January 31, 2007

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 05000289/2006006 AND NRC OFFICE OF INVESTIGATION REPORT 1-2006-11.

Dear Mr. Crane:

On December 31, 2006, 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 11, 2007, 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 inspection also reviewed your actions for an inattentive operator event that occurred on December 11, 2005. In response to this event, the NRC Office Of Investigations (OI) initiated an investigation on December 19, 2005 to determine if there was any wrongdoing regarding an apparently inattentive shift manager and the failure to report and/or document that situation in a timely manner. Based upon the evidence developed during the investigation, OI concluded that while violations of NRC requirements did occur, these violations were neither willful nor deliberate.

This report documents two self revealing findings of very low safety significance (Green). These findings were determined to involve violations of NRC requirements. Additionally, a licensee-identified violation which was determined to be of very low safety significance is listed in this report. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these 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.

Please note that final NRC documents, such as the OI report 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, Request For Records.

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 05000289/2006006
w/Attachment: Supplemental Information

cc w/encl:
Chief Operating Officer, AmerGen
Site Vice President - TMI Unit 1, AmerGen
Plant Manager - TMI, Unit 1, AmerGen
Regulatory Assurance Manager - TMI, Unit 1, AmerGen
Senior Vice President - Nuclear Services, AmerGen
Vice President - Mid-Atlantic Operations, AmerGen
Vice President - Operations Support, AmerGen
Vice President - Licensing and Regulatory Affairs, AmerGen
Director Licensing - AmerGen
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|---|---|
| Docket No: | 05000289 |
| License No: | DPR-50 |
| Report No: | 050000289/2006006 |
| Licensee: | AmerGen Energy Company, LLC (AmerGen) |
| Facility: | Three Mile Island Station, Unit 1 |
| Location: | PO Box 480<br>Middletown, PA  17057 |
| Dates: | October 1, 2006 - December 31, 2006 |
| Inspectors: | David M. Kern, Senior Resident Inspector<br>Javier M. Brand, Resident Inspector<br>Andrew Rosebrook, Project Engineer<br>Amar Patel, Reactor Inspector<br>Stephen Pindale, Senior Reactor Inspector<br>Ronald M. Nimitz, Senior Reactor Inspector<br>Pat Finney, Reactor Inspector |
| Approved by: | Ronald R. Bellamy, Ph.D., Chief<br>Projects Branch 7<br>Division of Reactor Projects (DRP) |

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## SUMMARY OF FINDINGS

IR 05000289/2006006; 10/1/2006 - 12/31/2006; AmerGen Energy Company, LLC; Three Mile Island, Unit 1; 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) 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. NRC-Identified and Self-Revealing Findings

**Cornerstone: Mitigating Systems**

- **Green**. A green self-revealing non-cited violation of Technical Specification 6.8.1 occurred on Sunday, December 11, 2005, at 3:45 a.m., when the on duty Operations Shift Manager, a licensed Senior Reactor Operator, was observed by three control room operators to be momentarily inattentive while on duty in an office in the control room complex. An NRC Office of Investigations (OI) investigation (1-2006-011) was initiated on December 19, 2005, to determine if any willful violations had occurred. The investigation report concluded that although the Shift Manager was inattentive, it was not a willful act. The Shift Manager was subsequently relieved of duty, Fitness For Duty (FFD) tested, and the licensee increased backshift monitoring by senior management, as well as conducted site-wide training on fatigue-related FFD issues.

  The inattentiveness of the Shift Manager was a performance deficiency because the Shift Manager was not attentive to the conditions of the plant at all times, which limited his ability to monitor safe operation of the plant and the conduct of personnel activities on the site. Also, the licensee had the ability to foresee and prevent this issue due to information the shift manager had appropriately provided to station management. The issue is more than minor because it affected the Human Performance attribute of the Mitigating Systems cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Licensed reactor operators provide important event mitigation capabilities. Also, if left uncorrected, this issue could become more safety significant. Furthermore, the NRC confers upon all reactor operator license holders a special trust and confidence in the safe operation of nuclear power reactor facilities, and all license holders are expected to maintain a level of performance that is above reproach. This includes an expectation to remain alert and attentive at all times to ensure protection of the public health and safety. The licensee has the obligation to take steps to ensure it's operators are not challenged to maintain this level of performance. The performance of the Operations Shift Manager and AmerGen management, in this instance, failed to meet that standard even though the NRC determined that the Shift Manager's actions were not willful.

There is no current SDP that specifically applies to licensed operator Fitness for Duty issues and traditional enforcement does not apply. However, because this issue involved the serious matter of inattentiveness on the part of a licensed Senior Reactor Operator, it was reviewed by NRC management in accordance with the provisions provided under IMC 0612, Section 05.04c and IMC 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria." NRC management carefully reviewed and considered all of the qualitative factors involved with this particular finding. The finding was also reviewed in relation to past historical regulatory precedence. Ultimately, NRC management concluded that the finding is more than minor but not greater than Green. (Section 4OA5)

- **Green**. A green self-revealing non-cited violation of 10 CFR Part 26, "Fitness for Duty Programs" occurred on Sunday, December 11, 2005, when three control room operators did not follow station procedures to initiate prompt actions to have an inattentive Operations Shift Manager relieved of licensed duties and escorted while in the protected area until FFD testing could be completed. The control room operators also did not promptly notify station management of the concern. Each of the control room operators had observed the Operations Shift Manager in an inattentive position in an office within the control room complex.

  The NRC Office of Investigations (OI) conducted an investigation (1-2006-011) to determine if any willful violations had occurred. The report concluded that three licensed operators did not carry out or complete FFD procedure requirements in a timely manner, but they did not do so in a willful manner because they were unaware that operator inattentiveness was a potential FFD issue.

  The failure of the licensed operators to adhere to the station FFD procedure, which implements the requirements of 10 CFR Part 26, was a performance deficiency. The issue is more than minor because it affects the Human Performance attribute of the Mitigating Systems cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Licensed operators provide important event mitigation capabilities and their performance affects the mitigating systems cornerstone.

  Since there is no current SDP that applies to Fitness for Duty or fatigue-related events, the finding was reviewed by NRC management in accordance with the provisions of IMC 0612, Section 05.04c and IMC 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria." NRC management carefully reviewed all of the qualitative factors involved with this particular finding, and the finding was reviewed in relation to past historical regulatory precedence. Ultimately, NRC management concluded that the finding is more than minor but not greater than Green.

  This finding also has cross-cutting aspects in problem identification and resolution for operating experience because Amergen did not effectively evaluate and communicate relevant external operating experience in a timely manner to train operators on fatigue-related FFD issues. (Section 4OA5)

B. Licensee Identified Violations

A violation of very low significance identified by the licensee was reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. The violation and corrective actions are listed in Section 4OA7 of this report.

