January 23, 2004

Mr. Christopher M. Crane
President and CEO
AmerGen Energy Company, LLC
200 Exelon Way, KSA 3-E
Kennett Square, PA 19348

SUBJECT: THREE MILE ISLAND STATION, UNIT 1 - NRC INTEGRATED INSPECTION REPORT 05000289/2003005

Dear Mr. Crane:

On December 31, 2003, the Nuclear Regulatory Commission (NRC) completed an inspection at Three Mile Island, Unit 1 (TMI) facility. The enclosed report documents the inspection findings that were discussed January 16, 2004 with Mr. Bruce Williams and other members of your staff.

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

The report documents one self-revealing finding and four NRC identified findings of very low safety significance (Green). The findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating them as non-cited violations (NCV) consistent with Section VI.A of the NRC Enforcement Policy. If you contest the NCVs, you should provide a response within 30 days of the date of this inspection report (IR), with the basis of your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspectors at Three Mile Island.

Since the terrorist attacks on September 11, 2001, the NRC has issued five Orders and several threat advisories to licensees of commercial power reactors to strengthen licensee capabilities, improve security force readiness, and enhance controls over access authorization. In addition to applicable baseline inspections, the NRC issued Temporary Instruction 2515/148, "Inspection of Nuclear Reactor Safeguards Interim Compensatory Measures," and its subsequent revision, to audit and inspect licensee implementation of the interim compensatory measures required by order. Phase 1 of TI 2515/148 was completed at all commercial power nuclear power plants during calendar year 2002 and the remaining inspection activities for TMI were completed during calendar year 2003. The NRC will continue to monitor overall safeguards and security controls at TMI.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARs) component of the NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

We appreciate your cooperation. Please contact me at 610-337-5234 if you have any questions regarding this letter.

Sincerely,

/RA by Richard S. Barkley, P.E. Acting For/

Peter W. Eselgroth, Chief
Reactor Projects Branch 7
Division of Reactor Projects

Docket No: 50-289
License No: DPR-50

Enclosure: Inspection Report 05000289/2003005
w/Attachment: Supplemental Information

<u>cc w/encl</u>:
Chief Operating Officer - AmerGen
Site Vice President - TMI Unit 1, AmerGen
Plant Manager - TMI Unit 1, AmerGen
Regulatory Assurance Manager - TMI Unit 1, AmerGen
Senior Vice President - Nuclear Services, AmerGen
Vice President - Mid-Atlantic Operations, AmerGen
Vice President - Operations Support, AmerGen
Vice President - Licensing and Regulatory Affairs, AmerGen
Director Licensing - AmerGen
Manager Licensing - TMI, AmerGen
Vice President - General Counsel and Secretary, AmerGen
Correspondence Control Desk - AmerGen
Chairman, Board of County Commissioners of Dauphin County
Chairman, Board of Supervisors of Londonderry Township
R. Janati, Chief, Division of Nuclear Safety
M. Schoppman, Framatome ANP
J. Johnsrud, National Energy Committee
E. Epstein, TMI-Alert (TMIA)
D. Allard, PADER

<u>Distribution w/encl</u>:
Region I Docket Room (with concurrences)
D. Kern, DRP - Senior Resident Inspector
H. Miller, RA
J. Wiggins, DRA
P. Eselgroth, DRP
R. Barkley, DRP
J. Jolicoeur, OEDO
R. Laufer, NRR
T. Colburn, NRR
D. Skay, PM, NRR
R. Architzel, NRR

ADAMS ACCESSION #
DOCUMENT NAME: C:\ORPCheckout\FileNET\ML040230730.wpd
After declaring this document “An Official Agency Record” it **<u>will</u>** be released to the Public.
To receive a copy of this document, indicate in the box: "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

OFFICE	RI/DRP		RI/DRP		RI/DRP			
NAME	DKern/***RSB for***		RBarkley***/RSB***		PEselgroth***/RSB for***			
DATE	1/20/04		1/22/04		1/23/04			

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION

REGION 1

Docket No:	05000289
License No:	DPR-50
Report No:	050000289/2003005
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Three Mile Island Station, Unit 1
Location:	PO Box 480 Middletown, PA 17057
Dates:	September 28, 2003 - December 31, 2003
Inspectors:	David M. Kern, Senior Resident Inspector Javier M. Brand, Resident Inspector Jamie C. Benjamin, Reactor Inspector, DRS Fred L. Bower, Senior Reactor Inspector, DRS Thomas F. Burns, Reactor Inspector, DRS Anthony Dimitriadis, Physical Security Inspector, DRS Paul R. Frechette, Physical Security Inspector, DRS Joseph T. Furia, Senior Health Physicist, DRS Jorge E. Hernandez, Reactor Engineer, DRP Nancy T. McNamara, EP Inspector, DRS David M. Silk, Senior Emergency Preparedness Specialist, DRS
Approved by:	Peter W. Eselgroth, Chief Projects Branch 7 Division of Reactor Projects (DRP)

CONTENTS

SUMMARY OF FINDINGS

IR 05000289/2003005; 09/28/2003 - 12/31/2003; AmerGen Energy Company, LLC; Three Mile Island, Unit 1; Refueling and Outage Activities and Other Activities (Reactor Containment Sump Blockage - NRC Bulletin 2003-01).

The report covered a thirteen-week period of inspection by resident inspectors and announced inspections by regional inspectors. Five Green non-cited violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Rev. 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Barrier Integrity

- Green. A self-revealing non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," was identified for failure to identify and correct reactor coolant system (RCS) pressure boundary leakage in a timely manner. Failure to identify the leakage during the previous refueling outage resulted in continued RCS barrier degradation and power operation from November 2001 until October 2003 with non-isolable RCS strength boundary leakage.

  The issue is more than minor because it adversely affected the barrier integrity cornerstone in that it reduced the likelihood that the physical RCS design barrier would protect the public from radio nuclide releases. In addition, if left uncorrected, the issue could become a more significant safety concern (i.e., RCS inventory loss). The inspectors determined this finding is of very low safety significance (Green) because the RCS leakage was small, the likelihood of a rapid increase in RCS leak rate was small due to the robust cover plate design, the remaining mitigation functions were unaffected, and the containment barrier remained fully functional to prevent radio nuclide release to the public. (Section 1R20)

- Green. The inspectors identified a non-cited violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," for failure to implement proper corrective actions to prevent corrosion of the containment liner. The corrosion resulted in reduced liner wall thickness that exceeded the ASME XI acceptance criteria.

  This issue affected the barrier integrity cornerstone and is more than minor because the condition impacted configuration control in that containment barrier wall thickness design parameters were not maintained. In addition, if left uncorrected, the condition could have affected the availability and reliability of the safety-related containment liner to protect the public from radio nuclide

release. This finding is of very low significance since the issue did not involve an actual open pathway in the physical integrity of the containment. (Section 1R20)

- Green. The inspectors identified a non-cited violation for failure to comply with 10 CFR 50, Appendix B, Criterion X, “Inspection.” This violation involved the installation of a floor grating for a permanent structure inside the containment that did not meet the required separation distance to the containment liner per structural drawing 421054. Station personnel failed to identify this degraded condition during containment inspections. The inadequate structural clearance increased the likelihood that the safety-related containment liner would be damaged during a postulated seismic event.

  This finding affected the barrier integrity cornerstone and is more than minor because the condition impacted configuration control in that the containment design parameter for clearance between structures and the containment liner was not maintained. In addition, if left uncorrected, the condition could have affected the availability and reliability of the safety-related containment liner to protect the public from radio nuclide release. The finding is of very low safety significance because the issue did not involve an actual open pathway in the physical integrity of the containment. (Section 1R20)

- Green. The inspectors identified a non-cited violation of technical specification 6.8.1.a for failure to properly perform inspections to assess the overall health of coatings inside the containment as required by procedure EP-055T. This issue reflected deficient human performance and problem identification because the applicable station procedure was not used and numerous existing degraded containment coating conditions were not identified. The inspectors subsequently identified various degraded containment coating issues. Corrective actions included a complete reinspection of containment coatings, which resulted in identification and evaluation of 127 coating indications.

  This finding is greater than minor because it affected the barrier integrity cornerstone and if left uncorrected, the condition could have degraded further and affected the operability of the safety-related containment sump and liner. The finding is of very low safety significance since the issue did not involve an actual open pathway in the physical integrity of the containment or an actual blockage of the containment sump. (Section 1R20)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation for failure to identify, document, and assess conditions adverse to quality which had the potential to adversely affect emergency core cooling system (ECCS) containment sump availability. The inspectors observed numerous sources of debris within containment and sump screen conditions which had the potential to degrade ECCS performance. Station personnel saw most of these same conditions, but did not document or assess the associated impact on containment sump operability until the issue was raised by the inspectors. Failure to recognize and evaluate screen blockage and sources of continued debris within containment could lead to further containment sump degradation and make ECCS systems inoperable.

This finding affected the mitigating systems cornerstone and is more than minor because it had the potential to adversely impact equipment availability and reliability for multiple ECCS systems which are designed to respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding was of very low safety significance (Green) because subsequent engineering evaluations concluded that the adverse sump conditions would not cause an actual loss of safety function. (Section 4OA5)

B. Licensee-Identified Findings

None.

REPORT DETAILS

Summary of Plant Status

AmerGen Energy Company, LLC (AmerGen), operated Three Mile Island, Unit 1 (TMI) at reduced power (approximately 98 percent) due to end-of-cycle coastdown effects until October 16, 2003, when power was reduced to 55 percent to investigate a tube leak in the ‘A’ side of the non-safety related main steam condenser. A failed heater drain receiver tank level controller caused an unplanned power reduction to 45 percent. The unit was shut down on October 18, 2003 for the Cycle 15 refueling outage (1R15). Major outage activities included: reactor vessel head replacement and reactor vessel lower head penetration nozzles inspection.

Operators synchronized the unit to the electrical grid on December 5, 2003 completing a 48-day refueling outage. On December 7, 2003, a failure of the ‘A’ main feedwater pump coupling caused an unplanned power transient from 98 to 63 percent power (Section 1R14). Following repairs, operators restored the unit to 100 percent power on December 22, 2003.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R02 Evaluation of Changes, Test, or Experiments (71111.02)

1. Replacement Reactor Vessel Closure Head Design and Planning

a. Inspection Scope

The inspectors verified that the licensee reviewed and documented the reactor vessel closure head (RVCH) related design changes and modifications to components described in the Updated Final Safety Analysis Report (UFSAR) in accordance with 10 CFR 50.59. The inspectors also reviewed the adequacy of 10 CFR 50.59 applicability reviews, screening evaluations, or safety evaluations for various procedure changes, design changes, and modifications. The inspectors also verified that any safety issues pertinent to the changes were resolved. Complete listings of documents reviewed are included in the attachment. Additional inspections of the RVCH replacement project were documented in report Sections 1R17, 1R15, 4OA2 and 4OA5.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

Partial System Walkdown.

The inspectors performed two partial system walkdowns on the following systems and components:

- ‘A’ decay heat removal (DHR) train while the ‘B’ train was running for shutdown cooling
- ‘A’ emergency feedwater system

The partial system walkdowns were conducted on the redundant and standby equipment to ensure that trains and equipment relied on to remain operable for accident mitigation were properly aligned and protected. The following documents were used for this inspection.

- Emergency feedwater flow diagram 302-082, Rev. 22
- Surveillance procedure 1303-11.57, “EFW Flowpath Check,” Rev. 5, completed on November 22, 2003
- OP-TM-212-111, “Shifting DHR Train A From DHR Standby to DHR Operating Mode,” Rev.2, completed on October 10, 2003

Complete System Walkdown.

The inspectors performed complete system walkdowns on the following systems and components:

- On October 23, 2003, ‘B’ DHR train while it was being used to provide shutdown cooling
- On October 29, 2003, ‘B’ spent fuel pool cooling train while fuel was being transferred into the spent fuel pool

The inspectors conducted a detailed review of the alignment and condition of the associated components. The inspectors reviewed applicable flow diagrams 302-640 Rev. 78, 302-641 Rev. 6, and 302-645 Rev. 36 for the DHR system, and 302-630, Rev. 31 for the spent fuel pool cooling system. In addition, the inspectors reviewed and evaluated the open work orders and corrective action program condition reports for impact on system operation. The system health reports were also reviewed and open issues were discussed with the system engineers. The inspectors also verified system parameters were within the required band for current plant conditions as determined by TMI operating logs and procedures.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

1. Annual Drill Observation (71111.05A)

a. Inspection Scope

The inspectors observed the crew performance of an unannounced plant fire drill on October 10, 2003. The inspectors observed fire fighters donning protective clothing and self-contained breathing apparatus and observed the fire fighting techniques employed against the simulated fire. The inspectors evaluated the brigade leader's performance on the use of preplanned strategies and communications with the fire team members and the main control room. The inspectors attended the post-drill critique, and reviewed CR 180303 which evaluated minor discrepancies identified by the inspectors regarding problems opening fire door C104 and the fire brigade post drill critique of the issue.

b. Findings

No findings of significance were identified.

2. Area Walkdowns (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection inspections for the following plant zones:

- Zone CB-FA-3D, Control Building Elevation 338'-6", Relay Room
- Zone AB-FZ-2A, Auxiliary Building Elevation 281', Makeup & Purification Pump A
- Zone AB-FZ-2B, Auxiliary Building Elevation 281', Makeup & Purification Pump B
- Zone AB-FZ-2C, Auxiliary Building Elevation 281', Makeup & Purification Pump C
- Zone AB-FA-1, Auxiliary Building Elevation 261', DHR Pit A
- Zone AB-FA-2, Auxiliary Building Elevation 261', DHR Pit B
- Zone CB-FA-2F, Control Building Elevation 322', East Battery Room
- Zone CB-FA-2G, Control Building Elevation 322', West Battery Room
- Zone CB-FA-2A, Control Building Elevation 322', Switch Gear Room
- Electrical Penetration # 317, in Zone RB-FZ-1C, Reactor Building Elevation 281'
- Zone RB-FZ-1d, Reactor Building Elevation 281', Inside Secondary Shield East
- Zone RB-FZ-1e, Reactor Building Elevation 281', Inside Secondary Shield West during various pressurizer weld repair activities

The rooms and areas were selected based on enclosing equipment important to safety. The inspectors conducted plant walkdowns and verified the areas were as described in the TMI Fire Hazard Analysis Report (FHAR). The plant walkdowns were conducted throughout the inspection period and included assessment of transient combustible material control, fire detection and suppression equipment operability, and compensatory measures established for degraded fire protection equipment. In addition, the inspectors verified that applicable clearances between fire doors and floor met the specified criteria per Technical Evaluation CC-AA-309-101, "Fire Door Acceptance Criteria," Rev. 0. The inspectors observed several fire doors to vital areas which were repeatedly found unlatched. In each case, compensatory measures were initiated in accordance with station procedures. Station personnel initiated CRs to address the inspector's concern that this was a repetitive problem.

