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April 15, 2016

U.S. Nuclear Regulatory Commission
ATTN. Document Control Desk
Washington, DC 20555

Three Mile Island Nuclear Station, Unit 1
Renewed Facility Operation License No. DPR-50
NRC Docket No. 50-289

Subject: Biennial 10 CFR 50.59 and Commitment Revision Reports for 2014 and 2015

Enclosed are the 2014 – 2015 Biennial 10 CFR 50.59 and Commitment Revision Reports as required by 10 CFR 50.59(d)(2) and SECY-00-0045 (NEI 99-04).

There are no regulatory commitments contained in this transmittal.

If you have any questions or require additional information, please contact Mike Fitzwater, of Regulatory Assurance, at 717-948-8228.

Sincerely,

Tom Haaf
Plant Manager, Three Mile Unit 1
Exelon Generation Co., LLC

Enclosure:
BIENNIAL 10 CFR 50.59 AND COMMITMENT REVISION REPORTS

cc: USNRC, Regional Administrator, Region I
USNRC, Senior Project Manager, NRR
USNRC, Senior Resident Inspector, TMI

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NRR

THREE MILE ISLAND
UNIT 1
DOCKET NO. 50-289

BIENNIAL 10 CFR 50.59 AND COMMITMENT REVISION REPORTS

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THREE MILE ISLAND
UNIT 1
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BIENNIAL 10 CFR 50.59 REPORT

JANUARY 1, 2014 THROUGH DECEMBER 31, 2015

10 CFR 50.59 EVALUATION SUMMARIES

Modifications

Title: Digital Control Rod Drive Controls System Upgrade

Year Implemented: 2009 (Note: this 50.59 evaluation was an omission from the Biennial report submitted in 2010. The omission is captured in our Corrective Action Program under Issue Report # 2615755)

Evaluation Number: Engineering Change Request Nos. TM 07-01037, 07-01038, 07-01039

Brief Description:

This activity is the digital upgrade of the Control Rod Drive Control System (CRDCS). The existing analog CRDCS and six associated reactor trip breakers (2 AC and 4 DC) will be replaced with a digital CRDCS (DCRDCS) designed and supplied by AREVA, and four new AC reactor trip breakers. The new DCRDCS does not include controls and indications for the Group 8 Axial Power Shaping Rod Assemblies (APSRAs). The boron dilution interlocks for Regulating Rod Groups 5 and 6 will be eliminated. This activity includes the following:

- Removal of the CRDCS cabinet internals and installation of the DCRDCS electronics in the existing cabinets
- Replacement of the AC reactor trip breaker cabinets (including breaker test panels), removal of the DC breaker cabinets
- Installation of two Position Indication Flat Panel Monitors on and two computers in main control room panel "PC"
- Deletion of Group Average Meters from the MCR console. The Group Average indications will be moved to the Flat Panel Monitors.
- Addition of inverter-backed In-Limit LEDs on Position Indication Panel
- Modification of the ATWS / Diverse Scram System (DSS) to input the electronic trip into the DCRDCS and use the DCRDCS software to perform the trip
- Replace the CRD Group Power Supply bus duct with new power cables between the power supply cabinets and the transfer switch cabinets. This includes installation of new cable tray and supports.
- Install new fuses above single rod power supply (SRPS) cabinets
- Plant Process Computer (PPC) changes to implement the following:
    - DCRDCS interface to PPC via new servers
    - Modify FIDMS interface software to remove calculated outputs and alarms that are now provided by the DCRDCS
    - Modify existing applications and displays to reflect new control system and removal of Group 8
    - Modify the existing Beta (Overhead alarms) Operator Menu display to remove the Window / Point Cross reference from the menu (this is not

directly related to the control system changes, but deletes an unnecessary feature in the PPC displays)

This activity is planned for implementation via several change packages, as follows:
ECR TM 07-01037: Justifications, engineering analyses, and design basis for the modification

ECR TM 07-01038: Installation details and drawings for modifications to the power and system logic cabinets, the Main Control Room (MCR) Operator Control Panel (OCP), and the Rod Position Indication Panel.

ECR TM 07-01039: Wiring diagrams, installation instructions, and testing requirements for replacement of the CRD reactor trip breakers and associated equipment.

A License Amendment Request (LAR) will be submitted to the NRC to address the deletion of the controls and indications for the Group 8 APSRAs, the configuration change to the CRD reactor trip breakers (two AC and four DC breakers are replaced with four AC breakers), the removal of the group power supply SCRs and changes to the Technical Specifications resulting from the change to the DCRDCS (Reference Technical Specification Change Request No. 342). This 50.59 review addresses the remainder of the scope of this activity.

