UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PA 19406-1415

June 7, 2010

Mr. Michael Pacilio
Senior Vice President, Exelon Generation Company LLC
President and Chief Nuclear Officer, Exelon Nuclear
4300 Winfield Rd.
Warrenville, IL 60555

SUBJECT: THREE MILE ISLAND STATION, UNIT 1 – NRC EVALUATION OF CHANGES, TESTS, AND EXPERIMENTS AND PERMANENT MODIFICATIONS TEAM INSPECTION REPORT 05000289/2010006

Dear Mr. Pacilio:

On April 29, 2010, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Three Mile Island Station, Unit 1. The enclosed inspection report documents the inspection results, which were discussed on April 29, 2010, with Mr. Richard W. Libra, Plant Manager, 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 of significance were identified.

In accordance with 10 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 Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

Lawrence T. Doerflein

Lawrence T. Doerflein, Chief
Engineering Branch 2
Division of Reactor Safety

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

Enclosure: Inspection Report No. 05000289/2010006
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

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Sincerely,

/RA/

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U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No.:	50-289
License No.:	DPR-50
Report No.:	05000289/2010006
Licensee:	Exelon Generation Company, LLC (Exelon)
Facility:	Three Mile Island Station, Unit 1
Location:	Middletown, Pennsylvania
Inspection Period:	April 12 through April 29, 2010
Inspectors:	K. Mangan, Senior Reactor Inspector, Division of Reactor Safety (DRS), Team Leader D. Orr, Senior Reactor Inspector, DRS E. Burkett, Reactor Inspector, DRS N. Lafferty, Reactor Engineer, Division of Reactor Projects (in-training)
Approved By:	Lawrence T. Doerflein, Chief Engineering Branch 2 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000289/2010006; 4/12/2010 – 4/29/2010; Exelon Genration Company, LLC; Three Mile Island Station, Unit 1; 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 and one inspector in training. No findings of significance 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 4, dated December 2006.

A. <u>NRC-Identified and Self-Revealing Findings</u>

No findings of significance were identified.

B. <u>Licensee-Identified Violations</u>

None.

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 (29 samples)

a. Inspection Scope

The team reviewed six 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 with 10 CFR 50.59 requirements. In addition, the team evaluated whether Exelon had been required to obtain NRC approval prior to implementing the change. The team interviewed plant staff and reviewed supporting information including calculations, analyses, design change documentation, procedures, the UFSAR, the Technical Specifications (TSs), 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," 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 three 10 CFR 50.59 screenings for which Exelon had concluded that no safety evaluation was 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 that Exelon had performed and approved during the time period covered by this inspection (i.e., since the last modifications inspection) not previously reviewed by NRC inspectors. The screenings and applicability determinations were selected 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 those 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 of significance were identified.

.2 Permanent Plant Modifications (11 samples)

.2.1 Replacement of Caulking Material Inside Containment

a. Inspection Scope

The team reviewed a modification (Engineering Change Request (ECR) TM-07-00453) that replaced caulk used as a moisture barrier between the containment floor and wall. The modification removed all the old style caulk at the base of containment and replaced it with a new style caulking material. This change was performed because the new caulk was shown, via testing, to withstand design basis accident environmental conditions and the older caulk had not been tested to those worst case conditions. The team conducted the review to verify that the design bases, licensing bases, and performance capability of safety related structures, systems, and components (SSCs) were not degraded by the new caulking material.

The team discussed the modification and design basis with design and system engineers to assess the capability of the caulk to remain intact following a design basis accident (DBA). The team reviewed the manufacturer's recommended applications for the caulk to ensure that the calculated environmental conditions inside containment were enveloped by the manufacturer's guidelines. The team also reviewed the installation procedure to verify the caulk was installed in accordance with the manufacturer's requirements. Finally, the team verified that the new caulk would not change the assumptions used for the containment sump testing as discussed in the Exelon's Generic Letter (GL) 2004-02 response. 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 of significance were identified.

