January 30, 2006

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 05000289/2005009, and NRC Office of Investigation Reports 4-2004-14 and 4-2004-15

Dear Mr. Crane:

On December 31, 2005, the U. S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Three Mile Island, Unit 1 (TMI) facility. The enclosed inspection report documents the inspection results, which were discussed January 9, 2006, with Mr. Rusty West and other members of your staff.

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

This report documents three NRC identified findings. One finding was a Severity Level IV Violation and two other findings were evaluated to be of very low safety significance (Green). These findings were determined to involve violations of NRC requirements. However, because of the very low safety significance of each issue and because they were entered into your corrective action program, the NRC is treating all three violations as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, 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 Inspector at Three Mile Island.

The Region IV Field Office of the NRC Office of Investigations (OI) initiated two investigations (Case Nos. 4-2004-14 and 4-2004-15) on May 11, 2004, to determine if licensed operators knowingly failed to provide complete and accurate information regarding medical conditions to AmerGen, and whether AmerGen then failed to provide complete and accurate information to the NRC in a timely manner.

Based on the evidence developed during these investigations, the NRC did not substantiate that licensed operators knowingly failed to provide complete and accurate information to

AmerGen, nor that AmerGen failed to provide complete and accurate information to the NRC in a timely manner.

Please note that final NRC documents, such as the OI reports described above, may be made available to the public under the Freedom of Information Act (FOIA) subject to redaction of information appropriate under FOIA. Requests under FOIA should be made in accordance with 10 CFR 9.23, Requests for Records.

Also, in accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, and its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of 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 610 337-5200 if you have any questions regarding this letter.

Sincerely,

/RA/

Ronald R. Bellamy, Ph.D., Chief
Reactor Projects Branch 7
Division of Reactor Projects

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

Enclosure: Inspection Report 05000289/2005009
w/Attachment: Supplemental Information

<u>cc w/encl</u>:
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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
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U.S. NUCLEAR REGULATORY COMMISSION
REGION 1

Docket No:	05000289
License No:	DPR-50
Report No:	050000289/2005009
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Three Mile Island Station, Unit 1
Location:	Middletown, PA 17057
Dates:	October 1, 2005 - December 31, 2005
Inspectors:	David M. Kern, Senior Resident Inspector Javier M. Brand, Resident Inspector Thomas F. Burns, Reactor Inspector, DRS Patrick W. Finney, Reactor Inspector, DRS Gilbert A. Johnson, Operations Engineer, DRS Jeffrey A. Kulp, Reactor Inspector, DRS Jonathan M. Lilliendahl, Reactor Inspector, DRS Ronald L. Nimitz, Senior Health Physicist, DRS Anne E. DeFrancisco, Reactor Inspector, DRS Thomas P. Sicola, Reactor Inspector, DRS
Approved by:	Ronald R. Bellamy, Chief Projects Branch 7 Division of Reactor Projects (DRP)

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SUMMARY OF FINDINGS

IR 05000289/2005009; 10/1/2005 - 12/31/2005; AmerGen Energy Company, LLC; Three Mile Island, Unit 1; Refueling and Other Outage Activities, Problem Identification and Resolution Sample, and Other Activities.

The report covered a 13-week period of inspection by resident inspectors and announced inspections by regional inspectors. Two Green non-cited violations (NCVs), and one Severity Level IV NCV 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," Revision 3, dated July 2000.

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

Cornerstone: Mitigating Systems and Barrier Integrity

- **<u>Green</u>.** The inspectors identified a non-cited violation (NCV) of Technical Specification (TS) 6.8.1.a for multiple failures to properly implement procedural requirements and engineering instructions to ensure control of materials brought into the reactor building containment while the plant was at power. The procedural violation resulted in multiple deficient conditions that challenged plant safety, including; the potential for hydrogen generation beyond design due to significant amounts of stored scaffolding, aluminum toe plates, unqualified materials (lead insulation blankets, painted scaffolding, plastic bags) with potential for reactor building sump clogging, and unrestrained stored materials inside containment. The licensee entered these issues into the corrective action program (issue reports 387939, 388006, 388791, 388916, 388407, and 395100), performed a prompt investigation, an extent of condition review, and an operability determination for each of the issues identified.

  This finding is more than minor because it affected the reliability objective of the equipment performance attribute under the mitigating systems cornerstone. The finding is also associated with the barrier integrity cornerstone and the respective containment configuration control attribute. The finding is of very low safety significance because no equipment was rendered inoperable, and no actual open pathway in the physical integrity of the reactor containment occurred. The cause of the finding is related to the cross-cutting area of human performance, because station personnel did not comply with engineering instructions and established procedures for control of materials inside containment. (Section 1R20)

Cornerstone: Barrier Integrity

Green. The inspectors identified a non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” because station procedures did not contain controls to verify and/or maintain the required environmental qualification (EQ) configuration associated with motor-operated valve (MOV) actuator T-drains. As a result, four safety-related containment isolation MOV valve actuators did not have T-drains as required by TI-103, “TMI-1 Environmental Qualification Report,” Rev. 5. This finding has been entered into the licensee’s corrective action program (issue reports 238918, 267293, 273768, 391720, 391707, and 271819).

The inspectors determined this issue was more than minor because it affected the barrier integrity cornerstone objective and the containment barrier performance attribute. Specifically, the lack of T-drains may allow moisture to enter the motor housing due to a high temperature and pressure steam environment associated with a Loss of Coolant Accident. The moisture and subsequent condensation could electrically short out the motor, which would reduce containment isolation reliability. In addition, if left uncorrected, this issue could become a more significant safety concern in that without procedures to maintain the required EQ configuration, additional MOV actuators could be installed with no T-drains or in an incorrect orientation and thus lead to a failure of the valve to perform its design function. This finding is of very low safety significance because the specific component qualification deficiency did not result in a loss of safety function, and the degraded condition did not cause an actual open pathway in the primary containment. Therefore, system or component operability was not effected. The cause of the finding is related to the cross-cutting area of human performance, because AmerGen did not develop appropriate measures to ensure that required MOV T-drains were properly installed, maintained, and inspected. (Section 4OA2.4)

Cornerstone: Mitigating Systems

- **Severity Level IV** . The inspectors identified a Green (Severity Level IV) non-cited violation of 10 CFR 50.74 for failure to report changes in medical conditions per Section 3.2.1 of Exelon administrative procedure OP-AA-105-101, “Administrative Process For NRC License And Medical Requirements,” Rev. 8. As a result, potentially disqualifying medical conditions for three operators were not reported to the NRC within the required 30-day time frame. In addition, for one of the operators, the medical condition ultimately required a change on his license. The licensee promptly entered this issue into their corrective action program (issue reports 164042, 189592, and 195798).

  This violation is more than minor because it had the potential to impact the NRC’s ability to perform its regulatory function, and it was evaluated using the traditional enforcement process. This finding is of very low safety significance because at no time did the individual stand watch without the medical condition being satisfied. In addition, the facility licensee was timely in their reporting of the medical conditions to the NRC when they received the pertinent information. The cause of the finding is related to the cross-cutting area of corrective actions, because it occurred after completion of actions to address a previous NCV for the failure to notify NRC of change in medical status of licensed operators. The cause of the finding is also related to the cross-cutting area of human performance, because multiple station operators did not comply with established procedures for reporting of potentially disqualifying medical conditions. (Section 4OA5.2)

B. Licensee Identified Violations

None

REPORT DETAILS

Summary of Plant Status

Three Mile Island, Unit 1 (TMI) began the inspection period at 100 percent rated thermal power and gradually reduced power due to end-of-cycle fuel depletion. On October 23, operators began a plant shutdown from 87.3 percent rated thermal power. The turbine output breakers were opened early on October 24, beginning the 16th refueling outage (1R16). Major work accomplished during this refueling outage included ‘B’ 125/250 volt station battery replacement, installation of a 6th vital bus inverter, feedwater heater replacement, and steam generator tube inspections. The 25-day refueling outage was completed on November 18. The reactor achieved 100 percent rated thermal power on November 20 and remained at or near full power through the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 1 sample)

a. Inspection Scope

The inspectors performed one inspection sample. The inspectors walked down risk significant plant areas in December 2005 and assessed AmerGen’s protection for cold weather conditions. The inspectors were sensitive to outside instrument line conditions and the potential for unheated components and ventilation. The walkdown included the emergency feedwater system, the condensate storage tanks, the borated water and sodium hydroxide storage tanks, and the fuel handling building engineered safety feature ventilation system. The inspectors also reviewed implementation of procedures WC-AA-107, “Seasonal Readiness,” Rev. 1 and OP-AA-108-111-1001, “Severe Weather Guidelines,” Rev. 2 for cold weather conditions. Other documents that were reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04 - 4 samples)

a. Inspection Scope

Complete System Walkdown

The inspectors performed one complete system walkdown sample on the following system:

- On December 28, the inspectors verified configuration alignment of the decay heat closed cooling water system. The inspectors conducted a detailed review of the alignment and condition of the system using the applicable one-line diagram 302-645, “Decay Heat,” Rev. 37 and procedure OP-TM-543-000, “Decay Heat Closed System,” Rev. 4. In addition, the inspectors reviewed and evaluated the corrective action program reports for impact on system operation and interviewed the system engineer.

Partial System Walkdowns

The inspectors performed three partial system walkdown samples on the following systems and components:

- On October 26, the inspectors walked down portions of the reactor coolant system while the piping was at mid-loop (drain down) operation during 1R16. In addition, the inspectors interviewed plant operators, and reviewed procedure OP-1103-11, “RCS Water Level Control,” Rev. 63.

- On October 31, the inspectors walked down the ‘A’ low pressure injection (LPI) train and portions of the ‘A’ decay heat closed cooling systems, while the ‘B’ LPI train was running for shutdown cooling.

- On November 2 and 3, the inspectors walked down the ‘B’ spent fuel pool cooling train while fuel was being transferred into the spent fuel pool.

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. The following documents were used for this inspection.

- OP-TM-212-101, “Shifting DHR Trains A and B From ES Standby To DHR Standby,” Rev. 2

- OP-TM-212-112, “Shifting DHR Train B From DHR Standby To DHR Operating Mode,” Rev. 3

- OP-1103-11, “RCS Water Level Control,” Rev. 63

- Flow diagram 302-630, “Spent Fuel Pool Cooling,” Rev. 31

Other documents that were reviewed are listed in the attachment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05 - 8 samples)

a. Inspection Scope

The inspectors performed eight inspection samples. The inspectors conducted fire protection inspections for several plant fire zones, which were selected based on the presence of equipment important to safety within their boundaries. The inspectors conducted plant walkdowns to verify the areas were as described in the TMI Fire Hazard Analysis Report, and that fire protection features were being properly controlled per surveillance procedure 1038, “Administrative Controls-Fire Protection Program,” Rev. 63. 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 accordance with procedure OP-MA-201-007, “Fire Protection System Impairment Control,” Rev. 2. The inspectors also reviewed issue report (IR) 396452 which evaluated minor discrepancies identified by the inspectors regarding the Appendix R 20-foot separation criteria (Section 1R15). In addition, the inspectors verified that applicable clearances between fire doors and floors met the criteria of Attachment 1 of Engineering Technical Evaluation CC-AA-309-101, “Engineering Technical Evaluations,” Rev. 7. Fire zones and areas inspected included:

- RB-FZ-1A, Reactor Building, Elev. 281', Outside Secondary Shield Wall North
- RB-FZ-1B, Reactor Building, Elev. 281', Outside Secondary Shield Wall S E
- RB-FZ-1C, Reactor Building, Elev. 281', Outside Secondary Shield Wall S W
- RB-FZ-1D, Reactor Building, Elev. 281', Inside Secondary Shield Wall East
- RB-FZ-1E, Reactor Building, Elev. 281', Inside Secondary Shield Wall West
- RB-FZ-2, Reactor Building, Elev. 308', Outside Secondary Shield Wall
- RB-FZ-3, Reactor Building, Elev. 346', Operating Floor
- Fire Zone AB-FZ-6, Auxiliary Building - demineralizers and ‘A’ motor control center (305' elevation) and evaluation of degraded fire control damper AH-D-84A

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07B - 3 samples)

a. Inspection Scope

The inspectors performed three inspection samples. Based on plant specific risk importance and resident inspector input, the inspectors selected the ‘A’ and ‘C’ nuclear service closed cooling water heat exchangers and ‘A’ intermediate cooling system heat exchanger as samples for inspection.