The new DCRDCS developed by AREVA has been installed at Oconee Units 1 and 3. The hardware and firmware/software are therefore proven technology presently in use at another nuclear station.

The reason for the activity is due to the existing CRDCS including its power supply transfer system is aging, contains components that are obsolete, and is becoming increasingly difficult to maintain. The APSRAs have not been actually used for imbalance control since 1994. Internal operating experience has shown that the APSRAs, if not used properly, can exacerbate axial imbalance swings during a power transient. In the early 2000s, there were multiple fuel rod defects in various B&W units (including TMI-1) due to pellet-clad-interaction (PCI) during end-of-cycle APSRA withdrawal maneuvers. The boron dilution interlocks for Regulating Rod Groups 5 and 6 are no longer used. The connections on the CRD Group Power Supply bus bars have experienced overheating, which is difficult to solve in the existing physical configuration due to asbestos insulation and tight spacing.

The effect of this activity will be that it will no longer be possible to use the Group 8 APSRAs and there will be no more boron dilution control interlocks for Regulating Rod Groups 5 and 6. There will be four AC CRD reactor trip breakers instead of two AC and four DC breakers. The CRDCS will be a digital control system. The output of the ATWS/DSS electronic trip to the CRD reactor trip breakers will be input within the DCRDCS instead of upstream of this system, so it will no longer remove all power from the control system. The design bases functions of the CRD control system are maintained. The safety analyses as described in the UFSAR are not changed.

Summary of Conclusion for the Activity's 50.59 Review:

Screening Summary
With the exception of those items outside of the scope of this 50.59 review, which are addressed separately in the associated LAR, all existing UFSAR described CRDCS design functions are maintained within the replacement digital system. However, the change from an analog to a digital CRD control system, and installing a new Human Machine Interface are conservatively considered to be adverse changes to the CRDCS design functions, and to the way in which the system is controlled. There is no change to any element of an evaluation methodology used to establish the design bases, or in the accident analysis. This activity does not include any tests or experiments in which any SSC is utilized outside the reference bounds of design. As previously identified, a Technical Specification change is required to address the deletion of the controls and indications for the Group 8 APSRAs, the configuration change to the CRD reactor trip breakers (two AC and four DC breakers are replaced with four AC breakers), and changes to the Technical Specifications resulting from the change to the DCRDCS. A License Amendment Request (LAR) will be submitted to the NRC, and must be approved before this activity can be implemented.

Providing the ATWS/DSS trip signals to the DCRDCS controller, and utilizing this controller to generate an electronic trip increases the assumed time it takes to perform the ATWS trip from 1 second to 2 seconds. This timeline is not described within the UFSAR. As documented in the ECR TM 07-01037 design summary, the increase in DSS timeline has been evaluated (reference calculation C-1101-645-E610-022, Rev. 0, "Loss of Feedwater With Diverse Scram System At 2568 MW_t with 20% SG Tube Plugging") and found to be acceptable, because the reactor coolant system peak pressure remains bounded by the ATWS Acceptance Criterion 1 limit for peak pressure.

Evaluation Summary
Malfunction of the CRD control system (or operator error) is an initiator of the startup and rod withdrawal accidents. The new DCRDCS meets the design requirements for the CRD control system. Thus there is no increase in the frequency of occurrence of an accident previously evaluated in the UFSAR. The modified Diverse Scram System (DSS) design utilizes the same power sources as the existing DSS, which are independent of reactor trip (RPS) related power sources. There is no change to the DSS logic circuitry. The DCRDCS meets all design requirements, including redundancy of critical functions and isolation from safety related systems. Therefore there is no increase in the frequency of occurrence of a malfunction of equipment important to safety. The CRDCS is not required for accident mitigation, post accident response or offsite release mitigation. It does not perform any

plant protective functions. The action of the RPS to trip the CRD reactor trip breakers, to remove power from the control rod drives, and drop the rods into the core, remains independent of the digital CRDCS. Thus, the new DCRDCS does not increase the consequences of an accident because the DCRDCS has no role in the mitigation of the consequences of any accident previously described in the UFSAR. Because the ATWS Acceptance Criterion 1 limit for reactor coolant system peak pressure remains bounded, there is no increase in the consequences of an AWTS event. Neither the existing CRDCS nor the replacement digital CRDCS has any required function to mitigate the consequences of a malfunction of an SSC important to safety previously evaluated in the UFSAR. Failure of the DCRDCS would not prevent safety related SSCs from performing their design function. A Failure Modes and Effects Analysis (FMEA) was performed for the new DCRDCS to determine if adverse effects (i.e., loss of reactor control, uncontrolled rod withdrawal, reactor trip, or prevention of reactor trip) could result from the credible failure of a single component. The results of the FMEA were that "All operation critical to the safe and effective performance of the DCRDCS maintains sufficient redundancy such that no credible single failure can compromise the design" (reference Attachment 5 of ECR 07-01037). Thus, the DCRDCS does not increase the consequences of a malfunction of an SSC important to safety evaluated in the UFSAR. This activity does not introduce the possibility for a malfunction of an SSC with a different result because the activity does not introduce a new failure result. The reviewed activity does not change any plant safety limits, setpoints or design parameters. There is no change to the fuel, fuel cladding, reactor coolant system, or containment integrity.

