



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

March 23, 2016

Mr. Bryan Hanson
Senior Vice President, Exelon Generation
President and Chief Nuclear Officer, Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

**SUBJECT: THREE MILE ISLAND STATION - EVALUATION OF CHANGES, TESTS
OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS
TEAM INSPECTION REPORT 050000289/2016007**

Dear Mr. Hanson:

On February 12, 2016, 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 February 12, 2016, with Mr. E. Callan, 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. In conducting the inspection, the team reviewed selected procedures, calculations and records, observed activities, and interviewed station personnel.

Based on the results of this inspection, no findings were identified.

In accordance with Title 10 of the *Code of Federal Regulations* (CFR) 2.390 of the NRC's "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's Public Document Room or from the Publicly Available Records component of the NRC's Agencywide Documents Access and Management 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/

Mandy K. Halter, Chief
Engineering Branch 2
Division of Reactor Safety

Docket No. 50-289
License No. DPR-50

B. Hanson

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Enclosure: Inspection Report 05000289/2016007
w/ Attachment: Supplemental Information

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B. Hanson

-2-

Letter to Mr. Bryan Hanson from Ms. Mandy K. Halter dated March 23, 2016

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**U.S. NUCLEAR REGULATORY COMMISSION
REGION I**

Docket No.: 50-289

License No.: DPR-50

Report No.: 0500089/2016007

Licensee: Exelon

Facility: Three Mile Island Nuclear Power Plant

Location: Middletown, PA

Inspection Period: January 25 through February 12, 2016

Inspectors: D. Kern, Senior Reactor Inspector, Division of Reactor Safety
(DRS), Team Leader
J. Richmond, Senior Reactor Inspector, DRS
T. O'Hara, Reactor Inspector, DRS

Approved By: Mandy K. Halter, Chief
Engineering Branch 2
Division of Reactor Safety

SUMMARY

IR 05000289/2016007; 1/25/2016-2/12/2016; Three Mile Island Unit 1 (TMI); Engineering Specialist Plant Modifications Inspection.

This report covers a two week on-site inspection period of the evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by three region based engineering inspectors. No findings were identified. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 5, dated February 2014.

Other Findings

A violation of very low safety significance identified by Exelon was reviewed by the inspectors. Corrective actions taken or planned were entered into Exelon's corrective action program. This violation and corrective action tracking number are listed in Section 4OA7 of this report.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R17 Evaluations of Changes, Tests, or Experiments and Permanent Plant Modifications (IP 71111.17)

.1 Evaluations of Changes, Tests, or Experiments (26 samples)

a. Inspection Scope

The team reviewed four safety evaluations to determine whether the changes to the facility or procedures, as described in the Updated Final Safety Analysis Report (UFSAR), had been reviewed and documented in accordance Title 10 of the Code of Federal Regulations (10 CFR) 50.59 requirements. In addition, the team evaluated whether Exelon had been required to obtain U.S. Nuclear Regulatory Commission (NRC) approval prior to implementing the changes. The team interviewed plant staff and reviewed supporting information including calculations, analyses, design change documentation, procedures, the UFSAR, Technical Specifications (TS), and plant drawings to assess the adequacy of the safety evaluations. The team compared the safety evaluations and supporting documents to the guidance and methods provided in Nuclear Energy Institute (NEI) 96 07, "Guidelines for 10 CFR 50.59 Evaluations," Revision 1, as endorsed by NRC Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," to determine the adequacy of the safety evaluations.

The team also reviewed a sample of twenty-two 10 CFR 50.59 screenings, applicability reviews, item equivalency changes, commercial change packages, equivalency change packages, temporary configuration change packages, and commercial grade dedications for which Exelon had concluded that a safety evaluation was not required. These reviews were performed to assess whether Exelon's threshold for performing safety evaluations was consistent with 10 CFR 50.59. The sample included design changes, calculations, and procedure changes.

The team reviewed the safety evaluations and screenings that Exelon had performed and approved during the time period covered by this inspection not previously reviewed by NRC inspectors. All safety evaluations since the last modifications inspection were reviewed, and the screenings and applicability determinations selected were based on the safety significance, risk significance, and complexity of the change to the facility.