## REPORT DETAILS

Summary of Plant Status

Three Mile Island, Unit 1 (TMI) began the period at 100 percent rated thermal power. On November 2, 2006, a turbine trip initiated an automatic reactor trip while instrumentation and control (I&C) technicians were performing scheduled calibration of the non-safety related 'B' condenser hood vacuum pressure transmitters. These transmitters form part of the digital turbine control system (see Section 4OA3.1). The plant was returned to 100 percent power on November 6, after completion of troubleshooting activities and implementation of corrective actions. On December 13, an automatic reactor trip was initiated by the reactor protection system (RPS). The trip resulted from an offsite power transmission grid disturbance (see Section 4OA3.2). The plant was returned to 100 percent power on December 16, after completion of troubleshooting activities and corrective actions. The plant operated at or near 100 percent rated thermal power for the remainder of the inspection period.

# 1. REACTOR SAFETY

**Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity**

1R01 Adverse Weather Protection (71111.01 - 1 samples)

a. Inspection Scope

On November 16 and December 1, 2006, the inspectors reviewed AmerGen's procedures for adverse weather, relative to the protection of safety-related structures, systems, and components from the effect of high winds. The inspection was performed immediately prior to and during periods of high wind warnings and tornado watches. The inspectors discussed station precautions with the control room shift manager, observed operators performing related activities, and performed station exterior walkdowns to verify materials and job activities were properly secured. Procedure 1202-33, "Tornado/High Winds," Rev. 27 was reviewed for this inspection.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04 - 4 samples)

a. Inspection Scope

Partial System Walkdowns

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

- On October 18-19, 2006, 'A' train of the building spray system while the 'B' train of the building spray and decay heat removal systems were unavailable due to planned maintenance.

- On October 18-19, ‘A’ train of the decay heat removal (DHR) system while the ‘B’ train of the DHR and building spray systems were unavailable due to planned maintenance.

- On October 19, ‘A’ train of the make-up system while the ‘B’ train of the DHR and building spray systems were unavailable due to planned maintenance.

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. Additional documents reviewed during the inspection are listed in the Attachment.

Complete System Walkdown

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

- On October 23 and November 7, the inspectors conducted a detailed review of valve and component alignment and material condition of the DHR system. Documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05 - 7 samples)

a. Inspection Scope

The inspectors conducted fire protection inspections for several plant fire zones which were selected based on the presence of important to safety equipment within their boundaries. The inspectors conducted plant walkdowns and verified 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. 66. 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. 3. 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:

- Fire Zone AB-FA-1, Auxiliary Building Elev. 261, Decay Heat Removal Pit ‘A’;
- Fire Zone AB-FA-2, Auxiliary Building Elev. 261, Decay Heat Removal Pit ‘B’;

- Fire Zone AB-FZ-2A, Auxiliary Building Elev. 281, Make-up and Purification Pump ‘A’;
- Fire Zone AB-FZ-2B, Auxiliary Building Elev. 281, Make-up and Purification Pump ‘B’;
- Fire Zone AB-FZ-2C, Auxiliary Building Elev. 281, Make-up and Purification Pump ‘C’;
- Fire Area ISPH-FA-2, Intake Screen Pump House Elevation 308', Diesel Fire Pump Room; and
- Fire Zone ISPH-FZ-1, Intake Screen Pump House Elev. 308', IR SWGR & Pump Area.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 1 samples)

a. Inspection Scope

The inspectors performed visual inspections of flood barriers, system boundaries, and waterline break sources located in portions of the auxiliary building where internal flooding could adversely affect safety-related systems needed for safe shutdown of the plant. The review included (1) the ‘A’ and ‘B’ decay heat removal system vaults, and (2) the ‘A’ and ‘B’ building spray vaults. Documents used to support this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

On November 27, the inspectors observed licensed operator requalification training at the control room simulator for the ‘C’ 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 TMI Operational Simulator Scenario TQ-TM-106-SRU-046, “RCP Seal Failure, Small Break LOCA, & Loss of Subcooling Margin,” Rev. 0. 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. The inspectors attended the post-drill critique 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. 4. Additional documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12 - 2 samples)

a. Inspection Scope

The inspectors evaluated Maintenance Rule (MR) implementation for the issues listed below. Specific attributes reviewed included MR scoping, characterization of failed structures, systems, and components (SSCs), MR risk categorization of SSCs, SSC performance criteria or goals, and appropriateness of corrective actions. The inspectors verified that the issues were addressed as required by 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” NUMARC 93-01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants” and AmerGen procedure ER-AA-310, “Implementation of the Maintenance Rule,” Rev. 5.

- Issue Report (IR) 513923 describes an engineering safeguards actuation system relay contact (63Z-2E/R-B1A) that failed to close.
- IR 461358 described a failure of the fuse clip for the ‘A’ emergency feedwater pump (EF-P-2A) due to clip distortion as a result of aging.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13 - 4 samples)

a. Inspection Scope

The inspectors reviewed the scheduling and control of maintenance activities in order to evaluate the effect on plant risk. This review was against criteria contained in AmerGen Administrative Procedure 1082.1, “TMI Risk Management Program,” Rev. 5 and WC-AA-101, “On-Line Work Control Process,” Rev. 12. The inspectors reviewed the routine planned maintenance, restoration actions, and/or emergent work for the following equipment removed from service:

- On October 3, 2006, atmospheric steam relief valve MS-V-4A was removed from service to repair a small air actuator air leak. This condition elevated the on-line maintenance risk profile to yellow (Risk Document 1219, Rev. 2).

- On October 4 and 5, the ‘A’ decay heat removal and building spray system trains were removed from service for scheduled maintenance activities. This condition elevated the on-line maintenance risk profile to orange (Risk Document 1183, Rev. 2).

- On October 10, the ‘B’ train for the decay heat closed cooling and decay heat river water systems were removed from service for scheduled maintenance activities. This condition elevated the on-line maintenance risk profile to orange (Risk Document 831, Rev. 16).

- On October 18, the ‘B’ decay heat removal and building spray system trains were removed from service for scheduled maintenance activities. This condition elevated the on-line maintenance risk profile to orange (Risk Document 831, Rev. 16).

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 4 samples)

a. Inspection Scope

The inspectors reviewed operability evaluations for the following degraded equipment issues. The inspectors verified that degraded conditions 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 IMC 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. Additional documents reviewed during this inspection are listed in the Attachment.

- On October 19, the inspectors identified a minor discrepancy involving the level set point for the constant level oilers for both spent fuel cooling pumps (SF-P-1A and 1B). Engineers determined the oiler for SF-P-1A was set at the minimum level and initiated actions to adjust the oiler set point and to perform an extent-of-condition review (IR 546175). The inspectors verified that operability of the pumps was not affected.