<u>License Amendment Request</u>
This activity includes an associated License Amendment Request to address the deletion of the controls and indications for the Group 8 APSRAs, the configuration change to the CRD reactor trip breakers (two AC and four DC breakers are replaced with four AC breakers), and changes to the Technical Specifications resulting from the change to the DCRDCS. Thus, these items are not included within the scope of this 50.59 review. The scope of the 50.59 review is restricted to the remainder of the modification to replace the CRDCS with a new digital system. Subsequent to approval of the LAR, the full activity may be implemented without further permission from the NRC.

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Title: Control Rod Survivability UFSAR Markups

Year Implemented: 2014

Evaluation Number: AT 971475-05; TMI-14-E-0001

Brief Description:

The TMI UFSAR will be updated with a new analysis for control rod melt survivability that was performed for TMI, replacing the older hand calculation based analysis with a more robust computational analysis based on the current LOCA Evaluation Model (BAW-10192-PA). Furthermore, that UFSAR section (14.2.2.3.2.C.2) will be cleaned up to remove antiquated references to Zircaloy clad materials. Lastly, Table 14.2-14 (Sheet 1 of 2) with table notes (Sheet 1 only) is updated to the current LOCA model for TMI (Areva letter 86-9111507-000). The changes to Table 14.2-14 (Sheet 1 of 2) with table notes (Sheet 1 only) was previously evaluated under 50.59 09-00173 Rev 0 TMI-1 Cycle 18 Core Reload Design. Furthermore, the PCT values from Table 14.2-14 (Sheet 1 of 2) are covered by the 10 CFR 50.46 annual report.

The reason for the new analysis is to update the previous documented analysis with the updated control rod and fuel materials, specifically the M5 zirconium-based cladding and ELCRA inconel control rod cladding. This analysis is using updated techniques to replace the antiquated analysis which was based on hand calculations and primitive computer codes. At the time the fuel and control rod materials were changed, the control rod survivability discussion in the TMI UFSAR was seen as historic content and was not updated since control rod melting was believed to be precluded by adherence to the 10 CFR 50.46 Peak Clad Temperature (PCT) limit of 2200 °F.

In April 2006, AREVA identified in Condition Report (CR) 2006-0439 that the possibility for control rod absorber and clad melting at temperatures below 2200 °F was not addressed for newer cladding and control rod cladding materials (e.g., M5 fuel rod cladding and Inconel 625 control rod cladding for TMI). A PWR Owners Group technical position was presented to the NRC on 11/14/2006 and was documented in PWROG Letter OG-07-15, Assessment of Post LOCA Control Rod Survivability, PA-ASC-0313, January 2007. The technical position demonstrated that control rods would survive post LOCA design basis scenarios. The NRC did not challenge this position but requested the analyses be documented in accordance with 10 CFR 50 Appendix B. This activity updates the TMI UFSAR with results from AREVA analyses that have been prepared and approved in accordance with their10 CFR 50 Appendix B program.

This effort poses no impacts to plant operations or design bases. The impact to safety analyses re-validates assumptions to the LOCA analysis that control rods will not melt if PCT is maintained below the 10 CFR 50.46 limit of 2200 °F. This activity requires updates to the UFSAR sections 14.2.2.3.2 and 14.2.2.4.2.

Summary of Conclusion for the Activity's 50.59 Review:

A 50.59 Screening determined that the use of an alternative evaluation methodology required a 50.59 evaluation. All other questions screened out as no physical change is being made to the plant, no changes are being made to procedures controlling plant SSCs, no test or experiment is being performed, and no change to TMI Technical Specifications or Operating License are being made.

A 50.59 Evaluation was required and performed due to the reasoning below.