The inspectors reviewed the testing and cleaning methods to ensure heat removal capabilities were consistent with commitments made in response to Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment” and accepted industry practices. The inspectors determined that acceptance criteria were consistent with design basis values. Also, the inspectors reviewed methods for monitoring and controlling biotic and macro-fouling to verify that they were implemented effectively.

The inspectors completed walk downs of the selected components and the associated service water intake and discharge structures to assess the general material condition of the selected heat exchangers and associated service water components. The inspectors reviewed a sample of IRs related to the selected heat exchangers to ensure that problems related to these components were appropriately identified, characterized, and corrected. Additional documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (ISI) (71111.08 - 4 samples)

a. Inspection Scope

The inspectors observed selected samples of nondestructive examination (NDE) activities in process and reviewed documentation 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 review was performed to determine that the activities were performed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements and that AmerGen has met the inspection commitments made in their response to NRC Bulletin 2004-01.

The inspectors reviewed a sample of inspection reports and IRs shown in Attachment A-2 that were initiated as a result of problems identified during ISI examinations. Also, the inspectors evaluated effectiveness in the resolution and corrective action of problems identified during ISI activities for selected samples.

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

The inspectors observed manual ultrasonic testing (UT) and magnetic particle (MT) testing and reviewed inspection documentation of liquid penetrant (PT) and visual testing (VT) activities to determine 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 and MT tests performed on FW0037 in the Main Feedwater System (System 422). The inspectors reviewed the NDE report of the PT of the Pressurizer Vent Pipe and DH0504, DH0505, and DH0506WA welds in the Low Pressure Injection Decay Heat Removal System (System 212). In addition, the inspectors reviewed several issue reports (IRs 00389766, 00389688, 00389690, 00389697 and additional listed in Attachment A-2) which identified boric acid leaks found during the boric acid corrosion control program inspection. The predominate nature of the leaks were determined to be from mechanical seals or failed valve packing. The inspectors reviewed the disposition of these reports to determine that the identification, characterization, and repair instructions were complete and were captured in the corrective action program.

The inspectors evaluated implementation of the steam generator inspection program by review of specific portions of the steam generator management plan and the condition monitoring and final operational assessment of 1R15 activities. The inspectors reviewed plant specific steam generator design information, tube inspection criteria, control and monitoring of foreign objects, integrity assessments, degradation modes, and tube plugging criteria. The inspectors determined through examination of calibration documentation that the eddy current testing (ECT) probes and related inspection equipment in use had been calibrated and qualified for the expected types of active tube degradation. The inspectors determined that the licensee had performed the required review of the equipment calibration documentation and had accepted the equipment for service. Personnel training and qualification documentation was reviewed by the inspectors to determine that test examiners had been trained and tested in the eddy current inspection process.

The inspectors observed AmerGen's performance of portions of the 100 percent bobbin inspection of selected tubes for their entire length in both steam generators. The inspectors reviewed the ECT plan to determine whether the plan met technical specification (TS) requirements, EPRI Guidelines, and commitments made to the NRC.

The inspectors reviewed the steam generator inspection plan to determine whether the plan identified areas of potential tube degradation (based on site-specific and industry experience) to be inspected with special attention to areas that are known to represent potential ECT challenges.

No tubes were repaired during the period the inspectors were onsite. The tube inspection was still in progress at the completion of the inspection period. However, several tubes had been identified for plugging at the conclusion of the tube inspection activity. No tubes were identified as candidates for in-situ pressure testing during the inspection period. The inspectors reviewed data which indicated that steam generator leakage of greater than three gallons per day had not occurred during this operating cycle or noted during the post-shutdown visual inspection of the tube sheet faces.

The inspectors evaluated the implementation of the steam generator inspection program by conducting interviews with data management and acquisition personnel, data analysts and resolution analysts. The inspectors interviewed the licensee's independent qualified data analyst, and reviewed selected samples of eddy current data and data analysis of selected tubes within the ‘A’ and ‘B’ steam generators. Also, the inspectors noted that AmerGen had revised their steam generator degradation assessment in response to the identification at other plants of a degradation mechanism not previously noted in once-through steam generators of similar design as TMI.

The inspectors reviewed welding activities associated with the replacement of the thermal sleeve in the High Pressure Injection (HPI) Makeup and Purification system (system 211) and the replacement of the high point vent in the pressurize head. This review was conducted to determine if welding activities were performed in accordance with the requirements of ASME Section IX and XI. The inspectors reviewed the instructions for the replacement of the thermal sleeve and the high point vent and reviewed the welding procedures, welding procedure qualifications, NDE requirements and the test results of the completed weld. The inspectors reviewed welding procedure specification (WPS) WP8/43, “Manual Gas Tungsten Arc Welding of P8 to P43 Material,” used to replace the HPI thermal sleeve and the supporting procedure qualification records (PQR) (7211 and 7213) for compliance with the requirements of ASME Section IX. In addition, the inspectors reviewed WPS 43/43, “Manual Gas Tungsten Arc Welding of P43 Material to Itself,” used in the replacement of the pressurizer high point vent and supporting PQR 7072 for compliance with the requirements of ASME IX.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q - 1 sample)

a. Inspection Scope

On November 28, the inspectors observed licensed operator requalification training at the control room simulator for the ‘B’ operator crew. 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 No. 4, “Loss Of Feedwater Requiring High Pressure Injection PORV Cooling,” Rev. 11. 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 transients and verified the fidelity of the simulator to the actual plant. The inspectors evaluated training instructor effectiveness in recognizing and correcting individual and operating crew errors, including post-training remediation actions. The inspectors attended the post-drill critiques in order to evaluate the effectiveness of problem identification. The inspectors verified that emergency plan classification and notification training opportunities were tracked and evaluated for success in accordance with criteria established in Nuclear Energy Institute 99-02, “Regulatory Assessment Performance Indicator Guideline,” Rev. 3.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12 - 2 samples)

a. Inspection Scope

The inspectors performed two inspection samples. 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. 3.

- IR 426075 described a safety relay (63Z-2A/R-C3B) fire that occurred on November 20. The actual fire lasted for a few seconds before it was extinguished. The inspectors interviewed the engineered safeguard actuation system engineer and reviewed AR-2129116, which documented the engineering maintenance rule functional failure (MRFF) evaluation. The inspectors verified that appropriate corrective actions were initiated, and that an extent-of-condition review was initiated for all similar relays. The evaluation determined this was a MRFF because the specific relay function was affected, but the system remained in the acceptable performance (a) (2) category of the maintenance rule.

- IR 297543 described a failure of the 'B' control building ventilation fan AH-E-19B that was identified February 2. The failure involved a cracked hub which rendered this component inoperable and resulted in a reportable issue for a condition prohibited by TMI's TS. The inspectors evaluated AmerGen's response to this failure from a maintenance rule perspective. The inspectors verified that engineers properly categorized this failure as an MRFF and also as a maintenance preventable functional failure. The inspectors also verified that this failure did not change the current Maintenance Rule (a)(2) categorization of this system.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 3 samples)

a. Inspection Scope

The inspectors performed three inspection samples. 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 Administrative Procedure 1082.1, "TMI Risk Management Program," Rev. 4 and WC-AA-101, "On-Line Work Control Process," Rev. 10. The inspectors reviewed the routine planned maintenance, restoration actions, and/or emergent work for the following equipment removed from service.

- From October 19-24, the boric acid concentration for boric acid mix tank was reduced below the core operating limits report specified value (Risk document 1032, Rev. 2)

- On November 29 and December 1 respectively, station personnel determined that snubbers MU-H-319 and MU-H-311 were inoperable. Both snubbers provide seismic support for portions of the reactor coolant system (RCS) letdown line. The snubbers were replaced and engineering evaluations of potential damage were completed as required by TSs 3.16 and 4.17.

- On December 2, reactor river water pump RR-P-1A was declared inoperable due to excessive seal leakage identified during a quarterly surveillance test. This emergent condition elevated on-line maintenance risk to 'Yellow' until completion of repairs later the same day.

b. Findings

No findings of significance were identified.

1R14 Operator Performance During Non-Routine Evolutions and Events (71111.14 - 2 samples)

a. Inspection Scope

The inspectors performed two inspection samples. 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. The inspectors evaluated whether the evolutions were properly implemented according to the applicable procedures and TS limiting conditions for operation.

- On November 4, the station experienced a loss of instrument air during a transfer test of the 'B' auxiliary transformer (IR 395476). The plant was shutdown with all fuel assemblies offloaded into the spent fuel pool (SFP) at the time of this event. An electrical transient caused the operating air compressor IA-P-4 to trip, IA-P-1B had no electrical power due to the test alignment, and IA-

P-1A did not load due to a malfunction of the unloader valve. The IA-P-1A unloader valve stuck open, which increased plant air usage and reduced both station air and instrument air system pressures more quickly. The inspectors responded to the air compressor locations in the plant and to the control room to monitor operator actions and plant conditions. Operators implemented OP-TM-AOP-28, “Loss of Instrument Air,” Rev. 0. The inspectors verified SFP cooling was not interrupted during this event. Operators successfully restarted IA-P-4 and restored air system pressure approximately one hour into the event.

- On October 26, the inspectors observed main control room and field operators perform an RCS cooldown and draindown to a mid-loop condition. The inspectors reviewed operating procedure 1103-11, “RCS Water Level Control,” Rev. 63, the evolution plans, the risk assessment, applicable contingency plans, interviewed operators, and performed walkdowns of redundant RCS level indication instruments and tygon tubing. 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 (IR 390703).

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 3 samples)

a. Inspection Scope

The inspectors performed three inspection samples for the degraded equipment issues. The inspectors verified that degraded conditions in question were properly characterized, operability of the affected systems was properly evaluated, that applicable extent of condition reviews were performed, and no unrecognized increase in plant risk resulted from the equipment issues. The inspectors referenced NRC Inspection Manual Chapter Part 9900, “Operable/Operability-Ensuring the Functional Capability of a System Component” and AmerGen procedure LS-AA-105, “Operability Determinations,” Rev. 1, to determine acceptability of the operability evaluations.

- On September 13, the inspectors identified increased room temperatures in the turbine driven emergency feedwater pump (EF-P-1) room and questioned the effects on the environmental qualifications of safety-related components in the room and adjacent areas. The inspectors reviewed IRs 373226, 374105, and 374114 which documented and evaluated the inspectors’ concerns, and verified that safety-related equipment operability was not significantly affected by the elevated temperatures. In addition, the inspectors performed an extent-of-condition walkdown and inspected all possibly affected areas. The inspectors also interviewed operators, the TMI environmental qualification (EQ) engineer, and the motor-operated valves and emergency feedwater system engineers.

- On October 19-20, engineering and maintenance personnel evaluated concerns raised by the inspectors regarding the controls for loading of materials into the reactor building containment in preparation for 1R16, and identified several discrepancies. This resulted in multiple deficient conditions that challenged plant safety, including increased hydrogen generation due to large quantities of stored aluminum, potential for reactor sump clogging due to unqualified materials (lead insulation blankets, painted scaffolding, plastic bags), and seismically due to unrestrained materials. The inspectors reviewed the engineering operability evaluation documented in IR 388006, and reviewed other applicable IRs (387939, 388791, 388916, 388407, 395100). The evaluation concluded that operability of the multiple safety systems involved was not impacted (Section 1R20).

