adequately address the DR and SR piping.

.3 Annual Sample: Foxboro Power Supply 10 CFR 21 Corrective Actions

a. Inspection Scope

The inspectors performed an in-depth review of Exelon's assessment and corrective actions in response to a reported deficiency (10 CFR 21) of Foxboro Nuclear SPEC-200 power supply potential failures due to defective cable ties and cable tie anchors (EN 48863). TMI identified that their current in-service and in-stock Foxboro power supplies were subject to the reported 10 CFR 21 deficiencies. Specifically, the cable ties and associate anchors had been known to degrade and fall into the power supply chassis, potentially causing a malfunction of the power supply. Exelon performed a review of the current power supply population at TMI and developed corrective actions to address the issue under IR 1496437.

The inspectors assessed Exelon's problem identification threshold, cause analyses, extent of condition reviews, compensatory actions, and the prioritization and timeliness of Exelon's corrective actions to determine whether Exelon was appropriately identifying, characterizing, and correcting problems associated with this issue and whether the planned and completed corrective actions were appropriate and timely. The inspectors compared the actions taken to the requirements of Exelon's CAP and 10 CFR 50, Appendix B. In addition, the inspectors performed field walkdowns and interviewed engineering personnel to assess the effectiveness of the implemented corrective actions.

b. Findings and Observations

No findings were identified.

In 2013, Integrated Resources Inc. reported the discovery of repeated defects in the Foxboro power supplies used in Foxboro SPEC-200 cabinetry and reported the defects in a 10 CFR 21 report (EN 48863). The vendor identified that the tie wraps used to hold and route the power supply cable bundles throughout the chassis had experienced age related failures. In addition, the metallic tie wrap anchors adhesive would degrade over time and the failure of both the tie wrap and the anchor adhesive allowed the metallic anchor to fall into the chassis and potentially damage a circuit.

Exelon performed an immediate review of the Foxboro power supply 10 CFR 21 report. Consequently, the defective components were identified during refurbishment of TMI Foxboro power supplies. Therefore, TMI was susceptible to the failure mechanism reported. Exelon performed an immediate operability review for the Foxboro power supplies currently in use in safety related applications and determined that no immediate operability concerns existed. However, expeditious replacement of the power supplies was warranted. Exelon entered the condition adverse to quality into the CAP under IR 1496437. Corrective actions include the removal and refurbishment of the in-service power supplies at the next opportunity commensurate with plant risk and operating mode. In addition, the warehouse stock of power supplies was refurbished to preclude the introduction of future failures.

The inspectors reviewed Exelon's prompt operability determination, list of applicable power supplies in-service and actions to preclude future failures of the Foxboro power supplies. The inspectors determined that Exelon's actions were reasonable and completed commensurate with the safety significance of the condition. In addition, the inspectors reviewed the refurbishment work order activity that resolved the condition adverse to quality and identified no issues of significance.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153 – 3 samples)

.1 Plant Events

a. Inspection Scope

For the plant event listed below, the inspectors reviewed and/or observed plant parameters, reviewed personnel performance, and evaluated performance of mitigating systems. The inspectors communicated the plant events to appropriate regional personnel, and compared the event details with criteria contained in IMC 0309, "Reactive Inspection Decision Basis for Reactors," for consideration of potential reactive inspection activities. As applicable, the inspectors verified that Exelon made appropriate emergency classification assessments and properly reported the event in accordance with 10 CFR Parts 50.72 and 50.73. The inspectors reviewed Exelon's follow-up actions related to the events to assure that Exelon implemented appropriate corrective actions commensurate with their safety significance.

- Through-Wall Leak on High Pressure Injection (HPI) 'A' Train Root Valve MU-V-1034 Socket Weld on July 10, 2014

b. Findings

No findings were identified.

.2 Notice of Enforcement Discretion (NOED) 14-1-03: NOED for Exelon Generation Company, LLC Regarding Three Mile Island Unit 1

a. Inspection Scope

On July 10, 2014, at 5:30 p.m., the 'A' train of HPI was declared inoperable, and a 72-hour limiting condition for operation (LCO) time clock was entered, when a small leak (1 drop every 2 minutes) was identified from a welded connection upstream on the line-side of an instrument root isolation valve (MU-V-1034). TMI Technical Specification 3.3.2 allows one train of HPI to be removed from service for maintenance during reactor operations for no more than 72 consecutive hours. During the execution of system isolation and non-destructive evaluation, Exelon determined that additional equipment was needed to obtain the necessary isolation of the system to complete the repairs. Exelon requested enforcement discretion of TMI's Technical Specification 3.3.2 for 46.5 hours past the original expiration time (5:30 p.m. July 13, 2014) to affect repairs and restore the system.

The NRC staff reviewed Exelon's request, which adequately addressed IMC 0410 criteria, and verbally granted the NOED during a telephone call at 5:19 p.m. on July 13, 2014. Exelon subsequently submitted a letter (ADAMS ML14197A293) on July 14, 2014, documenting information previously discussed with the NRC on telephone conferences held on July 13, 2014. Following completion of repairs and post-maintenance testing, Exelon exited Technical Specification LCO 3.3.2 at 9:53 a.m. on July 15, 2014, incurring 40 hours and 23 minutes of the 46.5 hours granted. The NRC documented the NOED details and confirmation that Exelon's letter was consistent with the verbal NOED request in a letter dated July 17, 2014 (ADAMS ML14198A494).

Inspector activities during the NOED process included review and evaluation of technical documents, participation in teleconferences concerning the NOED request, verification, to the extent practicable, of Exelon's oral assertions before the NOED was granted, and verification of Exelon compensatory actions to reduce the risk associated with plant configurations during HPI weld repairs.

The inspectors monitored licensee activities throughout the socket weld repairs including system isolation, troubleshooting, repair, and restoration. The inspectors also observed control room activities including use of procedures to configure the systems, technical specification adherence, NRC notification of the event, and operator actions during the system restoration.

b. Findings and Observations

No findings were identified.

Exelon entered the issue into their CAP (IR 1680680) and performed an Equipment Apparent Cause Evaluation to determine the cause of the failed welded connection upstream on the line-side of an instrument root isolation valve (MU-V-1034) and evaluate the cause of delays in achieving system isolation and plant conditions to allow leak repair. The most-probable cause of the weld leak was determined to be stress corrosion cracking. The cause of the delays in achieving conditions to commence repairs was determined to be due primarily to challenges in the availability of adequate freeze seal jackets and vender technical support in determining an adequate freeze seal. The inspectors determined that the actions taken by Exelon to achieve early isolation for repairs, though initially unsuccessful, were timely and appropriate based on operating experience and plant conditions. The inspectors concluded there was no performance deficiency related to causes which led to the need for the NOED.