The current analysis for control rod melt survivability listed in the UFSAR relies on hand calculations and primitive computer codes used to predicted the maximum control rod temperature during a LBLOCA event. They are also based on older fuel materials that are no longer employed in the core and older control rod materials that are being phased out of use. This analysis is being replaced by detailed analytical results based on the NRC-approved LOCA Evaluation Model (BAW-10192-PA) using the NRC-approved RELAP5/MOD2-B&W computer code with significant conservatisms applied in the form of bounding inputs and assumptions (86-9144404-001).

This new analysis, using more conservative inputs/assumptions and a more robust calculation of the transient, produced more conservative results than the existing analysis. For the LBLOCA, the maximum transient control rod absorber temperature is lower than the Ag-In-Cd melt temperature and the maximum control rod cladding and guide tube temperatures remain well below the lowest eutectic temperature for melt of these materials and will therefore remain intact. For the SBLOCA, the maximum transient control rod absorber temperature is just at the Ag-In-Cd melt temperature of 1470 °F, although time at that temperature is short-lived, and the maximum control rod cladding and guide tube temperatures remain well below the lowest eutectic temperature for melt of these materials and will therefore remain intact. These new conservative results continue to meet the acceptance criteria for ensuring structural stability of the control rods by ensuring absorber material is maintained within the control rod cladding.

Because this new analysis is based on the NRC-approved LOCA Evaluation Model using the NRC-approved RELAP5/MOD2-B&W computer code and produces more conservative results then the previous primitive hand calculation based analysis, this activity is acceptable under 10 CFR 50.59 and does not require NRC approval.

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Title: Liquid Radwaste / Concentrated Waste Piping

Year Implemented: 2014

Evaluation Number: Engineering Change Request No. TM 14-00245; TMI-14-E-0002

Brief Description:

The proposed Engineering Change Request (ECR) performs the following items:

1) Physical abandonment of the WDL-T-6A/B Concentrated Waste Storage Tanks removes the Miscellaneous Waste Evaporator (WDL-Z-1B, MWEvap) concentrated waste storage capacity from the MWEvap discharge path to the Hittman liner.

2) Physical abandonment and partial removal (henceforth described as "abandonment") of the Concentrated Waste Storage Tanks (CWSTs, WDL-T-6A/B); Concentrated Radwaste Pumps (WDL-P-12A/B), and associated components on the Liquid Radwaste Disposal (WDL) system. This activity pertains to valves, instrumentation, as well as piping. This physical abandonment consists of permanent isolation that impacts transfer capabilities supporting other WDL functions. Abandonment activities by cutting and capping connecting piping to WDL subsystems includes being able to transfer inventories between:

    a. Concentrated Radioactive Waste Pumps (WDL-P-12A/B)
    b. Reclaimed Boric Acid Tanks (RBATs)
    c. Boric Acid Recycle Pumps (WDL-P-13A/B)
    d. Transfer Piping to the Hittman Building

   Due to the abandonment of the CWSTs per Activity 1, the WDL-P-12A/B pumps no longer provide an active function as no concentrated wastes are stored to require transfer. The design function of these pumps no longer exists, so these pumps are abandoned and partially removed as part of this Activity. The RBATs, WDL-P-13A/B, and processing capabilities at the Hittman Building remain fully capable of performing their radwaste system functions.

3) Interconnecting systems to the CWSTs are being cut and isolated. This includes the following system interconnections:

    a. Instrument Air (IA) to the CWST diaphragm valves,
    b. Auxiliary Steam (AS) supply lines providing heating steam to the heat coils as well as the direct heating of the contents and steam cleaning via the steam sparger
    c. Reclaimed Water (CA) supply to the CWSTs
    d. Electrical Heat Trace (HT) to the associated CWST piping and components

4) Repurpose air-operated diaphragm valves WDL-V-256 (MWEvap/RCEvap Transfer to Solidification Valve Alley) and WDL-V-418 (MWEvap/RCEvap Transfer to

Solidification Valve Alley) to accommodate the transfer of liquid radioactive waste from the Miscellaneous Waster Evaporator (MWEvap, WDL-Z-1B) and Reactor Coolant Evaporator (RCEvap, WDL-Z-1A) bottoms directly to a waste disposal liner in the Hittman Building.

5) Installation of a new flow path from the discharge of the Boric Acid Recycle (RBAT) Pumps, WDL-P-13A/B, to a portion of the existing flow path between air-operated diaphragm valves WDL-V-256 and WDL-V-418. The new flow path will include 1 check valve and 2 manual isolation valves. The installation will also include new heat tracing on the new piping.

The activity is being performed because the WDL-T-6A/B tanks, WDL-P-12A/B pumps, associated valves, instruments, and piping are no longer required for liquid radwaste processing. The abandonment and partial removal of these associated components is expected to reduce the long-term radiological dose in the immediate vicinity.