.2.2 Nuclear River Water Outlet Isolation Valve Installation

a. Inspection Scope

The team reviewed a modification (ECR-TM-08-00251) that installed a new isolation valve in the nuclear river (NR) water system. Exelon implemented this modification in order to allow for maintenance on individual heat exchangers while maintaining adequate flow to other system loads allowing them to remain available for operation. In addition to the valve replacement, Exelon also replaced several sections of NR piping under the same work order. The team conducted the review to ensure that the design bases, licensing bases, and performance capability of the NR system had not been adversely affected by the modification.

The team reviewed the associated work order packages and conducted interviews with design and system engineers regarding the design, installation, and testing of the valve and piping to verify that the modification was adequate. The review included determining the safety classification of the new valve, piping and welding material. The team also reviewed the post maintenance testing of the welds and piping to verify it met the requirements of the American Society of Mechanical Engineers (ASME) code. Additionally, the team verified that the hydraulic and seismic calculations had been revised to address the effects the new equipment had on the design of the system. Finally, the team walked down the new valve and piping to assess the system configuration. 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 of significance were identified.

.2.3 Auxiliary Building Flood Analysis

a. Inspection Scope

The team reviewed a change to the flooding analysis (ECR-TM-06-00097) performed by Exelon to address a change in operating procedures associated with building spray (BS) pump operation. For certain design basis events, Exelon now operates the BS system so the analysis must assume a failure could occur in the system. This failure could result in flooding of certain safety related SSCs. Exelon's calculation revision and associated analysis determined that there was sufficient time for operators to respond to a failure such that no other equipment would be impacted. The team evaluated the change to confirm that the design bases, licensing bases, and performance capability of the safety related SSCs would not be affected by the change.

The team reviewed the calculation and associated analysis to verify the assumptions used in the calculation were valid. Additionally, the team walked down portions of the system to assess the configuration of the system. The team reviewed surveillance procedures used to verify that the drainage paths and associated water detection alarms were being adequately maintained. Finally, the team interviewed design engineers to review the calculation assumptions, applicable operating procedures, and the calculation's methodology to determine if the analysis showed adequate time was available for operator action. 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 of significance were identified.

.2.4 Decay Heat (DH) Removal System Valves (DH-V-6A and DH-V-6B) Motor Replacement

a. Inspection Scope

The team reviewed a modification (ECR-TM-07-00792) that replaced the motors and thermal overload heaters for valves DH-V-6A and 6B. Exelon implemented the modification following an analysis performed to evaluate the potential for thermal binding of the valves following changes associated with the reactor building (RB) sump modification performed in refueling outage 1R17. Since the analysis showed thermal binding conditions may exceed the capability of the motor operated valve (MOV) actuators, Exelon implemented the modification which installed larger motors with a higher starting torque, installed new thermal overload heaters, and changed the instantaneous setting on the DH-V-6A/B breakers to account for the higher inrush currents. The team conducted the review to verify that the design bases, licensing bases, and performance capability of valves DH-V-6A and B had not been adversely affected by the modification.

The team reviewed associated drawings, calculations, and maintenance procedures to ensure they had been properly updated to incorporate the changes to the DH-V-6A and 6B motors. In addition, post-modification testing was reviewed to verify proper operation of the MOVs. The team also discussed the modification and design basis with design engineers to assess the adequacy of 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 of significance were identified.

.2.5 Install New Air Handling Unit Starters and Components

a. Inspection Scope

The team reviewed a modification (ECR-TM-08-00607) that replaced the starters and associated components in the control circuits for each of the three RB recirculation air handling units (AH E-1A, -1B, -1C). Exelon implemented the modification because the existing parts were no longer supported by the vendor. The modification included replacing components in the motor control center (MCC) for each air handling unit, specifically the fast and slow speed starters, auxiliary contacts, and associated thermal overload relays and heaters. The team conducted the review to verify that the design bases and performance capability of the RB recirculation air handling units had not been adversely affected by the modification.