In addition, the team compared Exelon's administrative procedures used to control the screening, preparation, review, and approval of safety evaluations to the guidance in NEI 96-07 to determine whether the procedures adequately implemented the requirements of 10 CFR 50.59. The reviewed safety evaluations and screenings are listed in the Attachment.

b. Findings

No findings were identified.

.2 Permanent Plant Modifications (11 samples)

.2.1 Bypass of the Closed Torque Switch for MU-V-2B Letdown Cooler Outlet Isolation Valve

a. Inspection Scope

The team reviewed design change package (DCP) 13-00200, which electrically bypassed the closed torque switch contact in motor operated valve (MOV) MU-V-2B, 'B' Letdown Cooler Outlet Isolation Valve, until hard seat contact was obtained. The valve was normally open and provided a flow path for the Makeup and Purification System. The safety function was to automatically close on an Engineered Safeguards Actuation Signal to provide reactor building containment isolation. Exelon implemented the modification to ensure full motor capability was available to close the valve in order to improve the design margins. The increased margin allowed Exelon to decrease the test frequency and thereby decrease personnel radiation exposure (i.e., MOV was located in a high radiation area).

The team reviewed the modification to verify the design and licensing bases, and performance capability of the MOV had not been degraded by the modification. The team interviewed design engineers, and reviewed design drawings, engineering evaluations and analyses, MOV calculations, and diagnostic test results to determine whether the modification satisfied design and licensing requirements. In addition, the team reviewed post modification test (PMT) results and associated maintenance work orders to verify whether Exelon had appropriately implemented the modification and maintained adequate configuration control.

The team also reviewed corrective action issue reports (IR) and the Makeup and Purification System health report to determine whether there were reliability or performance issues that may have resulted from the modification. The 10 CFR 50.59 screening determination associated with this modification was also reviewed, as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

2.2 Fuse Replacements in the Emergency and Station Blackout Diesel Generators

a. Inspection Scope

The team reviewed equivalent change package (ECP) 15-00159, which replaced the potential transformer (PT) primary fuses for the emergency diesel generators and the station blackout diesel generator. The diesel generator control circuits used a PT which was protected by primary current limiting fuses. During a review of industry operating experience, Exelon identified manufacturing defects in certain manufacturer type fuses and determined that the existing diesel generator primary PT fuses were the same manufacturer type. Although there had been no related fuse failures at TMI, Exelon determined that replacement of the suspect fuses with a different manufacturer type was a reasonable action to preclude a future diesel failure. This modification replaced the existing PT fuses with a Non-Like-for-Like fuse type because no suitable like-for-like fuse type was commercially available.

The team reviewed the modification to verify the design and licensing bases, and performance capability of the diesel generators had not been degraded by the modification. The team interviewed design engineers, and reviewed design drawings, engineering evaluations and analyses, and over-current coordination calculations to determine whether the modification satisfied design and licensing requirements. In addition, the team reviewed PMT results and associated maintenance work orders to verify whether Exelon had appropriately implemented the modification and maintained adequate configuration control. The team also assessed Exelon's treatment of unverified design assumptions during the fuse selection, qualification, and installation processes.

The team also reviewed IRs and system health reports for the emergency and station blackout diesel generators to determine whether there were reliability or performance issues that may have resulted from the modification. The 10 CFR 50.59 screening determination associated with this modification was also reviewed, as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

2.3 Fuse Additions to Previously Unfused Direct Current Motor Control Circuits

a. Inspection Scope

The team reviewed DCP 14-00279, which installed fuses in the control circuits for GN-P-2, Generator Emergency Seal Oil Pump, and LO-P-9A/B, Feedwater Pump Emergency Oil Pumps. During a review of industry operating experience, Exelon determined that the direct current (DC) control circuits for the above non-safety related pump motors did not have adequate over-current protection. Exelon postulated that fire damage to the unprotected DC circuits could cause secondary fires in other fire areas and, as a consequence, adversely affect fire-safe shutdown equipment. This modification added over-current protection fuses to the affected motor control circuits.