Due to abandonment of the CWSTs and associated piping, alternate flow paths are needed to transfer MWEvap and RBAT contents to the Hittman liner. To restore the ability to transfer MWEvap to the Hittman liner, the piping including valves WDL-V-256 and WDL-V-418 will be restored to service. To allow transfer of RBAT inventory, a new path from near the WDL-P-13A/B pumps to WDL-V-256 is installed. Isolation valves are added to the new RBAT path piping to ensure cross-contamination of RBAT and MWEvap water inventories is prevented, and heat trace is added to inhibit boron recrystallization in the new pipe run.

This ECR facilities removal of associated components to reduce radiation and contamination sources remaining in the vicinity, and is intended to reduce long-term personnel dose.

This activity physically abandons and partially removes the CWST and related components and removes the MWEvap concentrated waste storage capacity. This storage capacity is described as a system function described in the UFSAR, and therefore represents an adverse change described in Screening Question 1. Otherwise, the physical installation of this modification does not impact plant operation, design bases and safety analyses as described in the UFSAR, and does not impact 10CFR20 or 10CFR100 dose analyses. The modifications do not result in a change to Technical Specifications.

The proposed Activity does adversely affect how the UFSAR describes the WDL system as the concentrated waste storage capability is affected, and the UFSAR will require to be corrected for sections that continue to reference the Concentrated Waste Storage Equipment that was not deleted in ECR 10-00286 or ECR 13-00139.

Screening Question 1 has been answered "Yes" due to the elimination of the design function to store concentrated waste. Storage of concentrated liquid waste is identified as a component function attributed to the CWSTs which are being abandoned. Questions 2 through 5 have been answered "No" for the Screening. Therefore, a 50.59 Evaluation is required. The 50.59 Evaluation has all Questions answered "No" so this activity can be completed without prior NRC approval.

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Title: Polar Crane Up Limit Switch Bypass Temporary Modification

Year Implemented: 2015

Evaluation Number: Engineering Change Request No. TM 15-00424; TMI-15-E-0001

Brief Description:

Install an electrical jumper within the main hoist circuit to bypass the upper limit switch (MIS-A-MH-LS) on the Reactor Building Polar Crane (MIS-A-1) by a temporary modification. The upper limit switch prevents the main hoist block from continuing to travel into the pulley of the Reactor Building Polar Crane by interrupting the control signal to the hoist coil when the upper limit is reached. The installation of the electrical jumper prevents the limit switch from interrupting the hoist control circuit and allowing the main hoist block to continue to rise past the upper limit point.

During operation of the Reactor Building Polar Crane, the main hoist would intermittently not raise but lower. It was determined by troubleshooting there was an interruption in the control circuitry between the Main Hoist panel and the upper limit switch on the crane. To allow the continued operation uninterrupted of the main hoist, an installation of an electrical jumper will bypass the upper limit switch and allow the crane main hoist to be raised without intermittent interruptions.

The addition of the electrical jumper to bypass the upper limit switch is adverse to commitments regarding heavy load handling. The UFSAR section 9.7.1 (Design Bases for Heavy Loads and Nuclear Fuel Handling) states that heavy load handling operations are performed in accordance with NUREG-0612 which is documented in the TMI Heavy Load Submittal dated February 25, 1984. Section 2.1.7 of NUREG-0612 NRC/SER (TER-C5506-397) states that cranes shall be inspected, tested and maintained in accordance with Chapter 2-2 of ANSI B30.2. Based on ASME B30.2, section 2-3.5, the upper limit switch shall be tested in a no load condition prior to initial use of each shift. The installation of the electrical jumper would prevent the upper limit switch from interrupting the hoist control circuit and testing of the upper limit switch could not be performed. The activity is adverse to heavy load program requirement to test the polar crane in accordance with ASME B30.2.

A 50.59 Evaluation is required based on the above effect of the activity. NRC approval is not required because additional personnel will be stationed to stop uncontrolled raising of the load by polar crane malfunction.

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Procedure Changes

There were no 10 CFR 50.59 required procedure changes for this reporting period.

End of 10 CFR 50.59 Revision Report

EXELON CORPORATION
THREE MILE ISLAND
UNIT 1
DOCKET NO. 50-289

BIENNIAL COMMITMENT REVISION REPORT

JANUARY 1, 2014 THROUGH DECEMBER 31, 2015

Letter Source: NUREG – 1928 Concrete Containment Tendon Prestress

Exelon Tracking No.: 603573-38

Nature of Commitment: The TMI-1 Concrete Containment Tendon Pre-stress aging management program is an existing program that is part of the TMI-1 ASME Section XI, Subsection IWL Program. The program is based on the 1992 Edition, with 1992 Addenda, of the ASME Boiler and Pressure Vessel Code, Section XI, and includes confirmatory actions that monitor loss of containment tendon pre-stressing forces during the current term and will continue through the period of extended operation.