- On October 27, during a containment walkdown, the inspectors identified that instrumentation lines for several pieces of plant equipment, including the pressurizer (PZR) and steam generators (OTSG) level transmitters, and RCS pressure and temperature transmitters, did not have the 20 foot separation criteria required per 10 CFR 50, Appendix "R' Subpart G.2.d. The inspectors interviewed plant operators and engineering personnel, consulted with NRC regional and headquarter specialists, and reviewed the engineering evaluation documented in IR 396452. The engineering evaluation concluded that with the exemption of the PZR level instrumentation, there was sufficient train separation and redundancy, and that operability of the equipment was not affected. The evaluation also determined that it was unlikely for a fire to occur due to the very small amount of combustible materials in the area. In addition, the evaluation concluded that although no redundancy existed for the PZR level instrumentation, the instrumentation was not needed for safe shutdown of the plant, and that operators could safely shutdown the plant per abnormal operating procedure OP-TM-AOP-043, "Loss Of Pressurizer (Solid OPS Cooldown)," Rev. 0.

b. Findings

No findings of significance were identified.

1R16 Operator Work-arounds (71111.16 - 1 sample)

a. Inspection Scope

The inspectors performed one inspection sample. The inspectors reviewed the cumulative effects of the existing operator work-arounds (OWAs), the list of operator challenges, existing operator aids and disabled alarms, and the list of open main control room deficiencies to identify any effect on emergency operating procedure operator actions, and impact on possible initiating events and mitigating systems. The inspectors evaluated whether station personnel were identifying, assessing, and reviewing OWAs as specified in AmerGen administrative procedure OP-AA-102-103, "Operator Work-Around Program," Rev. 1.

Additionally, the inspectors attended the Unit Restart Review Meeting discussion of OWAs at the completion of the refueling outage. The inspectors also toured the control room, and discussed the following items of particular concern with the responsible system engineers and operators to ensure the items were being addressed on a schedule consistent with their relative safety significance:

- Workaround AR-A2045366, Modification to Prevent Oil Drain-down in AH-C-4B
- Challenge AR-A2025300, EHC-HPU Cooling
- Challenge AR-A2103106, CW-V-3 Will Not Open >50% in Auto
- Challenge AR-A2108357, FS-P-4 Did Not Start on Low Pressure Set Point of 108#
- Challenge AR-A2087712, Replace Aux Boiler Controls to Improve Challenges
- Challenge AR-A2083733, CW-P-4A (Amertap Recirculation Pump) Tripping
- Control Room Deficiency AR-A2128458, ICS/NNI Power Loss Alarm

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19 - 3 samples)

a. Inspection Scope

The inspectors reviewed and/or observed three post-maintenance test (PMT) samples 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 and/or evaluated:

- On November 1, procedure OP-TM-216-202, “Pressure Isolation Test of CF-V-4B, CF-V-5B, and DH-V-22B,” Rev. 5 was performed following maintenance to address excessive leakage from CF-V-5B.

- On November 10, procedure OP-TM-211-211, “High Pressure Injection (HPI) Test,” Interim Change 19260 was performed following replacement of HPI injection valves MN-V-16A/B/C/D.

- On November 11, procedure 1303-11.11, “Station Battery Load Test,” Rev. 30 was performed following replacement of the ‘B’ 120/250 volt station battery in accordance with work order C2007871.

b. Findings

No findings of significance were identified.

1R20 Refueling and Other Outage Activities (71111.20 - 1 sample)

a. Inspection Scope

The inspectors performed one inspection sample. Station personnel conducted the 1R16 from October 24 to November 18. The inspectors reviewed selected reactor shutdown, refueling, outage maintenance, and reactor startup activities to determine whether shutdown safety functions (e.g., reactor decay heat removal, reactivity control, electrical power availability, reactor coolant inventory, spent fuel cooling, and containment integrity) were properly maintained as required by TSs and AmerGen Topical Report 097, “TMI-1 Outage Fuel Protection Criteria,” Rev. 10. 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 RCS inventory due to the associated increase in shutdown risk. In addition, the inspectors attended the morning plant status meetings, and reviewed IR-297485, which provided engineering limits for containment loading while at power. The inspectors reviewed AmerGen Administrative Procedure AP-1015, “Equipment Storage Inside Class I Buildings,” Rev. 2, and IRs 387939, 388006, 388791, 388916, 388407, and 395100, which documented and evaluated several discrepancies identified by the inspectors and by plant personnel, regarding deficient controls of materials brought into containment. The inspectors also performed inspections of all accessible areas inside containment, interviewed applicable engineers, supervisors and plant operators, and consulted with NRC specialists. Additional documents reviewed during the inspection are listed in the Attachment. Specific activities evaluated included:

- Safety Shutdown Review Board review of the TMI-1 1R16 Outage Risk Profile conducted on October 14, 2005
- Plant cooldown per procedure 1102-11, “Plant Cooldown,” Rev. 135
- RCS drain to mid-loop per procedure 1103-11, “RCS Water Level Control,” Rev. 63
- Fuel offload and reload, per refueling procedure 1505-1, “Fuel and Control Component Shuffles,” Rev. 44
- End of cycle 15, fuel assembly grid strap damage evaluation 51-9005734-000 (IR 393915) and post-core reload grid strap debris evaluation 51-9006115-000 (IR 396846)
- Reactor building emergency core cooling system sump inspection
- Reactor building walkdown to inspect for indication of RCS leakage and boric acid corrosion
- Plant heatup per procedure 1102-1, “Plant Heatup to 525 Degrees F,” Rev. 164
- Restoration of containment integrity in accordance with procedure 1101-3, “Containment Integrity and Access Limits,” Rev. 83
- Operating procedure 1102-2, “Plant Startup,” Rev. 144
- Power ascension per procedure 1102-4, “Power Operation,” Rev. 110
- AmerGen procedure OP-AA-108-108, “Unit Restart Review,” Rev. 5, following completion of 1R16

The inspectors also performed visual inspections of the reactor building containment liner during 1R16 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.

b. Findings

Introduction. The inspectors identified a Green non-cited violation (NCV) of TS 6.8.1 for multiple failures to properly implement procedural requirements and engineering instructions to ensure adequate controls of materials brought into the reactor building containment while the plant was at power. Specifically, maintenance personnel did not complete the Equipment Storage Data Sheets (ESDS) required per AmerGen procedure AP-1015, prior to placing the equipment inside containment. In addition, most of the containment loading limits established by engineering instructions documented in IRs 297485 and 345746 were not maintained.

Description. On October 6, the inspectors attended the morning plan-of-the-day meeting, and questioned discussions involving materials that would be brought into the reactor building (RB) containment in preparation for the upcoming refueling outage. Specifically, the inspectors questioned the controls in place to ensure plant design limits were not impacted, including the presence of zinc and potential for hydrogen generation due to scaffolding materials. Engineers initiated IRs 297485 and 345746 to address the inspectors' concerns. The inspectors noted that the engineers had established several limits for loading of materials in containment, but it was unclear how these limits were controlled. The limits established by engineering included: 1) all materials shall be stored in accordance with AP-1015, 2) an ESDS shall be initiated and processed per AP-1015, 3) scaffolding materials shall not be stored in the reactor building with unqualified coatings, 4) no scaffold materials made of aluminum are permitted in reactor building during power operations, 5) all lead shielding shall be stored in closed boxes, and 6) items shall be properly secured or otherwise proper distance shall be maintained to safety-related SSCs (including the RB containment liner).

Engineers determined that there was no supporting documentation to ensure proper controls of scaffolding materials and other materials that had already been loaded into the RB containment, and initiated IR 387939. Further engineering review identified that approximately 26,000 pounds of lead blankets which had been loaded into containment were not properly stored in closed metal boxes. A prompt investigation was initiated to address a RB sump operability concern due to potential sump screen blockage that could be caused by failure of the lead blanket vinyl covering material during a design basis loss of coolant accident (LOCA). Based on vendor-provided information regarding the lead blanket material properties and onsite exposure heat testing, engineers concluded that RB sump operability was not affected because the expected post-accident temperatures would not negatively affect the lead blanket coverings.

On October 20, the inspectors accompanied plant personnel during the extent-of-condition walkdowns. The licensee identified several other discrepancies with the procedural requirements of AP-1015, and most of the containment material load limits established by IRs 297485 and 345746 had not been properly implemented. These deficiencies increased the potential for increased combustible gas generation, RB sump blockage, and equipment damage during a seismic event. During these walkdowns, the inspectors identified a weakness in the licensee's extent-of-condition walkdown inspections, in that they did not notice several thousand pounds of aluminum toe kick plates which were specifically prohibited by engineering instructions documented in IR 345746. Other deficiencies or materials brought into containment included: unqualified coating materials used in scaffolding, tie-wraps, plastic bags, paper work orders, scaffold identification tags, electrical cables in plastic wrapping, several aluminum fiberglass ladders, and improperly tied scaffolding materials (IRs 395100, 388916, and 388407).

An operability evaluation of all these deficient conditions was documented in IR 388006. The engineering evaluation concluded that due to the stacked loading configuration of the aluminum plates, only a small fraction (24 pounds) of aluminum and zinc inventories brought into containment would generate hydrogen during a LOCA. This amount reduced the available equivalent margin specified in TMI calculation C-1101-901-5360-007, "Hydrogen Generation In Containment," (40 pounds of aluminum), by only half. Therefore, the evaluation concluded that the reactor containment remained operable. In addition, the engineering evaluation determined that the deficiencies did not impact the seismic analysis of the plant, and that the increased material would not have resulted in containment sump blockage.

Analysis. The inspectors determined that the failure to properly implement procedural and engineering instructions to ensure control of materials brought into the RB containment while the Unit was at full power operation, was more than minor because it affected the reliability objective of the equipment performance attribute under the mitigating systems cornerstone. The finding also affected the barrier integrity cornerstone objective for the containment barrier to protect the public from radionuclide releases by affecting the configuration control attribute of containment design. The inspectors evaluated the risk significance of this finding using NRC Manual Chapter 0609, "Significance Determination Process," Appendix A, Phase 1. The finding screened to very low safety significance (Green) because no equipment was rendered inoperable and no actual open pathway in the physical integrity of the reactor containment occurred. The inspectors also determined that the cause of the finding is related to the cross-cutting area of human performance, because station personnel did not comply with engineering instructions and established procedures for control of materials inside containment.

Enforcement: TS 6.8.1.a requires that written procedures shall be properly implemented covering 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, including containment and containment integrity. Contrary to this requirement, maintenance personnel did not properly implement procedural requirements and engineering instructions to ensure adequate control of materials brought into the RB containment while the plant was at full power operation. Because this violation was of very low safety significance and was entered into the TMI corrective action program (IRs 387939, 388006, 388791, 388916, 388407, and 395100), this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy. **NCV 05000289/2005009-01, Deficient Procedural Compliance Resulted in Inadequate Control of Materials Brought into the Reactor Building Containment**.

1R22 Surveillance Testing (71111.22 - 5 samples)

a. Inspection Scope

The inspectors performed five inspection samples. 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. Inspection activities included review of previous surveillance history to identify previous problems and trends, observation of pre-evolution briefings, and initiation/resolution of related IRs for selected surveillances. Additional documents reviewed during the inspection are listed in the Attachment.