The team reviewed the modification to verify the design and licensing bases, and performance capability of the affected equipment had not been degraded by the modification. The team interviewed design engineers, and reviewed design drawings, engineering evaluations and analyses, and over-current coordination calculations to determine whether the modification satisfied design and licensing requirements. In addition, the team reviewed PMT results and associated maintenance work orders to verify whether Exelon had appropriately implemented the modification and maintained adequate configuration control. The team also assessed Exelon's treatment of unverified design assumptions during the modification process.

The team also reviewed IRs and system health reports for the affected systems to determine whether there were reliability or performance issues that may have resulted from the modification. The 10 CFR 50.59 screening determination associated with this modification was also reviewed, as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

.2.4 Control Rod Drive Mechanism Replacement Project

a. Inspection Scope

The team reviewed engineering change request (ECR) 12-00217, which replaced the original, obsolete Type A control rod drive mechanisms (CRDM) with new TYPE C CRDMs. Other changes included in this modification were: (a) a new fiberglass position indicator tube, (b) an additional limit switch to the in-limit and out-limit circuits, and (c) added in-limit and out-limit test points to allow individual testing of each of the two out-limit switches. At TMI the limit test points are not required. AREVA had removed the in-limit test point wiring but not the out-limit test point wiring that goes to the digital control rod drive control system (DCRDCS). The out-limit test points use a previously spared wire that is grounded to the DCRDCS ground bus. This shorting to ground prevented the out-limit circuit from performing its intended function. This revision to the CRDM design removed the ground on that wire, removed two other spare wires, and revised affected drawings. The 50.59 screening and testing requirements were not affected by this revision because the intended function was restored. Additionally, Exelon replaced the previous CRDM support structure because AREVA changed the support plate clamping mechanism to the CRDM.

The team reviewed the modification to verify the design and licensing bases, and performance capability of the replacement CRDMs had not been degraded by the modification. The team interviewed design engineers, and reviewed design drawings, engineering evaluations and analyses, and diagnostic test results to determine whether the modification satisfied design and licensing requirements. In addition, the team reviewed PMT results and associated maintenance work orders to verify whether Exelon had appropriately implemented the modification and maintained adequate configuration control.

The team also reviewed IRs to determine whether there were reliability or performance issues that may have resulted from the modification. The 10 CFR 50.59 screening determination associated with this modification was also reviewed, as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

2.5 Replacement Middle and Lower Pressurizer Heater Bundles

a. Inspection Scope

The team reviewed ECR 12-00278, which replaced the middle and lower pressurizer heater bundles (PHB) to address corrosion and heater capacity issues. Additionally, the modification documented fatigue analysis and installation of shims to address bearing stress issues. The TMI pressurizer has three (3) heater bundles; an upper, middle, and lower. The diaphragm plates for the original TMI pressurizer heater bundles were fabricated from SB-168 (Alloy 600) and seal welded with Alloy 82/182 weld filler material to the stainless steel weld buttering applied to the pressurizer vessel carbon steel base material. Alloy 600 and associated weld filler materials have shown a propensity for primary water stress corrosion cracking (PWSCC), especially in components subjected to higher operating temperatures such as the pressurizer.

In 2013, modification ECR 12-00278 was implemented to replace the lower PHB with an AREVA Replaceable Element Pressurizer Heater Bundle (REPHB), whose diaphragm plate was fabricated from SA-240, Type 304 stainless steel. The middle PHB was replaced with a new PHB design which uses SA-182, Grade F304 stainless steel diaphragm plates and Alloy 52M seal welds. The materials for each heater bundle were selected, in part, because they were not susceptible to PWSCC. Cover plate shims were installed with the new PBHs to ensure American Society of Mechanical Engineers (ASME), Section III, NB-3227.1 bearing stress requirements were met. Installation of the new pressurizer heater bundles also restored full heater capacity.

In August 2012, the original upper PHB was replaced under ECRs 04-00675 Revision 1 (electrical portion) and 04-00375 Rev. 1 (mechanical portion), due to leakage from the diaphragm plate seal weld. The new heater bundle was an AREVA REPHB, whose diaphragm plate was fabricated from SA-240, Type 304 stainless steel and not susceptible to PWSCC. Cover plate shims were installed to address a bearing stress design issue. Modification ECR 12-00278 scope included documentation of the associated engineering analysis to ensure ASME Code allowable bearing stress requirements were met.