The inspectors identified a minor violation of NRC requirements while monitoring licensee activities during the initial response to the weld leak. Specifically, on July 11, 2014, during the draining of the ‘A” HPI header, in preparation for welding activities, operators exceeded the total maximum allowable leakage of 15 gallons per hour (GPH) into the auxiliary building sump as specified in Technical Specification Surveillance Requirement 4.5.4.1. The crew took immediate corrective actions to station a dedicated operator for prompt isolation, as needed, and continued to monitor sump in-leakage during draining. The inspectors determined the issue to be minor based on in-leakage did not exceed the 30 gallons per hour value assumed in the UFSAR (section 14.2.2.5) and that it had low safety impact considering the affected system (‘A’ HPI) was out of service. This issue was documented in IR 1681075. The failure to comply with Technical Specification Surveillance Requirement 4.5.4.1 constitutes a minor violation that is not subject to enforcement action in accordance with the NRC’s Enforcement Policy.

In most cases, the inspectors would open an unresolved item (URI) when the NOED was granted in accordance with IMC 0410, “Notices of Enforcement Discretion.” The purpose of the URI would be to determine if there is a performance deficiency for causes which led to the need for the NOED. Because this inspection activity occurred during the same quarter in which the NOED was granted and the inspectors did not identify any performance deficiency associated with the NOED, the inspectors determined that an URI was not required for this issue.

.3 <u>(Closed) Licensee Event Report (LER) 05000289/2013-001-00 and Supplement 01</u>: Reactor Coolant ‘B’ Cold Leg Drain Line Flaw

a. <u>Inspection Scope</u>

During the November 2013 refuel outage (1R20) a flaw, a short crack approximately 66% thru wall, was identified in the reactor coolant "B" cold leg 2-inch drain line elbow to pipe weld RC-289. No leakage was associated with the flaw. As this flaw represented a degraded reactor coolant system pressure boundary, the condition was reported to the NRC as required by 10 CFR 50.72(b)(3)(ii)(A), and LER 2013-001-00, "Reactor Coolant "B" Cold Leg Drain Line Flaw" was initiated by the TMI Staff. The area surrounding the flaw was removed and provided for metallurgical laboratory analysis, and replacement to American Society of Mechanical Engineers Code standards of the material in the flaw area was completed. The pipe section was destructively tested for analysis and results documented in supplemental LER 2013-001-01.

As part of the in-service inspection by the NRC during the 1R20 refueling outage, the ultrasonic examination results were observed by the inspector and the previous weld examination results and history were reviewed. Additionally, the inspector reviewed the initial root cause report dated December 20, 2013, the LERs dated January 6 and June 20, 2014, and the final root cause report dated May 16, 2014, that included the metallurgical laboratory results. The inspectors did not identify any new issues during the review of the LERs. These LERs are closed.

.4 (Closed) LER 05000289/2014-002-00: Through-Wall Leak on High Pressure Injection (HPI) 'A' Train Root Valve MU-V-1034 Socket Weld

The description of the event related to this LER is documented in section 4OA3.2, NOED 14-1-03: NOED for Exelon Generation Company, LLC Regarding Three Mile Island Unit 1. As this represented a condition prohibited by plant technical specifications, it was reported to the NRC as required by 10 CFR 50.72(a)(2)(i)(B), and LER 2014-002-00. The LER was reviewed and no new findings were identified. However, the inspectors identified a minor violation of Technical Specification Surveillance Requirement 4.5.4.1 which is documented in section 4OA3.2. This LER is closed.

4OA6 Meetings, Including Exit

Quarterly Inspection Report Exit

On October 17, 2014, the inspectors presented the inspection results to Mr. Rick Libra, TMI Site Vice President, and other members of the TMI staff. The inspectors verified that no proprietary information was retained by the inspectors or documented in this report.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel	
R. Libra	TMI-1, Site Vice President
M. Newcomer	TMI-1, Plant Manager
K. Aleshire	Exelon Corporate Emergency Preparedness Manager
T. Alvey	Manager, Chemistry
D. Atherholt	Manager, Regulatory Assurance
R. Campbell	Manager, Site Security
E. Carreras	TMI-1 Shift Manager
V. Cwietniewicz	Mid-Atlantic Corporate Emergency Preparedness Manager
D. Divittore	Manager, Radiological Engineering
M. Fitzwater	Senior Regulatory Assurance Engineer
L. Friante	Exelon Steam Generator Corporate Engineer
R. Green	Programs Engineer
R. Harris	Control Room Supervisor
T. Heindl	Exelon Site Steam Generator Engineer
M. Jesse	Regulatory Assurance Manager
R. Mas oero	TMI-1 Design Engineer
F. McGuire	Design Engineer
J. Morrissey	Work Week Manager
R. Myers	TMI-1 Fire Marshal
J. Piazza	Senior Manager, Design Engineering
B. Price	Shift Manager
E. Sharkey	Chemistry Technician
B. Shumaker	Manager, Emergency Preparedness
G. Smith	Director, Maintenance
T. Stertzel	Systems Engineer
M. Sweigart	Chemistry Lab Supervisor
S. Taylor	Fire Protection Program Engineer
M. Torborg	Manager Engineering Programs
NRC	
M. Gray	Chief Engineering Branch 1, Region I
K. Karwoski	Senior Technical Advisor SG, Division of Engineering, NRR
E. Murphy	Senior Steam Generator Engineer, Division of Engineering, NRR
A. Johnson	Steam Generator Engineer, Division of Engineering, NRR
Other Personnel	
D. Dyckman	Nuclear Safety Specialist Pennsylvania Department of Environmental Protection Bureau of Radiation Protection

LIST OF ITEMS OPENED, CLOSED, DISCUSSED, AND UPDATED

Opened/Closed

05000289/2014004-01	NCV	Inadequate Evacuation Time Estimate Submittals (Section 1EP5)

Closed

05000289/2013-001-00,01	LER	Reactor Coolant "B" Cold Leg Drain Line Flaw (Section 4OA3.3)
05000289/2014-002-00	LER	Through-Wall Leak on High Pressure Injection (HPI) 'A' Train Root Valve MU-V-1034 Socket Weld (Section 4OA3.4)

LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures
SDBD-T1-122, Flood Protection Systems, Rev. 3
OP-TM-AOP-002, Flood, Rev. 11

Drawings
1E-122-01-1002, TMI Flood Barrier System Control Building, Rev. 2

Miscellaneous
IR 1684837

Section 1R04: Equipment Alignment

Procedures
1104-25, Instrument and Control Air System, Rev. 149
IC-152, Instrument Air Walkdown, Rev. 5
OP-TM-211-000, Makeup and Purification System, Rev. 27
OP-TM-424-000, Emergency Feedwater System, Rev. 12

Drawings
302-082, Emergency Feedwater Flow Diagram, Rev. 12
302-270, Instrument Air, Rev. 5
302-271, Instrument and Station Service Air, Rev. 72
302-272, Backup Instrument Air, Rev. 22
302-277, Instrument Air Loads, Shts 1 thru 13, Rev. Various
302-661, Make-up and Purification, Rev. 61

Miscellaneous

IRs:	2380995	2344991	2178778	2344720	2178776	
WOs:	R2013013	R2218974	R2227036	C2020938	14500953A1	14500088A1

Section 1R05: Fire Protection

Procedures
1038, Administrative Controls-Fire Protection Program, Rev. 76
1038, Fire Protection Program, Rev. 80
1303-12.18.1, Fire System Nozzle Flow Test – Diesel Generator Cooling Air Intake Deluge, Rev. 1
CC-AA-309-101, Engineering Technical Evaluations, Rev. 11
OP-AA-201-003, Fire Drill Record for 3/1/D/14, dated July, 31, 2014
OP-MA-201-007, Fire Protection System Impairment Control, Rev. 6