- On October 13, procedure 1303-11.3, “Main Steam Safety Valves,” Interim Change 19064 (IR 385339)
- On October 29, procedure OP-TM-213-203, “Core Flood Train ‘A’ Flow Test,” Rev. 2
- On November 6, OP-TM-212-213, “DH-P-1A Refueling IST,” Rev. 3
- On November 11, testing of the ‘C’ nuclear service closed cooling pump in accordance with OP-TM-541-209, “IST Of NS Pumps During Cold Shutdown,” Rev. 0
- On November 15, procedure OP-TM-421-241, “Shutdown IST Of FW And S/U Valves,” Rev. 3, after replacement of leaking volume booster for main feedwater valve FW-V-16B

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 1 sample)

a. Inspection Scope

The inspectors selected one sample for review. The inspectors reviewed the following temporary modification (TM) and associated implementing documents to verify the plant design basis and the system or component operability was maintained. Procedures CC-AA-112, “Temporary Configuration Changes,” Rev. 8 and CC-TM-112-1001, “Temporary Configuration Change Implementation,” Rev. 1, specified requirements for development and installation of TMs.

- TM 05-00737, “FH-A-4A Limit Switch,” Rev. 0. This modification substituted operator-controlled manual switches on the east fuel transfer carriage, in place of an automatic limit switch which had failed, to permit continued reactor fuel offload activities until a permanent repair could be made.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness [EP]

1EP6 Drill Evaluation (71114.06 - 1 sample)

a. Inspection Scope

The inspectors performed one inspection sample. The inspectors observed an emergency event training evolution conducted at the Unit 1 control room simulator to evaluate emergency procedure implementation, event classification, and event notification. The event scenario involved multiple safety-related component failures and plant conditions warranting a simulated General Emergency event declaration. The licensee counted this training evolution for evaluation of Emergency Preparedness Drill/Exercise Performance Indicators (PIs). 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 Drill/Exercise PIs were properly evaluated consistent with Nuclear Energy Institute 99-02, “Regulatory Assessment Performance Indicator Guideline,” Rev. 3.

- On November 28, the inspectors evaluated Emergency Preparedness Training Drill No. 4, “Loss Of Feedwater Requiring High Pressure Injection PORV Cooling,” Rev. 11, from the control room simulator.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

Documents reviewed during the inspection are listed in the attachment.

2OS1 Access Controls (71121.01 - 21 samples)

a. Inspection Scope

The inspectors reviewed selected activities and associated documentation in the below listed areas. The evaluation of AmerGen's performance in these areas was against criteria contained in 10 CFR 20, applicable TSs, and applicable AmerGen procedures.

Inspection Planning - Performance Indicators (PIs)

The inspectors selectively reviewed PIs for the occupational exposure cornerstone. The inspectors also discussed and reviewed current performance with cognizant AmerGen personnel. (See Section 4OA1)

Plant Walkdowns, Radiation Work Permit (RWP) Reviews, and Jobs in Progress Reviews

The inspectors walked down selected radiological controlled areas and reviewed housekeeping, material conditions, posting, barricading, and access controls to radiological areas. The inspectors reviewed exposure-significant work areas to determine if radiological controls were acceptable and conducted selective radiation surveys with a survey instrument. The inspectors selectively walked down these areas to determine the adequacy of radiological controls (surveys, postings, barricades).

The inspectors selectively reviewed the adequacy of applied radiological controls performance for tasks completed during the outage relative to results achieved. Activities reviewed included reactor disassembly, ISI activities, refueling activities, valve work activities, turbine work activities, steam generator work activities including nozzle dam installation, shielding, radiation protection job coverage, thermal sleeve work, pressurizer work activities, and in-core detector changeout. The reviews included evaluation of the adequacy of all applied radiological controls, including radiation work permits, procedure adherence, radiological surveys, job coverage, system breach surveys, airborne radioactivity sampling and controls, and contamination controls. The inspectors reviewed exposure results for previous (1997) in-core detector change-outs. The reviews included, where applicable, barrier integrity and the application of engineering controls for potential airborne radioactivity areas. The reviews included evaluation of controls based on radioactive source term and radiation levels present. The inspectors also selectively reviewed radiological data associated with reactor building entries at power.

The inspectors also reviewed applicable RWPs and electronic personnel dosimetry alarm setpoints (both integrated dose and dose rate) to verify that the setpoints were commensurate with ambient/expected conditions, radiation work permits, plant policy, and were appropriate for the conditions. The inspectors selectively interviewed workers to verify if workers knew what actions were required when their dosimeters alarmed and if the workers knew the working ambient radiological conditions. The inspectors observed portions of the worker briefings for work activities.

The inspectors reviewed, observed, and discussed ongoing work in TS controlled High Radiation Areas, including the reactor building. The inspectors reviewed radiation protection job coverage, including use of audio and visual surveillance.

The inspectors reviewed work activities with radiation dose rate gradients, as applicable, to verify that AmerGen had applied appropriate radiological controls, including use of multiple dosimeters or repositioning of dosimetry, as appropriate, to accurately measure radiation doses. The inspectors reviewed posting and locking of entrances to high dose rate and very high radiation areas, as appropriate. The inspectors selectively reviewed high radiation area controls for underwater work.

The inspectors reviewed and discussed internal dose assessments for 2005, including the outage, to identify any apparent actual occupational internal doses greater than 50 millirem committed effective dose equivalent. The review also included the adequacy of evaluation of selected dose assessments, as appropriate, and included selected review of the program for evaluation of potential intakes associated with hard-to-detect radionuclides (e.g., airborne transuranics).

<u>High Risk Significant, High Dose Rate, High Radiation Area and Very High Radiation Area Controls</u>

The inspectors discussed procedure changes for High Radiation Area Access controls since the last inspection with the Radiation Protection Manager and selected supervisors to determine if the changes resulted in a reduction in the effectiveness and level of worker protection. The inspectors conducted a selective review of High Radiation Area controls (e.g., adequate posting and locking of entrances). The inspectors discussed controls for High and Very High Radiation Areas also with radiation protection technicians. The inspectors reviewed the key inventory for High and Very High Radiation Areas.

<u>Radiation Worker/Radiation Protection Technician Performance and Radiation Protection Technician Proficiency</u>

The inspectors observed radiation worker performance with respect to stated radiation protection requirements to determine if workers were aware of significant radiological conditions in their work place, the RWP controls/limits in place, and that their performance took into consideration the levels of radiological hazards present. The inspectors also evaluated radiation protection technician performance and proficiency relative to control of hazards and work activities, as applicable. In addition, the inspectors reviewed problem reports to identify problems with worker or radiation protection technician performance. The inspectors questioned both radiation workers and radiation protection personnel regarding on-going activities and knowledge of controls and conditions as applicable. The inspectors also reviewed post-outage problem reports to identify problems with worker or radiation protection technician performance.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02 - 13 samples)

a. Inspection Scope

The inspectors conducted the following activities to determine if AmerGen was properly implementing operational, engineering, and administrative controls to maintain personnel occupational radiation exposure as low as is reasonably achievable (ALARA). Implementation of these controls was reviewed against the criteria contained in 10 CFR 20, applicable industry standards, and applicable AmerGen procedures.

Inspection Planning, Radiological Work Planning

The inspectors reviewed pertinent information, since the previous inspection, regarding plant collective exposure history, current exposure trends, and ongoing and planned activities in order to assess current performance and exposure challenges. The inspectors determined the plant's current three-year rolling average collective exposure for the period January 2002 - December 2004. The inspectors evaluated site specific trends in collective exposures (using NUREG-0713 and plant historical data).

The inspectors reviewed planning and preparation for the outage. The inspectors selected work activities likely to result in the highest personnel collective exposures and selectively reviewed the planning and preparation for those work activities. The inspectors evaluated the level of detail associated with projected dose estimation. The work activities reviewed included reactor disassembly, scaffolding, radiological controls coverage, steam generator work activities, and pressurizer work activities.

The inspectors reviewed site specific procedures associated with maintaining occupational exposure ALARA, including processes used to estimate and track work activity specific exposures.

The inspectors selectively compared person-hour estimates for work activity planning with actual work activity time requirements. The inspectors evaluated the accuracy of these time estimates. The work activities reviewed included reactor disassembly, scaffolding, radiological controls coverage, steam generator work activities, and pressurizer work activities.

Job Site Inspections and ALARA Controls

The inspectors reviewed ongoing outage work activities and selected work activities likely to result in the highest personnel collective exposures or presented challenges for ALARA control, and reviewed the current and expected collective radiation exposure for these work activities. The work activities reviewed included reactor disassembly, scaffolding, radiation protection job coverage, in-service inspection, steam generator work activities, and pressurizer work activities. The inspectors also reviewed work activities that presented potential unusual conditions or situations (i.e., in-core detector changeout). The inspectors selectively reviewed mock-up training and use of mock-ups. The inspectors selectively reviewed implementation of applicable ALARA plans and procedures for these activities including tracking of exposures. The inspectors reviewed ALARA work activity evaluations, exposure estimates, and mitigation requirements. The inspectors evaluated the adequacy of AmerGen's engineering and work controls and the grouping of the activities relative to work activity. The inspectors reviewed the integration and implementation of ALARA requirements into procedures and RWP documents.

The inspectors toured selected areas of the radiologically controlled area, including the RB, and observed ongoing radiological work activity. The inspectors evaluated whether workers were using low dose waiting areas, were effective in maintaining their doses ALARA, and received appropriate on-the-job supervision to ensure ALARA requirements were met. The inspectors made independent radiation dose rate level measurements to evaluate ambient radiological conditions and dose reduction efforts.

The inspectors reviewed exposures of individuals from selected work groups and evaluated variations in exposure results to identify significant variations indicating potential ALARA work planning issues. The work activities reviewed included reactor disassembly, scaffolding, radiation protection job coverage, in-service inspection, steam generator work activities, and pressurizer work activities.

Verification of Dose Estimates and Exposure Tracking

The inspectors reviewed AmerGen's method for adjusting exposure estimates, or replanning work, when unexpected changes in scope, radiation levels, or emergent work were encountered to determine if the adjustments were based on sound radiation protection and ALARA principles. The inspectors also reviewed the frequency of these adjustments to evaluate the original ALARA planning process. The inspectors reviewed re-forecasts of work activity dose estimates as a result of lower radiation dose rates encountered during plant shutdown.

The inspectors reviewed the refueling outage revised dose estimates. The inspectors compared the results achieved (person-rem, dose rates) with estimated exposures and determined the reasons for inconsistencies between intended and actual exposure. The comparison included evaluation of person-hour estimates, expected dose rates, emergent work, and use of supplemental shielding, as necessary. The inspectors evaluated the reasons for inconsistencies between intended and actual work activity doses.

The inspectors determined if work activity planning included consideration of the benefits of dose rate reduction activities, such as shielding provided by water filled components/piping, job scheduling, and shielding and scaffolding installation and removal activities.

The inspectors evaluated the interfaces between operations, radiation protection, maintenance, maintenance planning, scheduling, and engineering groups for interface problems or missing program elements.

Source-Term Reduction and Control

The inspectors reviewed and discussed AmerGen's understanding of the Unit 1 plant source-term, including knowledge of input mechanisms to reduce the source term; and the source-term control strategy in place. The inspectors selectively reviewed and discussed AmerGen's cobalt reduction strategy designed to minimize the source-term external to the core. The inspectors discussed reasons for significantly lower effective radiation dose rates as compared to initial estimates. Also reviewed were fluid clean-up methods used to remove radioactivity. The inspectors evaluated dose reduction results achieved against priorities since the last refueling cycle.

The inspectors reviewed Station ALARA Council Meeting Minutes for 2005.

The inspectors reviewed and discussed the effectiveness of AmerGen's supplementary shielding, flushing strategies, filtration efforts, and work control/deferrals to minimize the impacts on person-rem. The inspectors reviewed the site ALARA procedures including job exposure estimates and tracking. The inspectors evaluated AmerGen's use of engineering controls to achieve dose reductions.