Based on the high combustible loading in the relay room, the inspectors conducted walkdowns of the area and compared the transient combustibles in the relay room against the combustibles listed in Exelon Training Guidance OP-TM-201-009-1001, "Transient Combustible Controls," Rev. 0. This was accomplished to ensure that TMI had maintained the combustible loading in this risk significant area in accordance with the design and licensing bases as described in the FHAR. The inspectors also reviewed CRs 189585 and 180346 which documented minor discrepancies identified by the inspectors regarding a latching concern with fire door (C-311) and the transient combustible materials listed in OP-TM-201-009-1001.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed AmerGen's internal flooding mitigation strategy for protection of the emergency feedwater pumps and nearby components following a main feedwater pipe break. The inspectors walked down the emergency feedwater pump rooms and the area enveloped by a circular retaining wall surrounding the reactor containment (commonly known at TMI as the "Alligator Pit"), interviewed the flood protection engineer and his supervisor, and reviewed the following documents:

- UFSAR Section 2.6.4, "Flood Studies"
- UFSAR Appendix 14A, "Design Review for Consideration of Effects of Piping System Breaks Outside Containment"
- AmerGen Technical Data Report, TDR-250, "Review of Intermediate Building Flooding Following a Feedwater Line Break in the Intermediate Building of TMI Unit 1," Rev. 3
- Calculation C-11-1-424-E540-064, "Flooding Due to a Postulated Pipe Break in the Intermediate Building," Rev. 1

1R07 Heat Sink Performance (71111.07B)

a. Inspection Scope

The inspectors reviewed the heat removal capability of the safety-related decay heat service closed cooling water Coolers (DC-C-2A/B) and the station blackout (SBO) diesel generator lube oil cooler, air cooler, and jacket water heat exchanger. The inspectors compared recent surveillance test data to the test acceptance criteria which had been developed through engineering calculations. The inspectors compared the bounding calculations and acceptance criteria to the licensing and design bases to ensure the minimum design basis assumptions were appropriately incorporated. The inspectors also verified that the number of tubes plugged in the DC-C-2A/B exchangers were accounted for in current calculations. The inspectors performed a walk down of the DC-C-2A/B heat exchangers and SBO diesel to assess their current material condition. In addition, the inspectors interviewed key personnel responsible for oversight of the heat exchangers to assess the adequacy of performance monitoring.

The inspectors reviewed a sample of condition reports (CRs) over the past two years related to equipment, programs, and performance of the DC-C-2A/B and SBO diesel heat exchangers. The inspectors reviewed the CRs to ensure that equipment deficiencies were being properly identified and evaluated, and that corrective actions were effective.

The inspectors reviewed the processes and programs used to treat the DC-C-2A/B and SBO diesel generator heat exchangers. Chemical addition processes to control fouling were reviewed for their effectiveness to ensure heat removal capabilities. The inspectors conducted interviews with knowledgeable personnel to assess challenges with various bio-fouling mechanisms. Additionally, the inspectors reviewed the Microbiologically Influenced Corrosion (MIC) program to determine the current effectiveness and future direction to mitigate MIC on heat exchanger performance.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (71111.08)

a. Inspection Scope

The inspectors observed selected samples of nondestructive examination (NDE) activities in process. Also, the inspectors reviewed selected additional samples of completed NDE and repair/replacement activities. The sample selection was based on the inspection procedure objectives and risk priority of those components and systems where degradation would result in a significant increase in risk of core damage. The observations and documentation reviews were performed to verify the activities were performed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements. The inspectors reviewed a sample of inspection reports and CRs initiated as a result of problems identified during inservice inspection (ISI) examinations. Also, the inspectors evaluated effectiveness in the resolution of problems identified during selected ISI activities.

The inspectors observed the performance of two NDE activities in process and reviewed documentation and examination reports for an additional four NDE activities. The inspectors reviewed four samples of welding activities on a pressure boundary, one ASME repair package for a repair performed this operating cycle, and one repair package for a repair performed during the previous operating cycle.

The inspectors observed manual ultrasonic testing (UT) and visual examination (VT) activities to verify effectiveness of the examiner, process, and equipment in identifying degradation of risk significant systems, structures and components and to evaluate the activities for compliance with the requirements of ASME Section XI of the Boiler and Pressure Vessel Code.

The inspectors observed the UT performed on reactor coolant system (RCS) nozzle to pipe weld 1D-ISI-RC-002 and the visual examination of the reactor pressure vessel (RPV) lower head instrumentation nozzles. The inspectors reviewed the examination reports of liquid penetrant testing (LP) of RCS nozzle to pipe weld B5.130, decay heat (DH) elbow to pipe welds 0012, 0013, and 0014. Also, the inspectors reviewed the LP test report of the pressurizer nozzle to pipe weld (SR0010BMWELD). The inspectors also reviewed the radiographs and the examiners' interpretation of indications observed within field welds 442, 445, 447 and the subsequent weld repair (442R1) in the makeup system (MU). The inspectors verified that the identification, characterization, disposition and repair of the indications were appropriate.

The inspectors evaluated implementation of the steam generator program by reviewing specific portions of the outage 1R15 steam generator management plan, condition monitoring, and final operational assessment. The inspectors reviewed plant specific steam generator information, tube inspection criteria, control and monitoring of foreign objects, integrity assessments, degradation modes, and tube plugging criteria. The eddy current test (ECT) probes and equipment were qualified for the expected types of active tube degradation.

The inspectors verified the licensee was performing a 100 percent bobbin inspection for the entire tube length in both generators. The inspectors confirmed that the ECT scope and expansion criteria met technical specification (TS) requirements, Electric Power Research Institute (EPRI) Guidelines, and commitments made to the NRC.

The inspectors confirmed that areas of potential degradation (based on site-specific and industry experience) were being inspected, with special attention to areas that are known to represent potential ECT challenges.

No tubes were repaired during this inspection period. No tubes were identified as candidates for in-situ pressure testing during the inspection period. The inspectors confirmed that steam generator leakage was minor and did not exceed greater than three gallons per day during the previous operating cycle or during post shutdown visual inspection of the tube sheet face.

To evaluate specific implementation of the steam generator inspection program, the inspectors interviewed data management and acquisition personnel and resolution analysts. The inspectors interviewed the licensee's independent qualified data analyst, and reviewed selected samples of the eddy current data acquisition and analysis of selected tubes within the 'A' and 'B' steam generators. Also, the inspectors verified that the licensee had revised their steam generator degradation assessment to address a degradation mechanism recently identified at a similar plant, which had not previously been identified at TMI.

The inspectors reviewed welding activities associated with the repair of selected components to verify the activities were performed in accordance with the requirements of ASME Sections IX and XI. The inspectors reviewed selected portions of work documents WOC2004373 and TM02-01194, Rev. 1 which provided the instructions for the replacement of valve MU-V-20 and MU–260 in the MU system. The inspectors reviewed the weld history record, welding instructions, welding procedure, welding procedure qualification, NDE requirements, and the test results of the completed welds. The inspectors reviewed welding procedure specification 821-TMI and the supporting procedure qualification records for compliance with the requirements of ASME Section IX.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

The inspectors observed licensed operator requalification training at the control room simulator. The inspectors reviewed the operators' ability to correctly evaluate the simulator training scenario and implement the emergency plan. The inspectors observed the operators' simulator drill performance and compared it to the criteria listed in simulator scenario "12/08/03 Dual Station Drill (TMI/LGS)." The inspectors observed supervisory oversight, command and control, communication practices, and crew assignments to ensure they were consistent with normal control room activities. The inspectors observed operator response during the simulator drill transient and verified the fidelity of the simulator to the actual plant. The inspectors observed the effect training evaluators had in recognizing and correcting individual and operating crew mistakes including post-training remediation actions. The inspectors attended the post-drill critique in order to evaluate the effectiveness of problem identification.

b. Findings

No findings of significance were identified.

1R12 Maintenance Implementation (71111.12)

a. Inspection Scope

The inspectors evaluated Maintenance Rule (MR) implementation for the issues listed below. Specific attributes reviewed included MR scoping, characterization of failed structures, systems, and components (SSCs), MR risk categorization of SSCs, SSC performance criteria or goals, and appropriateness of corrective actions. The inspectors verified that the issues were addressed as required by 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Rev. 2, and AmerGen procedure ER-AA-310, "Implementation of The Maintenance Rule, " Rev. 2.

- CR 186781, which evaluated the failure of pressurizer makeup valve MU-V-17 to open during plant cooldown. The inspectors reviewed applicable TS (Sections 3.5.5 and 3.1.12), and interviewed the system engineer. In addition, the inspectors verified that engineers properly categorized this failure as a maintenance rule functional failure.

- The nuclear services closed cooling water (NS) system was classified as maintenance rule category (a)(1) in May 2001 due to excessive unavailability. The principal causes were (1) ineffective scheduling and resource management which permitted extended component unavailability, (2) human performance errors during maintenance, and (3) mechanical seal and motor degradation. These issues were documented and evaluated in corrective action program documents (CAP) T2001-0431, 2001-347, and T2001-0552. The inspectors performed a partial system walkdown, reviewed the system health report and long term plan, and reviewed pump performance data to determine whether corrective actions were appropriate to improve the effectiveness of maintenance on the NS system.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13)

a. Inspection Scope

The inspectors reviewed the scheduling and control of maintenance activities in order to evaluate the effect on plant risk. This review was against criteria contained in AmerGen Topical Report 097, "TMI Outage Fuel Protection Criteria," Rev. 7 and AmerGen Administrative Procedure, "TMI Risk Management Program," Rev. 4. The inspectors reviewed the routine planned maintenance, restoration actions, and/or emergent work for the following equipment removed from service:

- On October 22, 2003, engineers evaluated the risk of performing 1D and 1E 4160 volt bus overcurrent relay trip testing while the station was in "Orange" shutdown risk since the reactor building equipment hatch was removed. The relay testing was deferred until the plant was at a lower risk condition.

- On November 1, 2003, the station entered the "Orange" shutdown risk category for approximately four hours to perform planned maintenance on the '1B' engineered safeguards motor control center and the 'S' 480v engineered safeguards electrical bus.

1R14 Personnel Performance During Non-routine Plant Evolutions (71111.14)

a. Inspection Scope

The inspectors reviewed human performance during the following non-routine plant evolutions to determine whether personnel performance caused unnecessary plant risk or challenges to reactor safety.

- On October 18, 2003, the inspectors observed main control room operators perform a plant cooldown in preparation for start of 1R15. The inspectors reviewed operating procedure 1102-11, "Plant Cooldown," Rev. 130, performed control room walkdowns, and interviewed plant operators. The inspectors also verified that operators properly identified minor discrepancies and entered them into the corrective action process.

.

- On October 21, 2003, the inspectors observed main control room operators perform an RCS cooldown and draindown to a mid-loop condition. The inspectors reviewed operating procedure 1103-11, "RCS Water Level Control," Rev. 60, the evolution plans, applicable contingency plans, observed crew briefings, and interviewed operators. The inspectors also verified that operators properly identified minor discrepancies which occurred during the draindown to mid-loop operation and entered them into the corrective action process (CRs 182296, 182711, 182858, 182957, 183072, 184936).

- At 1:50 a.m. on December 7, 2003, the 'A' main feedwater pump (FW-P-1A) to turbine coupling failed. The deformed coupling shaft caused sparks when it came in contact with the protective rotating equipment guard, which ignited oil that had collected at the base of the pump. Operators promptly responded by securing FW-P-1A, reducing power from 98 to 63 percent in accordance with OP-TM-MAP-M0101, "FWP 1A Trip," Rev. 0, and mobilizing the station fire brigade. The fire was extinguished within 10 minutes. Surveillance testing on emergency diesel generator EG-Y-1A was in progress at the time of the event. Operators secured EG-Y-1A and restored it to a standby status to minimize the number of ongoing activities which could distract from assessment and response to the partial loss of feedwater event.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

1. Routine Inspection

a. Inspection Scope

The inspectors reviewed operability evaluations for the following degraded equipment issues:

- Boric acid buildup on containment building electrical penetration 317E
- Three axial cracks identified during routine video camera inspection of the ‘B’ cold leg high pressure injection (HPI) makeup nozzle (MU/HPI) thermal sleeve

The following documents were reviewed and/or referenced for these inspections:

- AR A2074168 and WO-C2006594, dated November 10, 2003, issued to clean, inspect, and evaluate boric acid buildup on penetration 317E
- WO-C2001884, dated November 30, 2001, issued to clean, inspect, and evaluate prior boric acid buildup on penetration 317E
- UFSAR Section, 5.2.2.4.8.d, “Electrical Penetrations”
- UFSAR Section, 5.2.2.4, “Liner Plate and Penetrations”
- UFSAR Section, 5.2.2.5, “Corrosion Protection”
- Engineering Evaluation 5015728-03, “Engineering Determination for Continued Operation of TMI-1 With Observed Crack (s) in the MU/HPI TS,“ dated November 1, 2003
- AmerGen Transmittal of Inspection Results and NDE Report 5970-2003-024, dated October 31, 2003

The inspectors verified the degraded conditions were proper characterized, the operability of the affected systems was properly justified, and no unrecognized increase in plant risk resulted from the equipment issues. The inspectors performed several field walkdowns, interviewed plant engineers and technicians, reviewed applicable NDE inspections and video tape of the MU/HPI thermal sleeve inspection, and consulted with regional NRC specialists. The inspectors also referenced IMC Part 9900, “Operable/Operability-Ensuring the Functional Capability of a System Component” and AmerGen procedure LS-AA-105, “Operability Determination,” Rev. 1, to determine acceptability of AmerGen’s operability evaluations.

b. Findings

No findings of significance were identified.

2. Polar Crane Cracks

a. Inspection Scope

The inspectors reviewed one operability evaluation (OPE) to assess the technical adequacy of the evaluation, the use and control of compensatory measures, and compliance with the licensing and design basis. The inspectors’ review included a verification that the operability determination was made as specified by Exelon’s Procedure LS-AA-105. The technical content of the OPE was reviewed and compared to the technical specifications (TS), the UFSAR, and associated design and licensing basis documents. A listing of documents reviewed is included in the Attachment. The following evaluation was reviewed:

- OPE-03-031, “Polar Crane Cracks in Welds and Base Metal of Longitudinal Braces and Loose Bolted Connections in Lateral Bracing,” Rev. 1. The OPE removed conservative inputs and assumptions made during the design process and re-performed selected portions of the design basis calculation and its addendums to demonstrate that the polar crane would meet its design rating under safe shutdown earthquake conditions.

The inspectors verified that the licensee had initiated action tracking item 00181799-10 to generate the required configuration control documents (calculations, supporting engineering change request, and UFSAR revisions) if the polar crane braces were not restored to the original design configuration. The inspectors also verified that an Assignment Report (AR A2073913) was initiated and entered into the corrective action program to evaluate whether the polar crane would perform its design function without restoring the cracked braces to their original configuration.

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds (71111.16)

a. Inspection Scope

The inspectors reviewed the cumulative effects of the operator workarounds (OWAs) and two individual OWAs; A/R A2012675 and A/R A2034564. The workarounds were reviewed to identify any effect on emergency operating procedure (EOP) operator actions, and impact on possible initiating events and mitigating systems. The inspectors evaluated whether station personnel were identifying, assessing, and reviewing operator workarounds as specified in AmerGen administrative procedure OP-AA-102-103, “Operator Work-Around Program, “ Rev. 0.

Additionally, the inspectors the eight (8) outstanding operator challenges and six (6) items previously classified as operator burdens, a classification that was recently eliminated. The inspectors reviewed the status of planned and ongoing efforts to resolve these operator workarounds and challenges with the coordinator responsible for this program. The inspectors also reviewed the list of main control room distractions and toured the control room to evaluate the status and impact of these items, most of which involved inoperable chart recorders and radiation monitors. Items of particular concern were discussed with the responsible system engineers to ensure the items were being addressed on a schedule consistent with their relative safety importance. AmerGen’s main control room distraction reduction plan was also reviewed as were the results of AmerGen’s most recent quarterly assessment of operator workarounds and challenges.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17)

1. Replacement RVCH Design and Planning

a. Inspection Scope

Due to industry events involving Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 600 and prior repairs of the RVCH at TMI, AmerGen planned the replacement of the RVCH during the fall 2003 refueling outage. The design of the new RVCH is similar to the existing RVCH except for the replacement of the Alloy 600 nozzle and weld material with Alloy 690 material and other minor improvements. The inspectors reviewed design change packages listed below for technical adequacy and to verify that the design bases, licensing bases, and performance capability of the modified risk significant components were not degraded through the modifications. The inspectors reviewed the function of each changed component, the change description and scope, and the associated 10 CFR50.59 screening evaluations.