The team reviewed associated drawings and calculations to ensure they had been properly updated to incorporate the changes associated with the starters and related components of the RB recirculation air handling units. The team interviewed the responsible design and system engineers to understand the implementation of the modification and subsequent system performance. The team also reviewed condition reports (CRs) associated with the air handling units to verify that the modification had addressed identified deficiencies with the starters. The team reviewed the post modification test plan and results to ensure appropriate acceptance criteria had been established and met. 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 of significance were identified.

.2.6 Pressurizer Surge Nozzle and Decay Heat Drop Line Nozzle Weld Overlays

a. Inspection Scope

The team reviewed a modification (ECR-TM-07-00369) that applied a full structural weld overlay (SWOL) to the pressurizer surge nozzle to safe-end weld and a full SWOL to the decay heat drop line nozzle to safe-end weld. These welds were originally constructed using alloy 82/182 dissimilar metal weld material which industry experience has since shown to be susceptible to primary water stress corrosion cracking (PWSCC). Exelon implemented this modification to mitigate the effects of PWSCC. The team conducted the review to verify that the design bases, licensing bases, and performance capability of the pressurizer and decay heat removal system had not been adversely affected by the modification.

The team reviewed drawings and calculations to ensure they had been properly updated to incorporate the addition of the full SWOLs. The team also reviewed associated analyses to evaluate if the weld overlay would degrade the design basis of the pressurizer and to determine if the requirements of the ASME Boiler and Pressure Vessel Code were met. In addition, the team reviewed completed work orders and post modification testing to verify the acceptance criteria were met. The team also discussed the modification and design basis with design and system engineers to assess the adequacy of 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 of significance were identified.

.2.7 Pressurizer Surge Line Restraint Modification

a. Inspection Scope

The team reviewed a modification (ECR-TM-07-00453) that changed the dimension of the pressurizer surge line restraint. Exelon implemented this modification in conjunction with the pressurizer surge nozzle weld overlay modification. The modification involving the SWOL was performed to add weld material to the pressurizer surge line nozzle. This required the inner diameter of the surge line restraint to be increased. The team conducted the review to verify that the design bases and performance capability of the restraint had not been adversely affected by the modification.

The team reviewed drawings and calculations to ensure they were properly updated with the new dimensions of the restraint. The team reviewed the associated analysis to verify the structural integrity of the modified restraint was maintained. In addition, the team reviewed completed work orders to verify the appropriateness of the testing acceptance criteria and evaluated test results against the criteria. The team also discussed the modification and design bases with design engineers to assess the adequacy of 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 of significance were identified.

.2.8 Main Steam Atmospheric Dump Valve Replacements

a. Inspection Scope

The team reviewed a modification (ECR-TM-08-00145) that replaced the 'A' and 'B' main steam atmospheric dump valves (ADVs) and actuators. The new actuator design also required a modification to the instrument air system. The ADVs were replaced to improve reliability associated with seat tightness. In addition to evaluating valve capabilities and design characteristics, the modification evaluated and made changes to the instrument air system piping and pneumatic controllers. The ADVs are credited to mitigate several DBAs and are automatically operated by the integrated control system that can be remotely or locally operated for decay heat removal. The team conducted the review to ensure that the design bases, licensing bases, and performance capability of the ADVs had not been adversely affected by the modification.

In addition to reviewing the engineering analyses and supporting information within the modification package, the team reviewed calculations to verify that the two-hour backup instrument air system capacity requirements and time to reach cold shutdown were not adversely impacted by the modification. The team verified that Exelon evaluated the automatic operation of the ADVs as controlled by the integrated control system. The team verified that the post-modification test results were acceptable and that surveillance procedures, operating procedures, and the vendor manual were appropriately revised. The team also verified that operating personnel were appropriately trained on local operation of the replacement ADVs. Finally, the team walked down the 'A' and 'B' ADVs to identify abnormal conditions and reviewed operating procedures posted for local operation and discussed the modification and design basis with design engineers to assess the adequacy of 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 of significance were identified.