The inspectors selectively reviewed and discussed AmerGen's source term reduction strategy designed to minimize the source-term external to the core and results achieved for the outage. The inspectors discussed reasons for significantly lower effective radiation dose rates as compared to initial estimates.

Declared Pregnant Workers

The inspectors determined if there were any declared pregnant workers during the assessment period to evaluate exposure controls.

Radiation Worker/Radiation Protection Technician Performance

The inspectors selectively observed radiation worker and radiation protection technician performance in the area of ALARA practices to identify acceptable performance in areas of greatest radiological risk to workers. The inspectors selectively questioned workers and radiation protection personnel in-the-field to evaluate their understanding of ambient radiological conditions. The inspectors evaluated performance to determine whether the training/skill level was sufficient with respect to the radiological hazards involved.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03 - 5 samples)

a. Inspection Scope

The inspectors selectively reviewed radiation monitoring/measurement instrumentation in the below listed areas. The review was against criteria contained in applicable TSs and station procedures.

Inspection Planning/Identification of Additional Radiation Monitoring Equipment

The inspectors selectively reviewed the station's Updated Final Safety Analysis Report (UFSAR) to identify applicable radiation monitoring equipment for review and evaluation. The inspectors identified types of portable radiation detection instrumentation used for job coverage of high radiation area work, temporary radiation monitors, and air monitoring equipment.

Verification of Instrument Calibration, Operability, and Alarm Setpoint Verification

The inspectors reviewed calibration and operability check records for a variety of radiological survey instrumentation in use for radiological job coverage and area monitoring during the outage. The instrumentation included portable survey meters, scaler-counters, and portable area radiation monitors. The inspectors evaluated the adequacy of calibration sources used relative to the in-plant source term. The review include actions taken for out-of-specification instruments. (Ludlum 102763, 99198; Bicron RSO-50E B864W, SAC-4 794, 394; NRD G40862)

The instrumentation reviewed included risk significant area radiation monitors (RM-G-09, RM-G-10, RM-G-11, RM-G-12), high range containment monitors (RMG-22, RMG-23), personnel air samplers (12015, 1518, 2152), personnel alarming dosimeters (28548, 35205, 30013, 37697), gamma spectroscopy source calibration (No. 3), area airborne radioactivity monitor (6087-22), and personnel contamination monitors (PCM1B 1226, 1555; PM-7 448).

Radiation Protection Technician Instrument Use

The inspectors selectively verified the calibration expiration and source check response on radiation detection instruments staged for use for the outage. The inspectors observed radiation protection technicians for appropriate instrument selection and use, including self-verification of instrument operability.

Problem Identification and Resolution

The inspectors reviewed problem reports in this area since the last inspection to determine if AmerGen was including instrument deficiencies and issues in its corrective action program (Section 4OA2). The review included self-assessments, audits and corrective action reports.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation (71122.02 - 1 sample)

a. Inspection Scope

The inspectors selectively reviewed the preparation for shipment and visually inspected an exclusive use, non-excepted shipment (RS-05-173-I, LSA II) of material. The inspectors reviewed the preparation and shipment relative to applicable Department of Transportation requirements. The inspectors reviewed shipment paperwork and conducted independent radiation surveys of the shipment.

b. Findings

No findings of significance were identified

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151, 71121.01 - 2 samples)

.1 Occupational Exposure Control Effectiveness

a. Inspection Scope

The implementation of the Occupational Exposure Control Effectiveness Performance Indicator Program was reviewed. The inspectors reviewed corrective action program records for occurrences involving High Radiation Areas, Very High Radiation Areas, and unplanned personnel radiation exposures since the last inspection in this area. Data reviewed included the period November 2004 to October 2005. The inspectors also selectively reviewed exposure records. The review was against the applicable criteria specified in NEI 99-02, “Regulatory Assessment Performance Indicator Guideline,” Rev. 3. The purpose of this review was to verify that occurrences that met NEI criteria were recognized and identified as Performance Indicator occurrences.

b. Findings

No findings of significance were identified.

.2 RETS/ODCM Radiological Effluent Occurrences

a. Inspection Scope

The implementation of the radiological effluent treatment system/offsite dose calculation manual (RETS/ODCM) Performance Indicator (PI) was reviewed. The inspectors reviewed corrective action program records and projected monthly and quarterly dose assessment results due to radioactive liquid and gaseous effluent releases; for the fourth quarter 2004 to the fourth quarter 2005. The inspectors also evaluated potential for unmonitored releases and selectively reviewed the 2004 and 2005 Annual Effluent Release Reports. The review was against the applicable criteria specified in NEI 99-02. The purpose of this review was to verify that occurrences that met NEI criteria were recognized and identified as PI occurrences.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Review of Issue Reports and Cross-References to PI&R Issues Reviewed Elsewhere

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 a list of daily issue reports, by reviewing selected issue reports, attending daily screening meetings, and accessing the licensee's computerized database. Documents reviewed are listed in the Attachment.

Section 1R20 describes a finding for multiple failures to properly implement procedural requirements and engineering instructions to ensure control of materials brought into the reactor building containment while the plant was at power. Problem resolution of this finding was deficient in that station personnel did not identify large quantities of aluminum materials stored in containment during their extent-of-condition walkdowns to address the finding.

Section 4OA2.3 describes a finding for station procedures not containing controls to verify and/or maintain the required EQ configuration associated with MOV actuator T-drains. Problem resolution of this finding was deficient because the engineering evaluation did not identify the cause that resulted in the T-drains for four MOVs not being installed.

.2 Semi-Annual Review to Identify Trends

a. Inspection Scope

The inspectors performed a semi-annual review of common cause issues in order to identify any unusual trends that might indicate the existence of a more significant safety issue. This review included an evaluation of repetitive issues identified via the corrective action program, self revealing issues, and issues evaluated using programs supplemental to the formal corrective action program such as the maintenance rule program and corrective maintenance program. The results of the trending review were compared with the results of normal baseline inspections.

b. Assessment and Observations

NRC Inspection Report 05000289/2005005 previously documented a trend of procedure quality and usage deficiencies. Although the number of procedure problems has lessened during the past 6 months, it remains a continuing problem. Examples are: (1) deficient procedures for implementing NRC GL 89-13 inspection program requirements including visual inspections of the reactor building emergency air coolers (IR 371356) and acceptance criteria for the nuclear service system heat exchanger cleanliness (IR 431684); (2) deficient procedure for maintaining and verifying safety related motor operated valve (MOV) environmental qualification (four MOV actuators were missing t-drains) (IRs 238918, 267293, 273768, 391720, 391707, and 271819); (3) violation of station procedures to install charcoal filters in the safety related control building ventilation system (IR 349025); (4) noncompliance with station scaffold procedures (IR 388416); (5) violation of station fire protection program fire watch procedures (IRs 399726, 428991, and 429891); (6) deficient procedure usage level for swap of nuclear river water pumps led to unplanned pump inoperability; (7) lack of procedural controls during insulation removal to verify environmental qualification is maintained (IR 374104); (8) deficient procedure for periodic main steam safety valve (MSSV) setpoint test resulted in operations personnel not recognizing that a MSSV was inoperable (IR 385339); (9) deficient procedure for CF-V-5A(B) leak testing did not document duration of leakage collection to support leak rate calculation; and (10) deficient procedural compliance for control of materials brought into containment prior to the 1R 16 refueling outage. Several of these issues were documented as NRC findings. The inspectors discussed these and similar procedure issues with station management throughout the inspection period.

One causal factor for the elevated number of procedure errors is that the station has multiple (four) standard procedure formats currently in use for active procedures. Each format provides a different level of procedure instruction. Several formats are hold-overs from previous station owners. TMI recently implemented a standard procedure writers' guide and several procedure upgrade projects to standardize and improve the quality of procedures. The inspectors reviewed schedules for the Operations and Maintenance Department procedure upgrade projects. The inspectors noted that an integrated long term plan for the project(s) does not exist. Instead, each department develops a plan annually for the procedure upgrade work to be performed in that year. The station currently plans to standardize/upgrade all procedures by 2009.

Station management increased the number of supervisory field observations and has emphasized the importance of procedure usage during routine supervisory briefings. Station personnel adopted a lower threshold for identifying procedure quality or usage deficiencies during the last half of 2005. Accordingly, a larger number of items are being identified and addressed through the corrective action program. The inspectors observed improved procedure compliance in the field during inspection samples. However, the inspectors observed that the licensee has not performed or scheduled effectiveness reviews to reassess appropriateness of (or redirect) corrective actions, and the procedure upgrade schedule appeared lengthy. These observations were discussed with station management.

Additionally, the inspectors noted several deficiencies in the area of problem resolution. AmerGen evaluation of several degraded equipment conditions was either not timely or was too narrowly focused. Examples included (1) incomplete assessment of the cause of the AH-F-3B filter failure (IR 3490254), (2) lack of control of as-found test conditions to evaluate degradation of startup feedwater regulating valve FW-V-16B (IR 391999), (3) incomplete assessment of the cause of missing T-drains from four safety related MOV actuators (IRs 238918, 271819, 291707, 291720 and 441946), (4) incorrect safety system functional failure assessment of AH-E-18B (IR 434685), and (5) extent of condition walkdown inspections for materials improperly brought into containment in preparation for 1R16 did not identify several thousand pounds of aluminum toe kick plates which were specifically prohibited by engineering instructions (IR 345746).

.3 Annual Sample: Non-1E Electrical Equipment Powered from 1E Sources

a. Inspection Scope

The inspectors reviewed two IRs related to control room data recorders being powered from safety related (1E) or non-safety related sources. Specifically, the inspectors reviewed IR 289346 which documented that non-safety related Nuclear Instrument recorders (NI3NIR and NI5NIR), were powered from 1E power supplies. The inspectors also reviewed IR 290797 which identified that a safety related recorder (FW-LR-1083) had been improperly declassified as non-safety related during a previous design modification. The inspectors verified that this recorder was properly connected to a safety related (1E) power supply. The inspectors reviewed the IRs to ensure the full extent of the issues as identified and that appropriate corrective actions were specified. In addition, the inspectors reviewed pertinent engineering documentation and interviewed station personnel. Additional documents reviewed during the inspection are listed in the Attachment.

b. Findings and Observations

No findings of significance were identified. However, the inspectors noted that the extent-of-condition evaluations and the engineering assessments documented in the IRs lacked sufficient detail. This created ambiguity with regard to actions taken to correct the deficiencies. As a result, the deficiency which resulted in IR 290797 was not properly captured and was a duplicate of the original IR (289346). The inspectors determined that despite the administrative inaccuracies, the actions taken for the IRs adequately encompassed the necessary actions to correct the deficiencies discovered. Therefore, this issue is considered minor and no findings of significance were identified.

.4 Annual Sample: Equipment Qualification Not Maintained on Containment Isolation Valves Due to Deficient Procedures Or Instructions

a. Inspection Scope

Issue Report 238918, documented an inspector identified concern, regarding the lack of procedures or instructions to ensure that T-drains for environmentally qualified (EQ) motor-operated valves (MOVs) were properly installed and inspected. The inspectors reviewed AmerGen’s evaluation and corrective actions regarding four containment isolation valves that did not have the required environmental qualification T-drains. This issue was selected based on its potential for impacting the reactor building Barrier Integrity and Mitigating Systems cornerstones. The inspectors performed field walkdowns of several MOV’s inside the reactor building containment, interviewed the MOV and EQ engineers, interviewed plant operators, and reviewed the following documents:

- AmerGen procedure MA-AA-723-301, “Periodic Inspection of Limitorque Model SMB/SB/SBD-000 Through 5 Motor Operated Valves,” Rev. 0
- TI-103, “TMI-1 Environmental Qualification Report,” Rev. 5
- IRs 238918, 267293, 273768, 391720, 391707, and 271819

b. Findings and Observations

Introduction. The inspectors identified a non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” because station procedures did not contain controls to verify and/or maintain the required EQ configuration associated with MOV actuator T-drains. As a result, four safety-related MOV valve actuators used for containment isolation did not have required T-drains.