- Engineering Change Request (ECR) TM 02-01410, “Reactor Vessel Head Replacement,” Rev. 1
- ECR TM 02-01411, “Reinstall Service Structure on New Reactor Vessel Head,” Rev. 1

No major structural modification was performed for the reactor vessel (RV) head replacement activity and the licensee did not need any temporary modifications (TM) to the containment for access. Additional inspections of the RVCH replacement project were documented in Report Sections 1R02, 1R15, 4OA2, and 4OA5. Additional documents reviewed are listed in Attachment 1.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed and/or observed several post-maintenance tests (PMTs) to ensure: 1) the PMT was appropriate for the scope of the maintenance work completed; 2) the acceptance criteria were clear and demonstrated operability of the component; and 3) the PMT was performed in accordance with procedures. The following PMTs were observed:

- Pressurized visual leakage inspection, performed on December 3, 2003, in accordance with OP-TM-220-261, “Reactor Coolant System VT-2 Exam,” Rev. 1 - Interim Change 15231 following replacement of the leaking pressurizer heater bundle and diaphragm plate.

- Pressurized visual leakage inspection, performed on December 3, 2003, in accordance with OP-TM-220-261 following replacement of the reactor vessel head.

- Stroke testing of reactor building emergency cooling water valve RR-V-6, performed on November 17-18, 2003, in accordance with procedure 1303-11.9, “Reactor Building Emergency Cooling System,” Rev. 63 - Interim Change 15095 following repairs to actuator air supply valves IA-V-1622A, IA-V-1626A, and IA-PI-1010.

- “A” station battery load test performed on November 5, 2003, in accordance with procedure 1303-11.11, “Station Battery Load Test,” Rev. 30, following complete battery replacement. The inspectors also attended the pre-job brief, and interviewed electrical technicians and the system engineer.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

Station personnel conducted the TMI Unit 1 15th refueling outage (1R15) from October 18 to December 5, 2003. The inspectors reviewed selected reactor shutdown, refueling, outage maintenance, and reactor startup activities to determine whether shutdown safety functions (e.g., reactor DHR, reactivity control, electrical power availability, reactor coolant inventory, spent fuel cooling, and containment integrity) were properly maintained as required by technical specifications and AmerGen Topical Report 097, “TMI-1 Outage Fuel Protection Criteria,” Rev. 7. Specific attributes evaluated included configuration management, communications, instrumentation accuracy, and identification and resolution of problems. The inspectors closely evaluated configuration and inventory control during periods of reduced reactor coolant system inventory due to the associated increase in shutdown risk. Specific activities evaluated included:

- TMI-1 1R15 Outage Shutdown Safety Management Plan
- Operating procedure 1102-11, “Plant Cooldown,“ Rev. 130
- Reactor Coolant System (RCS) drain to mid-loop per procedure 1103-11, “RCS Water Level Control,” Rev. 58
- Fuel offload on October 27-28, 2003, per AmerGen refueling procedure 1505-1, “Fuel and Control Component Shuffles,” Rev. 42
- NRC Temporary Instruction (TI) 2515/152, “Reactor Pressure Vessel Lower Head Penetration Nozzles (NRC Bulletin 2003-02),” dated September 5, 2003
- NRC TI 2515/153, “Reactor Containment Sump Blockage (NRC Bulletin 2003-01),” dated October 3, 2003
- Operating procedure 1102-2, “Plant Startup,” Rev. 142
- Operating procedure 1102-4, “Power Operation,” Rev. 104

- Reactor system integrity and leak tightness per OP-TM-220-261, “Reactor Coolant System VT-2 Exam,” Rev. 1
- HPI Thermal Sleeve Inspections
- Expanded Scope Steam Generator Lower Tubesheet Inspections to Address Recent Industry Experience
- AmerGen procedure OP-AA-108-108, “Unit Restart Review,” Rev. 0, following completion of 1R15

The inspectors also performed visual inspections of the reactor building containment liner during 1R15 to ensure that the liner surface was free of defects that could affect either the structural integrity or leak tightness of the containment, and to assess the condition of the safety-related coatings inside containment. The inspectors reviewed controls of transient equipment and outage activities to protect the liner and the liner coating from damage. In addition, the inspectors reviewed engineering procedure EP-055T, “Monitoring and Tracking of Coatings in Containment,” Rev.1, performed during 1R15, interviewed the structural engineering supervisor, and reviewed the corrective actions for identified discrepancies. The following documents were also reviewed.

- CR 185821 and ECR TM 03-00921 which evaluated liner corrosion identified by the inspectors
- CR 183212 which evaluated several cases identified by the inspectors of metal components contacting the containment liner
- CR 189173 which evaluated a condition identified by the inspectors, regarding a large steel grating impacting the reactor building (RB) liner
- CR 187846 which evaluated a condition identified by the inspectors, regarding inadequate liner coating inspections
- Technical Report TR-536910-00014-01-SE, “Reactor Building Steel Liner,” dated October 21, 1999, which evaluated several areas of containment liner degradation which engineers identified during the primary Containment Section XI IWE Program Inspections
- Assignment Report (AR) A2075019, “1R15 RB Liner Coating Repairs”
- AmerGen NDE Containment Liner Ultrasonic Thickness Data Report 2003-041-002, dated November 14, 2003
- Containment Liner and Moisture Barrier Visual NDE Inspection Data Report 2003-041-001, dated November 14, 2003
- UFSAR Sections 5.2.1.2, 5.2.3, 5.2.2.4, and 5.2.2.5

b. Findings

RCS Pressure Boundary Leakage

Introduction. A self-revealing non-cited violation (NCV) of 10 CFR 50 Appendix B, Criterion XVI, “Corrective Action” was identified for failure to identify and correct RCS pressure boundary leakage in a timely manner. The issue was of very low safety significance (Green).

Description. On October 18, 2003, engineers performed a pressurized RCS walkdown following reactor shutdown for 1R15. Several soft piles of boric acid crystals were observed on a horizontal cable tray cover near one of three pressurizer heater bundle penetrations. Engineers determined the boric acid piles came from a leak on a nearby instrumentation line compression fitting (near RC-V-1044). The same fitting had been unsuccessfully repaired during the previous refueling outage (1R14). Further inspection of the affected area revealed large boric acid deposits extruding from a pressurizer heater bundle flange. Engineers subsequently determined the boric acid deposits were the result of RCS pressure boundary leakage from a pressurizer heater bundle diaphragm plate. Engineers classified this as leakage of an RCS strength boundary through a non-isolable fault. This condition was properly reported as required by 10 CFR 50.72, as a condition not permitted by technical specifications.

Engineers determined the cause of the cracks in the diaphragm was PWSCC. Additionally, boric acid corrosion caused by the diaphragm leak damaged the carbon steel diaphragm cover plate and minor surface etching of the surrounding bolts which provide structural support to the diaphragm. The most extensive cover plate damage was approximately 1.35 inches deep and 7 inches across. Engineers determined that the remaining cover plate and bolting material were sufficient to perform the structural support function. Initial repair efforts on the existing diaphragm failed the PMT during RCS pressurization. Station personnel subsequently installed a new pressurizer heater bundle, diaphragm plate, and cover plate which successfully passed the PMT.

Dry boric acid deposits, indication of RCS leakage, were identified on the heater bundle flange during 1R14. Engineers failed to make full use of industry operating experience and misdiagnosed the deposits as inactive leakage from a previous event. Corrective actions did not include removing the heater bundle cover plate to inspect the diaphragm seal weld to determine the actual leak path, nor any kind of test, such as Penetration Test (PT) or Ultrasonic Test (UT), to identify any flaw on the weld or heater penetrations. Chemical analysis of the dry boric acid crystals during 1R15 determined that the leak existed since about 1998. Failure to adequately evaluate and correct indications of RCS barrier degradation prior to 1R15 permitted continued RCS barrier degradation (e.g., pressurizer heater bundle diaphragm crack growth and boric acid corrosion of the cover plate and bolting). Engineers concluded that failure of initial repair efforts during 1R15 indicated that the diaphragm cracks were most likely worse than originally identified.

Analysis. Engineers failed to adequately evaluate and correct indications of RCS barrier degradation identified during the 1R14 outage. Consequentially, the reactor operated the entire 14th operating cycle with RCS pressure boundary leakage, contrary to TS 3.1.6.4. This finding affected the Barrier Integrity cornerstone. The issue is more than minor because it adversely affected the RCS equipment and barrier performance attribute in that it reduced the likelihood that the physical RCS design barrier would protect the public from radio nuclide releases. In addition, the issue, if left uncorrected, could become a more significant safety concern (i.e., RCS inventory loss). The inspectors processed this finding through Phase I and Phase II of the NRC IMC 609, “Significance Determination Process (SDP).” Phase I screening directs that a Phase II analysis be performed because the performance issue degraded the RCS barrier.

Assumptions used for this analysis included:

- The RCS barrier leakage existed for greater than 22 months and was unrecoverable by operator action.
- A catastrophic RCS barrier failure was not likely due to the slow growth rate associated with PWSCC and the robust design (cover plate thickness and bolting design) of the cover plate and bolting assembly.
- Further RCS barrier degradation was likely to reveal itself to plant operators through gradual increases in RCS leakage and reactor building airborne radioactivity.
- The reactor building containment and fuel cladding barriers were unaffected by this performance issue.
- Postulated RCS leakage from a catastrophic failure of the pressurizer heater bundle diaphragm would not adversely affect other mitigating systems.

The performance issue existed for greater than 22 months and increased the likelihood of a plant transient (e.g., TS required shutdown due to RCS pressure boundary leakage). Therefore, based on discussion with the NRC Region I Senior Risk Analyst, the inspectors raised the initiating event likelihood by one order of magnitude and analyzed the issue using Table 3.1 “TMI -Transients,” Rev. 1. The inspectors determined this issue was of very low safety significance because the RCS leakage was small, the likelihood of a rapid increase in RCS leak rate was small due to the robust cover plate design, the remaining mitigation functions in the Table 3.1 event sequences were unaffected, and the containment barrier remained fully functional to prevent radio nuclide release to the public.

Enforcement. 10 Code of Federal Regulations (CFR) 50 Appendix B, Criterion XVI further requires that (1) “Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected” and, (2) “In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.” Technical Specification 3.1.6.4 requires that the reactor be placed in cold shutdown within 24 hours of detecting any reactor coolant leakage through a non-isolable fault in an RCS strength boundary. Contrary to the above, after observing boric acid crystal buildup on the pressurizer heater bundle flange on October 11, 2001, engineers failed to identify and correct a significant condition adverse to quality, RCS pressure boundary leakage, until October 2003. Additionally, from December 6, 2001 until October 18, 2003, TMI Unit 1 operated at power with RCS strength boundary leakage through a non-isolable fault. Because the failure to adequately evaluate and correct the pressurizer heater diaphragm leak is of very low safety significance and has been entered into the corrective action program (CRs 181732 and 184753), this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000289/ 2003005-01, Failure to Evaluate and Correct Reactor Coolant System Pressure Boundary Leak in a Timely Manner.

Reactor Building Containment Liner Corrosion

Introduction. The inspectors identified that TMI failed to implement proper corrective actions to prevent corrosion of the safety-related containment liner. Specifically, TMI failed to address the consequence of leakage from non-safety related chemical addition valves (CA-V-1, 3, and 13) that was identified during the November 2001 refueling outage. This condition resulted in reduced liner wall thickness that exceeded the ASME XI IWE-3122.4 inspection acceptance criteria. This issue was assessed as having very low safety significance (Green) and was determined to be an NCV of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action."

Description. The inspectors identified corrosion and damage of the moisture barrier in several areas of the cylindrical portion of the containment liner at the 281'-0" elevation. The affected area was at the concrete to moisture barrier interface and measured approximately 2 to 4 inches wide and 20 feet long. In addition, the protective liner coating was chipped and there was evidence of boric acid crystals and moisture buildup on the exposed carbon steel liner. The inspectors also noted large areas of discolored water stains and boric acid buildup running down the liner from the floor above. The TMI reactor building, including its liner, is classified as a "Seismic I" structure in accordance with UFSAR Section 5.1.1.1a. The carbon steel liner ensures a high degree of leak tightness (vapor barrier) during operating and accident conditions. The nominal liner plate thickness is .375 inches for the cylindrical section. Per ASME XI, Subsection IWE-3122.4, a 10 percent liner thickness reduction is acceptable without further evaluation.

Ultrasonic testing of the degraded containment liner (CR 185821 and ECR 03-00921) determined that four liner sections had reduced liner wall thickness (ranging from .308" to .316") that exceeded the 10 percent wall reduction (to .338" thickness) prescribed by ASME XI acceptance criteria. Engineers determined that the reduced liner wall thickness did not impair the structural integrity of the liner or containment. Corrective actions included a 360° NDE inspection (ultrasonic testing) of the containment liner (limited to a four foot wide band), cleanup of the corrosion layer, re-application of the protective coating, and replacement of the damaged moisture barrier. Engineers also determined that the liner corrosion was caused by moisture and leakage from components inside containment.

The inspectors reviewed work order C2001884, dated November 3, 2001, which documented leaking non-safety related chemical addition valves (CA-V-1, 3, and 13) directly above the affected area. The actions specified in the work order addressed concerns with boric acid and water buildup inside an electrical penetration (Penetration 317), but did not consider the effects of the leakage on the safety-related containment liner and protective moisture barrier. Therefore, the corrective actions previously taken to address the effects of boric acid leakage on top of components were ineffective in preventing the containment liner corrosion and damage to the moisture barrier.

Analysis. Failure to consider the effects of borated water leakage on the safety-related containment liner and protective moisture barrier constitutes a performance deficiency. Corrective actions taken to address a leaking valve were narrowly focused in that they failed to consider potential degradation to safety-related components in the vicinity of the leak.

This issue affected the barrier integrity cornerstone and was considered more than minor because the condition impacted configuration control in that containment barrier wall thickness design parameters were not maintained. In addition, if left uncorrected, the condition could have affected the availability/reliability of the safety-related containment liner to protect the public from radio nuclide release. Using NRC IMC 0609, “Significance Determination Process,” Appendix A, Phase 1, this finding was determined to be of very low significance since the issue did not involve an actual open pathway in the physical integrity of the containment.

Enforcement. 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” requires that measures be established to ensure that conditions adverse to quality are promptly identified and corrected. Contrary to this requirement, station personnel failed to consider the effects of borated water leakage and did not implement adequate corrective actions to prevent degradation of the containment liner. Because this violation was of very low safety significance and TMI entered this issue into its corrective action program (CR 185821), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000289/2003005-02, Failure to Identify and Correct Boric Acid Corrosion of Reactor Building Containment Liner and Protective Moisture Barrier.

Permanently Installed Structure In Contact With Reactor Building Containment Liner

Introduction. The inspectors identified that station personnel failed to ensure that a structural component located inside containment was properly installed per structural drawing 421054 to prevent damage to the safety-related containment liner during a postulated seismic event. This issue was determined to be of very low safety significance (Green) and an NCV of 10 CFR 50 Appendix B, Criterion X “Inspection.”

Description. The inspectors identified that a floor grating for a permanently mounted structure (platform used for storage of the in-core cables during outages) came in direct contact with the containment liner, creating a potential concern for liner damage during a postulated seismic event (SSE). The metal floor grating which contacted the liner was able to be moved approximately 0.25 inches away from the liner. Engineers evaluated this condition under CR 189173 and determined that the design drawing (421054) indicates a structural clearance requirement of 1 inch. The evaluation also determined that the maximum relative seismic displacement between the containment building shell and the interior structure at this elevation was 0.316 inches. Therefore, contact with the liner during an SSE would occur. However, the evaluation concluded that the liner damage potential was low due to the limited energy that would have resulted in the impact during a seismic event, and the relative robust liner (3/8 inch thick carbon steel plate) and the thickness of the concrete behind it.