Miscellaneous
Fire Hazards Analysis Report, TMI-1, 990-1745, Rev. 26
TMI Unit #1 Fire Pre-Plan & Strategies, FH-FZ-5 (355'/365'), Rev. 4

IRs:	*01696237	*01696240				
WOs:	R2058224	C2032664	C2032402	C2028693	C2031830	R2185749

*IR generated as a result of NRC inspector observation

Section 1R06: Flood Protection Measures

Procedures
OP-TM-214-251, BS and DH Floor Drain Inspection, Rev. 1

Drawings
302-719, Sump Pump and Draining System, Rev. 63
302-690, RC Bleed Tanks and Misc Waste Collection, Rev. 26

Miscellaneous

IRs:	2178966	1424828	1424816			
WOs:	R2208327	R2208270	R2222087	R2227559	R2226572	C2028187

Section 1R08: In-service Inspection (also see Special Attachment at end of document list)

References
AREVA Document 51-9172250-000 "Condition Monitoring and Operational Assessment of TM1-1 Steam Generators at 1R19"
AREVA Document 51-9216830-000 "Condition Monitoring and Final Operational Assessment of TMI-1 Steam Generators at 1R20"
AREVA Document 32-9018798-002, "TMI-1 EOTSG Tube Support Plate/Support Rods Stress Analysis"
AREVA Document 32-9075055-000, ''TMI-1 EOTSG Lower Span Tie Rod Fatigue Analysis"
EPRI Technical Report 1025132, "Steam Generator Management Program: Steam Generator In Situ Pressure Test Guidelines, Revision 4", October 2012.

Section 1R11: Licensed Operator Requalification Program

Procedures
OP-AA-1, “Conduct of Operations“ Rev. 0
OP-TM-424-000, Emergency Feedwater System, Rev. 12
OP-TM-424-212, IST of EF-V-30s and EF-V-52s, Rev. 7

Drawings
302-082, Emergency Feedwater Flow Diagram, Rev. 12

Miscellaneous
TQ-TM-LRU-106-S039, TMI Operational Simulator Examination Scenario, Rev. 0
TQ-AA-155-F05, Simulator Evaluation Form, Rev. 2

Section 1R12: Maintenance Effectiveness

Procedures
EP-AA-113-F-10, TMI Assembly, Accountability and Evacuation Guidelines, Rev. D
ER-AA-310, Implementation of the Maintenance Rule, Rev. 8
EN-TM-405-0001, TMINS Environment and Waterway Management Inspections, Rev. 0

Miscellaneous
2013 Bridge Inspection Report, September 10-11, 2013
2014 Transformer Load Capacity Analysis, June 2014
Design Review Summary, Permanent Access Bridge, February 6, 1973
Topical Report 198, Underwater Bridge Inspection, Rev. 0

WOs: C2032537 C2027852 GA202503 R2231628

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures
1082.1, TMI Risk Management Program, Rev. 8
1302-5.31D, 4160V 1E Bus Loss of Voltage/Degraded Grid Timing Relay Calibration & Logic Check, Rev. 21
OP-AA-108-117, Protected Equipment Program, Rev. 4
WC-AA-101, On-Line Work Control Process, Rev. 18

Miscellaneous
Plan of the Day Meeting Agenda, September 11, 23 - 25, 2014
IRs: 1686104 1686110 1686273 1680680 1680742 953680 2208788
WOs: R2227538 R2169053

Section 1R15: Operability Evaluations

Procedures
OP-AA-108-115, Operability Determinations, Rev. 10
OP-AA-108-115-1002, Supplemental Consideration for On-Shift Immediate Operability Determinations, Rev. 2

Miscellaneous
NRC Event Notification and Supporting vendor documents for EN# 48976 and EN#50285
Westinghouse NSAL 14-1, “Impact of Reactor Coolant Pump No. 1 Seal Leakoff Piping on a Reactor Coolant Pump Seal Leakage During a Loss of All Seal Cooling,” February 10, 2014

IRs:	1625829	1624353	1682046	1682189	1508556	1527123

Section 1R18: Plant Modifications

Procedures
1104-25, Instrument and Control Air System, Rev. 149
1430-IC-1, Preoperational Startup Testing of Instrumentation and Control Equipment, Rev. 3

Miscellaneous
CC-AA-102, Design Input and Configuration Change Impact Screening, Rev. 20
CC-AA-103, Configuration Change Control, Rev. 21
ECR 12-00471, ESAS AC Relay Replacement, Rev. 3
ECR 14-00503, ESAS Cabinet 1A & 4A Relay Replacement, Rev. 4
ECR 14-00301, Generate AR for IA-Q-2 Controller Upgrade, Rev 0
PHC Presentation, IA-Q-2 Controller Upgrade Project, June 25, 2014
Smart Controls, Smart Control Board Simplified Hardware Overview

Clearances:	14500955	14500314	14500728		
WOs:	C2032380	C2031572	C2031573	C2031928	C2031929
IRs:	2384051	1539778	1602793	1624812	1641028
	1649620	1681648	2384989	2385006	2385017
	2386285	2386340	2386288	2385528	2385520
	2385409	2385378	1191892	1188866	1184881
	1476037	1466929	1475880		

Section 1R19: Post-Maintenance Testing

Procedures
MA-AA-743-310, Diagnostic Testing and Evaluation of Air Operated Valves, Rev. 6
OP-TM-424-212, IST of EF-V-30s and EF-V-52s, Rev. 7
OP-TM-823-251, Local Leak Rate Testing of Purge Exhaust Penetration Valves, Rev. 5

Drawings
Isometric Weld Map, MU-V-1034, Rev. 0

Miscellaneous
ECR 14-00346, Tech Eval: Freeze Seal on HPI Line for MU-V-1034 Repair, Rev. 1

AR:	A2358916					
IRs:	1690130	2178776	2178778	2344720	2344991	2380995
WOs:	R2198435	R2137266	R2203273	R2013013	R2205955	R2218974
	C2032503	C2032302	C2020938			

Section 1R22: Surveillance Testing

Procedures
1104-6, Spent Fuel Cooling System, Rev. 45
1300-3EB, IST of ‘B’ SF Pump and Valves, Rev. 7
1300-3Q.5, Quarterly Inservice Testing of CM-V-1/2/3/4 Valves During Normal Plant Operations, Rev. 4
1301-3, Reactor Coolant System Chemistry and Activity, Rev. 30
1302-5.18E, Calibration of ‘A’ Decay Heat Flow Transmitter DH-DPT-802, Rev. 1
ER-AP-331-1003, RCS Leakage Monitoring and Action Plan, Rev. 7
ER-TM-321-1041, TMI-1 1st Program Requirements, Rev. 003
N1807.1, Reactor Coolant & Pressurizer Sampling, Rev. 17
OP-TM-220-251, RCS Leak Rate Determination, Rev. 12
OP-TM-220-252, Primary to Secondary Leak Rate Determination, Rev. 10
OP-TM-541-000, Primary Component Cooling, Rev. 20
OP-TM-541-203, 1st of NS-V-52a/B/C and NS-V-53A/B/C, Rev. 4A
RP-aa-15, Radioactive Contamination Control Program Description, Rev. 2
WC-TM-430, Surveillance Testing Program, Rev. 0
WC-TM-430-1001, Surveillance Testing Program Database Interface and Maintenance, Rev. 1