The team reviewed the modification to verify the design and licensing bases, and performance capability of the replaced or repaired heater bundles were not degraded by the modification. The team interviewed design engineers, and reviewed design drawings, engineering evaluations and analyses, and over-current coordination calculations to determine whether the modification satisfied design and licensing requirements.

In addition, the team reviewed PMT results and associated maintenance work orders to verify whether Exelon had appropriately implemented the modification and maintained adequate configuration control. The 10 CFR 50.59 screening determination associated with this modification was also reviewed, as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

2.6 Condensate Storage Tank 'A' Buried Piping Mitigation

a. Inspection Scope

The team reviewed ECR 12-00370, which replaced condensate storage tank buried piping which was prone to leakage due to corrosion of the original carbon steel piping. The original carbon steel piping was replaced with stainless steel piping of the same size and wall thickness. Additionally, an epoxy coating was applied to the replaced piping exterior to further preserve and extend service life of the new piping. Exelon also completed a net positive suction head calculation to ensure the piping replacement had not affected the design flow of the new piping system.

The team reviewed the modification to verify the design and licensing bases, and performance capability of the affected equipment had not been degraded by the modification. The team interviewed design engineers, and reviewed design drawings, engineering evaluations and analyses. In addition, the team reviewed PMT results and associated maintenance work orders to verify whether Exelon had appropriately implemented the modification and maintained adequate configuration control. The team also assessed Exelon's treatment of unverified design assumptions during the modification process.

The team also reviewed IRs and the system health reports for the affected systems to determine whether there were reliability or performance issues that may have resulted from the modification. The 10 CFR 50.59 screening determination associated with this modification was also reviewed, as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

2.7 T1R20 Pressurizer Heater Bundle Replacements – Electrical

a. Inspection Scope

The team reviewed ECR 12-00441, which completed new cabling and electrical service to the Lower and Middle replacement pressurizer heater bundles, including cable trays and distribution breakers. The team reviewed the modification to verify the design and

licensing bases, and performance capability of the affected equipment had not been degraded by the modification. The team interviewed design engineers, and reviewed design drawings, engineering evaluations and analyses. In addition, the team reviewed PMT results and associated maintenance work orders to verify whether Exelon had appropriately implemented the modification and maintained adequate configuration control. The team also assessed Exelon's treatment of unverified design assumptions during the modification process.

The team also reviewed IRs and the system health reports for the affected systems to determine whether there were reliability or performance issues that may have resulted from the modification. The 10 CFR 50.59 screening determination associated with this modification was also reviewed, as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

2.8 Engineered Safeguards Actuation System Cabinet 1B and 5A Relay Replacement

a. Inspection Scope

The team reviewed ECR TM 12-00471, Revision 3 which evaluated replacement of 238 engineered safeguards actuation system (ESAS) relays. The team also reviewed ECR 13-00041, specifically written to replace the 18 ESAS relays located within ESAS Cabinets 1B and 5A and remove one spare relay. The affected ESAS functions included high pressure injection, low pressure injection, and containment isolation and cooling. The existing Joslyn-Clark control relays were obsolete and had experienced, armature misalignment, abnormal buzzing, high resistance contacts, and high plunger assembly friction. They were replaced with Cutler Hammer Type D26M relays. At the time of this inspection, approximately 160 relays had been replaced. Replacement of the remaining 80 ESAS relays was planned to complete later in 2016.

The team reviewed the modification to verify the design and licensing bases had not been degraded by the relay replacement modification. The team interviewed the responsible engineer and reviewed associated evaluations to verify that the modified configuration and relay operating characteristics were consistent with design assumptions. The team conducted a walkdown of the installed replacement relays in ESAS Cabinets 1B and 5A. Drawings requiring revision due to the modification were reviewed to verify appropriate changes were made, and PMT results were reviewed to verify that the testing confirmed acceptable relay installation and operation. The 10 CFR 50.59 screening determination associated with this modification was reviewed as described in section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

b. Findings

No findings were identified.