- On October 19, maintenance technicians and engineers identified an air void (approximately 10 cubic feet) in the ‘B’ DHR pump suction piping while the ‘B’ DHR train was out of service for maintenance. The air void was identified by ultrasonic testing (UT) performed as a post-maintenance test following installation of new high point vents. The high point vents were installed to address a previously identified deficiency involving the inability to properly fill and vent the system after it is drained for maintenance. The inspectors verified that proper extent-of-condition reviews were performed and that corrective actions including additional UT were performed to verify the air void was properly vented (IR 546188) prior to returning the ‘B’ DHR train to service.

- On December 13, an automatic plant trip occurred during a severe grid disturbance caused by the loss of a 230 KV line near the Middletown junction. The trip was initiated by the reactor protection system (RPS) upon inadvertently sensing a loss of power to the reactor coolant pumps (RCP). Engineers determined that a time delay relay in the RCP power monitoring logic circuit did not operate as anticipated to prevent the unnecessary indication of loss of power to the RCPs on electrical transients. The cause of the time delay relay malfunction remains under investigation. The inspectors verified the time delay relay is not required for protection of the reactor and the relay does not provide a safety function. The lack of a time delay feature is conservative with respect to reactor safety, because the RPS would trip sooner than expected on an RCP power supply transient (IR 569086).

- On October 27, Engineering revised Operability Evaluation, OPE-06-002, for the Reactor Building Purge Inlet Containment Isolation Valve, AH-V-1. This valve displayed elevated running friction during post-maintenance testing in March 2006. Engineering determined that the valve remained operable but degraded and generated corrective actions to restore the valve’s quality (IR 461841).

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17A - 1 sample)

a. Inspection Scope

The inspector reviewed the design change package, and work order associated with ECR 06-00297, "Install Vent Provision BS-V-83." The inspector also observed fabrication and non-destructive examination activities in the shop and installation activities in the plant.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19 - 5 samples)

a. Inspection Scope

The inspectors reviewed and/or observed the following 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.

- On October 3, maintenance technicians and plant operators performed testing in accordance with procedure OP-TM-411-204, “Quarterly Test Of MS-V-4A and MS-V-4B Valves During Normal Plant Operations For IST,” Rev. 3, following corrective maintenance to repair a small actuator air leak for main steam atmospheric relief valve MS-V-4A.

- On October 6, operators performed testing in accordance with procedure OP-TM-212-201, “IST Of DH-P-1A And Valves From ES Standby Mode,” Rev. 5, following scheduled preventive maintenance.

- On October 6, operators performed testing in accordance with procedure OP-TM-214-201, “IST Of BS-P-1A And Valves From ES Standby Mode,” Rev. 5, following scheduled preventive maintenance.

- On November 14, operators performed testing in accordance with procedure OP-TM-424-202, “IST OF EF-P-2B,” Rev. 2, following scheduled preventive maintenance and lubricating oil replacement.

- On December 20, operators performed testing of the station blackout diesel generator in accordance with procedure 1107-9, “SBO Diesel Generator”, Rev. 54, following scheduled preventive maintenance.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 5 samples)

a. Inspection Scope

The inspectors observed and/or 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/or initiation and resolution of related IRs for selected surveillances as appropriate. Additional documents reviewed during the inspection are listed in the Attachment.

- On October 19, procedure OP-TM-212-202, “IST OF DH-P-1B And Valves From Standby Mode,” Rev. 5.

- On October 19, procedure OP-TM-214-202, “IST OF BS-P-1B And Valves,” Rev. 6.

- On November 30 and December 23, procedure OP-TM-541-208, “IST of NS-P-1A/B/C,” interim changes 19452 and 22002, respectively.

- On December 2, FS-P-2 testing per procedure 3303-A3, “Fire Pump Capacity Testing,” Rev. 8.

- On December 4, procedure OP-TM-424-201, “IST Of EF-P-2A,” Rev. 2.

b. Findings

No findings of significance were identified.

## 2. RADIATION SAFETY

**Cornerstone: Occupational Radiation Safety**

2OS1 Access Controls (71121.01 - 19 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 Technical Specifications, and applicable AmerGen procedures.

Inspection Planning - Performance Indicators

The inspectors selectively reviewed performance indicators (PIs) for the Occupational Exposure Cornerstone. The inspectors also discussed and reviewed current performance with cognizant AmerGen personnel. See Section 4OA1.

Plant Walkdowns and RWP 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 reviewed the radiological controls for work activities associated with reactor containment entry at power, and for completed work for repair of the miscellaneous waste evaporator, and inspection of the “A” concentrated waste storage tank. 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 and evaluated calculation of neutron exposure and noble gas exposure for entry to the Unit 1 containment at power. The reviews included, where applicable, barrier integrity and the application of engineering controls for potential airborne radioactivity areas and radioactive source term, and radiation levels present.

The inspectors reviewed applicable radiation work permits and electronic personnel dosimetry alarm setpoints (both integrated dose and dose rate) to verify that the set-points were commensurate with ambient/expected conditions, radiation work permits were appropriate for the conditions, and plant policy. The inspectors observed the radiological controls briefing for worker entry into the Unit 1 reactor containment at power.

The inspectors reviewed, and discussed High Radiation Area controls for access to the reactor building. The inspectors reviewed and discussed physical and programmatic controls for highly activated or contaminated materials (non-fuel) stored within the spent fuel pool, or other storage pools, as applicable.

The inspectors discussed controls for 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 also reviewed and discussed inter-comparison of electronic dosimeter and thermoluminescent dosimeter results to identify anomalies and licensee actions, as applicable.

The inspectors reviewed and discussed internal dose assessments for 2006 to identify any apparent actual occupational internal doses greater than 50 millirem committed effective dose equivalent (CEDE). 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). The inspectors reviewed 2006 whole body counter logs and data.

Problem Identification and Resolution

The inspectors selectively reviewed self-assessments and audits 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 Occupational Exposure Control Effectiveness performance indicator (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 (TEDE), 5 rem shallow dose equivalent (SDE), or 1.5 rem lens dose equivalent (LDE). 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.

In addition, the inspectors also reviewed problem 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. (See Section 4OA2)

High Risk Significant, High Dose Rate High Radiation Area (HRA) and Very High Radiation Area (VHRA) Controls

The inspectors discussed procedure changes for HRA 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 HRA controls (e.g., adequate posting and locking of entrances). The inspectors discussed controls for HRA and VHRAs with radiation protection technicians. The inspectors reviewed the access key inventory for HRA and VHRA access areas and conducted a key inventory.

Radiation Worker/Radiation Protection Technician Performance and Radiation Protection Technician Proficiency

The inspectors 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.

b. Findings

No findings of significance were identified.