Summary of Justification:

Commitment Change Evaluation Form 14-01 revised this commitment: The ASME Boiler and Pressure Vessel Code, Section XI code of record is updated every 10 years as specified by 10 CFR 50.55a. This change in commitment is made to reflect the current ISI program plan as approved by the NRC and eliminates the need for future commitment updates based on code year revisions.

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Letter Source: Ltr. 5211-87-2174 SER: IST Program SER for Second Ten Year Interval

Exelon Tracking No.: 1987T0053 / 1122355-26

Nature of Commitment: GPUN committed to the NRC that annual diesel generator inspections will identify any long term diesel head cylinder check valve seat burning and long term wear to the air start distributor that might result from a malfunction of check valves EG-V-17A/B.

Summary of Justification:

Commitment Change Evaluation Form 14-02 cancelled this commitment. Since the EG-V-17A/B valves are not safety class 1, 2, or 3, the ASME OM Code does not require them to be considered for testing under the IST program requirements. As such, they are not included in the current IST Program Plan. Other surveillances and PM activities on the diesel generator units ensure the proper monitoring of the condition of the valves, and functional testing of the units.

FSAR section 8.2.3.6 includes the following specific discussion regarding periodic inspections: "Diesel generator reliability is achieved with an inspection every 24 months (with a 25% allowable grace period) in accordance with procedures prepared in conjunction with the applicable recommendations of the Fairbanks Morse Owners Group and those of the manufacturer for this class of standby service.

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Letter Source: NF-86-4736 TMI-1 FA Holddown Spring Surveillance Program

Exelon Tracking No.: 1986T0044 / 1122355-21

Nature of Commitment: GPUN committed internally to perform hold-down spring inspection per procedure RP 1507-12 Section 7.4 to identify any indications of breaks, cracks, or excessive corrosion or crud.
Remarks: The inspection of the hold-down springs will be conducted during each outage for the foreseeable future. All data to be reported to B&W. The spring surveillance program is described in Memo NF-86-4736.

Summary of Justification:

Commitment Change Evaluation Form 14-03 cancelled this commitment. In GPU Nuclear letter 6710-97-2303, "TMI-1 Fuel Assembly Hold-down Spring Surveillance Program" dated 7/22/1997, GPU Nuclear informed the USNRC that TMI was discontinuing the hold-down spring surveillance program initially committed to in GPUN letter TLL-447, "TMI-1 Fuel Assembly Hold-down Springs, dated 9/10/1980. This commitment should have been deleted at that time. The justification for discontinuing the surveillance program and deleting the commitment was that the helical coil springs that were susceptible to cracking were no longer being used for TMI reloads. A shot peening process eliminated the cracking issue for helical springs and TMI ultimately transitioned to fuel assemblies with a leaf spring design in Cycle 11 (1995). Since helical springs are no longer used at TMI the surveillance is no longer required.

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Letter Source: IEB 80-15 Possible Loss of Emergency Notification System with LOOP

Exelon Tracking No.: 1980T0008 / 1122072-72

Nature of Commitment: Issue an Administrative procedure that required notification of the NRC Operations Center within one hour of the time one or more of the Emergency Notification Systems ENS extensions are found to be inoperable.

Summary of Justification:

Commitment Change Evaluation Form 14-04 cancelled this commitment. This concern is now covered under 10 CFR 50.72 (b) (3) (xiii)

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Letter Source: 5211-83-2098: OTSG Topical Report 008 Revision 2 Section IV.B Page 29

Exelon Tracking No.: 1983T0094 / 1122072-55

Nature of Commitment: Prevent the introduction of chemical contamination through administrative controls such as:

1) Clear labeling of the tanks in the chemical addition room.
2) Locking open the breakers to pumps CA-P-2, CA-P-3 and CA-P-4 and placing them under the administrative control of the locked valve and component list.
3) The review of applicable procedures to insure that adequate guidance is provided. Also, to improve bulk chemical specifications and administrative procedures to control chemical contamination.

Summary of Justification:

Commitment Change Evaluation Form 14-05 cancelled this commitment. The original TMI-1 steam generators were replaced in 2009. The new TMI-1 steam generators have Alloy 690 tubing that is considerably more corrosion resistant than the original steam generators, which had Alloy 600 tubing.

The above items, such as locked chemical addition tanks, improved component labelling, and the chemical control program, are now "standard operating procedures" for the Exelon fleet. These items will, arguably, remain in place irrespective of whether the commitment from 1983 remains active.