Drawings
302-728, Spent Fuel Pool and Fuel Transfer Canal Leak Detection, Rev. 1
302-630, Spent Fuel Cooling System, Rev. 32
302-640, Decay Heat Removal, Rev. 84
87N51539-CD-0001, Signal Conditioning Cabinet A3, Rev. 4

Miscellaneous

IRs:	1684444	1677762	1677767
WOs:	R2239222	R2203240	

Section 1EP4: Emergency Action Level and Emergency Plan Changes

Miscellaneous
Letter from J. Barstow (Exelon Generation Company, LLC) to: U.S. Nuclear Regulatory Commission, "10 CFR 50, Appendix E – Evacuation Time Estimate Analysis Information for Oyster Creek Nuclear Generating Station and Three Mile Island Nuclear Station," dated April 4, 2014 [ML14101A164]

Section 1EP5: Maintaining Emergency Preparedness

Miscellaneous
Letter from D. M. Gullott (Exelon Generation Company, LLC) to: U.S. Nuclear Regulatory Commission, "10 CFR 50 Appendix E - Evacuation Time Estimate Analysis for Three Mile Island Nuclear Station," dated December 12, 2012 [ML123550293]
Letter from D. M. Gullott (Exelon Generation Company, LLC) to: U.S. Nuclear Regulatory Commission, "10 CFR 50 Appendix E - Evacuation Time Estimate Analysis Checklists," date January 23, 2013 [ML13024A209]

Letter from J. Barstow (Exelon Generation Company, LLC) to: U.S. Nuclear Regulatory Commission, "10 CFR 50, Appendix E - Evacuation Time Estimate Analysis Supplemental Response for Braidwood Station, Byron Station, Clinton Power Station, Dresden Nuclear Power Station, LaSalle County Station, Limerick Generating Station, Oyster Creek Nuclear Generating Station, Peach Bottom Atomic Power Station, Quad Cities Nuclear Power Station, and Three Mile Island Nuclear Station," dated September 5, 2013 [ML13254A112]
Letter from J. Barstow (Exelon Generation Company, LLC) to: U.S. Nuclear Regulatory Commission, "10 CFR 50, Appendix E – Evacuation Time Estimate Analysis Information for Oyster Creek Nuclear Generating Station and Three Mile Island Nuclear Station," dated April 4, 2014 [ML14101A164]

IRs: 1525923 1578649

Section 4OA1: Performance Indicator Verification

Procedures
1107-3, Diesel Generator, Rev. 141
ER-AA-2008, Mitigating System Performance Index Monitoring and Margin Evaluation, Rev. 3
LS-AA-2200, Mitigating System performance Index Data Acquisition and Reporting, Rev. 5

Miscellaneous
ARs: A1873950
IRs: 1581323 1654405 1660323

Section 4OA2: Problem Identification and Resolution

Procedures
LS-AA-115-1003, Processing of Level 3 OPEX Evaluations, Rev. 3
OP-AA-108-111, Adverse Condition Monitoring and Contingency Plan ("A" DR Train), Rev. 9
OP-TM-533-251, DR Train A Leakage Exam – System Leakage Test, Rev. 10

Drawings
1E-133-07-015 "A" DR (Decay River Line) Leak in Underground Piping

Miscellaneous
1310997-37"A" DR Leak Treatment Delay Evaluation
13-00390, Update VM-TM-0674 as Requested per IR 1496437, Rev. 0
Customer Advisory # 2013027abi, Invensys Response to US NRC 10CFR21 EN 48863 as Reported by Integrated Resources Inc., June 7, 2013
LS-AA125-1003 R 10 Apparent Cause Report-Equipment
Operability Evaluation IR1310997, OpEval # OPE-12-001 System #533 Decay River System, Rev. 2
R2107510 Work Order DR Leakage Exam and Quarterly IST DR Train "A"
R2214119 Work Order DR Train "A" Leakage Exam
R2155147 Work Order Surveillance Test Procedure OP-TM-533-251on DR Train "A"
R2195393 Work Order Surveillance Test Procedure for IST of Train "A"
System 533 Decay Heat (DR) River Health Reports for Quarters 2012, 2013 and 2014
SI 0-00616, Power Supply Cable Tie Replacement for Heatsink Mounted Cable Harness, May 2013
Stock Code 200 05063, Inspection Criteria, March 15, 2014
TMI-00160 Failure Analysis of "A" Decay River Concrete Pipe Liner AR/CR/WO: 1310997
TMI BPRWCP Program TMI Buried Pipe Program Health Reports (2011, 2012, 2013 & 2014)

ARs:	*01310997	A2333606				
IRs:	1688163	1537537	1496437	1537552	1532653	1494443
	1537546	1499751				
WOs:	C2032369	R2141026	R2141555	R2141028	R2141558	R2141560

*Issue Report Assigned Actions of 01 thru 47 (latest 03/26/2014)

<u>Section 4OA3: Followup of Events and Notices of Enforcement Discretion</u>

<u>Procedures</u>

CC-AA-501-1025, Exelon Nuclear Welding Program Weld End Preparation and Joint Details, Rev. 52
ER-AA-335-002, Non-Destructive Examination, Rev. 006
MA-AA-736-610, Application of Freeze Seal to All Piping, Rev. 5
OP-AA-106-101-1001, Event Response Guidelines, Rev. 6
PI-AA-125-1003, Apparent Cause Evaluation, Rev. 1

<u>Drawing</u>

302-661, Make-Up & Purification Flow Diagram, Rev. 61

<u>Miscellaneous</u>

ASME XI, Section B31.7
BOP-PT-2014-008, NDE Report for MU-V-1034
Equipment Apparent Cause Evaluation 1680680
Engineering Change Request 14-00346, Tech Eval for Freeze Seal on HPI Line for MU-V-1034 Repair, Rev. 1
Licensee Event Report 2013-001-00, "Reactor Coolant "B" Cold Leg Drain Line Flaw"
Licensee Event Report 2013-001-01, "Reactor Coolant "B" Cold Leg Drain Line Flaw"
Licensee Event Report 2014-002-00, “Through-Wall Leak on High Pressure Injection (HPI) ‘A’ Train Root Valve MU-V-1034 Socket Weld”
Plant Computer Trend for Aux Building Sump Level (C4053), dated July 11, 2014.
Plant Oversight Review Committee Minutes for 2014-12 (LER 2014-002-00), dated September 9, 2014
Root Cause Report, Elbow to pipe weld RC-289 flaw, dated December 20, 2013
Root Cause Report, Elbow to pipe weld RC-289 flaw, dated May 16, 2014
Shift Operations Logs Dated July 10 – 15, 2014
TMI-14-102, Request for Enforcement Discretion for Technical Specification (TS) 3.3.2, “Emergency Core Cooling, Reactor Building Emergency Cooling and Reactor Building Spray Systems” dated July 14, 2014 (ML14197A293)
TMI-1 Technical Specifications, Amendment 278
Weld Map and Record for C2032503-WM-1
Updated Final Safety Analysis Report, Chapter 6 and 14, Rev. 22

Clearance:	14500903				
IRs:	1681075	1681639	1694480	1684217	1680680
	1680742	1681137	1681212	1682046	1682189
	1682298	1682386	1682410	1682825	1683017
	2384792				
WOs:	C2032503	A2358916			

SPECIAL ATTACHMENT for Section 1R08: In-service Inspection

Question 1 - What were the lengths of the flaws in tube 49-119 at TSPs 10, 11, and 12? Characterize the flatness of these flaws or, alternatively, provide the burst equivalent depths and lengths.