2.9 Reactor Coolant Pump RC-P-1A/B/C/D Low Leakage Seal Replacement

b. Inspection Scope

The team reviewed modification 13-00099, which replaced existing Westinghouse 93AS cartridge seals with Flowserve N-9000 low leakage seals and associated instrumentation on all four reactor coolant pumps (RCP). The existing seals were vulnerable to excessive leakage during certain events (e.g., station blackout, loss of component cooling water, fire). The new seals were designed to minimize seal leakage after a loss of RCP seal cooling event. The new seal assembly has three redundant stages, each of which can handle full reactor coolant system pressure. The new seal assembly also includes an abeyance seal designed to actuate and significantly limit seal assembly leakage in a scenario where the three primary seals have failed. Consequently, the new seal design extends the coping time following a beyond design basis external event, such as extended loss of all AC power concurrent with a loss of ultimate heat sink. Installation of this modification is a portion of TMI's strategy to comply with NRC Order EA 12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," issued following the 2011 Fukushima Daiichi accident. Additionally, this modification enabled the RCP seal replacement interval to be extended (from every 4 years to once every 10 years) and reduced reliance on operator action in the event of a fire or loss of RCP seal cooling.

The team reviewed the modification to verify the design and licensing bases, and performance capability of the affected equipment had not been degraded by the modification. The team interviewed design engineers, and reviewed design drawings, engineering evaluations and analyses, and performed in-plant walkdowns of associated instrumentation and procedures to determine whether the modification satisfied design and licensing requirements. In addition, the team reviewed PMT results and associated maintenance work orders to verify whether Exelon had appropriately implemented the modification and maintained adequate configuration control.

The team also reviewed IRs and system health reports for the affected systems to determine whether there were reliability or performance issues that may have resulted from the modification. The 10 CFR 50.59 screening determination associated with this modification was also reviewed, as described in Section 1R17.1 of this report. The documents reviewed are listed in the Attachment.

.2.10 Commercial Grade Dedication of Reactor Protection System Time Delay Relay

a. Inspection Scope

The team reviewed commercial grade dedication (CGD) CGD-T1-93-0016, which certified commercially procured time delay relays for use in safety related applications including the reactor protection system RCP power monitor system. The RCP power monitor system provides a reactor trip signal when a sustained low voltage or loss of power condition to a RCP is detected, and a heat sink protection system actuation signal upon sensing loss power to all four RCPs.

The CGD technical evaluation identified six critical characteristics to be verified for acceptance of the fuses. TMI staff performed testing to verify that the relays met the specified critical characteristics. Procurement engineers reviewed the test results and implemented this CGD on September 3, 2014 to certify two relays.

On September 3, 2014 one of the RCP power monitor time delay relays failed during periodic surveillance testing. A commercial grade dedicated 120 volt, 0.1 to 3.0 sec model SSC22AAA Agastat time delay relay was issued for WO C2032806-01, Replace RC-P-1C Power Monitor 1 Timer Relay, to replace the failed relay. Material inventory records indicated one of the relays was installed in the plant and the other was returned to stock.

The team reviewed the modification to verify that the design bases, licensing bases and performance capability of the RCP power monitor system had not been degraded by the modification. The team interviewed procurement engineering staff and reviewed technical evaluations, industry test standards, and test results associated with the CGD to determine if the RCP power monitor system would function in accordance with design assumptions. The team reviewed the associated work order to verify that maintenance personnel implemented the modification as designed. The team reviewed the associated post-modification test results and performed a field verification of the fuse installation in control tower cabinet 1CB322200 to verify the fuses were properly installed and no abnormal visual characteristics were present. The documents reviewed are listed in the attachment.

b. Findings

No findings were identified.

.2.11 Commercial Grade Dedication of Engineered Safeguards Actuation System Time Delay Fuses

a. Inspection Scope

The team reviewed CGD T1-93-0031, which certified commercially procured dual element, time delay, UL Class RK-5 Fusetron fuses for use in safety related applications. The CGD technical evaluation identified five critical characteristics to be verified for acceptance of the fuses. The critical characteristics and acceptance criteria were developed based on TMI site operating experience and industry standards. A 10 CFR 50, Appendix B certified facility was contracted to perform testing and evaluation to verify the fuses met the specified critical characteristics. Procurement engineers reviewed the test results and implemented this CGD on November 20, 2014 to certify twenty fuses.