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Letter Source: 5211-85-3106: SER 1984 TMI-1 Steam Generator Tube Indications

Exelon Tracking No.: 1985T0097 / 1122355-16

Nature of Commitment: GPUN had established the following guidelines to maintain protective conditions for cold layup conditions:

(1) The pH of the reactor coolant was elevated to above 7.2 using Ammonia.
(2) Chlorides and sulfates were controlled to below 100 ppb as they were during operation.
(3) The oxygen concentration was kept under 100 ppb when the system was filled and able to be pressurized. In cases where the Reactor Coolant System (RCS) was open and oxygen could not be excluded, air saturated conditions were specified.
(4) To prevent local buildup of contaminants, the OTSG water inventory in the primary side was periodically mixed by recirculating during layups.

Summary of Justification:

Commitment Change Evaluation Form 14-06 cancelled this commitment. The original TMI-1 steam generators were replaced in 2009. The new TMI-1 steam generators have Alloy 690 tubing that is considerably more corrosion resistant than the original steam generators, which had Alloy 600 tubing.

Today the steam generators' primary sides are not typically in wet layup. In the early 1980's, when the subject commitment was authored, the TMI-1 plant was in an extended wet layup period awaiting permission to restart the plant after the TMI-2 accident. Today, the steam generators' primary sides are drained during refueling outages (to perform eddy current) and then refilled for plant restart. Extended time periods in wet layup are no longer commonplace, or expected, as they were in the 1980's.

Currently the plant's procedure for primary chemistry (CY-AP-120-105) is based on the EPRI Pressurized Water Reactor Primary Water Chemistry Guidelines. All U.S. PWRs have committed to follow chemistry programs consistent with those guidelines, which did not exist in the 1980's.

In summary, the subject commitment is no longer applicable to TMI-1. TMI-1 is no longer a plant with Alloy 600 steam generator tubes that spends long periods of time with primary water in a static, cold wet layup.
TMI-1 is now a plant with Alloy 690 steam generator tubes – and where primary coolant is either drained, or is recirculated by the Decay Heat System, during refueling outages every 2 years.

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Letter Source:	NRC Bulletin 2005-02, Emergency Preparedness and Response Actions for Security-Based Events
Exelon Tracking No.:	5928-05-20234.001 / 1122072-05
Nature of Commitment:	The Emergency Plan and the EAL-set will be updated to reflect the information provided in Bulletin 2005-02, Attachment 2, "Examples of Acceptable Changes to Emergency Classification Levels and Emergency Action Levels for Security Events," Page 6, "NUMARC/NESP-007 guidance." Acts of civil disobedience or felonious acts that are not part of a concerted attack on the station will be addressed by non-hostile action-based EALs and EALs addressing other hazards to station operation continue to apply. Other modifications to the EAL-set will be made to maintain internal consistency, e.g ., EALs for classification based on judgment.

Summary of Justification:

Commitment Change Evaluation Form 14-07 cancelled this commitment. Rev 5 to NEI 99-01 represents several years of use and implementation of the NEI 99-01 methodology. Initially, portions of Rev 4 were superseded by NRC Bulletin 2005-02 "Emergency Preparedness and Response Actions for Security-Based Events" to immediately implement changes to the security philosophy following the events of September 11, 2001. This process was accomplished using a NEI White Paper "Enhancements to Emergency Preparedness [programs For Hostile action", May 2005 (Revision November 18, 2005) and endorsed by the NRC in RIS 2006-12 on July 19, 2006. The security changes are formalized with rev 5.

In order to address development and implementation issues, a FAQ process was used to take input from the industry and the NRC. The NEI 99-01 EAL FAQ Task Force evaluated each concern presented and provided an industry perspective to each. The Task Force presented the recommendations to the NRC for consideration for approval. FAQ that were acceptable are incorporated with the change.

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Letter Source: NRC Bulletin 2005-02, Emergency Preparedness and Response Actions for Security-Based Events

Exelon Tracking No.: 5928-05-20344.006 / 1122072-10

Nature of Commitment: The Emergency Plan will be revised to include provisions for drills and exercises using terrorist-based events described in Bulletin 2005-02, Attachment 6, "Examples of Acceptable Changes to Security Related Drill and Exercise Program," as follows:

-after the NRC pilot, a terrorist-based scenario will be conducted as scheduled in conjunction with the FEMA Region III scheduling conferences;
-following the NRC inspected plant demonstration, terrorist based scenarios will be incorporated into the 6-year exercise program as scheduled in conjunction with the FEMA Region III scheduling conferences;
-full implementation will include engagement of offsite responders and FEMA.