The inspectors also observed over 10 examples where outage related transient metal components came in direct contact with the containment liner. In some cases contact between the containment liner and the components resulted in minor damage (scratches and gouges) to the liner. The identified components included: several scaffold poles, two heavily loaded four-wheeled carts, and several large pieces of metallic insulation from the reactor vessel head. Engineers evaluated these issues under CR 183212 and determined that no significant damage to the containment liner occurred. The evaluation also determined that there were no specified criteria for a standoff distance between the liner and transient materials. A corrective action to prevent damage to the containment liner from transient materials was initiated to provide guidance to include a spacing clearance of at least 1 inch from the containment liner.

Analysis. Failure to ensure that a floor grating for a specified structural component located inside containment was properly installed per the applicable structural drawing 421054 constitutes a performance deficiency.

This issue affected the barrier integrity cornerstone and was considered more than minor because the condition impacted configuration control in that the containment design parameter for clearance between structures and the containment liner was not maintained. In addition, if left uncorrected, the condition could have affected the availability/reliability of the safety-related containment liner to protect the public from radio nuclide release. Using NRC Manual Chapter 0609, “Significance Determination Process,” Appendix A, Phase 1, this finding was determined to be of very low significance since the issue did not involve an actual open pathway in the physical integrity of the containment. Using NRC Manual Chapter 0609, “Significance Determination Process,” Appendix A, Phase 1, this finding was determined to be of very low significance since the issue did not involve an actual open pathway in the physical integrity of the containment.

Enforcement. 10 CFR 50, Appendix B, Criterion X, “Inspection,” requires, in part, that a program for the inspection of activities affecting quality be established and executed to verify conformance with documented instructions, procedures, and drawings. Contrary to this requirement, station personnel failed to ensure that a structural component located inside containment was properly installed per design drawing 421054. Because this violation was of very low safety significance and TMI entered this issue into its corrective action program (CR 189173), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy:
NCV 05000289/2003005-03, Failure to Maintain Structural Design Clearances Inside Reactor Building Containment.

Coating Inspection Inside Reactor Building Containment

Introduction. The inspectors identified a non-cited violation (NCV) for engineering failure to properly implement inspections to assess the overall health of coatings inside the containment as required by procedure (EP-055T). This condition resulted in a complete reinspection and evaluation of 127 coating indications. This issue was assessed as having very low safety significance (Green).

Description. The inspectors reviewed the 1R15 containment coatings inspection performed per engineering procedure EP-055T, “Monitoring and Tracking of Coatings in Containment,” Rev. 1. The engineering inspection was required by CR 171425 to assess the overall health of the coating inside containment, because scheduled coating maintenance was deferred from 1R15 to the next refueling outage (1R16). Periodic coating repairs/replacement ensure that the amount of service level I coating which may be susceptible to detachment during a loss of coolant accident (LOCA) is minimized. The initial inspection was completed on November 20, 2003, by the qualified coating specialist and identified little to no discrepancies. The inspectors performed an independent assessment of the coating inside containment and identified many discrepancies at all elevations inside containment that were not identified or evaluated by the engineering inspection. These discrepancies required evaluation with regards to potential liner degradation or reactor building sump performance per procedure EP-055T guidelines. The discrepancies included surface rust, buildup of boric acid crystals, liner scratches or gauges, and blistering, peeling and chipping of the coating.

Engineers evaluated this condition under CR 187846 and performed a complete re-inspection of the coatings inside containment. A total of 127 discrepancies were identified. The evaluation also determined that the engineer performed the initial inspection without the procedure, and that the inspection was ineffective due to poor human performance. This condition was documented under CR 187352. The evaluation determined that the indications and discrepancies did not challenge the operability of the containment sump or liner. The inspectors concluded that the corrective actions to address this issue were satisfactory.

Analysis. Failure to perform the required coating inspection per procedure EP-055T is considered a performance deficiency. This finding affected the barrier integrity cornerstone and was considered more than minor because if left uncorrected it would have degraded further and adversely impacted availability/reliability of the safety-related containment sump and liner.

Using Appendix A, Phase 1 of IMC 0609, this finding was determined to be of very low significance since the issue did not involve an actual open pathway in the physical integrity of the containment or an actual blockage of the containment sump. The inspectors determined that the subsequent extent of condition review, evaluations, and corrective actions were adequate to address the condition of the containment building coating.

Enforcement. Technical specification 6.8.1.a requires in part that written procedures shall be established, implemented and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Rev. 2, February 1978. Regulatory Guide 1.33, Appendix A, recommends procedures for safe operation and shutdown of safety-related systems. Contrary to this requirement, TMI engineers failed to properly perform inspections to assess the overall health of coatings inside containment as required by procedure EP-055T. Because this violation was of very low safety significance and TMI entered this issue into its corrective action program (CR 187846), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000289/2003005-04, Failure to Properly Perform Reactor Building Containment Coating Inspections.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors observed and reviewed the following operational surveillance tests, concentrating on verification of the adequacy of the test to demonstrate the operability of the required system or component safety function.

- OP-TM-212-218, “DH-V-6B to Reactor Building Sump Leak Check and VT-2,” Rev. 2 - Interim Change 14943
- Procedure 1303-11.9, “Reactor Building Emergency Cooling System,” Rev. 63 - Interim Change 15095
- Procedure 6610-OPS-4550.03, “Reactor Building Floor Drain Housekeeping,” Rev. 1
- Surveillance procedure 1301-5.8, “Station Battery Quarterly,” Rev. 28, completed on November 4, 2003. The inspectors also interviewed electrical technicians and the system engineer.
- Surveillance procedure 1303-11.3, “Main Steam Safety Valves,” Rev. 30, completed on October 15, 2003

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed temporary modifications (TMs) and associated implementing documents to verify the plant design basis and the system or component operability was maintained. Nuclear Power Division Administrative Procedure (NPDAP) 7.4, “Temporary Modifications,” Rev. 8, specified requirements for development and installation of TMs. The inspectors reviewed the following TM.

- TM 03-00620, Rev. 0, which installed temporary equipment to record data during stroke testing of air operated reactor river valve RR-V6. This TM was issued to support troubleshooting and investigation of slower valve stroke times identified during IST testing.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness (EP)

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

a. Inspection Scope

A regional in-office review was conducted of licensee-submitted revisions to the emergency plan, implementing procedures, and EALs which were received by the NRC during the period of April - December 2003. The review included plan aspects related to the risk significant planning standards (RSPS), such as classifications, notifications, and protective action recommendations. A cursory review was conducted for non-RSPS portions. These changes were reviewed against 10 CFR 50.47(b) and the requirements of Appendix E and they are subject to future inspections to ensure that the combination of these changes continues to meet NRC regulations. In addition, in January 2003, the licensee generated a consolidated Emergency Plan for all Exelon sites (Peach Bottom, Limerick, TMI) and an Annex Plan related specifically to TMI. The 10 CFR 50.54(q) reviews associated with the specific changes/deletions made from the original Plan to the current Plans will be reviewed and assessed during the next EP program inspection to ensure that Exelon did not decrease the effectiveness of the original Plan during the transition. The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 4, and the applicable requirements in 10 CFR 50.54(q) were used as reference criteria.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed and emergency event training evolution conducted at the Unit 1 control room simulator to evaluate emergency procedure implementation, event classification, event notification, and protective action recommendation development. The inspectors also observed emergency response organization activities at the Technical Support Center and Operations Support Center. The event scenario involved multiple safety-related component failures and plant conditions warranting simulated Alert, Site Area Emergency, ane General Emergency event declarations. The licensee counted this training evolution for evaluation of Emergency Preparedness Drill/Exercise Performance (DEP) Indicators. The inspectors observed the drill critique to determine whether the licensee critically evaluated drill performance to identify deficiencies and weaknesses. Additionally, the inspectors verified the DEP performance indicators (PIs) were properly evaluated consistent with Nuclear Energy Institute (NEI) 99-02, “Regulatory Assessment Performance Indicator Guideline,” Rev. 2. Additional documents used for this inspection activity included:

- Procedure 1202-33, “Tornado/High Winds,” Rev. 25
- Procedure 1202-29, “Pressurizer System Failure,” Rev. 60
- Procedure 1202-8, “Control Rod Drive Equipment Failure,” Rev. 54
- OP-TM-EOP-001, “Reactor Trip,” Rev. 4
- OP-TM-AOP-020, “Loss of Station Power,” Rev. 1
- OP-TM-EOP-005, “Once Through Steam Generator Tube Leakage,“ Rev. 1
- EP-AA-1009, “TMI Unit 1 Emergency Action Level Matrix,” Rev. 2

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control (71121.01)

a. Inspection Scope

During the period of October 27-30 and November 10-12, 2003, the inspectors reviewed exposure significant work areas (i.e., High Radiation Areas and Airborne Radioactivity Areas) in the plant and associated controls and surveys of these areas to determine if the controls (e.g., surveys, postings, barricades) were acceptable. For these areas, the inspectors reviewed radiological job requirements and attended job briefings to determine if radiological conditions in the work area were adequately communicated to workers through briefings and postings. The inspectors also verified radiological controls, radiological job coverage, and contamination controls to ensure the accuracy of surveys and applicable posting and barricade requirements. The inspectors determined if prescribed radiation work permits (RWPs), procedure and engineering controls were in place; whether surveys and postings were complete and accurate; and if air samplers were properly located. The inspectors conducted reviews of RWPs used to access exposure significant work areas to identify the acceptability of work control instructions or control barriers specified. The inspectors reviewed electronic pocket dosimeter alarm set points (both integrated dose and dose rate) for conformity with survey indications and plant policy. The inspectors reviewed dosimetry placement, discussed High Radiation Area controls, and reviewed controls for materials removed from the flooded reactor cavity. The controls implemented were compared to those required under plant TS 6.12 and 10 CFR 20, Subpart G, for control of access to high and locked high radiation areas.

The primary focus of this inspection was the unit refueling outage. Outage activities in exposure significant areas observed included: eddy current testing in both steam generators; defueling; reactor coolant pump work; inservice inspection; weld repairs related to the pressurizer; and reactor reassembly.

This inspection activity represents the completion of 12 samples relative to this inspection area.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspectors reviewed ALARA job evaluations, exposure estimates, and exposure mitigation requirements and compared ALARA plans with the results achieved. A review was conducted of: the integration of ALARA requirements into work procedures and RWP documents; the accuracy of person-hour estimates and person-hour tracking; and generated shielding requests and their effectiveness in dose rate reduction.

A review of actual exposure results versus initial exposure estimates for current work was conducted including: comparison of estimated and actual dose rates and person-hours expended; determination of the accuracy of estimations to actual results; and determination of the level of exposure tracking detail, exposure report timeliness, and exposure report distribution to support control of collective exposures to determine conformance with the requirements contained in 10 CFR 20.1101(b).

The exposure goal for 1R15 was established at 123 person-rem with a stretch goal of 115 person-rem. Major work activities and their dose goals included: steam generator testing (19 person-rem); reactor coolant pump maintenance (2.4 person-rem); reactor disassembly/reassembly (15.15 person-rem); routine inservice inspection (6.709 person-rem); temporary shielding (1.498 person-rem); radwaste (3.5 person-rem); and, scaffolding (13.14 person-rem). Through the first three weeks of the outage, site-wide outage exposures were tracking above estimates by approximately 2 person-rem. Emergent work involving the pressurizer (weld repairs on the surge line and heater seal) had added approximately 6.5 person-rem to the outage exposure; expanded eddy current testing in the steam generators added 1.3 person-rem; and, repairs to the reactor building sump were expected to add 1.96 person-rem.

This inspection activity represents the completion of five samples relative to this inspection area.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation (71121.03)

a. Inspection Scope

The inspectors reviewed field radiological controls instrumentation used by radiation protection (RP) technicians and plant workers to measure radioactivity, including portable field survey instruments, friskers, and portal monitors. The inspectors conducted a review of selected radiation protection instruments observed in the radiologically controlled area (RCA). Items reviewed were verification of proper function, certification of appropriate source checks, and calibration for these instruments used to ensure that occupational exposures are maintained in accordance with 10 CFR 20.1201.

The inspectors reviewed portions of the internal exposure monitoring program, including the most recent annual system calibration of the whole body counter and daily whole body counter performance tests. The inspectors evaluated actions taken by the licensee if survey instruments are determined to be greater than 50 percent out of specification when brought in for calibration. The inspectors also evaluated the licensee’s ability to fill and transport breathing air bottles to personnel during emergency conditions.

This inspection activity represents the completion of two samples relative to this inspection area.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

1. Occupational Radiation Safety Cornerstone

a. Inspection Scope

The inspectors reviewed a listing of licensee CRs for the period January 1, 2003 through November 8, 2003 for occurrences involving High Radiation Areas, Very High Radiation Areas, and unplanned personnel exposures against the applicable criteria specified in NEI 99-02, “Regulatory Assessment Performance Indicator Guideline,” Rev. 2, to verify that conditions meeting the NEI criteria were recognized and identified as PI occurrences, as appropriate.

b. Findings

No significant findings or observations were identified.

2. Public Radiation Safety Cornerstone

a. Inspection Scope

The inspectors reviewed a listing of CRs for the period January 1, 2003 through December 1, 2003 for radiological effluent technical specifications (RETS) and offsite dose calculation manual (ODCM) occurrences. The review was performed to determine if TMI experienced any radiological effluent release occurrences meeting the PI dose criteria specified in NEI 99-02 for the previous four quarters.

b. Findings

No significant findings or observations were identified.

3. Physical Protection Cornerstone

a. Inspection Scope

The inspectors performed a review of PI data submitted by the licensee for the physical protection cornerstone. The review was conducted of the licensee's programs for gathering, processing, evaluating, and submitting data for the Fitness-for-Duty, Personnel Screening, and Protected Area Security equipment PIs to verify these PIs had been properly reported as specified in NEI 99-02.

The review included the licensee's tracking and trending reports, personnel interviews, and security event reports for the PI data collected from the 4th quarter of 2002 through the 3rd quarter of 2003. The inspectors noted from the licensee's submittal that there were no reportable failures to properly implement the requirements of 10 CFR 73 and 10 CFR 26 during the entire reporting period. Based on the data reviewed and interviews with personnel, the inspectors concluded that the personnel screening and the fitness-for-duty programs functioned as intended. This inspection activity represents the completion of three samples relative to this inspection area; completing the annual inspection requirement.

b. Findings

No findings of significance were identified.

4. Safety Systems Functional Failures

a. Inspection Scope

The inspectors reviewed the PI assessment for safety system functional failures to determine whether the NRC approved guidance, provided in NEI 99-02, was properly implemented. Verification included review of the data collected, definitions, data reporting elements, calculation methods, definition of terms, and use of clarifying notes. The inspectors verified accuracy of the reported data, through reviews of Licensee Event Reports submitted during the period October 2002 through September 2003.

b. Findings

No findings of significance were identified.

5. Residual Heat Removal Safety System Unavailability

a. Inspection Scope

The decay heat system at TMI Unit 1 provides both (1) the post-accident DHR and low pressure injection functions and (2) the normal shutdown cooling function for long term heat removal following plant shutdown. The inspectors reviewed operating logs, maintenance rule records, and selected surveillance procedures to verify whether the data was reported accurately, in accordance with NEI 99-02, during the period July 2002 through September 2003. In addition, the following procedures and documents were reviewed to evaluate determination of availability:

- OP-TM-212-201, “IST of DH-P-1A and Valves From ES Standby Mode,” Rev. 4
- OP-TM-212-202, “IST of DH-P-1B and Valves From ES Standby Mode,” Rev. 4
- OP-TM-533-471, “Backwashing DC-C-2A,” Rev. 2
- OP-TM-533-472, “Backwashing DC-C-2B,” Rev. 2
- 1104-65, “River and Circulating Water System Macrofouling Treatment,” Rev. 2
- TMI IST Test Program for 3rd Ten-Year Interval, dated September 21, 1995

b. Findings

No findings of significance were identified.