Description. Engineering document, for MOVs, TI-103, “TMI-1, Environmental Qualification,” Rev. 5, Section A, SRS A-1, states in part, that T-drains must be installed to reflect the tested configuration for containment applications with Class RH insulation motors (Ref. EQ-T1-103-04, Paragraphs 4.1.2 and 8.0).” This document also states that T-drains are used to allow water (condensation) to drain out of the motor compartment following exposure to a postulated harsh environment, and to allow the motor to “breath.”

In July 2004, the inspectors requested information from the TMI environmental qualification engineer regarding controls and procedures used to ensure that the required T-drains were properly installed, maintained, and inspected. In addition, the inspectors questioned if TMI had specific controls to ensure that during maintenance activities, the required T-drains were installed in the proper orientation and on the lowest point of the assembly for proper draining of condensation. The engineering review of applicable information to address the inspectors' questions determined that there were a total of 21 MOVs at TMI with Class RH insulation that required T-drains, and that these MOVs were inspected per AmerGen procedure MA-AA-723-301, "Periodic Inspection of Limitorque Model SMB/SB/SBD-000 Through 5 Motor Operated Valves," Rev. 0. The engineer also determined that there were no specific procedures, instructions or drawings to ensure that the required T-drains were installed, nor to ensure that during maintenance activities, the required T-drains were installed in the proper orientation and on the lowest point. The engineer initiated IR 238918 to address these issues. In addition, a series of documentation reviews and actions were initiated to perform visual inspections and an extent of condition review of all applicable MOVs inside containment. These activities and subsequent discrepancies identified were documented and evaluated per IR's 391720, 391707, and 271819. The engineering walkdowns identified that the following four containment isolation MOVs did not have the required T-drains installed.

a. Make-up valve MU-V-2A, Reactor coolant pump letdown containment isolation
b. Make-up valve MU-V-2B, Reactor coolant pump letdown containment isolation
c. Make-up valve MU-V-25, Reactor coolant pump seal return containment isolation
d. Reactor coolant drain tank vent valve WDG-V-3 containment isolation

An engineering assessment determined that during a bounding accident condition such as a large break LOCA, the containment isolation safety function for all the valves would have been automatically achieved via a high reactor containment pressure signal prior to the containment environment reaching the peak temperature and pressure. Therefore, the valves remained operable. However, the inspectors noted that reliability of the MOV's was impacted in that for small break LOCA conditions, where the MOVs would also be exposed to harsh environments (high temperature and pressure) the pressure would not reach the automatic set point value and manual operator action would be required to perform the containment isolation function. The inspectors verified that appropriate corrective actions have been implemented to correct the degraded condition which included: replacement of three valve motor actuators that did not have ports to install the required T-drains (MU-V-2B, MU-V-25 and WDG-V-3), and installation of the required T-drain for valve MU-V-2A. Additionally, the engineer initiated actions to ensure proper orientation and installation of required T-drains during future motor replacement activities.

Analysis. The lack of instruction or procedures to maintain the required EQ configuration for MOV actuators located inside the reactor building containment, and to ensure that the T-drains were properly oriented when motor replacement activities are performed, constituted a performance deficiency. As a result, four containment isolation MOVs were installed inside the reactor building without the required EQ T-drains.

The inspectors determined this issue was more than minor because it affected the barrier integrity cornerstone objective and the containment barrier performance attribute. Specifically, the lack of T-drains may allow moisture to enter the motor housing on a high temperature and pressure steam environment during a LOCA and electrically short out the motor, which reduces containment isolation reliability. In addition, if left uncorrected, this issue could become a more significant safety concern in that without procedures to maintain the required EQ configuration, additional MOV actuators could be installed with no T-drains or with T-drains in the incorrect orientation and thus lead to a failure of the valve to perform its design function. The inspectors evaluated the risk significance of this finding using NRC Manual chapter 0609, "Significance Determination Process," Appendix A, Phase 1. The finding screened to very low safety significance (Green) because the specific component qualification deficiency was determined not to result in an actual open pathway in the physical integrity of the reactor containment. This finding has been entered into the licensee's corrective action program as IRs 238918, 267293, 273768, 391720, 391707, and 271819.

The cause of the finding is related to the cross-cutting area of human performance, because AmerGen did not establish appropriate measures to ensure that required MOV T-drains were properly installed, maintained, and inspected. The inspectors noted that the engineering review and evaluation of this issue did not identify the cause(s) that resulted in the T-drains not been installed, and questioned the accuracy of the extent-of-condition review for EQ requirements. IR-441946 was initiated to address this issue.

Enforcement. 10 CFR 50, Appendix A, Criterion IV, "Environmental and Dynamic Effects Design Bases," requires, in part, that structures, systems and components (SSCs) important to safety shall be designed to be compatible with the environmental conditions associated with postulated accidents, including a LOCA. 10 CFR 50, Appendix B, Criterion III, "Design Control," requires that measures be established to assure the design basis for SSCs are correctly translated into specifications, drawings, procedures, and instructions. Contrary to the above, AmerGen did not provide instructions or procedures to maintain the required EQ configuration associated with T-drains for safety-related MOVs inside the reactor building containment. Consequently, four MOVs did not have T-drains. Because this violation was of very low safety significance and was entered into the TMI corrective action program (IRs 238918, 391720, 391707, and 271819), this violation is being treated as an NCV consistent with Section VI.A of the NRC Enforcement Policy. **NCV 05000289/2005009-02, Equipment Qualification Not Maintained On Four Containment Isolation Valves, due To Deficient Procedures or Instructions.**

.5 Radiation Safety

a. Inspection Scope

The inspectors selectively reviewed issue reports, self-assessments, post-outage self-assessments, and audits of access controls and the ALARA program since the previous inspection to determine if identified problems were entered into the corrective action program for resolution. The inspectors evaluated the database for repetitive deficiencies or significant individual deficiencies to determine if self-assessment activities were identifying and addressing the deficiencies. The review also included evaluation of data to determine if any problems involved PI events with dose rates greater that 25 R/hr at 30 centimeters, greater than 500 R/hr at 1 meter or unintended exposures greater than 100 millirem total effective dose equivalent, 5 rem shallow dose equivalent, or 1.5 rem lens dose equivalent. The inspectors also reviewed the corrective action database for non-PI radiological incidents to determine if follow-up activities were being conducted in an effective and timely manner consistent with radiological risk.

The review also included a review of issue reports since the last inspection which involved potential radiation worker or radiation protection personnel errors to determine if there was an observable pattern traceable to a similar cause. The review included an evaluation of corrective actions, as appropriate. Additional documents reviewed during the inspection are listed in the Attachment. This review was against criteria contained in 10 CFR 20, TSs, and the station procedures.

b. Findings

No findings of significance were identified.

.6 In-Service Inspection

a. Inspection Scope

The inspectors reviewed a sample of corrective action reports shown in Attachment 1. The inspectors selected IR 394447 to assess the “accept as is” disposition of a flaw discovered during this outage. The inspectors determined the condition identified during non-destructive testing (remote visual examination) was reported, characterized, and evaluated with engineering input to accept the condition without repair for an additional cycle with the stipulation that the component (thermal sleeve) would be examined during the next refuel outage.

b. Findings

No findings of significance were identified.

4OA4 Cross-Cutting Aspects of Findings

Human Performance

Section 1R20 describes a finding for multiple failures to properly implement procedural requirements and engineering instructions to ensure control of materials brought into the reactor building containment while the plant was at power. This finding is a cross-cutting issue in the area of human performance, because station personnel did not comply with engineering instructions and established procedures for control of materials inside containment.

Section 4OA2.4 describes a finding regarding four containment isolation MOV actuators that did not include the required EQ T-drains. This finding is a cross-cutting issue in the area of human performance, because AmerGen did not develop appropriate measures to verify and/or maintain the required EQ environmental configuration associated with MOV actuator T-drains.

Section 4OA5.2 describes a finding in which potentially disqualifying medical conditions for three licensed operators were not reported to the NRC. This finding is a cross-cutting issue in the area of human performance, because multiple station operators did not comply with established procedures for reporting of potential disqualifying medical conditions.

4OA5 Other Activities

.1 TI 2515/160 - Pressurizer Penetration Nozzles and Steam Space Piping Connections in U. S. Pressurized Water Reactors (PWR) (NRC Bulletin 2004-01)

a. Inspection Scope

The inspectors reviewed AmerGen’s response to NRC Bulletin 2004-01, which alerts PWR licensees of the potential need to supplement current inspection methods with additional measures to detect and adequately characterize flaws due to primary water stress corrosion cracking in pressurizer penetrations and steam space piping connections. The inspectors reviewed AmerGen’s pressurizer penetration examination procedure to determine that it provides adequate guidance and examination criteria to implement the examination plan. The inspectors interviewed examination personnel and reviewed training and qualification records to determine whether the personnel qualification process adequately prepared the assigned staff to perform the examination and determine relevance of indications identified.

The inspectors observed a portion of AmerGen’s inspection activities to assess performance of the pressurizer penetration examination procedure. The inspectors also reviewed photographs and examination reports to determine whether procedure implementation was effective for detection of leakage from the pressurizer penetrations and/or shell locations examined.

The inspectors selected ten penetrations (PORV nozzle, two safety relief nozzles, high point vent and spray nozzles, upper and lower level taps and three heater penetrations) to determine if the intersection of each penetration with the shell could be fully accessed to reliably perform a 360 degree examination of the intersection region. The inspectors determined by direct visual observation and review of photographs that the locations examined were free of dirt, debris, insulation, significant oxidation, and any material that could obstruct a 360 degree viewing of the penetrations and their intersection with the pressurizer shell. Residual insulation debris was noted in various locations but did not impede the ability to perform the inspection. The inspectors observed foreign material in the vicinity of the thermowell nozzle, which was further examined by additional VT-2 examination and determined to be splatter from the adjacent D-ring wall. No boron was noted as leaking from the thermowell nozzle.

The inspectors determined 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. Findings

No findings of significance were identified.

The specific reporting requirements of TI 2515/160 are documented in Attachment A-1.

.2 (CLOSED) URI 2004006-01: Potential Violation for Failure to Notify the NRC Regarding the Change in the Medical Status of a Licensed Operator as Required by 10 CFR 50.74

The inspectors reviewed URI 2004006-01 which concerned the late reporting of medical conditions for three licensed operators. This issue had been documented in NRC Inspection Report No. 05000289/2004006. Specifically, during a PI&R inspection, three IRs (164042, 189592 and 195798) documented examples where individuals did not disclose information related to potentially disqualifying conditions to the license coordinator.

Introduction. The inspectors identified that AmerGen did not notify the NRC of changes in the medical status of three licensed operators within 30 days. This finding was determined to be a non-cited Severity Level IV violation of 10 CFR 50.74.

Description. URI 05000289/2004006-01 described the late reporting of medical issues determined to be “disqualifying conditions” in ANS/ANSI 3.4. The requirement to report changes in medical conditions is contained in Section 3.2.1 of procedure OP-AA-105-101, “Administrative Process for NRC License and Medical Requirements,” Rev. 8. This procedure implements requirements for 10 CFR Part 55 and 50.74. Each of the individual licensees has a license condition to “observe the operating procedures and other conditions specified in the facility license.”