Exelon Response

Table 1 summarizes the axial lengths, and burst equivalent depths and lengths for the flaws at TSPs 10, 11 and 12 in tube 49-119 in SGB. These measurements are from the array probe exams.

Table 1: SG "B" TSP 10, 11 and 12 Sizing Results

Support	Flaw	Shape	Axial Length (in) (Note 1)	Burst Equivalent Depth (%TW}	Burst Equivalent Length (in)
12S	Flaw1	Tapered	1.83	33.4	0.65
	Flaw2	Tapered	0.68	13.9	0.47
IIS	Flaw1	Tapered	1.74	33.4	0.79
	Flaw2	Tapered	1.39	31.0	0.44
	Flaw3	Tapered	0.32	16.0	0.23
IOS	Flaw1	Tapered	1.77	54.5	0.82
	Flaw2	Tapered	0.35	12.2	0.30

Note 1: These are the total lengths of the flaws as measured with array probes. Note that several of these measurements are longer than the 1.18" thickness of the TSPs and is, hence, considered to be conservative. This overestimate of the lengths is believed to be attributable to the look-ahead and look-behind of the coil.

Question 2: For TSP indications found during 1R20 to exceed the plugging limit, how many were new indications, what was the depth, structural depth, and structural length of these new indications, and how many other indications at other TSPs were found on the same tube?

Exelon Response

A total of 14 TSP bobbin wear indications found during 1R20 met the technical specification plugging criterion of equal to or exceeding 40% thru-wall (TW), all of which were in SG B. Nine of these indications are "new" and were not detected during the previous inspection. However, of these nine indications, only four indications had maximum depths greater than or equal to 40%TW when sized with array probes. The array depths are generally lower due to the fact that the array coil can distinguish between multiple flaws at different lands at the same intersection. The bobbin coil cannot distinguish between different flaws in the same horizontal plane and, hence, combines the multiple flaws into a single, larger signal. In most of these cases, the bobbin depth will be deeper than the array depth. The bobbin depths were used in the OA analyses (i.e., no credit was taken for the conservative bobbin sizing in cases where there are multiple indications at the same TSP intersection).

Table 2 - below summarizes all of these new indications that exceeded the plugging criterion. This table includes the 1R20 Bobbin depth, the maximum array depth, the burst equivalent (i.e., structural) depths and lengths from the array exams, and the number of other TSP wear indications in the same tube as reported with bobbin probes.

Table 2: New Indications Meeting the Plugging Criterion of 40%TW

SG	Row	Tube	Elevation	1R20 Bobbin Depth (%TW)	Notes	Maximum Array Depth (%TW)	Burst Equivalent Depth (%TW)	Burst Equivalent Length (in)	# of Other Indications Detected by Bobbin
B	48	118	13S +0.00	40	-	36.2	31.0	1.53	91nds at 8 TSPs
			IOS +0.24	53	-	49.2	44.5	0.63	
B	49	118	12S +0.00	45	Note 1	40.1	36.9	0.66	81nds at 5 TSPs
						27.5	23.4	0.37	
			13S +0.00	46	Note 1	37.2	33.2	1.35	
						29.3	26.0	0.57	
B	49	119	11S +0.00	48	Note 1	37.9	33.4	0.79	61nds at 5 TSPs
						34.3	31.0	0.44	
						18.0	16.0	0.23	
			IOS -0.71	58	Note 2	60.1	54.5	0.82	
			IOS +0.37	44		14.0	12.2	0.30	
			12S +0.29	40	Note 1	36.6	33.4	0.65	
						16.3	13.9	0.47	
			13S +0.00	63 (Array)	-	63.3	61.1	1.26	

Notes:

1. Multiple indications were confirmed (i.e., at more than one trefoil land) and measured with array at these locations.

2. Due to the length, shape, and position of these flaws, bobbin identified two separate and relatively deep indications. However, the array probe showed that the indication at the upper edge of the TSP was much shallower than what was measured with the bobbin probe. Since the deep indication spanned the entire thickness of the TSP, the bobbin signal at the upper edge was enhanced by the deeper (lower edge) flaw and isn't representative of the depth of the shallow flaw at the upper edge.

Question 3 - Table 3-5 of the CMOA report indicates that a maximum depth of 58.3% had been projected for new indications at the 1R20 inspection. What associated lengths and burst equivalent lengths and depths had been projected?

Exelon Response

The 58.3%TW value was a deterministically projected maximum depth for the deepest new indication. This value was documented in the Operational Assessment performed after the 1R19 outage in the fall of 2011 (Reference 1).

No specific burst equivalent lengths and depths for this flaw were documented in the previous operational assessment. However, using the same assumptions used for structural length and structural depth ratios, a flaw with a structural length of 1.2 inches and a structural depth of 52.5%TW is obtained. Note that the previous operational assessment used a fixed structural length of 1.2".

Question 4 - The Condition Monitoring curves in Figure 4-1 of the CMOA account for yield plus ultimate strength uncertainty, burst pressure prediction model uncertainty, and eddy current measurement error. Was this curve established deterministically or probabilistically? If deterministically, what statistical bounds for each of these parameters were used?

AREVA Response

The condition monitoring curves shown in Figure 4-1 were generated probabilistically.

Question 5 - Please confirm that the 1.4 x MSLB criterion was met by analysis for tube 49-119 at TSP 13.

AREVA Response

Yes, the deep flaw at TSP 13 in tube 49-119 in SGB passes the 1.4 x MSLB criterion (3605 psi) analytically. The Condition Monitoring limit for a flaw with a structural length of 1.26" is 63.6%TW versus the measured structural depth of 61.1%TW.

Observation #1 - The CMOA analysis appears to take no credit for the radial restraint against burst provided by the TSP. The TSP does constrain the tube from bursting during the in-situ pressure test, given the limited amount of axial offset which occurs between hot and cold conditions. This appears appropriate if one is demonstrating that the 3 delta P criterion is met. However, in-situ pressure testing may not be valid for purposes of demonstrating that the 1.4 x MSLB criterion is met, unless it can be demonstrated that the TSP will not undergo significant axial displacement relative to the flaw location.