6. Emergency Preparedness

a. Inspection Scope

The inspectors reviewed the licensee’s process for identifying the data that is utilized to determine the values for the three EP performance indicators (PI) which are (1) Drill and Exercise Performance, (2) Emergency Response Organization (ERO) Participation, and (3) Alert Notification System (ANS) Reliability. The review assessed data submitted to the NRC for the fourth quarter of 2002 (since the last EP PI verification inspection) up to, and including, the third quarter of 2003. Classification, notification, and protective action opportunities were reviewed from licensed operator simulator sessions and site ERO drills and exercises. Attendance records for drill and exercise participation was reviewed for verification purposes. Test results of the ANS testing were reviewed for accuracy and completeness. The inspectors reviewed this data using the criteria of NEI 99-02.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

1. Routine Inspection Activities

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems,” and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee’s corrective action program. This review was accomplished by reviewing hard copies of each condition report, attending daily screening meetings, and accessing the licensee’s computerized database.

b. Findings

Section 1R05.1 describes a performance deficiency in which the fire brigade was unable to unlock a secured fire door, as credited in the station FHAR, during a fire drill training evolution. Identification and corrective action for this deficiency were ineffective and the problem reoccurred the following month.

Section 1R05.2 describes that the inspectors found several vital fire doors unlatched. Station personnel had not initiated a condition report to investigate and correct this repetitive problem until informed by the inspectors.

Section 1R20 describes a finding involving failure to properly identify and resolve pressurizer heater bundle leak indications during the last refueling outage (1R14). The failure to properly disposition these indications was a violation of 10 CFR 50 Appendix B, Criterion XVI.

Section 1R20 describes a finding involving failure to fully evaluate the effects of borated water leakage and implement adequate corrective actions to prevent degradation of the containment liner. This issue was a violation of 10 CFR 50 Appendix B, Criterion XVI.

Section 1R20 describes a finding involving failure to maintain design clearances between the containment liner and structural components within the reactor building. This issue along with numerous observations of transient material in contact with the containment liner indicated deficient problem identification on the part of station personnel.

Section 1R20 describes a finding involving failure to properly inspect containment coatings. This issue was a violation of TS 6.8.1 and demonstrated deficient human performance and problem identification. The applicable station procedure was not used and numerous existing degraded containment coating conditions were not identified.

Section 4OA5.1 describes a finding involving failure to identify and evaluate degraded conditions which had the potential to affect ECCS containment sump availability. This issue was a violation of 10 CFR 50 Appendix B, Criterion XVI.

2. Reactor Vessel Closure Head Manufacture and Replacement

a. Inspection Scope

The inspectors reviewed a selected sample of Framatome Advanced Nuclear Power (FANP) condition reports (CRs), non-conformance reports (NCR) and contract variation approval requests (CVARs) to ensure that FANP and their subcontractors appropriately identified, evaluated, and initiated actions to correct problems associated with the manufacture and replacement of the replacement RVCH. The inspectors also reviewed a selected sample of conditions adverse to quality documented in AmerGen CRs to verify that corrective actions were identified and to verify completion of a selected sample of corrective actions.

The inspectors reviewed NRC Information Notice (IN) 2003-20, “Derating Whiting Cranes Purchased Before 1980", and discussed the IN with station personnel and noted that it was applicable to TMI. The issue was evaluated and corrective actions were specified in assignment report (AR) 00142012. AmerGen personnel stated that IN 2003-20 was applicable to the auxiliary hook on the Reactor Building Polar Crane which had been derated until repairs could be performed. The inspectors verified that the IN was not applicable to the polar crane main hook that is used to perform heavy lifts such as the RVCH.

A complete listing of documents reviewed is included in the attachment. Additional inspections of the RVCH replacement project were documented in Report Sections 1R02, 1R15, 1R17 and 4OA5.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

1. NRC Temporary Instruction (TI) 2515/153, Reactor Containment Sump Blockage (NRC Bulletin 2003-01)

a. Inspection Scope

NRC Bulletin 2003-01, “Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors,” dated June 9, 2003, requested licensees to either (1) perform a plant-specific evaluation to confirm compliance with 10 CFR 50.46(b)(5) or (2) implement interim compensatory measures to reduce the potential risk due to post-accident containment sump debris blockage pending completion of the plant specific evaluation. AmerGen chose the second option and described their interim compensatory measures in their response to NRC Bulletin 2003-01. The inspectors interviewed station personnel, reviewed records, and inspected areas within the reactor building containment to determine whether the station personnel effectively implemented reasonable compensatory measures as committed to in their bulletin response dated August 6, 2003.

b. Findings

Documentation of NRC Temporary Instruction (TI) 2515/153 - Bulletin 2003-01 Line Item Reporting Requirements on ECCS Sump Performance

a. Not applicable.
b. TMI Unit 1 completed a refueling outage during this inspection period. A containment walkdown to quantify the potential debris sources was not conducted. However, station personnel conducted an as-left containment walkdown immediately prior to unit restart to remove or otherwise secure potential sources of debris.

c. Not applicable.

d. The inspectors and station personnel performed a detailed walkdown of the containment sump in accordance with procedure OP-TM-212-218, “DH-V-6B to Reactor Building Sump Leak Check and VT-2,” Rev. 2 - Interim Change 14943. The containment sump and mesh screen configuration was verified to meet the configuration specified in drawing 1D-572,36-1000, “Reactor Building Sump - Sump Screen Cage Assembly,” Rev. 0. Sump mesh screen wire spacing (1/8”) was verified and inspection for gaps in the sump's screened flowpath was completed. Approximately 15 square inches of bypass area between the mesh screen and the sump walls, minor areas of wire deformation, and approximately a one-inch deep layer of muck at the bottom of the sump was identified. Corrective maintenance was promptly performed to restore design mesh screen wire spacing, eliminate the bypass area, and remove all dirt/debris from the sump.

e. No.

f. Status of specific interim actions stated in AmerGen response to NRC Bulletin 2003-01 follows:
   - Operator training on indications and response to sump clogging was completed.
   - Licensed Operator Requalification Training on enhanced emergency core cooling system (ECCS) throttling criteria was completed.
   - Procedure OP-AA-108-108-1001, “Drywell/Containment Closeout,” Rev. 0 was enhanced to focus on loose debris that could affect the ECCS sump. While the procedure does address loose debris, it does not specifically address dirt and dust as stated in the licensee response to NRC Bulletin 2003-01.
   - The reactor building containment was designated as a foreign material exclusion area upon restart from 1R15.
   - Procedure 6610-OPS-4550.03, “Reactor Building Floor Drain Housekeeping,” Rev. 1 was established and performed to clean and inspect the floor drains in containment. The procedure also verified the flow path from the floor drains to the reactor building containment sump was not blocked. However, the inspectors observed a layer of dirt & small debris, approximately 1” deep, remained present throughout the network of 4” piping connecting the floor drains after completion of the procedure. The inspectors concluded the sump remained operable.

g. The reactor building containment sump is the source of water for several accident mitigation systems (decay heat (DH), building spray (BS), and HPI via piggy-back operation mode). The TMI Unit 1 sump design is somewhat unique in that the ECCS sump also serves as the single collection point for all dirt and debris collected by the containment floor drain system.

Introduction. The inspectors identified an NCV for failure to identify, document, and assess conditions adverse to quality which had the potential to adversely affect ECCS containment sump availability. This issue was of very low safety significance (Green) because subsequent evaluations concluded the ECCS containment sump remained operable and capable of performing its safety function.

<u>Description</u>. On October 21, 2003, the inspectors toured containment to evaluate the as-found sump condition and look for additional debris sources within containment which could potentially block the flow of water through the sump. Station personnel had already placed a protective tarp over the sump grating to prevent additional debris from falling into the sump during outage maintenance activities. The inspectors looked beneath the tarp and observed that approximately a quarter of the visible sump mesh screen area was blocked with a scum-like coating that appeared to be a combination of dirt, oil residue, and biological growth. Additionally, small amounts of paper towels, tape, and labels were observed at the surface of the water (approximately 2 feet deep) in the sump. The inspectors also identified various debris (i.e., paper, cloth rags, insulation, tape, nails, bolts, nuts, paint chips, containment liner coating material, etc.) in cable trays, reactor building floor drains, and other containment areas which could potentially be transported to the containment sump and degrade its performance. The inspectors noted that although station personnel had seen most of these same conditions, the conditions were not documented, and no as-found assessment of containment sump operability was performed.

On October 28, radwaste personnel cleaned the sump screen to reduce area radiation levels and began removing debris from the containment floor drain boxes. The inspectors accompanied station personnel during sump cleaning, inspection, and repair activities from October 30 to November 20. The cleaning process uses a portable pump to transfer the sump water, silt, and debris to the auxiliary building sump without quantification. When pumping was complete, approximately one inch of silt/muck remained at the bottom of the containment sump. Despite failure to quantify the material removed from the sump, the inspectors expressed concern that the cumulative debris (scum layer on screen, debris floating in sump, loose debris in containment, debris in floor drains, debris pumped from the containment sump to the auxiliary building sump, one inch layer of sump muck, 15 square inches of screen gap, and minor debris found within the containment sump mesh screen) had not been evaluated and could potentially affect the operability of DH, BS, and HPI systems. Engineers subsequently evaluated the conditions discussed above and concluded the containment sump had remained operable during the previous operating cycle.

<u>Analysis</u>. Station personnel failed to identify degraded containment sump conditions. Failure to recognize and evaluate screen blockage and sources of continued debris within containment could lead to further containment sump degradation which could make ECCS systems inoperable. This performance deficiency affected the Mitigating Systems cornerstone and was more than minor because it had the potential to adversely impact equipment availability and reliability for multiple ECCS systems which are designed to respond to initiating events to prevent undesirable consequences (i.e., core damage). The finding was of very low safety significance (Green) because it did not represent an actual loss of safety function.

<u>Enforcement</u>. 10 CFR 50, Appendix B, Criterion XVI “Corrective Action” requires in part that measures be established to assure that conditions adverse to quality are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective

action taken to preclude repetition. The condition, the cause, and the corrective action taken shall be documented and reported to appropriate levels of management. Contrary to the above, station personnel failed to establish measures to identify, document, and evaluate as-found debris within the reactor building containment and containment sump which had potential to adversely affect operability of the DH, BS, and HPI ECCS systems until the issue was raised by the inspectors. Because this issue was of very low safety significance and has been entered into the corrective action program (CRs 184313, 183711, and 189051), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000289/2003005-05, Failure to Identify, Document, and Evaluate Conditions Adverse to Quality Which Had the Potential to Adversely Affect ECCS Containment Sump Availability.

2. TI 2515/152 - Reactor Pressure Vessel (RPV) Lower Head Penetration (LHP) Nozzles (NRC Bulletin 2003-02)

a. Inspection Scope

The inspectors reviewed the licensee's response to NRC Bulletin 2003-02 which described the RPV lower head penetration inspection program. The inspectors reviewed the licensee's LHP nozzle examination procedure to determine whether it provided adequate guidance and examination criteria to implement the licensee's examination plan. The inspectors interviewed examination personnel, reviewed training and qualification records to verify the licensee personnel qualification process adequately prepared the assigned staff to perform the examination, and to disposition the deficiencies identified.

The inspectors observed the licensee's inspection activities to verify proper performance of the procedure. The inspectors also reviewed photographs and examination reports to verify that the procedure implementation was effective for detection of leakage from the RPV and (LHP) nozzles and/or corrosion of the lower head.

The inspectors selected eight penetration nozzles to evaluate the effectiveness of the VT examination to verify that the penetration intersection location could be fully accessed to reliably perform a 360 degree examination of the intersection region. The inspectors verified by direct observation and review of photographs that the RPV lower head was free of dirt, debris, insulation, significant oxidation, and any material that could adversely affect viewing of all penetrations (360 degrees around the circumference of the nozzles) and the vessel head in its entirety. The inspectors observed boron deposits on the inside surfaces of the mirror insulation vertical panels and reviewed the licensee's evaluation that the origin of the deposits was from a location above the lower head (cavity seal plate), which had since been corrected.

The inspectors verified that the procedure used for the inspection provided adequate guidance for the recording, evaluation, and documentation of the disposition of discrepancies identified during the examination.

b. <u>Findings</u>

No findings of significance were identified.

The specific reporting requirements of TI 2515/152 are documented in the attachment.

3. <u>Reactor Vessel Closure Head Replacement Project</u> (71007)

a. <u>Inspection Scope</u>

Recent industry events involving PWSCC of Alloy 600 at other plants, along with prior repairs at TMI, prompted AmerGen to replace the TMI-1 RVCH during their Fall 2003 refueling outage. The design of the new RVCH is similar to the existing RVCH except for the replacement of the Alloy 600 nozzle material and Alloy 600 weld material with a new and improved PWSCC resistant material (Alloy 690) and other minor improvements. The new RVCH was made as a single forging and clad with stainless steel on the inside, was machined and fabricated with welded control rod drive mechanism (CRDM) and thermocouple (TC) nozzles, and was hydrostatically pressure tested at the FANP facilities in France prior to being shipped to TMI for installation. A new control rod drive service structure (CRDSS) support skirt and insulation package was also purchased. AmerGen determined that the CRDMs would be transferred from the existing RVCH to the replacement RVCH and reused.

From September 25 to November 13, 2003, the inspectors reviewed the TMI RVCH Replacement Project using the guidance in NRC Inspection Procedure 71007, “Reactor Vessel Head Replacement Inspection.” Additional inspections of the RVCH replacement project were documented in Report Sections 1R02, 1R15, 1R17, 4OA2 and 4OA5.

A list of documents reviewed was included in the attachment.

<u>Design and Planning Inspections</u>

From September 29 to October 17, 2003, the inspectors conducted in-office and onsite reviews of the design change packages which described the replacement RVCH and the new control rod drive service structure (CRDSS) support skirt. The inspectors also reviewed design calculations and analyses for sizing of the new RVCH, validation of closure stud tensioning for the replacement RVCH, reactor vessel closure analysis for the replacement RVCH, and CRDM and TC nozzle analysis.

The inspectors inspected the planning and preparation activities for RVCH replacement. The inspection included review of the design evaluation (ECR TM 03-00202) that included plans for moving the replacement RVCH from temporary storage in the lay-down area to the head stand in containment. The inspectors reviewed analyses, design calculations, and evaluations for RVCH drop, the RVCH lay-down areas and the safe load path for replacement and existing RVCH movement and storage in containment prior to the replacement head installation on the reactor and existing head removal from containment. ECR TM 03-00202 evaluated the planned loads and acceptability of simultaneously having two reactor heads on the reactor building floor. The inspectors also reviewed the analysis of the potential impact of load handling activities on the reactor core, spent fuel cooling, and other plant support systems and the consequence of any impact loading of structures, systems, and components due to an RVCH drop accident. The 10 CFR 50.59 screening evaluations were included in the applicable ECRs.

Reactor Vessel Head Fabrication Inspections

The replacement RVCH was manufactured by FANP in France to the 1989 Edition of ASME Section III Code. The inspectors performed an onsite review of the AmerGen receipt inspection of the replacement RVCH to verify that the receipt activities conformed to site administrative procedures. The inspectors confirmed that the AmerGen receipt inspectors verified that Framatome was on the Approved Vendors List. The receipt inspectors also confirmed the receipt of the following documents from Framatome: (1) manufacturing documents for RVCH Serial Number CC/TM001 from Framatome - France; (2) Certificate of Conformance (C of C) for the RVCH; (3) C of C for the CRDSS support skirt; (4) data package for the RVCH; and, (5) radiographs for all the welds in the RVCH.