2OS2 As Low As Is Reasonably Achievable (ALARA) Planning and Controls (71121.02 - 4 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 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 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 3-year rolling average collective exposure for the period January 2003 through

December 2005. The inspectors evaluated site specific trends in collective exposures (using NUREG-0713 and plant historical data). The inspectors discussed proposed occupational radiation exposure estimates for 2007.

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 containment entry at power, repair of the miscellaneous waste evaporator, and inspection of the “A” concentrated waste storage tank. The inspectors reviewed the integration and implementation of ALARA requirements into procedures and radiation work permit (RWP) documents.

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

Job Site Inspections and ALARA Controls

The inspectors observed the radiological controls briefing for worker entry into the reactor containment at power. The inspectors reviewed exposures of individuals from selected work groups to identify significant exposure variations which may exist among workers.

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 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 scaffolding installation and removal activities.

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. Fluid clean-up methods used to remove radioactivity were reviewed. The inspectors evaluated dose reduction results achieved against priorities since the last refueling cycle. The inspectors discussed the TMI five year source term reduction plan. The inspectors also reviewed Station ALARA Council Meeting Minutes for 2006.

Declared Pregnant Workers

The inspectors reviewed and evaluated, as applicable, radiation exposure controls for declared pregnant workers.

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.

Problem Identification and Resolution

The inspectors selectively reviewed problem reports in this area since the last inspection to determine if AmerGen was including ALARA deficiencies and issues in its corrective action program, as applicable. (See Section 4OA2.)

The review included self-assessments, audits, and corrective action reports related to the ALARA program since the last inspection to determine if the follow-up activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk.

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03 - 3 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 Technical Specifications and station procedures.

Inspection Planning/Identification of Additional Radiation Monitoring Equipment

The inspectors selectively reviewed the station's 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 selectively 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 following instruments were reviewed: electronic dosimeters (ED) 78496, 36241; air monitor AMS-4 1207; RM-14 3656; personnel monitors PM-7 Nos. 482 and 446; PCM1B No. 1058; RSO50 No. 76472; low-volume air sampler No. 1333; lapel sampler Nos. 2231 and 2218; survey meter ASP-1 No. 1332; contamination monitor SAM-11 No. 173; and Telepole No. 77834.

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. (See Section 4OA2). The review included self-assessments, audits and corrective action reports.

b. Findings

No findings of significance were identified.

## 4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151 - 3 samples)

.1 Safety System Functional Failures

a. Inspection Scope

The inspectors reviewed the PI assessment for safety system functional failures (SSFFs) to determine whether the SSFFs had been accurately reported to the NRC as required by Nuclear Energy Institute (NEI) 99-02, “Regulatory Assessment Performance Indicator Guideline,” Rev. 4. Verification included review of the data collected, definitions, data reporting elements, calculation methods, definition of terms, and use of clarifying notes. The inspectors verified accuracy of the reported data through review of selected station operating logs, system health reports, and SSFF databases, and Licensee Event Reports for the period October 2003 through September 2006.

b. Findings

No findings of significance were identified.

.2 Occupational Exposure Control Effectiveness

a. Inspection Scope

The implementation of the Occupational Exposure Control Effectiveness Performance Indicator (PI) Program was reviewed. The inspectors reviewed corrective action program records for occurrences involving HRAs, VHRAs, and unplanned personnel radiation exposures since the last inspection in this area. The inspectors reviewed individual radiation exposure results and selectively reviewed exposure records and associated radiation work permits. 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.

.3 Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual (RETS/ODCM) - Radiological Effluent Occurrences

a. Inspection Scope

The implementation of the 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 previous four quarters. The inspectors selectively reviewed the 2005 Annual Effluent Release Report. The inspectors also reviewed and discussed potential abnormal releases via groundwater or effluents. The review was against the applicable criteria specified in NEI 99-02, Regulatory Assessment Performance Indicator Guideline," Rev. 4. 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 Problem Identification and Resolution 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.

Section 4OA5 describes a finding when three licensed operators observed the Shift Manager being inattentive to duty. Problem identification of this finding was deficient in that the three operators did not recognize this was a fitness for duty issue and did not follow applicable procedure requirements.

.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

No findings of significance were identified. However, the inspectors made the following observations. NRC Inspection Reports 05000289/2005005 and 05000289/2005009 previously documented a trend of procedure quality and adherence deficiencies. In addition, the NRC problem identification and resolution team inspection completed in May 2006 (NRC Inspection Report 05000289/2006007) identified continued challenges in the area of procedure adequacy and adherence. Station management has implemented numerous actions between 2005 and 2006 to address these trends, including initiation of an Accountability Review Board for evaluating human performance events, line ownership through the TMI First line Supervisors Peer Group, and the development of monthly performance indicators. The inspectors reviewed the status of applicable corrective actions and the results of the most recent licensee self assessment to review the progress accomplished by end of year 2006 in procedure quality and adherence. The inspectors also discussed the results-to-date with station personnel.

The inspectors noted that through the development and trending of measurable matrixes, TMI has shown positive indications of workforce behavior improvements in procedure use and adherence. Direct indications of improvements include; 1) a reduction of Significance Level 1, 2, and 3 Issue Report events attributed to procedure adherence issues from 50 events in 2005 to 20 events in 2006, 2) a lower number of configuration control issues, and 3) an increase in the number of procedure change requests. In addition, the inspectors observed improved procedural compliance and quality during field inspections and noted a decrease in the number of NRC Findings related to procedure quality and adherence.

Notwithstanding the noted progress toward improved procedure quality and adherence, the inspectors noted examples where established programs were not properly implemented. Recent examples include a security violation regarding protected area access controls (IR 565138), fire seal inspection procedure was not clear for collective assessment of sample results to identify trends or abnormal degradation, an interim procedure change issued as a corrective action was inadvertently dropped during a procedure revision (IR 522409), incorrect RCS power flow trip setpoint acceptance criteria, and deficient inspection criteria for periodic visual inspection of backup river water discharge line. AmerGen has identified procedure quality and adherence as a continued station focus area for 2007.

Additionally, the inspectors noted areas for improvement in the area of problem identification and resolution, including use of industry operating experience (OE). AmerGen evaluation of several degraded equipment conditions was either not timely or was too narrowly focused. Examples included (1) main feedwater check valve IST failure (IR 481851), (2) licensed operator fitness-for-duty (IR 432733), (3) untimely corrective action to borated water storage tank level alarm instrument drift (IR 523284 and 525514), (4) no interim process in place to perform ultrasonic testing for air voids after draining and refilling safety systems pending installation of high point vents (IR 537432), (5) corrective actions to add high point vents to address a previous air void problem in the decay heat system not fully effective (IR 5467188), (6) review did not identify deficient inspection criteria for backup river water discharge line (IR 542822), and (7) post-trip review process did not question effect of open main steam safety valve on RCS sub-cooling. The inspectors discussed the above examples with the plant manager who acknowledged that TMI's use of industry OE may need further self assessment.