AREVA Response

The EPRI In Situ Pressure Test Guidelines [5] explicitly discuss the effects of tube supports on the in situ pressure test results. Specifically, Section 0.3.1 discusses the inadequacy of burst tests at demonstrating structural integrity due to the restraint provided by drilled supports; however, the guidelines also address broached supports, stating in the Note at the end of 0.3.1 that: "Degradation at lattice type eggcrate TSPs is not exempt from in situ proof testing, as it has been shown that these supports provide little strengthening to regions with axial degradation.

The same is true of broached TSP supports." In addition, the vertical displacement of the flaws between steady state operating conditions and cold test conditions provides some assurance that a portion of the flaw was outside of contact with the TSP. Finally, since leakage due to TSP wear occurs coincidently with tube burst, it can be concluded that successful completion of the in situ proof test per the passage from the EPRI Guidelines cited above, also validates that accident induced leakage criteria has also been met.

Also, as discussed in the response to Question 5, condition monitoring was met analytically for the worst-case flaw at TSP 13 in Tube 49-119 in SGB. Hence, in situ pressure testing was not required to confirm condition monitoring at 1.4 x MSLB conditions.

Observation #2 - Given that the 3 delta P Condition Monitoring limit is a structural (burst equivalent) depth of 61.1% for a flaw with a structural (burst equivalent) length of 1.18-inches, then the comparable Condition Monitoring depth limit for 1.4 x MSLB is approximately 63.5% (burst equivalent). This is very close to what was measured for tube 49-119 at TSP 13.

AREVA Response

As discussed in the response to Question 5, condition monitoring was met analytically at 1.4 x MSLB for the deep flaw in Tube 49-119 in SGB. The calculated Condition Monitoring limit at 1.4 x MSLB conditions is 63.6%TW versus a measured structural depth of 61.1%TW.

Question 6 - Figure 6-2 of the CMOA report shows the distribution of indications by depth that were returned to service in SG B. Are these the measured maximum depths for each indication or structural depths?

Exelon Response

Figure 6-2 shows the maximum depths for each indication. The depths shown are those recorded by the bobbin inspections except the deep flaw in Tube 49-119 in SG B. The deep flaw in Tube 49-119 was sized at 63%TW maximum depth using an array probe. This flaw was sized with an array probe since it exceeded the deepest flaw in the wear calibration standard used with the bobbin coil examination. The array sizing made use of a qualified fixed calibration curve.

Question 7- Confirm that the projected end-of-cycle structural lengths for both indications that were returned to service and projected new indications are based on the lognormal distribution fit given in Figure 6-4 of the CMOA report. If not, please clarify how the end-of-cycle structural lengths were determined for both indications returned to service and projected new indications.

Exelon Response

The end-of-cycle structural lengths for both new and repeat indications were based on the lognormal curve shown in Figure 6-4. The minimum and maximum values of the lognormal curve were specified. A minimum value of 0.3" and a maximum value of 1.3" were used.
It should be noted that Figure 6-4 and the text preceding Figure 6-4 are correct and consistent with what was used in the analyses. However, the values in Table 6-1 of the operational assessment (Reference 2) were incorrect. A corrected table is provided below. Note that the values for the structural depth ratio were also incorrect in Table 6-1 of the operational assessment.

None of these errors affected the OA analyses. The analyses were performed with the intended distributions. Only the documentation in Table 6-1 of the report was in error.

The CMOA report is being revised to correct this error and will be made available to the NRC.

Table 3: Corrected Summary Table of Inputs for the Probabilistic OA

Parameter	Value
Mean of the sum of yield and ultimate strengths at temperature	130.4 ksi
Standard deviation of the sum of yield and ultimate strengths at temperature	2.184 ksi
3 X Normal Operating Pressure Differential	3830 psid
Tubing wall thickness	0.0368 in.
Tubing outer diameter	0.625 in.
Structural length lognormal mean	-0.8
Structural length lognormal standard deviation	0.52
Structural length maximum	1.3
Structural length minimum	0.3
Structural depth ratio mean	0.897
Structural depth ratio standard deviation	0.028
Structural depth ratio maximum	1
Structural depth ratio minimum	0.85
ETSS Technique	96043.1 Rev 1
ETSS NDE depth sizing parameters	Slope= 1.05 Intercept = -1.13 Standard Error= 1.85 (Technique) Standard Error= 0.925 (Analyst)
Number of Monte Carlo cycles	1000000 for Probability of Survival 10000 for Max Depth Projections
Length of Cycle 20	1.919 EFPY

Question 8 - Confirm that for each Monte Carlo sample projection of end-of-cycle wear depth that a corresponding structural length was assigned based on a random sampling of the log normal distribution for structural length in Figure 6-4. If yes, what was the justification for doing so as opposed to assuming a bias of larger lengths to relatively large depths? If not, please clarify how each Monte Carlo sample projection of end-of-cycle wear depth was paired with a corresponding structural length.

Exelon Response

Yes, structural lengths were randomly sampled from the lognormal curve shown in Figure 6-4 with the exception of the minimum and maximum criteria as discussed above. If a value greater than the specified maximum length is randomly selected, the structural length is set to the maximum value (i.e., the curve is not resampled to obtain a new structural length). Likewise, if a value less than the specified minimum is selected, the structural length is set to the minimum value.

The basIs for not considering a bias of longer lengths to deeper indications is provided in the response to Question 9.

Question 9 - Was the eddy current data reviewed to establish whether or not larger structural lengths are biased toward being associated with deeper flaws? If so, what trends were observed?

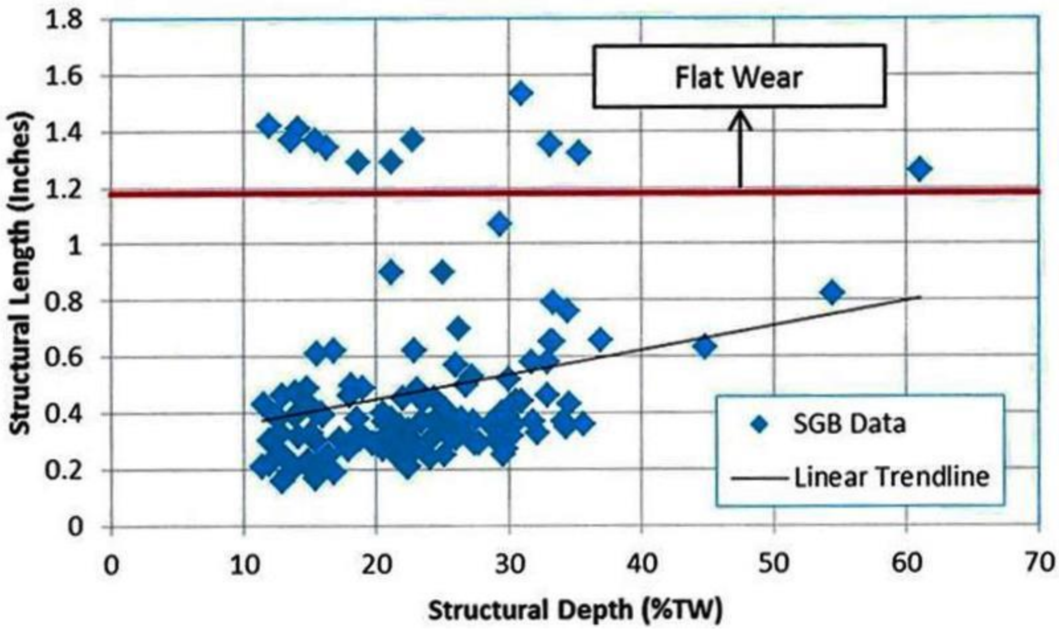
Exelon Response

Yes, there is a correlation between structural length and structural depth. However, this correlation is driven primarily by the shallow, tapered indications. To support the response to this question, additional line-by-line sizing was performed. This additional line-by-line sizing was focused on indications that the resolution analysts deemed to be flat as well as relatively shallow, tapered indications. This was done to complement the sizing that was performed during the outage which focused primarily on the deepest indications.