From October 7 to November 14, 2003, the inspectors performed in-office reviews of the FANP design specification (08-5014897) for the TMI replacement RVCH to verify that the material, design, fabrication, inspection, examination, testing, certification, documentation, packaging, shipping, and functional requirements specified were consistent with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division I for construction in accordance with NCA-3250 and other applicable requirements. The inspectors also reviewed the ASME Code reconciliation report (51-5025039), the ASME Code Form N-2 and the End of Manufacturing Reports (BUQRTM/NCC 0001 & 0003) for the replacement RVCH assembly that included the control rod drive service structure (CRDSS) support skirt and mounting flange. The reconciliation report addressed the specified design, materials, fabrication, and examination of the as-fabricated replacement RVCH. The end of manufacturing reports contained certified material test reports (CMTR), heat treatment records, weld records, non-conformance reports with corrective actions, nondestructive evaluations, and weld material acceptance tests for the manufacture of the replacement RVCH, CRDM and TC nozzles, and the CRDSS support skirt and mounting flange. The inspectors verified that an authorized nuclear inspectors (ANI) had inspected the replacement parts, reviewed the manufacturing reports, and certified that FANP fabricated the parts in accordance with the ASME Code, Section III, Division 1.

The specific attributes reviewed included:

- verification that material heat treatment was used to enhance mechanical properties of the reactor head material and connected appurtenances and verification that vendor heat treatment procedures were consistent with ASME III requirements;

- verification that manufacturing controls monitored that the NDT was performed in accordance with code and design specification requirements;
- verification that weld overlay cladding operations were performed to establish a layer of stainless steel on the inside of the RVCH head per specifications and drawings;
- verification of a sample of fabrication material CMTRs for the replacement RVCH to confirm that the materials did not contain deleterious substances;
- verification that repair procedures were consistent with ASME Code material specifications that repair welding was in accordance with ASME Code qualified procedures that welders were qualified, and that repair records were maintained;
- verification that programs and procedures were established for the preparation of CMTRs and the records of examinations and tests are traceable;
- verification that the original ASME records demonstrate that the N-stamp remains valid and that the replacement RVCH complies with appropriate Code requirements and recommendations;
- verification that the Stress Report and Design Specifications were adequate, complete, and signed by professional engineers competent in ASME Code requirements;
- verification that machining was performed under a controlled system and consistent with manufacturers' quality assurance (QA) program;
- verification that QA drawings and documents were appropriately controlled;
- verification that process controls maintained traceability throughout the RVCH manufacture;
- verification that the ASME Code, Section III data packages were supplemented by ASME Code, Section XI pre-service data packages; and,
- verification that selected manufacturing and inspection records of the finished replacement RVCH were documented.

The inspectors also reviewed the audit and surveillance of the FANP quality assurance programs (QAPs) at both the Paris/La Defense and Chalon/Saint Marcel, France facilities that were led by Exelon and performed by nuclear industry personnel. The inspectors also reviewed an audit that assessed the QAP at the Chalon plant and the interaction between the Chalon plant and the FANP facilities in Lynchburg, Virginia.

<u>Head Removal and Replacement Inspections</u>

<u>Observation of Movement of Replacement RVCH from Storage to Containment</u>

From September 29 to October 2, the inspectors reviewed ECR 03-00202, "Reactor Vessel Closure Head Replacement Hauling and Rigging," Rev. 1, and the activities associated with rigging and lifting of the new RV head. The inspectors reviewed: (1) the preparations and procedures for rigging and heavy lifting; (2) the planning and selection of required crane and lifting devices; (3) the procedures for inspection and testing of the cranes and lifting and rigging equipment; (4) documents for structural or equipment modifications, if required; (5) preparation of the lay-down area; and, (6) training of rigging personnel. The inspectors verified that the capability of the lifting equipment, including fixtures and rigging, had been analyzed and evaluated though engineering calculations and analyses. The review was focused on the applicable approved procedures and plans.

From October 20 to 24, the inspectors reviewed ECR 03-00202, Rev. 2, and the applicable work orders and procedures for transporting the replacement RVCH from the lay-down area outside the protected area and into containment. The inspectors walked down the vendor prepared hauling path; observed selected portions of the construction of the outside runway and its supporting structure; observed removal of the protective top shipping cover; observed the transport of the new RVCH from the lay-down area into the protected area; and observed selected portions of its alignment with the runway system. The inspectors noted that Exelon Nuclear Oversight personnel were conducting independent observation of selected portions of the hauling and rigging activities. The inspectors verified that no major structural modifications were performed for the RV head replacement activity. The inspectors also verified that no temporary modifications were needed for containment access to support the RVCH head replacement activity.

<u>Observation of CRDMs and Axial Power Shaping Rods (APSRs) Uncoupling</u>

From October 20 to 24, the inspectors also observed the preparations, pre-job briefing and a selected sample of work activities performed to uncouple the CRDMs and APSRs from the existing RVCH. The inspectors observed the FANP task leader pre-briefing the requirements for uncoupling CRDMs from the RVCH and verified that appropriate topics were discussed. The inspectors verified that refueling procedures 1504-12, “Shim Safety CRDM Uncoupling,” Rev. 17 and 1504-13, “APSR Lead Screw Uncoupling,” Rev. 24 were implemented during the observed portions of the uncoupling of lead screws from the CRDMs and APSRs. The inspectors noted that a shielded work platform (SWP) was used on top of the head to minimize personnel exposure. The inspectors verified that the progress of the uncoupling operations was communicated to the control room operators monitoring core activity. The inspectors verified that the maintenance technicians implemented the foreign material exclusion (FME) procedures during the uncoupling activities.

<u>Installation of CRDMs on the New RVCH</u>

The inspectors observed the implementation of procedure 03-5011503, “CRDM Replacement Procedure,” Rev. 2. The inspectors observed activities in progress both inside containment and via remote video monitor. Particular attention was devoted to the CRDM stator installation, lubrication of O-Rings, nozzle flange cleaning, and installation of hold down bolts. Hold down bolts that were removed from the existing RVCH were cleaned and subjected to visual and LP examinations prior to reuse. The inspectors verified the adequacy of the reuse acceptance criteria. The inspectors also verified the effectiveness of the communication systems that were established to facilitate test, inspection, and component tracking data transfer and remote supervision of the technicians working on the top of the replacement RVCH.

The inspectors reviewed completed procedure 03-5011503 and selected data sheets and records for the transfer and reinstallation of the existing CRDMs. The inspectors verified that the heat numbers, serial numbers, and inspection results were documented.

Removal of Old RVCH from Containment

From November 3 to 7, the inspectors reviewed selected applicable portions of the preparation for moving the old RVCH from the temporary head stand, out of containment and to a secure temporary lay-down area inside the protected area. In particular, the inspectors reviewed the technical evaluations associated with qualifying the alternate transportation path. The old RVCH was wrapped prior to being lowered to the equipment hatch area. The inspectors observed selected portions of the construction of the outside runway and its supporting structure. The inspectors also observed the sealing of the old RVCH in a qualified shipping bag and movement from containment to the transporter. The inspectors noted that additional precautions were taken due to inclement weather. The inspectors noted that Exelon Nuclear Oversight personnel were conducting independent observation of selected portions of the hauling and rigging activities. Senior licensee management personnel also observed selected rigging and hauling activities.

Post-Installation Verification and Testing Inspections

The inspectors accompanied TMI personnel during post-maintenance testing (PMT) of the installed replacement RVCH. The inspectors verified that the PMT was conducted in accordance with AmerGen procedures and independently verified the TMI personnel's conclusion that no leakage was observed from the replacement RVCH. The inspectors verified that questions related to the adequacy of ASME Code required testing were entered into the CAP as AR 186108.

b. Findings

No findings of significance were identified.

4. Reactor Vessel Head Replacement Radiation Protection

a. Inspection Scope

The inspectors reviewed work activities involving the replacement of the RVCH. Work activities observed included removal of the control rod drives and welding of lifting rig modifications. The old reactor vessel head in the head stand was controlled as a locked high radiation area, while the new reactor vessel head in the head stand was controlled as a high radiation area upon installation of the control rod drives. The controls implemented were compared to those required under plant TS 6.12 and 10 CFR 20, Subpart G, for control of access to high and locked high radiation areas.

The licensee established an exposure goal of 34.86 person-rem for reactor vessel head replacement, and through the first three weeks of the outage, exposures were tracking approximately six person-rem lower than estimated. Major work activities remaining at the time of this review involved the preparation of the old reactor head for transport. Work activities were evaluated against the requirements contained in 10 CFR 20.1101(b).

The inspectors reviewed the documentation associated with the shipment of the old reactor vessel head to a waste processor in Memphis, TN. This review was conducted against the requirements contained in 10 CFR Parts 20, 61, and 71, and 49 CFR Parts 100-177. The inspectors also selectively reviewed the licensee's program of training for personnel involved in the radwaste and radioactive materials transportation program with regard to the requirements contained in NRC IE Bulletin 79-19 and 49 CFR, Subpart H.

This inspection activity represents the completion of 1 sample relative to the inspectable area of radioactive material shipping.

b. Findings

No findings of significance were identified.

5. Reactor Vessel Closure Head Transportation Radiation Protection

a. Inspection Scope

The inspectors observed preparations for the RVCH shipment on December 2-3, 2003, including packaging of the head, loading onto the transport trailer, and shipping paperwork review, including the shipment manifest (Shipment RS-03-210-1). This review was conducted against the requirements contained in 10 CFR Parts 20 and 71, and 49 CFR Parts 100 -177. Specific requirements include:

- 49 CFR 172, Subpart C - Shipping Papers
- 49 CFR 172, Subpart D - Marking
- 49 CFR 172, Subpart F - Placarding
- 49 CFR 172, Subpart G - Emergency Response Information
- 49 CFR 172, Subpart H - Training
- 10 CFR 20, Appendix G - Uniform Manifest (NRC Forms 540 & 541)

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

Exit Meeting Summary

On January 16, 2004, the resident inspectors presented the inspection results to Mr. Bruce Williams and other members of his staff who acknowledged the findings. The regional specialist inspection results were previously presented to members of AmerGen management. Due to the type of security inspection conducted, no formal exit meeting was held; however, the inspectors did notify the licensee representatives of the preliminary findings at the conclusion of the inspection. The inspectors confirmed that proprietary information was not provided or examined during the inspection.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

<u>Licensee Personnel</u>
R. Atkinson, Replacement RVCH Project Manager
K. Bartes, Plant Operations Director
P. Bennett, Engineering
R. Brady, Emergency Preparedness Manager
M. Bruecks, Security Manager
R. Campbell, Operations
G. Chick, Director, Maintenance
L. Clewett, Director, Site Engineering
G. DeHoff, Security Operations Supervisor
R. Detwiler, Nuclear Oversight Supervisor
S. Dunkelberger, System Engineer
W. Eckman, Nuclear Oversight
L. Edwards, Radiological Engineer
E. Eisen, System Engineer
E. Fuhrer, Regulatory Assurance
G. Gellrich, Plant Manager
H. Langley, Site Emergency Preparedness Coordinator
D. Laudermilch, Security Analyst
L. Lucas, Chemistry Manager
M. Malloy, Security Shift Supervisor
S. Mannix, Facilities and Equipment Manager
D. Merchant, Radiation Protection Manager
A. Miller, Regulatory Assurance
P. Omaggio, Project Planner
J. Portz, Manager Supply Chain
S. Queen, Design Engineering Manager
G. Rombold, Regulatory Assurance
M. Sweigart, Radwaste Shipping
J. Tessmer, Operations Engineer
K. Tremblay, Lead Quality Receipt Inspector
B. Williams, Vice President, TMI Unit 1

<u>Framatome ANP</u>
W. Bryant, RVH Site Manager
D. Beckwith, Task Leader
M. Farlow, QA/QC Inspector
S. Horn, Task Leader
S. Lovelace, Project Coordinator
G. White, Weld Supervisor

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

2003005-01	NCV	Failure to Evaluate and Correct Reactor Coolant System Pressure Boundary Leak in a Timely Manner (Section 1R20)
2003005-02	NCV	Failure to Identify and Correct Boric Acid Corrosion of Reactor Building Containment Liner and Protective Moisture Barrier (Section 1R20)
2003005-03	NCV	Failure to Maintain Structural Design Clearances Inside Reactor Building Containment (Section 1R20)
2003005-04	NCV	Failure to Properly Perform Reactor Building Engineering Containment Coating Inspections (Section 1R20)
2003005-05	NCV	Failure to identify, Document, and Evaluate Conditions Adverse to Quality Which Had the Potential to Adversely Affect ECCS Containment Sump Availability (Section 4OA5.1)

LIST OF DOCUMENTS REVIEWED

Performance Indicator Report, Protected Area Security Equipment Performance, 3rd Quarter 2002 - 3rd Quarter 2003

Semiannual Fitness-for-Duty Data Summaries for July 1, 2002 - December 31, 2002 and for January 1, 2003 - June 30, 2003

LS-AA-2110, Monthly Data Elements for NRC Emergency Response Organization (ERO) Drill Participation, Rev. 6
LS-AA-2120, Monthly Data Elements for NRC Drill/Exercise Performance, Rev. 4
LS-AA-2130, Monthly Data Elements for NRC Alert and Notification System (ANS) Reliability, Rev. 4

Calculations and Drawings

Calculation C-1101-543-E540-014, TMI-1: DHCCW Heat Exchanger 12R Test Evaluation
Calculation C-1101-543-E410-015, TMI-1: DHCCW Heat Exchanger 13R Test Evaluation
Calculation C-1101-543-E540-013, TMI-1: DHCCW Heat Exchanger Test Evaluation
Drawing No. 302-357, Station Blackout Diesel Flow Diagram, Rev. 4

Procedures

M-164, Station Blackout (SBO) Diesel Generator Major Inspection (Mechanical), Rev. 12
M-166, Station Blackout (SBO) Diesel Generator Reduced Scope Inspection (Mechanical), Rev. 7

Notifications

AR 094206, DC-P-1B Oil Leak not Evaluated for MRFF
AR 095410, DHCCW System is Classified Maintenance Rule (a)(1)
AR 099304, DC-P-1B Line Brg. Horiz. Vibes Increased > 50%
AR 099632, MSDS Sheets not Readily Available for Decay Closed Chem Add
AR 100339, SBO Battery Load Test Deferral Evaluation Inadequate
AR 107885, Mrule Functions for SBO do not address energizing ES Bus
AR 109808, NRC Feels Operability Determination was Weak
AR 111949, Configuration for DC-PI-179, not in Accordance w/dwg
AR 116674, DC-P-1A/B Thrust Bearing Thermocouples Improperly Classified
AR 117863, Heighten Awareness of Integrated Risk from Emergent Work
AR 118154, Added 55 Gallons of Oil to EG-Y
AR 119531, DC-P-1B did not start using CS1 in Control Room
AR 121022, 1Y Transformer Rating Listed Incorrectly in Calculation
AR 121107, DC-V-65A/B Manual Engage Pin Lockwire
AR 121405, SBO & Substation Batteries Manufacturing Defect
AR 125365, Improper Installation of Paddle Assembly on DC-FS
AR 125525, DC-FS-237 Gives Computer Alarm after Completion of Work
AR 128135, SBO Output Power Indicator on CR has No Power