.3 Radiological Controls

a. Inspection Scope

The inspector selectively reviewed issue reports and self-assessments to determine if identified problems were entered into the corrective action program for resolution. The inspector selectively reviewed the reports to evaluate AmerGen's threshold for identifying, evaluating, and resolving problems. The review included a check of possible repetitive issues, such as worker or technician errors. (IRs 555220, 555330, 555425, 497744, 512903, 545989, 497977, 524054, 555635, 526008, 527425, 555564, 555425, 527091, 529633, 548051)

This review was against criteria contained in 10 CFR 20, Technical Specifications, and the station procedures.

b. Findings

No findings of significance were identified.

.4 Annual Sample: Control Building Ventilation and Chiller Design and Capability

a. Inspection Scope

The inspectors selected one sample for review, which included several IRs related to the Control Building Ventilation and Chiller design and capability. The inspectors reviewed AmerGen's responses to the IRs to ensure that the full extent of the condition was identified, appropriate evaluations were performed, and appropriate corrective actions were specified and prioritized. The inspectors walked down the system and interviewed relevant station personnel. Applicable calculations and safety evaluations were also reviewed. The specific documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified. However, the inspectors identified minor weaknesses with the calculation that analyzed the control building envelope heatup rates during a postulated loss of control building ventilation and/or cooling scenarios. In particular, the worst case allowable initial room temperature was not used to determine the final room temperature under certain assumed scenarios. Notwithstanding, based on additional technical review and discussions with design engineers, the inspectors determined that the final room temperature under the worst case initial conditions would not exceed the maximum temperatures specified in design basis documents. This issue was captured in IR 541610.

4OA3 Event Followup (71153 - 4 samples)

.1 Automatic Turbine Trip/Reactor Trip Due to Invalid Low Condenser Vacuum Signal

a. Inspection Scope

On November 2, 2006, at 1:34 p.m., the reactor automatically tripped from 100 percent reactor power due to a turbine trip in response to a low condenser vacuum protection signal. Plant systems responded as designed with no significant complications. Operators properly implemented OP-TM-EOP-001, "Reactor Trip," Rev. 7, in response to the trip and safely stabilized the plant in the Hot Shutdown mode.

The inspectors responded to the control room, the turbine building, and the intermediate building to evaluate plant equipment and mitigating system response to the trip; operator actions including communications and use of appropriate emergency operating procedures; and plant stabilization to a safe shutdown condition. The inspectors observed operator actions, reviewed various instruments and sequence of events recorders, and discussed the plant status with operators to verify safe plant conditions.

A surveillance test on the 'B' reactor protection system (RPS) and calibration of a main condenser pressure transmitter were in progress at the time of the trip. The cause of the trip was related to the ongoing pressure transmitter calibration. The inspectors verified the reactor trip was properly reported in accordance with 10 CFR 50.72. Following plant stabilization, the inspectors reviewed the event's risk significance with licensee personnel and the NRC regional senior risk analyst. The inspectors determined that the conditional core damage probability was very low and that no additional NRC reactive response was necessary. The cause of the trip and minor deficiencies associated with the 50.72 notification and simulator modeling of plant response were documented in IRs 552591, 553202, 554854, and 554865. The inspectors observed the plant readiness for restart assessment meeting to determine whether station personnel understood the event and had taken appropriate actions to reduce the likelihood of recurrence, prior to plant restart.

Engineers determined that the trip resulted from a design deficiency in the digital turbine control system (DTCS). At the time of the event, technicians were performing IC-192, "Calibration of Condenser Hood Pressure Transmitters," interim change 16173. The procedure provided instructions which inhibited (or blocked) a turbine trip signal emanating from the pressure transmitter being calibrated. This should have permitted operators to calibrate one set of pressure transmitters online, without generating a turbine trip signal to the protection logic. Vendor manuals indicated that the DTCS was single fault tolerant. After the event, engineers obtained and evaluated proprietary vendor drawings. They determined the most likely cause of the trip was electrical cross-talk to separate DTCS protection channels through common grounds paths when the pressure transmitter was powered up. The inspectors determined the existence of the design defect was beyond the level of knowledge station engineers could reasonably be expected to possess on this non-safety related control system. Station personnel initiated appropriate industry experience notifications to alert other power plants to this previously unknown latent DTCS design issue. Additional documents reviewed during the inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

.2 Automatic Reactor Trip in Response to Offsite Power Transmission Grid Transient

a. Inspection Scope

On December 13, 2006, at 5:48 p.m., the reactor automatically tripped from 100 percent reactor power in response to an instantaneous electrical transient on the offsite power transmission grid. Plant systems responded as designed with no significant complications. Operators properly implemented OP-TM-EOP-001, "Reactor Trip," Rev. 7, in response to the trip and safely stabilized the plant in the Hot Shutdown mode. Main steam safety valves briefly opened to relieve secondary system pressure and reseated, as designed.

Immediately prior to the reactor trip, the off-site transmission grid operator attempted to reenergize a nearby faulted 230 Kilovolt transmission line. This action resulted in a brief (60 millisecond) voltage transient which caused the offsite power supply voltage to the plant to drop by approximately 43 percent. The RPS sensed the degraded voltage and interpreted this as an imminent loss of power to the reactor coolant pumps (RCPs). The RPS consequently generated a “reactor power/number of RCPs running” protective trip signal and the reactor automatically shut down.

All four RCPs continued to run, providing reactor coolant flow during and after the transient. Station engineers determined that a valid RPS trip signal had been generated. However, a time delay circuit intended to preclude RPS trips due to brief power transients such as this one did not function properly. Following the trip, technicians functionally tested the associated RPS trip circuits and the time delay circuits. Each circuit functioned properly. Station personnel reviewed the test results and determined the RPS safety function to automatically shut down the reactor remained operable. The cause of the time delay relay malfunction remains under investigation. The inspectors verified the time delay relay is not required for protection of the reactor and the relay does not provide a safety function.