This schedule is based on meeting several milestones, including:

-development of draft industry guidelines by the end of 2005;
-conduct of industry tabletop drills; and
-NRC and FEMA endorsement of the new evaluated exercise process program.

NOTE: Scheduled Completion Date is "Upon NRC and FEMA endorsement of the new evaluated exercise process.

Summary of Justification:

Commitment Change Evaluation Form 14-08 cancelled this commitment. 10 CFR 50 Appendix E states the requirements for drill frequencies.

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Letter Source: 5211-85-3303: GL 83-28 Item 4.2.1 and 4.2.2 (Reactor Trip System Reliability SER)

Exelon Tracking No.: 1985T0099 / 1122355-17

Nature of Commitment: NRC stated within 12/12/85 SER that GPUN Committed to Maintain a 6 Month Maintenance Interval for CRD General Electric AK Breakers until sufficient information was available to assess Breaker performance. CRD Breaker maintenance interval may be changed to 12 months with acceptable breaker performance.

TMI-1 CRD Breaker periodic maintenance program include the following inspections:

1. Verification of breaker cleanliness and insulation
2. Verification of breaker physical condition to include wiring and terminations
3. Verification of proper manual operation (check for excessive friction, trip bar freedom, latch engagement, mechanism alignment and Under-Voltage Trip UVT Device freedom of movement).
4. Verification of Armature freedom of movement
5. Verification of Trip Latch Engagement per GE Service Advice 175-9.3S Item S2
6. Verification of Under-voltage pick up testing per GE Service Advice 175-9.3S Item S3
7. Verification of Trip Shaft Torque (Less than 1.5 pound-inches per GE Service Advice 175-9.3S Item S5
8. Verification of proper Trip response time per GE Service Advice 175-9.3S Item S6
9. Shunt Trip Attachment (STA) operation verification
10. Examination and cleaning of breaker enclosure
11. Functional Test of breaker prior to Return to Service

Preventive Maintenance Procedure E-36 Revision 17 dated 9/15/88 fulfills the above requirements

Summary of Justification:

Commitment Change Evaluation Form 14-09 cancelled this commitment. NRC Commitment 1985 (Reactor Trip System Reliability SER) and Issue Report 01122355 Assignment 17 is no longer required and is no longer applicable to TMI-1 plant. The Control Rod Drive System has been up-graded (ECR TM 07-01037) to a Digital Control System that utilizes Square D

Masterpact Model NT08N1 breakers. General Electric AK style Breakers are no longer used for the Control Rod Drive System. Maintenance Procedure E-36 has been deleted from EDMS.

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Letter Source: GL 83-28 Section 2.1 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (RTS)

Exelon Tracking No.: 1983T0118 / 1122073-57

Nature of Commitment: Required the confirmation that all components whose functioning was required to trip the reactor were identified as safety related on all documentation and in information handling systems that were used to control all activities performed on that safety related equipment. The NRC SER 5211-87-3170 accepted the GPUN response and closed the item.

Summary of Justification:

Commitment Change Evaluation Form 15-01 cancelled this commitment. The GL 90-03 states that the Vendor Equipment Technical Information Program (VETIP) described in the Nuclear Utility Task Action Committee Report, INPO 84-10 issued March 1984, meets the intent of Generic Letter 83-28, Item 2.2 part 2. The VETIP program includes the Nuclear Plant Reliability Data System and Significant Event Evaluation and Information Network, both managed by INPO. It also includes existing programs the utilities now conduct with vendors.

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Letter Source: GL 83-28 Item 4.2.1 Reactor Trip System Reliability

Exelon Tracking No.: 1986T0081 / 1122355-23

Nature of Commitment: GPUN committed to the NRC to revise procedure E-36 to further clarify the gap measurement of the under voltage trip device. Procedure E-36 step 6.5 requires the clearance between the rivet and the armature of the under voltage trip relay device be measure on the GE style breakers.

Summary of Justification:

Commitment Change Evaluation Form 15-02 cancelled this commitment. This commitment needs to be closed as the GE style breakers were replaced with Momter Pac breakers and is no longer applicable to the new style breaker.

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Letter Source:	NUREG 0612: Control of Heavy Loads
Exelon Tracking No.:	1984T0075 / 1122072-79
Nature of Commitment:	NUREG 0612, Control of Heavy Loads required commitment to be placed in 1506-8 Seal Plate Installation work.

Summary of Justification:

Commitment Change Evaluation Form 15-03 cancelled this commitment. The seal plate was permanently installed and no longer requires to be removed/installed.

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