Figure 1 below shows the structural length plotted against structural depth for all profiled indications in SG B. As shown in the figure, there were twelve indications that were clearly flat as evidenced by a structural length greater than 1.2". Eleven of these twelve indications had structural depths less than 40%TW and six of these indications had structural depths less than 20%TW.

TMI-1 1R20 Responses to NRC Questions on CMOA

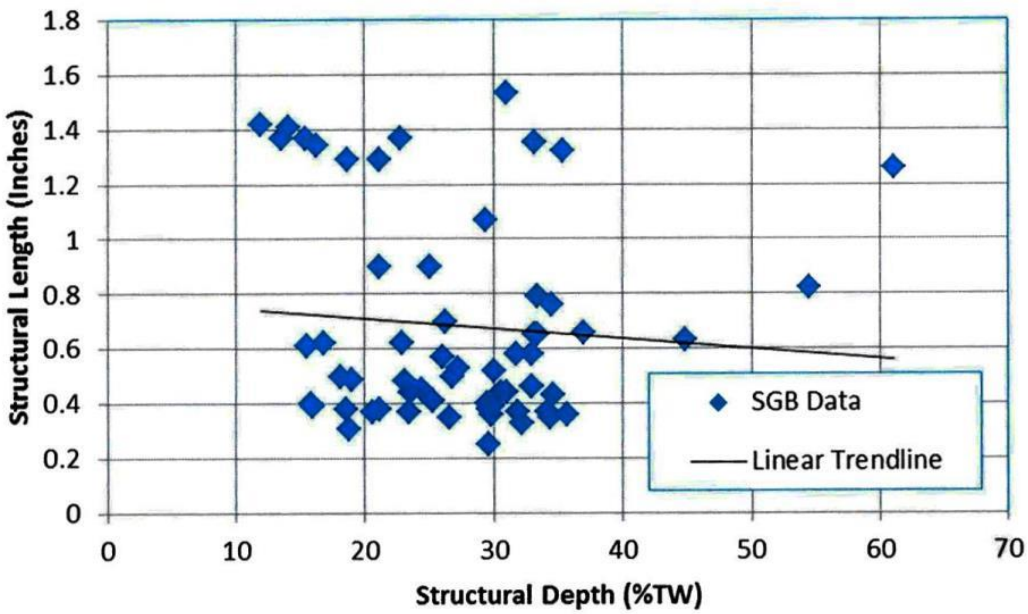
Figure 1: All Profile Results for SGB



As shown in Figure 1, there is some correlation between structural length and structural depth. However, as previously mentioned, this correlation is primarily driven by the shallow, tapered indications. When a tapered flaw first initiates, it will only be present at one edge of the support and may only have a total length of 0.4", for example. Obviously, if the total length is 0.4" and the flaw is tapered, its structural length will be less than 0.4". As the flaw grows in depth, its total length and structural length will increase provided the taper angle remains the same. This is the primary reason that there is an apparent relationship between structural length and depth.

In order to illustrate the effect that the shallow tapered indications have on this correlation, another curve was generated which includes only those profiled indications with total measured lengths greater than 1 inch. Applying this criterion effectively eliminates the shallow tapered indications since tapered flaws would need to grow to a relatively deep depth before they would reach a length of 1 inch. This relationship is shown in **Figure 2** as shown in the figure; there is no correlation of increasing structural length with increasing structural depth. Hence, for the flaws of most concern (flat or tapered and relatively deep), there is no correlation between structural length and structural depth.

Figure 2: Profile Results for Indications Greater Than 1 Inch Total Length



Question 10 - It is stated in the middle of page 50 of the CMOA report that "The probability of survival (POS) for each depth bin was calculated as the individual indication POS raised to the number of indications in that bin." This sentence and the example provided in the next sentence suggest that each indication in each depth bin is assumed to have the same POS. Please confirm or clarify. If this is the case, then each indication in each depth bin must have the same structural length. Why are the structural lengths for each of the indications in the bin the same rather than varying randomly from indication to indication?

Exelon Response

The probability of survival (POS) calculated for each depth bin includes random sampling of the various uncertainties including growth rates, non-destructive examination (NDE) uncertainties, structural lengths, and structural depth ratios. Therefore, the POS already includes a range of structural lengths (i.e., they are not assumed to be all the same). Since random structural lengths are included in the calculation of the POS, each indication in a given depth bin will have the same POS.

This method is statistically equivalent to the more traditional Monte Carlo approach where an end-of-cycle burst pressure is calculated for each flaw predicted at the end of the cycle.

Observation #3 - Operating experience over two cycles for the TMI SGs may indicate an approximate 0.5 probability of a wear flaw over the current operating cycle exhibiting a structural depth around 63% or more with a structural length of 1.18 - inches. Based on observation #2, this translates to a probability of approximately 0.5 of a tube reaching the 1.4xMSLB Condition Monitoring limit. That these are reasonable estimates is further supported by the fact that the current operating cycle is expected to be slightly longer than the previous cycles (i.e., 1.919 effective full power years (EFPY) versus 1.877 EFPY for cycle 19 and 1.719 for cycle 18) and that the number of new indications is predicted by the OA to continue to increase significantly compared to the previous cycle, creating more opportunity for at least one tube to exhibit extreme growth behavior.

Exelon Response

Based on the lack of a relationship between structural length and structural depth for the most significant flaws, the OA doesn't project a 0.5 probability of a flaw with structural depth of 63%TW and a structural length of 1.18 inches. As discussed in the response to Observation #4, the OA routinely predicts indications with maximum depths greater than or equal to 63%TW. However, it is a rarer occurrence that these deep flaws are paired with a long, flat wear shape.

It should also be noted that reaching the Condition Monitoring limit doesn't equate to an unacceptable probability of survival of less than 0.95. It only means that in situ pressure testing would be required. To illustrate this point, a flaw with a measured (array coil) structural length of 1.18 inches and a structural depth of 63.9%TW is at the 3 delta P Condition Monitoring limit and, hence, has a probability of survival of 0.95. However, a flaw with known (as opposed to measured) dimensions of the same length and depth has a probability of survival of 0.992.