The inspectors inspected plant equipment, reviewed various records and sequence of events recorders, and interviewed station personnel to verify safe plant conditions and determine the cause of the reactor trip. The inspectors confirmed that the reactor trip resulted from a valid RPS protective trip. The inspectors also verified the reactor trip was properly reported in accordance with 10 CFR 50.72. Following plant stabilization, the inspectors reviewed the event’s risk significance with licensee personnel and the NRC regional senior risk analyst. The inspectors determined that the conditional core damage probability was very low and that no additional NRC reactive response was necessary. The cause of the trip and associated follow-up actions, including discussion of grid operation protocols with the transmission grid owner, were documented in IRs 569086 and 569118. The inspectors observed the plant readiness for restart assessment meeting to determine whether station personnel understood the event and had taken appropriate actions to reduce the likelihood of recurrence, prior to plant restart. The inspectors also monitored plant restart. Additional documents reviewed during the inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

.3 (Closed) Licensee Event Report (LER) 05000289/2005001-01: Control Building Ventilation Fan Inoperable Due To Cracked Fan Hub

On April 18, 2006, AmerGen issued a revision (Supplement 1) to LER 05000289/2005001. The initial LER reported the event under 10 CFR 50.73 (a)(2)(i)(B), for a condition prohibited by the plant’s Technical Specification. The supplement was issued to document that the event was also reportable under criterion 10 CFR 50.73 (a)(2)(v), for a condition that could have prevented the fulfillment of the safety function related to post-accident control room habitability.

The condition involved a cracked fan hub on the control building ventilation fan (AH-E-19B) that was identified on February 3, 2005. An evaluation of past operability concluded that the Control Room Emergency Filtration System would not have been able to meet it's mission time of 30 days during several time intervals between August 20, 2003 and September 29, 2004, when a redundant fan was not available. This event and associated enforcement actions were previously documented in NRC Inspection Report No. 05000289/2004-005 (NCV 05000289/2004005-04, Failure To Timely Investigate And Repair a Degraded Control Building Ventilation Fan AH-E-19B). In addition, the initial LER was reviewed and closed out in NRC Inspection Report No. 05000289/2005-005 (Section 4OA3), as a licensee identified finding. No new performance issues were identified and the assessment of the event safety significance did not change as a result of the new information provided by the Supplement. This LER is closed.

.4 (Closed) LER 05000289/2006001: Design Change Error for the Decay Heat Valves Connecting the Borated Water Storage Tank (BWST) and the Reactor Building (RB) Sump Negatively Impacted the Fire Mitigation Strategy for an Auxiliary Building Fire Area.

On April 23, 2006, while performing reviews of fire abnormal operating procedures to assure compliance with the Fire Hazards Analysis Report (FHAR), engineers identified a control logic error in the elementary circuit drawings for the isolation valves (DH-V-6A and DH-V-6B) between the BWST and the RB sump. Technicians verified both valve control circuits were wired as shown in the elementary circuit drawings. The control logic error could allow DH-V-6A(B) to spuriously open due to a fire, thereby draining the BWST inventory to the RB sump. This in turn could cause a loss of high pressure injection makeup capability, resulting in an unanalyzed condition that significantly degrades plant safety.

The licensee determined the root cause to be insufficient technical rigor applied in the validation of design requirements for DH-V-6A(B) in the original 1985 10 CFR 50, Appendix R design package. Corrective actions included establishing a one hour fire watch in the auxiliary building and modifying the control circuitry for DH-V-6A(B) to prevent them from spuriously opening due to a hot short (IR 482679). The licensee determined this issue was in violation of 10 CFR 50, Appendix R, fire protection program requirements. This finding is more than minor because it has a credible impact on safety, in that inadvertent draining of the BWST would adversely impact reliability and availability of the high pressure injection mitigating system. The finding affected the mitigating systems cornerstone and was considered to have very low safety significance due to the existence of a fire detection system, low fire loading in the affected area, and based on the risk informed approach in NRC Regulatory Issue Summary (RIS) 2004-03. In this RIS, the NRC states that "Multiple high impedance faults are considered of very low likelihood." The enforcement aspects of the violation are discussed in Section 4OA7. This LER is closed.

4OA5 Other

.1 Inattentive Operations Shift Manager Follow-up

a. Inspection Scope

An NRC Office of Investigations (OI) investigation was initiated on December 19, 2005, to determine if there was any wrongdoing regarding an apparent inattentive Shift Manager, as well as the failure of licensed operators to report and document that situation in a timely manner. Investigation Report 1-2006-011, "Inattentive Control Room Shift Manager and Failure to Follow Procedures" was issued on September 14, 2006. In this report, it was concluded that (1) The Shift Manager had been inattentive on Sunday, December 11, 2005, and (2) Licensed control room operators did not follow FFD procedures on December 11 upon discovery of the inattentive Shift Manager. In both cases, the NRC determined that the involved operators did not willfully violate the applicable station procedures. The two issues are discussed below.

b. Findings and Observations

1) Inattentiveness to Duty by a Shift Manager

Introduction: A green self-revealing non-cited violation of Technical Specification 6.8.1 occurred on Sunday, December 11, 2005, when the on duty Shift Manager was observed by three control room operators to be inattentive to duty in an office in the control room complex.

Description: On Sunday, December 11, 2005, at 3:45 a.m., the on duty Shift Manager was observed by three control room operators (2 Senior Reactor Operators (SRO) and 1 Reactor Operator (RO)) to be apparently inattentive and/or asleep in an office in the control room complex. In response, one crew member acted promptly to arouse the Shift Manager by calling him on his cell phone. He responded promptly to the call. However, the three control room operators did not recognize that this event was an FFD issue and did not follow the station FFD procedure which required the individual suspected of being unfit to be temporarily relieved of licensed duties, and escorted while in the protected area until FFD testing can be completed. The procedure also required prompt notification of the FFD concern to the individual's immediate supervisor. The Shift Manager was neither relieved of licensed duties nor escorted while in the protected area. The issue was later brought to the attention of station management approximately 12 hours after the initial observation. Once informed, station management responded by relieving the Shift Manager of licensed duties and initiating FFD testing. The testing was performed approximately 19 hours after the initial observation. Licensee evaluation eventually concluded that the inattentiveness was caused by insufficient rest and that neither alcohol nor drugs were involved.

An NRC Office of Investigation (OI) investigation (1-2006-011) was initiated on December 19, 2005 to determine if any willful violations had occurred. The report concluded that the Shift Manager had been inattentive; however, OI determined that this was not the result of an intentional act. The report also revealed that the shift manager had informed station management of off duty commitments which could reasonably be concluded as having the potential to impact his readiness over a period of time. In subsequent actions, the licensee notified the NRC that the Shift Manager no longer had a need to maintain his individual operating license. In response to the notification, the NRC terminated the individual's operating license. The licensee also increased the number of backshift observations of crew performance conducted by station management and held site-wide training on fatigue-related FFD issues.

Analysis: The inattentiveness of the on-duty Shift Manager was a performance deficiency because the Shift Manager was not attentive to the conditions of the plant at all times, which limited his ability to monitor safe operation of the plant and the conduct of personnel activities on the site. Also, this deficiency was reasonably within the licensee's ability to foresee and prevent, because the shift manager had appropriately informed station management of other off duty commitments which could impact his FFD. The licensee did not initiate any compensatory measures to monitor this individual's activities prior to this event. This resulted in the licensee's behavior monitoring program not identifying that the shift manager was fatigued, and no action being taken prior to the shift manager becoming inattentive on December 11th.