The team identified that for three operators, potentially disqualifying conditions were not reported to the NRC within the required 30-day time frame. The time when the individuals were aware of these conditions until the reporting requirement was recognized and reported to the NRC ranged from three to six months. In all cases, the conditions were adequately controlled by medical treatment. The three individuals' medical conditions were discovered by the facility licensee during routine annual physicals greater than 30 days after the respective medical conditions actually occurred. The facility licensee reported these medical conditions within 30 days of learning of the condition. There is no requirement for the individual licensees to directly report changes in medical condition to the NRC.

Analysis

The performance deficiency involved untimely reporting of changes in the medical status of licensed operators within the time required by 10 CFR 50.74. However, it was noted that the facility licensee "was timely of their reporting the medical condition(s) to the NRC when they received the pertinent information." Further, this issue was self-identified and placed in their corrective action program.

Three licensed operators did not report changing medical conditions to the facility licensee as required in facility procedures. The requirement to report changes in medical conditions is contained in Section 3.2.1 of procedure OP-AA-105-101. This procedure implements requirements for 10 CFR Part 55 and 50.25. Each of the individual licensees has a license condition to "observe the operating procedures and other conditions specified in the facility license..."

Operator A required medication prescribed in April 2003. Furthermore, he unilaterally stopped taking the medication in June 2003. Operator B had a fainting spell at work on September 13, 2003 (apparently associated with pre-existing diabetes), and was taken to a hospital. The notification to the NRC was not made until December 18, 2003, and resulted in an additional restriction on his license. Operator C was diagnosed by his physician as having sleep apnea. He then had an operation to alleviate the condition in December 2003.

Traditional enforcement applies because it had the potential for affecting the NRC's ability to perform its regulatory function. NRC review determined the finding to be a Level IV violation consistent with Supplement I.D of the Enforcement Policy because there were multiple examples and Operator B required a change to his license. However, in no case did the operators stand watch without the medical condition being satisfied. The inspectors determined that the cause of the finding is related to the cross-cutting area of corrective actions, because it occurred after completion of actions to address a previous NCV for the failure to notify NRC of change in medical status of licensed operators. The inspectors also determined that the cause of the finding is related to the cross-cutting area of human performance, because multiple station operators did not comply with established procedures for reporting of potentially disqualifying medical conditions.

Enforcement. 10 CFR 50.74 requires, in part, that "each licensee shall notify the Commission within 30 days of a licensed operator or senior operator having a permanent disability or illness as described in 10 CFR 55.25." 10 CFR 55.25 requires "If during the term of the license, the licensee develops a permanent physical or mental condition that causes the licensee to fail to meet the requirements of 10 CFR 55.21 of this part, the facility licensee shall notify the commission of learning of the diagnosis." ANSI/ANS 3.4 identifies "disqualifying conditions" as medical conditions that, potentially, do not meet the requirements of 10 CFR 55.21. The facility licensee has committed to using Rev. 8 of ANSI/ANS 3.4 when conducting medical examinations to satisfy 10 CFR 55.21. All three operators had medical conditions that would be classified as "disqualifying conditions" in ANSI/ANS 3.4., Rev. 8.

Contrary to the above, on three occasions between April and December 2003, a "disqualifying condition," associated with a licensed operator existed, and the facility licensee did not notify the NRC within 30 days. This Severity Level IV violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Manual. The licensee took prompt corrective actions and entered this issue into their corrective action program (IRs 164042, 189592 and 195798). Although the violation was repetitive as a result of inadequate corrective actions, it was not initially identified by NRC, and it was not willful. In addition, the licensee conducted a 100 percent review of all operator medical records to address any other discrepancies identified.
NCV 05000289/2005009-03, Failure to Report Medical Conditions for Three Licensed Operators.

.3 <u>TI 2515/161 - Transportation of Reactor Control Rod Drives in Type A Packages</u> (1 sample)

a. <u>Inspection Scope</u>

The inspectors completed Phase 1 of the Temporary Instruction. The inspectors interviewed cognizant personnel and determined that AmerGen had undergone refueling/defueling activities between January 1, 2002 and present. However, it had not shipped irradiated control rod drives.

b. <u>Findings</u>

No findings of significance were identified.

.4 <u>Review of Institute of Nuclear Power Operations Plant Assessment and Operator Training Accreditation Review Update</u>

The Institute of Nuclear Power Operations (INPO) performed a TMI plant assessment during the period March 28 to April 8, 2005. The final INPO assessment report was issued on September 27, 2005. The inspectors reviewed the interim and final plant assessment reports. Problems identified in the reports were consistent with NRC findings and no new safety issues were identified. Additionally, on June 15, 2005, the INPO National Nuclear Accrediting Board reaccredited the TMI operator training programs. The reassessment and accreditation were completed following successful implementation of an operator training betterment program.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On January 9, 2006, the resident inspectors presented the inspection results to Mr. Rusty West and other members of the TMI staff who acknowledged the findings. The regional specialist inspection results were previously presented to members of AmerGen management. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified. Additionally, on January 10, 2006, NRC Region I staff from the Division of Reactor Safety conducted a telephone exit with Mr. Glenn Chick and other members of the TMI staff to debrief the inspection closure of URI 2004006-01.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

<u>Licensee Personnel</u>
S. Baker, Radiation Protection Manager
G. Chick, Plant Manager
A. Miller, Regulatory Assurance
D. Mohre, Nuclear Oversight Services Manager
C. Smith, Regulatory Assurance Manager
R. Walton, Chemistry Manager
C. Wend, Radiation Protection Manager
R. West, Vice President, TMI Unit 1

<u>Others</u>
M. Murphy, Pennsylvania Department Bureau of Radiation Protection

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Opened and Closed</u>

05000289/2005009-01	NCV	Deficient Procedural Compliance Resulted in Inadequate Control of Materials Brought into the Reactor Building Containment (Section 1R20)
05000289/2005009-02	NCV	Equipment Qualification Not Maintained on Four Containment Isolation Valves due to Deficient Procedures or Instructions (Section 4OA2.4)
05000289/2005009-03	NCV	Failure to Report Medical Conditions for Three Licensed Operators (Section 4OA5)

<u>Closed</u>

05000289/2004006-01	URI	Potential Violation for Failure to Notify the NRC Regarding the Change in the Medical Status of a Licensed Operator as Required by 10 CFR 50.74

LIST OF DOCUMENTS REVIEWED

<u>Section 1R01: Adverse Weather Protection</u>
OP-TM-108-111-1001, "TMI Site Inaccessibility Plan," Rev. 1
OP-AA-108-111-1001, "Severe Weather and Natural Disaster Guidelines," Rev. 2
WC-AA-107, "Seasonal Readiness," Rev. 1

Section 1R04: Equipment Alignment
AP-1015, “Equipment Storage Inside Class I Buildings,” Rev. 2

Section 1R07: Heat Sink Performance

Issue Reports	092697	119242	152766	183325	264456
264708	287025	294475	320086	320094	383582
391840	396021	429488	430624	431058	431684
431906					

1202-32, “Emergency Procedure: Flood,” Rev. 61
1202-33, “Emergency Procedure: Tornado/High Winds,” Rev. 26
ST 1301-6.7, “Monitoring of Silt Buildup in River Water Screen House,” Rev. 19, 14-Jul-05
ST 1301-9.7, “Intake Pump House Floor, Silt Accumulation and Inspections,” Rev. 21, 14-Oct-02
1D-541-29-001, “Heat Exchanger NS-C1A Nuclear Service Closed Cooling Water Tube Plugging Record,” Rev. 3, 11-Feb-99
1D-541-29-002, “Heat Exchanger NS-C1B Nuclear Service Closed Cooling Water Tube Plugging Record,” Rev. 3, 11-Feb-99
1D-541-29-003, “Heat Exchanger NS-C1C Nuclear Service Closed Cooling Water Tube Plugging Record,” Rev. 4, 26-May-99
1D-541-29-004, “Heat Exchanger NS-C1D Nuclear Service Closed Cooling Water Tube Plugging Record,” Rev. 4, 11-Feb-99
1D-542-29-001, “Cooler IC-C-1A Intermediate Cooling Tube Plugging Record,” Rev. 4, 10-Dec-99
1D-542-29-002, “Cooler IC-C-1B Intermediate Cooling Tube Plugging Record,” Rev. 5, 7-Jan-02
PID 302-202, “Nuclear Services River Water System Flow Diagram,” Rev. 69, 05-Oct-05
Letter 6710-96-2097, Generic Letter 89-13 Revised Response, 06-Jun-96
C-1101-531-5310-009, “Evaluation of Nuclear Services Closed Cooling Heat Exchangers,” Rev. 1, 30-Aug-96
C-1101-531-E410-019, “Nuclear River Water System (NR) Pipe-Flo Model,” Rev. 1, 19-May-02
C-1101-531-E540-011, “Analysis of NSCCW Heat Exchangers,” Rev. 3, 12-Nov-98
C-1101-541-5310-024, “NSCCW Hydraulic Analysis (Minor Revision),” Rev. 1A, 29-May-03
C-1101-541-5360-020, “Effect of 95F River Water on NSCCW,” Rev. 2, 18-Dec-98
C-1101-542-E210-015, “IC-C-0001A/B Tube and Shell Required Thickness,” Rev. 0, 16-Jan-98
M06, “Island Flood Control,” Rev. 8
M144, “Heat Exchanger Inspections and Cleaning,” Rev. 25
OP-TM-823-432, “Winter Operating Guidelines for Industrial Coolers,” Rev. 1
R1831795, Heat Exchanger Inspection and Clean (NS-C-1A), 22-Apr-04
R2011871, IC-C-1A: Heat Exchanger Inspection and Clean, 20-Nov-03
R2015103, NS-C-1C: Clean and Inspect (Eddy Current), 17-Mar-05
OP-TM-541-202, “IST of NSRW Pumps and Valves during Refuelings,” 28-Oct-05
TR-119, “Generic Letter 89-13 Program Description,” Rev. 3, 30-Mar-03
VM-TM-0041, “Yuba Heat Exchangers”
WC-AA-107, “Seasonal Readiness,” Rev. 1
Diagram, ICC1A 15R Eddy Current Inspection

Diagram, ICC1B 15R Eddy Current Inspection

Section 1R20: Refueling and Other Outage Activities
1103-2, “Vent of the Reactor Coolant System,” Rev. 82
1507-3, “Main Fuel Handling Bridge Operating Instructions,” Rev. 23
1507-5, “Spent Fuel Handling Bridge Operating Instructions,” Rev. 30
1507-7, “Fuel Transfer Systems Operating Instructions,” Rev. 24
OP-TM-212-217, “DH-V-6A to Reactor Building Sump Leak Check and VT-2,” Rev. 3
OP-TM-212-218, “DH-V-6B to Reactor Building Sump Leak Check and VT-2,” Rev. 3
OP-TM-108-108-1008, “TMI-1 Supplement to OP-AA-108-108,” Rev. 2
IR 391055, “Evaluation of Boric Acid Stains on Reactor Vessel Head”
IR 396846, “Evaluation of Debris (Part of Grid Strap Corner) Found in Reloaded Core”

AmerGen NDE Containment Liner Ultrasonic Thickness Data Report 2003-041-002, dated November 14, 2003

UFSAR Sections 5.2.1.2, 5.2.3, 5.2.2.4, and 5.2.2.5

Section 1R22: Surveillance Testing
IR 385339, “MS-V-18B As-Found Test >3% of Nameplate”
Prompt Investigation for MSSV Setpoint Exceeding ASME Code Acceptance Criteria, Rev. 1