Observation #4 - The licensee's OA predicts an overall POS (probability of survival; i.e., probability of satisfying performance criteria) of 0.968 for 3 delta P, taking no credit for constraint against burst provided by the TSPs. The CMOA report did not address 1.4xMSLB specifically, but the POS for this case would be even higher than the 0.968. Clearly, these results are at odds with the staff's observation #3. For the OA to be correct, the occurrence of the 63% deep, 1.18- inch long wear indication in tube 49-119 was simply extreme bad luck, defying the odds, and unlikely to happen again. However, given that the cause of the previously observed high growth rates is not well understood and that no corrective measures have been taken, this is a conclusion that is difficult to accept.

Exelon Response

The OA routinely predicts indications with maximum depths of at least 63%TW. An average of about two indications of this depth (or deeper) are projected in each Monte Carlo simulation. It is acknowledged that, in the OA analysis, pairing these deep flaws up with a long, flat indication is a more unusual occurrence. Applying the distributions for structural depth ratio and structural length leads to an indication meeting or exceeding the limiting 1R20 flaw (61.1%TW structural depth AND 1.26" structural length) in about 1.8% of the simulations. There are many other cases where either this structural length or structural depth (but not both) are exceeded. However, as discussed in the response to Question 9, there is no relationship between structural length and structural depth for the most significant flaws.

It should also be emphasized that the OA is predicting the likelihood of a Condition Monitoring failure (as opposed to the requirement of in situ pressure testing). To illustrate the difference, the limiting flaw at 1R20 is used as an example. Since the limiting flaw in 1R20 fell on the Condition Monitoring limit, it's POS (including NDE uncertainties) is about 0.95. However, a flaw of the same known dimensions (61.1%TW for a length of 1.26") has a POS of 0.994.

Observation #5 - A possible problem with the OA methodology may be the treatment of structural depth and structural length as totally independent parameters (if this is what was done (see question 8). The inspection data needs to be evaluated for any bias of longer structural lengths to the deepest flaws.

Exelon Response

See the response to Question 9 which discusses the lack of a relationship between structural length and depth for the most significant flaws.

Question 11 - Please confirm that the stresses in the tube supports have been assessed to ensure the supports will continue to perform their function during normal operating, transient, and accident conditions. In addition, please confirm that the tube supports will not impose unacceptable loads on the tubes in the event of design basis accidents given the locked condition of the supports.

Exelon Response

Assessment of inspection data, including vertical offsets of wear scars, gives evidence that the peripheral edges of the TSPs may be flexed upward during the time that the wear is accumulated. It is postulated that the TSP edges are frictionally bound to the adjacent shrouds as a result of unintended fabrication consequences. During heat-up, the shrouds move upward more than the TSPs (supported vertically by the tie rods) pulling upward on the TSP edges. Thus, for lateral loadings the flexed TSP edge introduces a geometric eccentricity that increases the stresses within the TSP beyond those of a flat TSP. Currently, the configuration of TSPs with edges flexed upward is not in the design basis.

To assess the impact of TSPs with flexed edges, the design basis analyses are visited. This assessment includes evaluation of the existing margins on the TSP criteria and estimation of effects due to the inclusion of the lateral load eccentricity.

The design analysis of the TSPs [3] lists the maximum range of displacement between the shroud and TSP as 0.58" [3, Section 4.3.1]. However, for the case of the TSP edge being pulled upward this amount, a resisting tension load is induced in the tie rods. Reference [4, Appendices A, B & C) performs an analysis of tie rod loads associated with three configurations of bound TSPs:

- All TSPs bound to the upper and lower shrouds
- All TSPs bound in the upper shroud (free TSPs in lower shroud)
- TSP 14 & TSP 15 bound in the upper shroud (all other TSPs are free)

The case with the minimum axial load in an outer tie rod for any of the three cases evaluated is 6000 pounds. Although this load was the result of a compression case, it did not result in tie rod bow, so it can be used to linearly estimate the behavior of the structure when the load is tensile. The 6000 pound tie rod load produces a calculated vertical stretch of 0.27". Therefore, the magnitude of eccentricity from the plane of the TSP (tie rod connections) to the vertical location shroud-to-bound TSP is 0.58"- 0.27" = 0.31". The thickness of the TSP is nominally 1.18". Therefore, the eccentricity is only approximately 1/4th of the TSP thickness.

Considering the magnitude of eccentricity associated with bound TSPs, the only significant influence is associated with large lateral loadings. Review of Table 4 - 2 and Section 4.3.2 [3] indicates the largest lateral loads correspond to faulted (Service Level D) conditions.

The design analysis [3] prudently analyzes a 'nominal' geometry and a 'corroded' geometry. Per [3, Section 4.8.1] for the faulted case, the applied conservatively- large lateral loads in the equivalent form of accelerations are 130 G's and 145 G's for the nominal and corroded geometries, respectively. However, Section 8.2.1 [3] indicates that the re-calculated applicable equivalent G-loads are 60.01 G's and 68.32 G's for the nominal and corroded geometries, respectively. Therefore, the loads reflected in the analysis are more than two-times the applicable values. Thus, the results listed in the design analysis are conservative by more than a factor of two based on consideration of loads alone.

The general methodology applied in the analysis of faulted conditions is 'elastic-perfectly plastic.' This method takes no credit for the inherent additional strength of the material beyond the specified yield stress. As noted in Section 4.10 [3], this method is conservative as compared to the more representative method of employing a 'true stress-true strain' curve to utilize the increment of strength when stressed above the yield.

The results of the very conservative elastic-perfectly plastic analysis indicate the worst case tube-TSP broached hole gap was reduced from a nominal value of 0.006" to 0.003" (only a 0.003" change under large loading and further conservative method).

Based on the discussion above, it is concluded that the effects due to the inclusion of the TSP edge flexure vertical eccentricity (-1/4 TSP thickness) is more than offset by the conservative loads used (>2x) and the conservative methodology (no strength beyond yield) applied in the design basis analysis. Therefore, the stresses in the tube supports (TSPs) have been assessed and ensured that they will continue to perform their function during normal operating, transient, and accident conditions. Furthermore, it is confirmed that the tube supports will not impose unacceptable loads on the tubes in the event of design basis accidents given the locked condition of the supports

LIST OF ACRONYMS

AC	Alternating Current
ADAMS	Agencywide Documents Access and Management System
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CMOA	Condition Monitoring and Operational Assessment
DR	Decay River
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECR	Engineering Change Request
EFFY	Effective Full Power Years
EP	Emergency Preparedness
EPIP	Emergency Preparedness Implementing Procedure
ESAS	Engineered Safeguards Actuation System
ETE	Evacuation Time Estimate
GPH	Gallons Per Hour
HPI	High Pressure Injection
IMC	Inspection Manual Chapter
IR	Issue Report
ISPH	Intake Screen and Pump House
LCO	Limiting Condition for Operation
LER	Licensee Event Report
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NOED	Notice of Enforcement Discretion
NRC	Nuclear Regulatory Commission
NSIR	Nuclear Security and Incident Response
OA	Operational Assessment
ORO	Offsite Response Organization
PARS	Publicly Available Records
POS	Probability of Survival
RCS	Reactor Coolant System
SDP	Significance Determination Process
SG	Steam Generator (i.e., Once Through Steam Generator)
SR	Secondary Service River Water
SSC	Structure, System, or Component
TMI	Three Mile Island
TSP	Tube Support Plates
TW	Through-Wall
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item