.2.9 Station Blackout (SBO) Battery Replacement

a. Inspection Scope

The team reviewed a modification (ECR-TM-08-00055) that replaced all cells that make up the 125Vdc SBO battery. The SBO battery was replaced to ensure continued battery reliability. The SBO battery is a necessary support system of the SBO diesel generator. The modification was performed using redesigned cells that offered improved battery performance because the original battery cells were no longer available. The team conducted the review to ensure that the design bases, licensing bases, and performance capability of the SBO diesel and its associated battery had not been adversely affected by the modification.

The team verified that Exelon appropriately evaluated all changes and characteristics of the new battery including cell internals and external connections to the battery jar. Additionally, the team reviewed the engineering analyses and supporting information within the modification package and battery capability calculations to verify that the SBO battery and SBO battery charger were appropriately sized to respond to a SBO event. The team verified that Exelon incorporated all vendor recommendations related to battery installation, maintenance, and testing into work orders and procedures. The team reviewed pre-installation, post-installation, and recent surveillance test results to verify that battery performance was consistent with vendor specifications and design assumptions. The team also walked down the SBO battery to identify abnormal conditions and determine if the installation was consistent with vendor recommendations. Finally, the team discussed the modification and design basis with design engineers to assess the adequacy of 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 of significance were identified.

.2.10 Prevention of Spurious Operation of Reactor Building Sump Isolation Valves

a. Inspection Scope

The team reviewed a modification (ECR-TM-08-00786) that made control circuit changes to the decay heat removal pump reactor building sump isolation valves, DH-V-6A and DH-V-6B. Spurious operation of the DH-V-6A and 6B due to electrical cable damage was determined by Exelon to be a credible scenario during certain fire scenarios. For fire areas outside the main control room, the circuit modification eliminated the possibility of a fire induced hot short that would spuriously operate DH-V-6A or 6B. In addition to wiring changes, the modification included new disable/enable switches located on the main control room panels. As a result of the modification, valve operation would require operation of both the associated new switch and the normal control switch in order to open each valve. The team conducted the review to ensure that the design bases, licensing bases, and performance capability of the decay heat removal system had not been adversely affected by the modification.

The team verified that Exelon reviewed all aspects of the circuit changes and failure modes to ensure reliable control room operation of DH-V-6A and 6B for non-fire scenarios was maintained. The team also verified that the circuit changes eliminated the possibility of a hot short outside the main control room from spuriously opening the valves. The team reviewed voltage drop calculations and post-modification test results to ensure the circuits operated as designed. The team reviewed operator training records and operating procedures to ensure the operators were appropriately trained and procedures were revised to incorporate the operation of the new switch when the decay heat valves were required to be repositioned. The team observed the new disable/enable switches on the main control board and verified the labeling and human factor considerations were adequate. The team also discussed the modification with design engineers to assess the adequacy of 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 of significance were identified.

2.11 Main Inverter Replacements

a. Inspection Scope

The team reviewed a modification (ECR-TM-08-00281) that replaced three electrical inverters to address obsolete parts issues and nuisance transfers from the preferred 480Vac power supply to the 125Vdc battery backup. Exelon purchased the new inverters such that they would be qualified to the original specifications and maintain the same footprint for plant installation. The inverters are designed to provide a reliable and redundant source of essential power to several vital 120Vac distribution panels. The team conducted the review to ensure that the design bases, licensing bases, and performance capability of the ac power system had not been adversely affected by the modification.