Section 2: Radiation Safety
Monthly Data Elements for radiological Occurrences (October 2004 - November 2005)
NOS Oversight Monthly Issues, September 2005, October, 2005, November 2005
NOS Quarterly Report - October 21, 2005
NOS Rapid Trending Report - October 30, 2005
NOS Pre-Outage Readiness Assessment
NOS Monthly Issues (August, September, October 2005)
NOSA-TMI-05-08, dated December 7, 2005, ODCM, REMP, Effluent Monitoring
Plant source term analysis data
Reactor coolant chemistry data for shutdown
Various radiation monitor calibration and operability check data
Various radiological survey records for ongoing outage work activities including records
Various radiation work permits for outage work activities and associated ALARA plans.
Various personnel whole body count data results
2004 Annual Radioactive Effluent Release Report
2005 T1R16 Outage ALARA Information
2005 T1R16 Job Specific Air Sampling- Selected Decontamination, System Breech, and Valve Work
2005-2007 Exposure Reduction Plan
AD-TM-101, “Exposure Control and Authorization,” Rev. 1
LS-AA-2150, “Monthly Data Elements for RETS/ODCM Radiological Effluent Occurrences,” Rev. 5
RP-AA-220, “Bioassay Program,” Rev. 2
RP-AA-250, “External Dose Assessments from Contamination,” Rev. 3

RP-AA-300, “Radiological Survey Program,” Rev. 1
RP-AA-301, “Radiological Air Sampling Program,” Rev. 0
RP-AA-400, “ALARA Program,” Rev. 3
RP-AA-400-1001, “Establishing Collective Radiation Exposure Estimates and Goals,” Rev. 0
RP-AA-400-1002, “Dose Equalization,” Rev. 0
RP-AA-220, “Bioassay Program,” Rev. 2
RP-AA-401, “Operational ALARA Planning and Controls,” Rev. 5
RP-AA-460, “Controls for High and Very High Radiation Areas,” Rev. 9
RP-AA-460-1001, “Additional High Radiation Exposure Control,” Rev. 0

<u>Section 4OA2: Identification and Resolution of Problems</u>

IRs 324668	393725	393912	393906	393303	387445
392879	392871	389884	398640	432328	433572
433567	289543	272362	270567	387440	395920
433853	434132	433572	433567	398170	273153
272362	387440	395920	289543	393906	440566
290797	289346				
AR Nos.	390002	390004	390010	390022	392213
	317137				

Drawing ECR-04-00247-001, “Heat Sink Protection System (HSPS) Control Panel Configuration,” Rev. 0
Drawing ECR-04-00247-002, “Heat Sink Protection System (HSPS) Control Panel Configuration,” Rev. 0
Drawing 201-174, “120/240 Volt AC Power System Electrical Arrangement,” Rev. 21
Drawing 201-188, “120 Volt AC Vital Power System Electrical Arrangement, 1C 120V Single Phase AC Distribution Panel VBC,” Rev. 17
Action Request A2083446, 02/24/04
Station Work Order C2008606
Station Work Order C2008605
Engineering Change Request TM-04-00248-001, NI3NIR/NI5NIR Digital Upgrades
Electrical Loading Data (TMI/OC) TI-077-04-012 - Replacement of Intermediate Range and Power Range Recorders NI3NIR & NI5NIR - 10/14/2004

LIST OF ACRONYMS

ADAMS	Agencywide Documents and Management System
ALARA	As Low As is Reasonably Achievable
AmerGen	AmerGen Energy Company, LLC
ASME	American Society of Mechanical Engineers
CFR	<u>*Code of Federal Regulations*</u>
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
ECT	Eddy Current Testing
EQ	Environmental Qualification

ESDS	Equipment Storage Data Sheet
HPI	High Pressure Injection
IMC	Inspection Manual Chapter
INPO	Institute of Nuclear Power Operations
IR	Issue Report
ISI	In-Service Inspection
LOCA	Loss of Coolant Accident
LPI	Low Pressure Injection
MOV	Motor-operated Valve
MR	Maintenance Rule
MRFF	Maintenance Rule Functional Failure
MSSV	Main Steam Safety Valve
MT	Magnetic Particle Testing
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
NRC	U. S. Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
ODCM	Off-site Dose Calculation Manual
OTSG	Once-Through Steam Generator
OWAs	Operator Work-around
PARS	Publicly Available Records
PI	Performance Indicator
PMT	Post-Maintenance Test
PQR	Procedure Qualification Records
PT	Penetrant Testing
PWR	Pressurized Water Reactor
PZR	Pressurizer
RB	Reactor Building
RCS	Reactor Coolant System
REMP	Radiological Environmental Monitoring Program
RETS/ODCM	Radiological Effluent Treatment System/Offsite Dose Calculation Manual
RR	Reactor Building Emergency Cooling Water System
RWP	Radiation Work Permit
SDP	Significance Determination Process
SFP	Spent Fuel Pool
SSCs	Structures, Systems and Components
TI	Temporary Instruction
TM	Temporary Modification
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
UT	Ultrasonic Testing
VT	Visual Testing
WPS	Welding Procedure Specification

ATTACHMENT A-1

Three Mile Island Station-Unit 1
ISI Activities Inspection, October 27 - November 8, 2005
TI 2515/160, Pressurizer Penetration Nozzles and Steam Space Piping Connections in U. S. Pressurized Water Reactors (NRC Bulletin 2004-01)

Reporting Requirements for TI 2515/160

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.

b. The areas examined were free of dirt, debris, insulation, significant oxidation and foreign material that could adversely affect viewing of the nozzles and its intersection with the pressurizer shell. Residual insulation was noted between the heater penetration strongbacks and the shell insulation and was identified as debris and not boric acid residue.

c. The visual inspection was conducted by direct visual observation by examination personnel.

d. Examination coverage was completed 360 degrees around the circumference of the nozzles and their intersection with the pressurizer shell..

e. The presence of small boric acid deposits representing reactor coolant leakage, as described in Bulletin 2004-01 could be identified and characterized.

f. No material deficiencies were identified. No indications were identified during the inspection period that required repair.

g. Selected insulation material was removed to facilitate the visual inspection of the nozzles and their intersection with the pressurizer shell. There were no impediments to the performance of the visual examination.

h. No indications were detected during the visual examination process.

i. No boric acid deposits were identified at the interface between the pressurizer shell and the nozzles examined. An unidentified white stain was noted on the D-Ring wall in the area above the pressurizer heaters and was identified on issue report AR 00391081 for evaluation, identification and determination of origin. This IR was dispositioned to "clean and re-examine" during the inspection period. The stain was re-examined and identified and characterized as grouting that had been painted over. The inspectors reviewed the disposition and resultant acceptance of the condition. The inspectors also interviewed personnel involved in the examination and disposition of the foreign material.

j. AmerGen conducted appropriate follow-on examinations for indications of boric acid leaks from pressure-retaining components in the pressurizer system and additional systems and components during the scheduled boric acid walkdown inspections performed during and after plant shutdown.

ATTACHMENT A-2

Three Mile Island Station
ISI Activities Inspection, October 27 - November 8, 2005
Inspection Procedure 71111.08, Inserview Inspection Activities,
TI 2515/160, Pressurizer Penetration Nozzles and Steam Space Piping Connections in U. S. Pressurized Water Reactors (NRC Bulletin 2004-01)

Documentation Review

Action Request/Condition Report

AR 00391081	Possible boron found on D-ring wall
AR 00390432	ASME Section III RPE certification not obtained
AR 00393197	Alloy 600 Scope Deferral From 1R16
AR 00394447	Surface Anomaly or Crack Evident in HP-C Thermal Sleeve
AR 00390199	Lack of Bond on the RB Moisture Barrier
AR A2101631	Gaps Found in RB Moisture Barrier
AR 00391232	BACC Boron Indications Found During NDE Inspection
AR 00391055	BACC Thin Boron Film on East Side RX Head Surface
AR 00395167	Lower Tube End Exam Scope Increase in “A” OTSG

Boric Acid Control Program Issue Reports

AR 00389766	Leaking Flange Downstream of WDL-V-339, System 232
AR 00389688	Boric Acid On RC 3B-PT-1
AR 00389690	Packing Leak on RC-V-1178
AR 00389697	Leaking Fitting - Elbow Upstream of RC-V-1037
AR 00389709	Fitting Leak on System 220
AR 00389712,9716	Fitting Leak at Pressure Transmitter Connection, System 220

NDE Examination Test Reports

C2009881-14	VT-2, RC-T-2 Sample Tap Safe End (PR-052N)
C2009881-07	VT-2, RC-T-2 Pressurizer Lower Level TAP (PR-049N)
C2009881-08	VT-2, RC-T-2 Pressurizer Lower Level TAP (PR-050N)
C2009881-09	VT-2, RC-T-2 Pressurizer Lower Level TAP (PR-051N)
C2009881-15	VT-2, RC-T-2 Thermowell Nozzle
1-NDE-770	PT, Pressurizer vent pipe refurb
2005-037-001	VT, Remote Video Exam of HPI-MU-V-86B Thermal Sleeve
C2009595-04	UT, FW0037 20" Elbow to Pipe
C2009595-07	MT, FW0037 Weld Elbow to Pipe
C2009277-03	PT, Welds DH0504, 0505 and 0506 for DH-6 Support
C2009542-05	UT, RCS weld overlay SR0010BMWELD

NDE/Miscellaneous Procedures

54-ISI-835-08	Ultrasonic Examination of Ferritic Piping Welds
ER-AA-335-003 R 1	Magnetic Particle Examination
ER-AA-335-002 R 2	Liquid Penetrant Examination
ER-AA-335-015 R 4	VT-2 Visual Examination
ER-TM-335-1005 R3	Analysis of OTSG Eddy Current Data at TMI
ER-AP-420-003 R1	TMI Unit 1: Steam Generator Eddy Current Activities
51-5005406-04	Qualified Eddy Current Examination Techniques for Three Mile Island 1R16

Drawings/Isometrics

ID-ISI-RC-012 R 1	Pressurizer RC-T2 Details
1272597B-0	OTSG EDM/ASME/Wear Calibration Standard
10005211	Expansion Standard W/EDM Notches
1272591B-0	OTSG EDM/ASME/WEAR CALIBRATION STANDARD AS BUILT

Welding Procedures

WP 8/43	Manual Gas Tungsten Arc Welding of P8 to P43 Materials
WP 43/43	Manual Gas Tungsten Arc Welding of P43 to P43 Materials
PQR 7211	Procedure Qualification Record Supporting WPS 8/43
PQR 7213	Procedure Qualification Record Supporting WPS 8/43
PQR 7072	Procedure Qualification Record Supporting WPS 43/43

Miscellaneous

NDE Contractor Ready to Work Checklist, T1R16

Areva, Alloy 600 Visual Examination Data Sheet, Work Order C2009881-03, C2009881-0-2,C2009732-01, C2009878-01

NEI-139/MRP-139 Primary System Piping Butt Weld Inspection and Evaluation Guideline for PWRs

AmerGen/Excelon letter to NRC dated 07/27/2004, Initial Response to NRC Bulletin 2004-01

AmerGen letter to NRC dated 08/16/2005, Response to Request for Additional Information concerning NRC Bulletin 2004-01

Exelon Nuclear Issue #00394077, Steam Generator Tube Examinations/Degradation

Eddy Current Examination Plan, Three Mile Island - Unit 1, October 2005 - 16R

Steam Generator Degradation Assessment, Rev. 0, for TMI Unit 1, Outage 1R16

Cycle 15 Refueling (T1R15) Inservice Inspection (ISI) Summary Report

TMI 1R16 Examination Technique Specification Sheet, 54-ISI-400-14, Bobbin Standard ASME Code Examination for Unsleeved Parent Tubing

TMI 1R16 Examination Technique Specification Sheet, 54-ISI-400-14, Rotating Probe (.115/+point/.080HF) Kinetic Expansion, Lane & Wedge, Dent/Ding, Crevice Region

Distribution and Characterization of Bobbin Indications SG”A” and “B”, 1R16

Personnel and Equipment Qualification and Calibration Certifications (Eddy Current Exam)

TMI - Unit 1 Steam Generator Tube Plugging Limit and Tube Plugging History