On November 24, 2014, while installing a new Cutler Hammer model D26MR44A relay for 63Y-2/RC-2B (ESAS high pressure injection trip function) under WO C2032695, technicians identified damaged relay protection fuses. Commercial grade dedicated fuses were supplied to replace the damaged fuses. Material inventory records indicated 2 of 20 commercially dedicated fuses were issued for installation in ESAS relay cabinet 2B.

The team reviewed the modification to verify that the design bases, licensing bases and performance capability of the ESAS had not been degraded by the modification. The team interviewed procurement engineering staff and reviewed technical evaluations, industry test standards, and test results associated with the CGD to determine if the ESAS and its support systems would function in accordance with design assumptions. The team reviewed the associated work order to verify that maintenance personnel implemented the modification as designed. The team reviewed the associated post-modification test results and performed a field verification of the fuse installation in ESAS cabinet 2B to verify the fuses were properly installed and no abnormal visual characteristics were present. The documents reviewed are listed in the attachment

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (IP 71152)

a. Inspection Scope

The team reviewed a sample of issue reports associated with 10 CFR 50.59 and plant modification issues to determine whether Exelon was appropriately identifying, characterizing, and correcting problems associated with these areas, and whether the planned and/or completed corrective actions were appropriate. In addition, the team reviewed IRs written on issues identified during the inspection to verify adequate problem identification and incorporation of the problem into the corrective action system. The IRs reviewed are listed in the Attachment.

b. Findings

No findings were identified.

4OA6 Meetings, including Exit

The team presented the inspection results to Mr. E. Callan, Site Vice President, and other members of Exelon's staff at an exit meeting on February 12, 2016. The team returned the proprietary information reviewed during the inspection and verified that this report does not contain proprietary information.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by Exelon and is a violation of NRC requirements which meets the criteria of the NRC Enforcement Policy for being dispositioned as a non-cited violation.

Title 10 CFR 50.55a (g)(4), In-service Inspection Requirements, requires in part, that throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) that are classified as ASME Code Class 1, must meet the requirements, except design and access provisions, and preservice examination requirements set forth in Section XI of editions and addenda of the ASME Boiler Pressure and Vessel Code (BPVC) that become effective subsequent to editions specified in paragraphs (g)(2) and (g)(3) of this Section, and that are incorporated by reference in paragraph (b) of this Section, to the extent practical within the limitations of design, geometry, and materials of construction of the components. Section XI of the ASME BPVC, 2001 Edition with 2003 Addenda, Table IWF-2500-1, Examination Category F-A Supports, requires VT-3 examination of 100 percent of the ASME Class 1 supports, other than piping supports, every ISI Interval (examination item F1.40), as modified by Notes 1, 2, 3 and 5 of Table IWF-2500-1.

Contrary to this requirement, from initial plant operation until November 14, 2015, (when Exelon staff completed the initial required VT-3 examination), Exelon failed to perform the required VT-3 examination of ASME Class 1 supports, other than piping supports, (i.e. seismic support plates and associated load path components) on the TMI control rod drive mechanism assemblies. Exelon staff entered the issue into their corrective action program as IR 01678190. The inspectors evaluated this finding using IMC 0609.04, Initial Characterization of Findings, IMC 0609, Appendix A, Exhibit 2, Mitigating Systems Screening Questions, and IMC 0609, Appendix A, Exhibit 3, Barrier Integrity Screening Questions. The finding is more than minor because it is associated with the protection against external factors attribute of the mitigating systems cornerstone and adversely affects the objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined that the finding was of very low safety significance (Green) because the finding did not involve the loss or degradation of equipment or function specifically designed to mitigate a seismic initiating event and was not associated with pressurized thermal shock of the reactor coolant system boundary.