The team reviewed the operating characteristics of the new and old inverters to ensure the quality of the new inverters was equal or better. The team noted that Exelon had identified that the minimum dc input voltage for the new inverters was slightly less. The team verified that Exelon appropriately evaluated the discrepancy via an engineering equivalency review. The team also reviewed the work orders that installed and tested the new inverters and ensured that all inverter test results were within specification and all installation issues had been appropriately evaluated and dispositioned. Additionally, the team verified that Exelon had revised its 120Vac breaker coordination study and evaluated the potential for the new inverter electrical characteristics impact on the 120Vac power system. The team verified that operators were appropriately trained on the new inverters and the operating procedures were properly updated. The team also discussed the modification and design basis with design engineers to assess the adequacy of the modification. Finally, the team walked down the inverters to identify abnormal conditions. 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 of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (IP 71152)

a. Inspection Scope

The team reviewed a sample of CRs 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 or completed corrective actions were appropriate. In addition, the team reviewed CRs written on issues identified during the inspection to verify adequate problem identification and incorporation of the problem into the corrective action system. The CRs reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

The team presented the inspection results to Mr. Richard Libra, Plant Manager, and other members of Exelon's staff at an exit meeting on April 29, 2010. The team returned the proprietary information reviewed during the inspection to the licensee and verified that this report does not contain proprietary information.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Exelon Personnel

R. Libra	Plant Manager
B. Carsky	Engineering Director
P. Bennett	Design Engineering Manager
W. McSorley	Mechanical Design Engineer
R. Ezzo	Manager, Electrical and I&C Design Engineering

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

None.

LIST OF DOCUMENTS REVIEWED

10 CFR 50.59 Evaluations
SE 06-00219, TMI Unit 1 Cycle 16 Cycle Management – RPS Reactor Coolant Pump Power Monitor Trip Total Delay Time, Rev. 1
SE-000534-012, Reactor Bldg. Emergency Cooling Design Flow Change, Rev. 0
SE-000572-001, Generic Letter 2004-02/GSI-191 Supplemental Response, Rev. 0
SE-000641-036, RPS Testing with Channel B in Manual Bypass, Rev. 0
TM-06-00550, Reanalysis of the Dropped Control Rod Accident for TMI-1, Rev. 0
TM-09-00524, Removal of RM-G-21, Rev. 0

10 CFR 50.59 Screened-out Evaluations
05-00177, C-1101-823-E610-014 and UFSAR Revision for EQ, Rev. 0
06-01135, Implementation of Mark-B-HTP Fuel Design, Rev. 0
07-00119, Technical Specification 4.6.1A Basis Change, Rev. 0
07-00310, Configuration Change for Conversion to S15 ULSD Fuel Oil, Rev. 0
07-00375, RCS Flow Loop Error Analysis, Rev. 0
07-00434, RCPPM Response Time, Rev. 0
08-00203, Installation of Reactor Building Liner Inspection Ports, Rev. 1
08-00210, Divert Condensation Away from Reactor Building Liner, Rev. 0
08-00771, RCS Pressure High, Low, and Shutdown Bypass RPS Setpoints, Rev. 0
08-00872, Spent Fuel Pool Re-Rack, Rev. 1
09-00268, New Style Fuel Grapple for TMI Fuel Handling Bridges, Rev. 0
09-00567, Reactor Head Drop Analysis, Rev. 0

Modification Packages (* designates Modification and 10 CFR 50.59 screen sample)
*ECR-TM-06-00097, Auxiliary Building Sump Flood Analysis, Rev. 1
*ECR-TM-07-00369, TMI-1 Pressurizer Surge Nozzle and Decay Heat Drop Line Nozzle Weld Overlays, Rev. 1
*ECR-TM-07-00453, Pressurizer Surge Line Restraint Modification, Rev. 1
*ECR-TM-07-00479, Reactor Building Approved Caulking Material, Outside D-ring, Rev. 0
*ECR-TM-07-00792, DH-V-6A/B Motor Replacement, Rev. 0
*ECR-TM-08-00055, Evaluate Replacement SBO Batteries, Rev. 0
*ECR-TM-08-00145, Replacement of Atmospheric Dump Valves MS-V-4A/B, Rev. 1
*ECR-TM-08-00251, ICCE River Water Outlet Isolation Valve Install, Rev. 2
*ECR-TM-08-00281, Replace 1A, 1B, and 1D Main Inverters, Rev. 2
*ECR-TM-08-00607, Install New AH-E-1A/B/C Starters and Components, Rev. 0
*ECR-TM-08-00786, Prevention of Spurious Operation DH-V-6A/B, Rev. 1
ECR-TM-05-00177, C-1101-823-E610-014 and UFSAR Revision for EQ, Rev. 0
ECR-TM-06-00219, TMI Unit 1 Cycle 16 Cycle Management – RPS Reactor Coolant Pump Power Monitor Trip Total Delay Time, Rev. 1
ECR-TM-06-01135, Implementation of Mark-B-HTP Fuel Design, Rev. 0
ECR-TM-07-00743, Generic Letter 2004-02/GSI-191 Supplemental Response, Rev. 0
ECR-TM-08-00872, Spent Fuel Pool Re-Rack Project – Phase 3, Rev. 1
ECR-TM-09-00567, Reactor Head Drop Analysis, Rev. 0