In this case, traditional enforcement does not apply because the NRC determined that the Shift Manager's actions were not willful, the actions did not have the potential for impacting the NRC's ability to perform its regulatory function, and the issue did not have actual safety consequences. The issue is considered more than minor since it affects the Human Performance attribute of the Mitigating Systems cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Licensed Operators are considered to be an important mitigating system. Furthermore, the issue is more than minor because if left uncorrected, the finding could become more safety significant.

There is no current SDP that applies specifically to licensed operator FFD or fatigue-related issues. However, because this issue involved the serious matter of inattentiveness on the part of a licensed Senior Reactor Operator, the finding was reviewed by NRC management in accordance with the provisions provided under IMC 0612, Section 05.04c and IMC 0609, Appendix M. Station procedure OP-AA-101-111, "Roles and Responsibilities of on Shift Personnel," Section 3.1 specifies that the Shift Manager is directly responsible for the safe operation of the plant. In addition, the Shift Manager was the designated Emergency Response Organization (ERO) Emergency Director and the senior licensee official on site when he was discovered to be inattentive to duty. The NRC confers upon all Reactor Operator license holders a special trust and confidence in the safe operation of nuclear facilities, and all license holders are expected to maintain a level of performance that is above reproach. This includes an expectation to remain alert and attentive at all times to ensure protection of the public health and safety. The licensee has the obligation to take steps to ensure it's operators are not challenged to maintain this level of performance. In this instance, both AmerGen management and the Shift Manager failed to meet this standard.

NRC management carefully reviewed the specific factors of this case in relation to past historical regulatory precedence in order to render a significance determination that was appropriate for this particular situation. Several qualitative circumstances associated with this finding were considered: 1) the importance and special trust conferred upon licensed individuals as noted above; 2) OI determination that the violation was not willful; 3) two qualified SROs remained on duty in the control room at the time of discovery; 4) the Shift Manager was not at the controls of the reactor at the time he was observed to be inattentive; 5) the Shift Manager was inattentive for a relatively short period of time (estimated 5-10 minutes); and 6) the Shift Manager responded to a phone call from a crew member in a timely manner which would support the conclusion that he could have acted in his capacity as the Emergency Director within 15 minutes, as required. In addition, there were no actual plant consequences associated with this issue. Based upon all of these factors and past historical regulatory precedence, NRC management concluded that the finding was more than minor, but not greater than Green.

Enforcement: Three Mile Island Unit 1 Technical Specification 6.8.1 requires the establishment and implementation of procedures covering the activities listed in Regulatory Guide 1.33, Revision 2, February 1978. Regulatory Guide 1.33 specifies, in part, that administrative procedures be developed and implemented covering authorities and responsibilities for safe operation and shutdown and for shift and relief turnover. Station Procedure OP-AA-101-111, "Roles and Responsibilities of on Shift Personnel," Revision 1, Section 3.4 states that, "all on shift licensed and non-licensed and operating supervisors are required to be aware of and responsible for the status of the plant at all times." Section 3.5 states, "Operations Personnel shall be attentive to the conditions of the plant at all times. They must be alert to ensure plant safety." Section 3.1 specifically states, "the Shift Manager during his or her shift is responsible for safe operation of the plant and the conduct of all personnel and all activities on site." Section 4.1 defines the roles and responsibilities of the Shift Manager, including designating the Shift Manager as the Emergency Director. Contrary to the above, on December 11, 2005, at approximately 3:45 a.m., the on duty Shift Manager was not attentive to the condition of the plant since he was observed to be inattentive/asleep by three other control room operators. Although prompt action was taken by the crew to arouse the shift manager, the FFD procedure requirements were not timely. The licensee subsequently removed the Shift Manager from licensed duties and later informed the NRC that the individual no longer had a need for an operating license. In response, the NRC terminated the license. The licensee also increased backshift monitoring of on shift-crew performance by senior management. Finally, because this finding was determined to be Green and entered into the AmerGen corrective action program (IR 432733), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000289/2006006-01, Inattentiveness to Duty by a Shift Manager)**

2) Failure to Follow FFD Procedures

Introduction: A green self-revealing non-cited violation of 10 CFR Part 26, "Fitness for Duty Programs" occurred on Sunday, December 11, 2005, when three NRC licensed control room operators observed an on-duty Operations Shift Manager inattentive to duty in an office in the control room complex and did not implement established FFD requirements. Each of the operators involved indicated that they were unaware that an FFD evaluation was necessary. The station FFD procedure, which implements the requirements of 10 CFR 26, specified that individuals presumed to be unfit for duty must be relieved of duties and escorted while in the protected area until the FFD testing is completed. The procedure also required that the immediate supervisor of the individual presumed to be unfit is notified of the occurrence.

Description: On Sunday, December 11, 2005, three control room operators (2 senior reactor operators (SRO) and 1 reactor operator (RO)) did not follow station FFD procedures and initiate actions to have an inattentive Shift Manager relieved of duty and escorted while inside the protected area until FFD testing could be completed. Furthermore, the three control room operators did not promptly notify station management of their observations. The three control room operators did not recognize that this event was an FFD issue and did not follow the FFD procedure which required the individual with an FFD concern to be relieved of duty and escorted while in the protected area until the concern is resolved. The Shift Manager was neither relieved of duties nor escorted while in the protected area and no formal FFD evaluation was performed. Operations Department management was not informed of the occurrence until approximately 12 hours after the event. When Operations Department management was informed, the Shift Manager was temporarily relieved of duties and FFD testing was conducted. However, this testing was not performed until approximately 19 hours after the initial observation.

An NRC Office of Investigations (OI) investigation (1-2006-011) was initiated on December 19, 2005, to determine if any willful violations had occurred. During the investigation, it was determined that the three licensed operators did not follow the FFD procedures; however, they did not do so in willful violation of these procedures. The investigation also revealed that there was a site-wide misconception that fatigue was not an FFD concern. This knowledge deficiency was not isolated to the three licensed operators. AmerGen conducted an internal investigation into this issue and conducted extensive site wide training on the FFD process as it relates to fatigue issues. AmerGen also conducted training using case studies of this and other past issues involving licensed and non-licensed individuals. Corrective actions also included increased back shift monitoring tours by senior management to monitor crew performance. AmerGen acknowledged that there had been a clear missed opportunity to train on fatigue-related FFD issues following a significant NRC enforcement action regarding an inattentive SRO at the Pilgrim Nuclear Power Station in 2005. AmerGen had received Industry Operational Experience (OE) on this issue but had not trained operations personnel on the lessons learned prior to December 11, 2005.