AR 140152, Procedure difficulties in OP-TM-543-473 (Decay Closed)
AR 141090, EG-V-97A Lift Prematurely Lifting at 250# Set Point 275#
AR 142924, DC-V-17A Leakage
AR 143271, DC-V-45B Malfunctioned Due to Lack of Lubrication
AR 143443, EG-V-97A Repeat Maint (Duplicate of CR 141090)
AR 144964, PM Task to Collect Vibrations Data Missed During EG-Y-4 Run
AR 153277, Reactor Heat Sink Performance Self Assessment
AR 157959, Conflicting Information in Chem Recommendation
AR 161500, SBO DG Overvoltage Trip
AR 161506, SBO DG Overspeed not performed, Vendor Rep. Not Present
AR 161624, Instrument Designations do not Agree with Print Designation
AR 161916, EE-G2-12-EX1 PTL Position Found Out of Position
AR 162328, EGY4 Potential Transformer Labels do not Agree with Drawings
AR 163234, Step Change in SBO Output Current
AR 163585, Air Cooler Temperatures Remain Higher than Desired
AR 166662, NRC SSDI Question 2003-90 and 2003-9 : DC Cooling Water Flow
AR 169949, Time to load SBO diesel exceeded UFSAR requirements
AR 175850, As-Found Set pressure of DC-V-17A low (74 vs. 77.6 psig)
AR 180432, Anti-Freeze Sample Results
AR 183330, Self Reading Dosimetry Lost in RB Sump

System Health Reports

System Health Report - System (Station Blackout Diesel Generator)
System Health Report - System (Decay Heat River Water System)
System Health Report - Service Water Corrosion/MIC/ER-AA-340/TDR 117/TDR 119
System Health Report - GL 89-13

Work Orders and Change Request

ECR 03-00794, Tech. Eval. For 1R15 DH-C-1B and DC-C-2B Cooler Testing
ECR 03-00804, TMI-1 : 1R14 DC-C-2A Performance Evaluation
WO R1178940, External Cooler Inspection (Decay Closed River)
WO R1831792, DC-C-2A Heat Exchanger Inspection & Clean
WO R1835824, SBO Diesel Generator Major Inspection (Mechanical)
WO R2039774, EG-Y-4 Operational Test
JO 00085087, SBO Diesel Generator Major Inspection (Mechanical)

TMI Unit 1 UFSAR

TMI-1 UFSAR 9.5 DHR System
TMI-1 UFSAR 9.6 Cooling Water Systems

Miscellaneous

Topical Report 117, Microbiologically Influenced Corrosion (MIC) Program Description, Rev. 1

Topical Report 119, Generic Letter 89-13 Program Description, Rev. 2
EPRI NP-7552, Heat Exchanger Performance Monitoring Guidelines
EPRI TR-107397, Service Water Heat Exchanger Test Guidelines
GPU Nuclear SDD-TI-700A, Station Blackout Modifications, Rev. 4
Section A-12, TMI Operations Plant Manual Site Blackout and Auxiliary Equipment
Letter from GPU Nuclear Corporation to U.S. Nuclear Commission, TMI-1 Generic Letter
89-13 Revised Response

Design Analysis, Calculations and Evaluations:

FANP 32-5024461, Reactor Vessel Head CRDM Analysis, Rev. 2
FANP 32-5024461, CRDM Nozzle Analysis, Rev. 0
FANP 32-5024373, Replacement Reactor Vessel Closure Head Sizing, Rev. 2
FANP 32-5025764, Service Structure Support Skirt Cover Hatch Analysis, Rev. 0
FANP 32-5025064, Validation of Stud Tensioning with Replacement Closure Head, Rev. 1
FANP 32-5025765, Reactor Vessel Closure Head Service Structure Support Skirt
Insulation Bracket Analysis, Rev. 1
FANP 32-5024750, Reactor Vessel Closure Analysis with Replacement Head, Rev. 1
FANP 32-5024750, Head Closure Analysis, Rev. 0
FANP 32-5024069, Thermocouple Nozzle Analysis, Rev. 1
FANP 32-5025780, ASME Design Report, Rev. 0
FANP 08-5014897, Design Specification, RVCH, TMI-1, Rev. 8
Appendix A Codes and Standards
Appendix B Document Submittal
Appendix C Design Conditions
FANP 08-5014897, Design Specification, Rev. 5
FANP 08-5014897, Reactor Vessel Closure Head Replacement - TMI-1, Rev. 4
FANP 18-5015842, Load Specification for TMI-1 Replacement RVCH, Rev. 2
FANP 18-5015842, Load Specification for TMI-1 Replacement RVCH, Rev. 1
TERA Corporation Design Calculation (DC)-83-25, Evaluation of Heavy Load Handling
Operations at TMI-1, Volume II, Reactor Building
OPE-03-031, Polar Crane Cracks in Welds and Base Metal of Longitudinal Braces and
Loose Bolted Connections in Lateral Bracing, Rev. 1
Gilbert Associates Inc. Calculation Number 1:01:05, Reactor Building - Polar Crane
Gilbert Associates Inc. Calculation Number 1:01:05, Appendix, Reactor Building -
Special Condition - Erection or Removal of Steam Generators
ECR TM 03-00202, Evaluation of New Reactor Vessel Head Movement, Rev. 0
ECR 03-00202, Reactor Vessel Closure Head Replacement Hauling and Rigging
Technical Evaluation, Rev. 2 [FANP Document No. 04-4882-02]
ECR 03-00281, Undergrounds Between Liberty Lane & Equipment Hatch for Truck Load, Rev. 0
ECR 01-00558, Underground Facility Along Alternate Component Travel Path, Rev. 0

Design Change Packages:

ECR TM 02-01410-001, Reactor Vessel Head Replacement
ECR TM 02-01411-001, Reinstall Service Structure on New RV Head

ECR 01-00314, Relocation of the Used Main Transformer, Rev. 0

BIGGE Power Constructors Rigging Calculations:

BIGGE Document 2061-D1, Engineering Design Basis, Reactor Vessel Closure Head (RVCH) Replacement Project, Three Mile Island Nuclear Plant
Bigge Calculation No. 2061-C1.1, Gantry Rigging and inside Rigging, Rev. 1
Bigge Calculation No. 2061-C3.2, Tie-down of Head to Transporter, Rev. 0

Original As-Built Drawings:

B&W 128769E, Closure Head Flange, Rev. 1
B&W 128770E, Closure Head Center Disc, Rev. 2
B&W 128771E, Control Rod Mechanism Housing, Rev. 9
B&W 128772E, Upper Shell Assembly, Rev. 5
B&W 128774E, Closure Head Sub-Assembly, Rev. 8
B&W 128775E, Misc Closure Head Details, Rev. 6
B&W 128776E, Closure Head Assembly, Rev. 7
B&W 128787E, Service Structure Support &Mating Flange, Rev. 3
B&W 51-00-017-01, Vessel - Head Flange Mating Surfaces, Sheet 6 of 13
Gilbert Associates Inc., E-521-011, Reactor Building - Steel Crane Runway, Rev. 8

Framatome ANP Drawings:

FANP 5025193D, TMI-1 RVCH Plant Site Haul Route, Rev. 0
FANP 5025194E, Sheets 1 & 2, TMI-1 RVCH Qualified Haul Route (Mechanical), Rev. 0
FANP 5025195E, Sheets 1 & 2, TMI-1 RVCH Qualified Haul Route (Electrical), Rev. 0
FANP 5020522E, Sheets 1 & 2, TMI-1 RVCH Replacement Floor Loading & Layout Plan, Elevation 346'
FANP 5020524B, Sheets 1, 2 & 3, TMI-1 RVCH Replacement Floor Loading & Layout Detail Plan, Elevation 346'
FANP 5029463B, Special Washer - TMI CRDSS, Rev. 0
FANP 5029102D, TMI-1 Service Structure Lifting Lug, Rev. 1
FANP 5029103E, TMI-1 Service Structure Lifting Lug Installation and Removal, Rev. 1
FANP 02-5015610E, Closure Head Specification Drawing, Rev. 8
FANP 02-5019935D, CRDM Nozzle Weld Preparation and Requirements, Rev. 1

BIGGE Drawings:

Calculation Drawing, C3.1, Sheets 1, 2 & 3, - Figure 6, Transporter/Prime Mover and Trailer
02E47-11, 3 Sheets, Inside Rigging at Containment Old\New RVH Rigging Detail, TMI Nuclear Station, Framatome ANP, Inc., Rev. 0
02E47-21, 3 Sheets, Plot Plan, Outside Rigging System, Old\New RVH Storage Area, TMI Nuclear Station, Framatome ANP, Inc., Rev. 1
02E47-31, 8 Sheets, Plot Plan, Runway System Set, Old\New RVH on Rail Beams, TMI Nuclear Station, Framatome ANP, Inc., Rev. 1

02E47-70, 3 Sheets, Plot Plan, Haul Route of RVH, TMI Nuclear Station, Framatome ANP, Inc., Rev. 0
02E47-71, 3 Sheets, Sections & Details, Old\New RVH Transport & Tie Down, TMI Nuclear Station, Framatome ANP, Inc.

NRC Inspection Procedures and Regulatory Documents:

71111.02, Evaluations of Changes, Tests, or Experiments
71111.15, Operability Determinations
71111.17, Permanent Plant Modifications
71007, Reactor Vessel Head Replacement Inspection
TS 3.12 (and Bases), Reactor Building Polar Crane
UFSAR 5.2.2.4, Liner Plate and Penetrations
UFSAR 5.2.1.2.12, Polar Crane Load
UFSAR 5.2.1.2.4, Live Load
UFSAR 5.1.1, Classes of Structures and Systems for Seismic Design
UFSAR Table 5.4.1, List of Class I Structures, Systems, and Components
Information Notice (IN) 2003-20, Derating Whiting Cranes Purchased Before 1980

Correspondence

GPU Letter 5211-84-2013 to NRC, regarding Control of Heavy Loads, dated February 21, 1984

GPU Letter 5211-84-2200 to NRC, regarding Control of Heavy Loads, dated August 16, 1984

GPU Letter C311-96-2167 to NRC, regarding GPU Response to NRC Bulletin 96-02, Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment dated May 14, 1996

NRC Letter (TAC M95651) to GPU Nuclear (1920-98-30259), regarding Completion of Licensing Action for NRC Bulletin 96-02, Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment, dated April 27, 1998

TMI-1 Commitments

1985T0027, Head & Internal Handling Fixture Weld ISI Program
1984T0132, Procedures for Heavy Loads in Reactor Building
1984T0075, Heavy Loads Procedure - Sling Selection/Approval
1984T0131, Signalman Inspection of Heavy Load Pathways
1982T0046, Reactor Vessel Missile Shield Handling
1985T0064, Heavy Loads Procedure - Head & Internals Handling
1984T0133, RV Head Removal & Installation Lift Height Restriction
1984T0137, RB & FHB Heavy Load Handling Area Layout Drawings

AmerGen and Exelon Procedures

MA-AA-716-021, Rigging and Lifting Program, Rev. 0
LS-AA-105, Operability Determinations, Rev. 1
HV-AA-1211, Pre-job Briefing Check List, Rev. 1
MA-AA-716-008, FME Program, Rev. 1
SM-AA-102, Warehouse Operations, Rev. 3
MA-TM-134-903, Reactor Vessel Disassembly, Rev. 1
MA-TM-134-904, Reactor Vessel Reassembly, Rev. 0
1504-13, APSR Leadscrew Uncoupling, Rev. 24
1504-12, Shim Safety CRDM Uncoupling, Rev. 17
1407-21, Frequent and Periodic Inspection of Lift/Rigging Equipment, Rev. 0
1507-1, Polar Crane, Rev. 22
1507-1, Polar Crane, Rev. 20
1507-1, Polar Crane, Rev. 21

AmerGen Work Orders and Associated Work Order Activities:

Work Order (WO) C2004750, Move New RVCH and SWP into Reactor Building.
    Work Order Activity (WOA)-03, Move Security Barriers - RVCH to Gate, August 30, 2003
    WOA-05, Install Runway (for New RVCH), August 30, 2003
    WOA-20, Rig/Lift/Set New Head on Transporter, October 20, 2003
    WOA-22, Rig/Lift and Remove RVCH Tophat, October 19, 2003
    WOA-36, Camera Security Inspection of the New RVCH, October 19, 2003
    WOA-06, Set New RVCH on Runway & Move to RB 308, October 10, 2003
    WOA-07, Lift/Rig/Move New RVCH - 308’ to 346’, October 10, 2003
    WOA-18, HLA - Head Lift to 346’, August 30, 2003.
    WOA-08, Head Inspection - 346’, September 26, 2003
    WOA-08, Head Inspection - 346’, October 10, 2003
    WOA-02, RVCH Temporary Head Stand Layout/Preps, October 19, 2003
    WOA-21, Move new Head-Laydown Area to Gate #11, August 30, 2003
    WOA-23, Offload/Runway/Fit-up & Gantry Assembly, October 6, 2003
    WOA-31, Revise Rx Bldg. Procedures/ECR 03-00202, September 16, 2003

WO C2004752, Prep Service Structure for Removal
    WOA-08, Rig/Lift/Load Test & NDE Temp Lift Lugs, September 4, 2003
    WOA-09, Layout/Weld Temp CRDSS Lifting Lugs, October 28, 2003
    WOA-12, CRDSS Flange Disassembly, October 28, 2003
    WOA-11, Install Rigging to CRDSS Lifting Lugs, October 24, 2003
    WOA-18, Inspect/Evaluate Unpainted CRDSS Areas, August 29, 2003
WO C2004754, Install Service Structure on New RVCH
    WOA-03, Rig/Move CRSS to New RVCH, October 28, 2003
    WOA-07, Modify Thermocouple Piping Contingency, October 28, 2003
    *****WOA-29, Review All Framatome SDCN, DCN, SI and NCR Documents for Impact to Modification Packages
WO R2011221, Perform Rx Vessel Disassembly, October 19, 2003

WO R2009396, MIS-A-1, Perform Inspection of Reactor Building Polar Crane, November 1, 2003

Note: " ***** " Indicates activity was generated as part of the inspection process.