Calculations & Analysis
2543C, Thermal Binding Analysis for 14-inch Aloyco Solid Wedge Gate Valves, DH-V-6A/B, Rev. 0
32-9051871, Pressurizer Surge Nozzle Weld Overlay Residual Stress Analysis, Rev. 1
32-9052931, Three Mile Island Unit 1 Pressurizer Surge Nozzle Weld Overlay Analysis, Rev. 2
32-9057426, TMI Unit 1 Pressurizer Surge Nozzle Restraint Loading Evaluation, Rev. 2
32-9107831-000, TMI-1 Cycle 18 Maneuvering Analysis, Rev. 0
86-9020266-001, TMI-1 Dropped Rod w/o Leadscrew Analysis Summary, dated 07/25/07
990-1745, Fire Hazards Analysis Report, Rev. 23
A2206841-19, SFP Re-rack Heavy Loads Technical Evaluation, dated 03/17/09
C-1101-210-E610-011, LPI and BS Pump NPSH Margin Available Following a LLOCA, Rev. 9a
C-1101-534-E410-020, TMI-1 RBEC Pump Curve Acceptance Requirements, Rev. 3
C-1101-573-E410-004, Auxiliary Building Sump Flood Analysis, Rev. 1
C-1101-641-E420-018, RCS Flow Loop Error Analysis, Rev. 0B
C-1101-641-E420-020, Reactor Coolant Pump Power Monitor Time Response, Rev. 0
C-1101-641-E510-021, RPS Trip Setpoints for RCS High Pressure, Low Pressure and Shutdown Bypass High Pressure, Rev. 0
C-1101-730-5350-002, GL 89-10 MOVs Thermal Overload Heater Determination, Rev. 4
C-1101-733-E510-021, 480 VAC MCC 120V Control Circuit Voltage Drop Calculation, Revs. 0 and 3
C-1101-735-5350-002, 120V Vital AC System Coordination Study, Rev. 3
C-1101-741-E510-005, Loading Summary of EDGs & Engineered Safeguards Buses, Rev. 4
C-1101-823-5450-001, TMI-1 LBLOCA EQ Temperature Profile Using Gothic Computer Code, Rev. 9D
C-1101-823-E610-014, TMI-1 MSLB Containment Response, Rev. 2A

C-1101-852-5360-004, Two Hour Backup Instrument Air System as-Built Capacity Calculation and Evaluation, Rev. 3
C-1101-862-5360-002, TMI-1 EDG Fuel Requirement, Rev. 3
C-1101-864-E420-001, SBO Battery and Charger Sizing and Hydrogen Generation Calculation, Rev. 0
C-1101-900-E410-049, Weak Link Calc for TMI GL 89-10 Valves, Rev. 7C
C-1101-900-E410-073, Limitorque Actuator Center of Gravity Corrections, Rev. 0A
C-1101-910-E410-001, TMI-1 RB Unqualified/Degraded Coating Inventory, Rev. 2
C-1101-911-E610-001, Time to Reach Cold Shutdown and Required Condensate Inventory, Rev. 4
EER 00169261, Inverter DC Input Undervoltage Setpoint Change, Rev. 0
NAI-1299-001, TMI Unit 1 Containment Sump Valve Thermal Response Analysis, Rev. 0A
TDR 083, Evaluation of Containment Isolation Signals, Rev. 3