**ATTACHMENT
SUPPLEMENTAL INFORMATION
KEY POINTS OF CONTACT**

Exelon Personnel

E. Callan, Site Vice President
T. Haaf, Plant Manager
D. Atherholt, Regulatory Affairs Manager
D. Auch, Procurement Engineer
P. Bennett, Manager, Mechanical Design Engineering
H. Crawford, Nuclear Fuels Engineer
S. Diven, System Engineer
R. Ezzo, Electrical Design Engineer
M. Fitzwater, Senior Regulatory Assurance Engineer
M. Grimm, ASME Section XI Program Manager
K. Heisey, Motor Operated Valve Program Engineer
D. Hull, Electrical Design Engineer
Y. Jaber, Electrical Design Engineer
W. McSorley, Senior Design Engineer
J. Sherk, Electrical Design Engineer
E. Showalter, Electrical Design Engineer
B. Wunderly, Director, Site Engineering

NRC

M. Farnan, Motor Operated Valve Specialist, NRR
R. Mathew, Electrical Branch, NRR
D. Werkheiser, Senior Resident Inspector

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

None

LIST OF DOCUMENTS REVIEWED

10 CFR 50.59 Safety Evaluations

AT 971475-05, Control Rod Survivability UFSAR Markups, Revision 0
SE-000622-003, Digital Control Rod Drive Controls System Upgrade, Revision 2.1.1
ECR 14-E-002, Liquid Radwaste / Concentrated Waste Piping Reroute, Revision 0
TCP 15-00424, Up Limit Switch Polar Crane Jumper, Revision 0

10 CFR 50.59 Screened-out Evaluations and/or Applicability Determinations

C-1101-642-E420-007, ESAS Block Loading Timers Uncertainty, Revision 0
C-1101-662-5350-047, TMI-1 RCS Pressure Wide Range Accident Loop Error Analysis, Revision 0
C-1101-700-5350-006, TMI-1 Short Circuit Study at Worst Case Grid Voltage, Revision 4
C-1101-741-D510-005, Loading Summary of Emergency Diesel Generators and Engineered Safeguards Buses, Revision 5D and 5E
DCP 12-00402, Air Intake Tunnel Flood Protection Boundary Change, Revision 0
DCP 13-00183, Control Room Habitability for Offsite Hazardous Chemicals Analysis, Revision 0
ECP 11-00149, EDG Speed Switch Equivalent Replacement, Revision 0
ECP 11-00425, MU17-PT Makeup Tank Pressure Transmitter Replacement, Revision 3
ECP 14-00159, Updated Seismic Analysis for MU-V-16A/B/C/D, Revision 0
ECR 08-00544, Commercial Grade Dedication – Control Relay 5U6-2-76, Revision 0
ECR 12-00240, Loss of Phase Protection Relays, Revision 1
ECR 13-00089, HD-V-4 Bypass Valve Modification, Revision 0
ECR 13-00327, 2USS Breaker Trip Unit, Revision 1
ECR 13-00405, Hydrogen Generation Calculation Revision, Revision 0
ECR 13-00555, DR-XJ-12A/B Commercial Grade Dedication, Revision 0
ECR 14-00060, Updated Reactor Building Purge Valve Closing Stroke Time, Revision 0
ECR 14-00147, Upgrade Reactor Head Fan Motors, Revision 0
ECR 14-00484, New Statistical Design Limit for Mk-B-HTP Fuel, Revision 0
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ECR 12-00441, T1R20 Pressurizer Heater Bundle Replacements – Electrical, Revision 0
ECR 12-00471, ESAS Relay Replacement Project, Revision 3
DCP 13-00041, ESAS Cabinet 1B and 5A Relay Replacement, Revision 7
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* IR written as a result of this inspection

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LIST OF ACRONYMS

10 CFR	Title 10 of the Code of Federal Regulations
AC	Alternating Current
ADAMS	Agency-Wide Documents Access and Management System
ASME	American Society of Mechanical Engineers
CGD	Commercial Grade Dedication
CRDM	Control Rod Drive Mechanism
DC	Direct Current
DCP	Design Change Package
DCRDCS	Digital Control Rod Drive Control System
DRS	Division of Reactor Safety
ECP	Equivalent Change Package
EPRI	Electric Power Research Institute
ESAS	Engineered Safeguards Actuation System
IR	Issue Report
MOV	Motor Operated Valve
NEI	Nuclear Energy Institute
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
PHB	Pressurizer Heater Bundle
PMT	Post-Modification Test
PT	Potential Transformer
PWSCC	Primary Water Stress Corrosion Cracking
RCP	Reactor Coolant Pump
REPHB	Replaceable Element Pressurizer Heater Bundle
TMI	Three Mile Island, Unit 1
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report