Framatome ANP Procedures and Working Instructions (WIs):

FANP 03-5011503, CRDM Replacement Procedure, Rev. 2, dated June 11, 2003
FANP Safety Document Change Notice, 30-5036110-00, for revisions to FANP 03-5011503, Rev. 2, dated October 31, 2003
FANP 03-5011503, CRDM Replacement Procedure, Rev. 2, completed record dated November 18, 2003
FANP 03-5025283, TMI Reactor Vessel Closure Head Replacement, Rev. 4, October 13, 2003
FANP 50-5024805, Removal/Reinstallation of TMI control Rod Drive Service Structure, Rev. 4, October 13, 2003
WI-1, Quality Control Activities, Rev. 11
WI-3, Project Control of Field Service Work, Rev. 5
WI-4, Training and Qualification of Personnel, Rev. 6
WI-5, Field Task Leader Responsibilities, Rev. 3
WI-9, Nonconforming Items, Rev. 9
Deviation Letter to WI-9, dated July 7, 2003
WI-13, Technical Requirements Document, Rev. 4
WI-14, Process Traveler, Rev. 5
54-ISI-240-41, Visible Solvent Removable Liquid Penetrant Examination, dated February 10, 2003
54-ISI-366-06, VT-1 and VT-3 Visual Examinations, dated July 30, 2003
Visual Examination Data Sheet

BIGGE Power Constructors Rigging Procedures:

No. 2062-P5, Procedure for Handling Old RVCH from Containment to Storage, Rev. 2
No. 2061-P4, Procedure for Rigging Inside Containment Building, Rev. 2, [FANP No. 01-5028075-02]
No. 2061-P1, Procedure for Transferring New Head to Platform Trailer, Rev. 2 [FANP No. 01-5028072-02]
No. 2061-P3, Procedure for Land Transport & Transporter to Runway Transfer, Rev. 2, [FANP No. 01-5028074-02]
No. 2061-P2, Procedure for Runway System Installation and Operation, Rev. 3 [FANP No. 01-5028073-03]

FANP Corrective Actions and Quality Assurance Documents

FANP 51-5025039, Replacement Reactor Vessel Closure Head Reconciliation, Rev. 0, and ASME Boiler and Pressure Vessel Code, Code Cases
ASME Code Form N-2 for the Replacement Head dated August 21, 2003
FANP BUQRTM/NCC 001, End Of Manufacturing Report for Replacement RVCH, dated September 24, 2003

FANP BUQRTM/NCC 0003, End of Manufacturing Report for Replacement Support Skirt and Mounting Flange, dated August 24, 2003
FANP Non Conformance Notification (NCN) 02/00001 Rev. 0
FANP NCN 02/00002 Rev. 0
FANP NCN 02/00007 Rev. 0
FANP NCN 03/00002 Rev. 0
FANP NCN 03/00003 Rev. 0 and Rev. 1
FANP NCN 03/00008 Rev. 1
FANP NCN 03/00009 Rev. 0
FANP NCN 03/00012 Rev. 0
FANP NCN 03/00015 Rev. 0
FANP NCN 03/00016 Rev. 0
FANP NCN 03/00021 Rev. 0
FANP Contract Variation Approval Request (CVAR) 87-5026683-00
FANP CVAR 87-5027787-00
FANP Condition Report (CR) 6028874, Insulation Ring Panel Installed by Transco Appears to Contact RVCH Thermocouple Nozzle, Rev. 1
FANP CR 602887, Replacement RVCH Insulation Ring Panel Installed by TRANSCO is in Close or Direct Contact of the Thermocouple Nozzle, Rev. 0
FANP CR 6029460, Worker Walked Under a Suspended Load During Removal of the Replacement RVCH “Top Hat” Assembly
FANP CR 6029446, RVCH Surveys Identified Dose Rates 2 to 3 Times Higher than Expected
FANP CR 6029262, Humidity of the RV Head Exceeds 60%, Rev. 0
FANP CR 6029613, As-Found Condition of the RVLIS and RCITS Anti-Rotation Collars Insufficient Clearance, Rev. 0
FANP CR 6029631, As-Found Condition of the RCITS Anti-rotation Collar Has Tap Broken off in Bolt Hole, Rev. 0
FANP NCR 6029491, Tolerance for the Pads for Leveling the West CRDM Rack Support Stand Was Not Met, Rev. 2
FANP CR 6029486, Bigge Rail System Had Improper Rail Set-up, Rev. 0
FANP NCR 6029497, Dropped PI Tube on the Service Structure, Rev. 0
FANP CR 6029536, As-found Condition during Layout of Temporary Lifting Lugs (Weld Joints where Lift Lug to be Welded, Rev. 0
FANP CR 6029606, 25% of Stator Thermocouple Connection Broken Off
FANP CR 6029607, One of Two Stator Thermocouples Failed

Exelon and AmerGen Corrective Actions and Quality Assurance Documents

Exelon Nuclear Oversight Plan for the TMI Reactor Head Replacement Project
Reactor Vessel Head Replacement Project, Nuclear Oversight Objective Evidence Report, (AR Tracking No. 00179545) dated October 17, 2003
Exelon Approved Vendors List
Exelon Audit Report No. SR-105380-02, Nuclear Utilities Procurement Issues Committee (NUPIC), Joint Quality Assurance Program Audit Report, Framatome ANP [Paris, La Defense, France; Chalon Plant (Saint Marcel), France] dated July 2, 2002, and

associated Exelon and FANP correspondence resolving the audit deficiency documented in Exelon CR #113945

Exelon Audit Report No. SR-137988B-03, Supplier Evaluation Services Department, Framatome ANP [Chalon/Saint Marcel, France] dated March 25, 2003, and associated Exelon and FANP correspondence resolving the audit deficiency documented in Exelon CR #150926

Exelon Surveillance Report No. SR-55-2003, Supplier Evaluation Services Department, Framatome ANP [Chalon/Saint Marcel, France] dated August 5, 2003, and attached End of Manufacturing Report (BUQRTM/NCC 0001), Rev. A, dated August 8, 2003

Action Tracking Item (Assignment Number) 00181779-04, Assure PIMS AR to Repair Crane Is Complete

Action Tracking Item (Assignment Number) 00181779-10, Generate Configuration Control Documents If Polar Crane Not Repaired

Assignment Report (AR) 0181799, Polar Crane Bridge Rail Inspection Found Loose Bolts & Cracks

AR 00142012, 10 CFR21 Notification of Whiting #25 Gear Case Hoist Bolt Overstress

AR A2073913, Evaluate Deteriorated Diagonal Braces for RB Polar Crane, October 28, 2003

AR 2025299, Evaluation 71, Review Storage Location/Path for Interim Storage of Old Head, November 4, 2003

AR 2025299, Evaluation 72, Review Revised Storage Location/Path for Storage of Old Head RVCH Inside Gate 10, November 4, 2003

AR 00142012, 10 CFR Notification of #25 Gear Case Hoist Bolt Overstress

AR A2016463, Perform Inspection of Reactor Building Polar Crane, November 3, 2003

AR 81482, Polar Crane Speed Control Problem Near Miss

AR 142012, 10 CFR 21 Notification of #25 Gear Case Hoist Bolt Overstress

AR 182189, Near Miss, Trolley Redundant Brake Link Fell Off Polar Crane

*****AR 184594, Less Than Timely Procedure Change for Polar Crane Rating

*****Condition Report (CR) 186032, Identified that FANP System Engineering Review of All FANP DCNs, SDCNs, SIs, CRs and NCRs were Missed for At Least One CR. Further Investigation Identified Additional Missed Reviews

Note: “ * “ Indicates AR was generated as part of the inspection process.

<u>10 CFR 50.59 Screens and Applicability Reviews</u>

50.59 Applicability Review for MA-TM-134-903, Reactor Vessel Disassembly, Rev. 1

50.59 Applicability Review for ECR# 01-00314, Rev. 0, Relocation of the Used Main Transformer

50.59 Applicability Review for ECR# 01-00558, Rev. 0, Underground Facility Along Alternate Component Travel Path

50.59 Applicability Review for Procedure 1507-1, Polar Crane Operation, Rev. 21, Procedure Change 13373

50.59 Applicability Review for Procedure 1507-1, Polar Crane Operation, Rev. 21, Procedure Change 14912

50.59 Screening for ECR 01-00314, Rev. 0, Relocation of the Used Main Transformer

50.59 Screening for ECR# 01-00558, Rev. 0, Underground Facility Along Alternate Component Travel Path
50.59 Screening for Procedure 1507-1, Polar Crane Operation, Rev. 21, Procedure Change 14912

Other Documents

TMI Reactor Vessel Closure Head (RVCH) Replacement Project Plan dated August 20, 2003
FME Plan for the T1R15 Refueling Floor and Spent Fuel Pool Activities, Project Date October 18, 2003, Rev. 0
Exelon Purchase Order (PO) 80009666, RVCH and CRDSS Support Skirt
FANP 18-1173549, Reactor Coolant System Functional Contract Specification, Rev. 2

LIST OF ACRONYMS

1R14	Cycle 14 Refueling Outage
1R15	Cycle 15 Refueling Outage
1R16	Cycle 16 Refueling Outage
ADAMS	Agencywide Documents and Management System
ALARA	as low as is reasonably achievable
AmerGen	AmerGen Energy Company, LLC
ANP	Advanced Nuclear Power
ANS	Alert and Notification System
APSR	Axial Power Shaping Rods
AR	Assignment Reports
ASME	American Society of Mechanical Engineers
BS	Building Spray
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CMTR	Certified Material Test Report
C of C	Certificate of Conformance
CR	Condition Report
CRDM	Control Rod Drive Mechanism
CRDSS	Control Rod Drive Service Structure
CVAR	Contract Variance Approval Request
DC-C-2(A/B)	Decay Heat Service Closed Cooling Water Coolers
DEP	Drill/Exercise Performance
DH	Decay Heat
DHR	Decay Heat Removal
DRP	Division of Reactor Projects
DRS	Division of Reactor Systems
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
ECR	Engineering Change Request
ECT	Eddy Current Testing
EOP	Emergency Operating Procedure
EP	Emergency Preparedness
EPRI	Electric Power Research Institute
ERO	Emergency Response Organization
FANP	Framatome Advanced Nuclear Power
FHAR	Fire Hazards Analysis Report
FME	Foreign Material Exclusion
HPI	High Pressure Injection
IMC	Inspection Manual Chapter
IN	NRC Information Notice
IR	Inspection Report
ISI	Inservice Inspection
LHP	Lower Head Penetration
LP	Liquid Penetrant
MIC	Microbiologically Influenced Corrosion

MR	Maintenance Rule
MU	Make-Up
NCN	Non-Conformance Notification
NCR	Non-Conformance Reports
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NS	Nuclear Services Closed Cooling Water
OPE	Operability Evaluation
PI	Performance Indicator
PMT	Post-Maintenance Test
PWSCC	Primary Water Stress Corrosion Cracking
QA	Quality Assurance
QAP	Quality Assurance Program
RB	Reactor Building
RCS	Reactor Coolant System
RPV	Reactor Pressure Vessel
RSPS	Risk Significant Planning Standards
RV	Reactor Vessel
RVCH	Reactor Vessel Closure Head
RWP	Radiation Work Permit
SBO	Station Blackout
SDP	Significance Determination Process
SWP	Shielded Work Platform
TC	Thermocouple
TM	Temporary Modifications
TMI	Three Mile Island, Unit 1
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Testing
VT	Visual Examination

TI 2515/152, REACTOR PRESSURE VESSEL LOWER HEAD PENETRATION NOZZLES (NRC BULLETIN 2003-02)

Reporting Requirements

a.1. The examination was performed by qualified and knowledgeable personnel with certification to the American Society of Mechanical Engineers (ASME), Section XI, Level II and Level III for visual examiners. In addition, Level II and Level III examiners had received a minimum period of training in this type of inspection. The training included a review of the penetration drawings, inspection techniques and use of visual aids, effects of surface conditions on detecting and evaluating indications, industry experiences, lessons learned, inspection results and procedure requirements.

a.2. The examination was performed using adequate procedures. The procedure specified the extent of the inspection required, provided detailed documentation requirements and provided clear inspection standards and acceptance criteria on which personnel were trained.

a.3. The examination was adequate to identify, resolve, and disposition deficiencies.

a.4. The examination performed was capable of identifying pressure boundary leakage and/or lower head corrosion described in the bulletin.

b. The reactor vessel lower head was free of dirt, debris, insulation, significant oxidation and any foreign material that could adversely affect viewing of the penetrations. No boric acid deposits were identified at the interface between the vessel and the penetrations.

c. The inspection was conducted by direct visual inspection by examination personnel and by the use of a video camera. The inspection effort achieved examination for 360 degrees around the circumference of all nozzles and the vessel bottom head in its entirety.

d. If present, small boric acid deposits representing reactor coolant leakage, as described in Bulletin 2003-02, could be identified and characterized.

e. No material deficiencies were identified. No indications were identified at the time this inspection was performed.

f. Selected insulation panels were removed to facilitate the visual inspection of the bottom head penetrations. There were no impediments to the performance of the visual examination.

g. No boric acid deposits were noted on the lower vessel head. Stains were noted in a small number of locations which were evaluated and concluded to be water stains. These stains were transparent with no visible solids.

h. There were no boric acid deposits on the lower vessel head. Analysis was not performed on protective coatings, tape residue and iron oxide (rust).

i. Selected penetrations intersecting the lower vessel head were cleaned of rust, dirt and tape residue using “scotch brite” pads and rinsed with water.

j. The licensee noted there were no boric acid deposits on the lower vessel head. The licensee reported boric acid deposits on the inside surfaces of the mirror insulation vertical panels and concluded the deposits had originated from leakage at the refuel cavity seal plate. This conclusion was based on examination of possible leakage sources above the vessel bottom. Based on the observations made by the inspector, interviews with examiners and review of pertinent data, the licensee’s conclusions appeared reasonable.

Three Mile Island Station
ISI Activities Inspection, October 19-31, 2003
Inspection Procedure 71111.08, Inservice Inspection Activities,
Temporary Instruction 2515/152, Reactor Pressure Vessel Lower Head Penetration Nozzles (NRC Bulletin 2003-002)

Documentation Review

Action Request/Condition Report

AR 00184368	Possible Loose Part Indications in OTSG ‘A’ Tubes
AR 00182877	Weld SR-0010BM Has 2 Recordable Indications
AR 00182174	RPV Lower Head Instrumentation Nozzle Inspection
AR 2044945	MU-V-26 Replacement of Valve and Actuator

NDE Examination Test Reports

2574-5	Ultrasonic Test of Nozzle to Pipe Weld, RCS, 1D-ISI-RC-002, R1
C200505005-1	Ultrasonic Test of Nozzle to Safe End, SR0010BMWELD, Pzr Surge Line
2573-1	Dye Penetrant of Nozzle to Pipe Weld, RCS, B5.130
W/OC2005617	Visual Exam Results RPV Lower Head Instrumentation Nozzles
C200505703-4	Liquid Penetrant, Elbow to Pipe DH0012, 13 and 14 Weld, Decay Heat
C200505004	Liquid Penetrant Nozzle to Pipe Weld, SR0010BMWELD, Pzr Surge Line

NDE Examination Procedures

51-5005406-02	Qualified Eddy Current Examination Techniques for TMI 1R15
54-ISI-240-40	Visible Solvent Removable Liquid Penetrant Examination
54-ISI-835-05	Ultrasonic Examination of Ferritic Piping Welds
54-ISI-270-41	Wet or Dry Magnetic Particle Examination
ER-AP-335-1012	Visual Examination of PWR Reactor Vessel Head Penetrations

Radiographic Examination Report

MU-V-26	Valve to Pipe Butt Weld No. 447, Valve No. 26 Make Up System
MU-V-20	Valve to Pipe Butt Weld No. 445, Valve No. 20 Make Up System
MU-V-20	Valve to Pipe Butt Weld No. 442, Valve No. 20 Make Up System
MU-V-20	Weld Repair of Weld No. 442 (442R1)

Repair-Replacement Work Order

TM02-01194	Replacement of Seal Injection and Seal Return Isolation Valves, Rev. 1
WOC2004373	Replacement of Vale MU-V-26 and Operator

Drawings/Isometrics

131949E	Assembly Details for 10" Surge Nozzle, Rev. 4
ID-ISI-MS-004	MS System Penetration 112 and 113, Rev. 1
135742E	Reactor Vessel Details, Rev. 1
154714E	Modified Instrumentation Nozzle, Rev. 4
302660	Make Up & Purification Flow Diagram, Rev. 38
302661	Make Up & Purification Flow Diagram, Rev. 52

Miscellaneous

Thirty Day Response to NRC Bulletin 2003-02, Leakage RPV Lower Head Penetrations
TMI-1 Steam Generator Inspection Degradation Assessment for Outage 1R15 (Rev. 1)
ER-TM-335-1005, Analysis of OTSG Eddy Current Data At TMI, Rev. 2
AES 01104524-1Q-1 Condition Monitoring and Operational Assessment, TMI, R14
ER-AP-420, Steam Generator Management Program Activities, Rev. 2
1246524A Instructions for Plug Inspection
ER-AA_2006, Lost Parts Evaluations, Rev. 0
FME Plan, TMI-1 Outage 1R15 Steam Generator Work, Rev. 0
ER-AP-331-1001, Boric Acid Corrosion Control (BACC) Inspection Locations, Implementation, and Inspection Guidelines, Rev. 0
ER-AP-331-1002, Boric Acid Corrosion Control Program Identification, Assessment, and Evaluation, Rev. 0
ER-TM-335-1006, TMI OTSG Site Specific Performance Demonstration Training Program, Rev. 0
OP-TM-220-405, Shutdown and Startup of the Loose Parts Monitoring System, Rev. 0
WPS 821-TMI, Weld Procedure Specification for P8 to P8 Material, Rev. 5