Corrective Action Reports (* denotes NRC identified during this inspection)

01062182*	00655718	00939776	01028756
01062006*	00668731	00952819	01044515
01061840*	00669036	00960743	01044845
01055951*	00677120	00967035	01044866
01056477*	00687244	00971644	01044935
01056691*	00694392	00995518	01055298
01057219*	00741357	00995532	01056337
01061840*	00820824	00995537	01061578
00268010	00831336	00998110	01062396
00319626	00831838	01010923	01063041
00626956	00918472	01025674	

Drawings
1E-611-07-012, Control Room Panel, Rev. 10
112794D, Sht. 2, Reactor Coolant System Foundation Loadings, Rev. 3
302511, Reactor Building Normal and Emergency Cooling Water System, Rev. 12
302640, Decay Heat Removal, Rev. 80
521029, Pressurizer Bottom Steel Restraint, Rev. 7
302-719, Sump Pump and Drainage System, Rev. 62

Surveillance and Modifications Acceptance Tests
FTP 212.01, DH-V-6A/B Control Circuit Response to a Hot Short Verification, performed 12/24/09
U-17, Zurn Floor Drain Inspection, Rev. 13
OP-TM-212-217, DH-V-6A and Associated Tests, performed 11/04/07
OP-TM-212-218, DH-V-6B and Associated Tests, performed 11/05/07

Procedures
1107-2B, 120 Volt Vital Electrical System, Rev. 24C
1107-2C, Vital DC Electrical System, Rev. 8
1302-5.2, RPS High and Low RC Pressure Channels, Rev. 34

1302-5.4A, RPS Channel A Reactor Coolant Flux Flow Comparator, Rev. 8C
1302-5.4B, RPS Channel B Reactor Coolant Flux Flow Comparator, Rev. 7C
1302-5.4C, RPS Channel C Reactor Coolant Flux Flow Comparator, Rev. 7C
1302-5.4D, RPS Channel D Reactor Coolant Flux Flow Comparator, Rev. 5C
1303-4.16, Emergency Power System, Rev. 122
1303-4.1A, RPS Channel A Test, Rev. 16
1505-1, Fuel and Control Component Shuffles, Rev. 44
E-135, SBO Diesel Storage Batteries Inspection, Rev. 8
E-26.1, Vital Power Inverter Maintenance, Rev. 3B
HI-2053400, Bulk and Local Thermal-Hydraulic Analysis for TMI-1 SFP, Rev. 0
HPP-1613-02, TMI Rack Handling and Installation Procedure, Rev. 1
HPP-90444-23, Post-Installation Drag Test Procedure for Spent Fuel Racks, Rev. 0
LS-AA-104, Exelon 50.59 Review Process, Rev. 6
LS-AA-104-1000, Exelon 50.59 Resource Manual, Rev. 5
MA-TM-134-902, Reactor Vessel Reassembly, Rev. 6
MA-TM-134-903, Reactor Vessel Disassembly, Rev. 8c
OP-TM-212-000, Decay Heat Removal System, Rev. 12
OP-TM-232-635, Draining the Reactor Building Sump, Rev. 2
OP-TM-411-451, Manual Control of TBVs/ADVs, Rev. 6
OP-TM-534-225, IST of RR-P-1A and Valves during Cold Shutdown, Rev. 0A
OP-TM-534-228, IST of RR-P-1B and Valves during Cold Shutdown, Rev. 0A
OP-TM-534-901, RB Emergency Cooling Operations, Rev. 9
OP-TM-AOP-062, Inoperable Rod, Rev. 2
OP-TM-EOP-010, Emergency Procedures Rules, Guides and Graphs, Rev. 11
OP-TM-EOP-020, Cooldown from Outside of Control Room, Rev. 10
OP-TM-LWDS-0101, Aux. Bldg. Sump Level above Norm, Rev. 1
OP-TM-LWDS-0102, Aux. Bldg. Sump Level Hi, Rev. 1
OP-TM-PLB-0405, 1A Decay Heat Pump Compartment Leak Detect, Rev. 3
OP-TM-PLB-0505, 1B Decay Heat Pump Compartment Leak Detect, Rev. 3
PLB-6-5, Panel Left Annunciator B, Rev. 10

<u>Work Orders</u>

C2014705	C2015607	C2018197	R2081891
C2014706	C2015782	C2019231	R2118461
C2014707	C2016190	C2020072	R2123540
C2014708	C2017910	C2020073	R2155237
C2014795	C2017911	R2065963	R2155574
C2014937	C2017912	R2077551	R2158122

<u>Vendor Manuals</u>
VM-TM-0953, Excide Stationary Lead Calcium Batteries, Rev. 2
VM-TM-2903, Enersys Flooded Lead-Acid Batteries C, D, E, F & G, Rev. 2
VM-TM-2961, Copes-Vulcan GS and SD Control Valves Instruction Manual, Rev. 2

Miscellaneous

80-025116, SBO Battery Purchase Order, Rev. 0

9027596-000, TMI-1 Specific POG Guidance - Dropped CRA due to Bayonet Failure, dated 05/18/07

ASME Code Case N-416-3, dated 09/07/01

Framatome ANP report to the NRC, Evaluation of a Potential Safety Significant Issue Pursuant to 10CFR21(a)(2), dated 11/23/04

Letter from TMI to USNRC, "Three Mile Island Unit 1 Response to Request for Additional Information Related to NRC Generic Letter 2004-02," dated 11/10/08

Letter from TMI to USNRC, "Three Mile Island Unit 1 Response to Request for Additional Information Regarding Generic letter 2004-02," dated 11/09/09

Letter from TMI to USNRC, "Three Mile Island Unit 1 Supplemental Response to NRC Generic Letter 2004-02", dated 12/28/07

License Amendment No. 164, dated 04/27/92

License Amendment No. 170, dated 02/17/93

NEI 08-05, Industry Initiative on Control of Heavy Loads, Rev. 0

Op Eval DOO-T1-106, MSLB Inside Containment EQ Evaluation, Rev. 0

Operator Training Module 11.2.01.559, 1R18 Outage Modifications, dated 08/03/09

Op Eval 05-004, CRDM Leadscrew Bayonet Coupling, Rev. 2

Pre-NRC Inspection: Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications Self Assessment, performed 3/1/10-3/19/10

Safety Evaluation by the Office of Nuclear Reactor Regulations Related to Holtec International Report HI-2022871 Regarding Use of Metamic in Fuel Pool Applications, dated 06/17/03

SP-1101-11-246, Specification for Procurement of New 15kVA Static Inverter, Rev. 2

Technical Specification Amendment No. 271, TMI-1 Cycle 18 Core Operating Limits Report, dated 01/10/10

Technical Specifications, Three Mile Island Nuclear Station Unit 1, Amendment 189

Three Mile Island Final Safety Analysis, Rev. 19

Three Mile Island Station Engineering Design Control Audit, performed 8/3/09-8/13/09

LIST OF ACRONYMS

ADAMS	NRC Document System
ADV	Atmospheric Dump Valve
ASME	American Society of Mechanical Engineers
BS	Building Spray
CFR	Code of Federal Regulations
CR	Condition Report
DBA	Design Basis Accident
DH	Decay Heat
DRS	Division of Reactor Safety
GL	Generic Letter
IP	Inspection Procedure
MCC	Motor Control Center
NEI	Nuclear Energy Institute
NR	Nuclear Riverwater
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
PWSCC	Primary Water Stress Corrosion Cracking
RB	Reactor Building
SBO	Station Blackout
SSC	Structure, System and Compenents
SWOL	Structural Weld Overlay
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
Vac	Alternating Current
Vdc	Direct Current