Analysis: The failure to adhere to station procedures for FFD, which implements the requirements of 10 CFR 26, was a performance deficiency. Three licensed operators observed the Shift Manager being inattentive to duty, but they did not recognize this was an FFD issue and therefore did not follow the station FFD requirements. The issue is more than minor because it affects the Human Performance attribute of the Mitigating Systems cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). The licensed operators provide important mitigating capabilities.

There is no current SDP that specifically applies to licensed operator FFD or fatigue-related issues. Therefore, the finding was reviewed by NRC management in accordance with the provisions of IMC 0612, Section 05.04c and IMC 0609 Appendix M, "Significance Determination Process Using Qualitative Criteria." NRC management concluded that the finding was more than minor but not greater than Green since there were no actual safety consequences associated with this issue, the licensee identified the issue, and the licensee took appropriate corrective actions once the issue was identified.

This finding has a cross-cutting aspect in the area of problem identification and resolution for operating experience because AmerGen did not effectively evaluate and communicate relevant external operating experience in a timely manner to train TMI operators on fatigue-related FFD issues.

Enforcement: 10 CFR Part 26, “Fitness for Duty Programs,” prescribes requirements and standards for the establishment and maintenance of certain aspects of FFD programs and procedures, and each licensee subject to this part is required to establish and implement written policies and procedures to meet these objectives. TMI Procedure SY-AA-102, ”Fitness for Duty,” Revision 9, Section 3.6.3 states, in part, that supervisors are required to act in a timely manner when a Fitness for Duty concern has been identified. If Fitness for Duty is questionable, the supervisor is required to immediately remove the person from work activities and the person is required to be escorted at all times until the concern is satisfactorily resolved or until the person leaves the protected area. The procedure also required that the individual’s immediate supervisor be notified of the suspicion of an FFD concern. Contrary to the above, the three control room operators who witnessed their Shift Manager being inattentive to his duties on December 11, 2005, did not ensure that the Fitness for Duty procedure was followed. The Shift Manager was not relieved of his duties and was allowed to complete his shift, no formal FFD evaluation was performed, and Operations management was not informed until approximately 12 hours after the initial observation. Formal FFD testing was not conducted until 11:00 p.m., on December 11, 2005, approximately 19 hours after the initial observation.

Corrective actions included night orders to the operators, reactive training using case studies of this issue and the 2005 Pilgrim issue, and increased back shift monitoring tours by senior management. Finally, because this finding was determined to be Green and was entered into the AmerGen corrective action program (IR 432733), this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. **(NCV 05000289/2006006-02, Failure to Follow FFD Procedures)**

.2 (Closed) NRC Temporary Instruction (TI) 2515/169 Mitigating Systems Performance Index Verification

a. Inspection Scope

The objective of TI 2515/169, "Mitigating Systems Performance Index (MSPI)," was to verify that licensees correctly implemented the MSPI guidance for reporting unavailability and unreliability of monitored safety systems as described in NEI 99-02. Safety systems monitored under the MSPI program at TMI included Emergency Alternating Current, High Pressure Injection, Emergency Feedwater, Residual Heat Removal, and cooling water support systems. The inspectors, on a sampling basis, selected key aspects of the MSPI to ensure that AmerGen followed the MSPI guidelines. The inspectors validated the unavailability and unreliability input data and verified accuracy of the reported results for the period April 1 through September 30, 2006. The inspectors performed the following activities:

- Reviewed TMI's MSPI basis document and compared the listed systems, boundaries, and components against the guidance contained in NEI 99-02, Rev. 4, "Regulatory Assessment Performance Indicator Guidance," to verify that AmerGen was monitoring the correct components;

- Reviewed surveillance test procedures, performed in-plant equipment walkdowns, and interviewed station personnel to confirm that equipment credited as remaining available was rendered unavailable only for a short duration, or can be rapidly restored to service using instructions provided in the procedures;

- Reviewed unavailability data for the MSPI target systems which was previously reported under the "Safety System Unavailability" PI for the period of 2002 through 2004. This review was performed to verify that the data was properly incorporated into the planned unavailability for MSPI;

- Reviewed selected work orders, corrective action documents, system health reports, operator logs, and completed surveillance tests reports for the period of January 2005 through September 2006 to verify planned and unplanned unavailability periods for the MSPI systems; and

- Reviewed TMI LERs and a list of corrective action condition reports for the period of January 2005 through September 2006, to identify failures of MSPI monitored components.

b. Findings and Observations

No findings of significance were identified.

The inspectors identified that the baseline planned unavailability hours for the MSPI systems were accurately documented. The inspectors determined that the actual unavailability hours were accurately documented. The inspectors determined that the actual unreliability was correctly documented for the samples selected. The inspectors did not identify any significant errors in the data used to calculate the MSPI value. The inspectors did not identify any significant discrepancies in the MSPI basis document.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On January 11, 2007, 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.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of 10 CFR 50, Appendix R requirements which meets the criteria of Section 8.1.7.1 (c) of the NRC Enforcement Manual, for being dispositioned as an NCV. See Section 4OA3.4 for additional details.

- 10 CFR 50, Appendix R, requires fire protection features to limit fire damage so that one train of systems necessary to achieve and maintain safe shutdown from either the control room or emergency control station is free of fire damage. Contrary to this requirement, on April 23, 2006, engineers identified a control logic error in the elementary circuit drawings for the isolation valves (DH-V-6A and DH-V-6B) which could allow DH-V-6A(B) to spuriously open due to a fire, thereby draining the BWST inventory to the RB sump. This finding is more than minor because it has a credible impact on safety, in that inadvertent draining of the BWST would adversely impact reliability and availability of the high pressure injection mitigating system. This issue was placed in AmerGen’s corrective action program as IR 482679. In addition, interim compensatory measures were promptly implemented, and the condition was corrected by implementation of a design modification to the DH-V-6A(B) control circuitry. The finding is of very low safety significance (Green) due to the existence of a fire detection system, low fire loading in the affected area, and based on the risk informed approach in NRC RIS 2004-03. In this RIS, the NRC states that “Multiple high impedance faults are considered of very low likelihood.”

ATTACHMENT: SUPPLEMENTAL INFORMATION

## SUPPLEMENTAL INFORMATION

## KEY POINTS OF CONTACT

Licensee Personnel
B. Carsky, Director, Operations
S. Baker, Chemistry, Radwaste, Environmental Manager
P. Bennett, Mechnical Design Engineer
T. Dougherty, Plant Manager
E. Eilola, Director, Site Engineering
E. Eisen, System Engineer
J. Heischman, Director, Maintenance
W. McSurley, Operations
A. Miller, Regulatory Assurance
C. Smith, Regulatory Assurance Manager
R. West, Vice President, TMI Unit 1
C. Wend, Radiation Protection Manager
J. Valent, System Engineering Manager

## LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

| | | |
|---|---|---|
| 05000289/2006006-01 | NCV | Inattentiveness To Duty By an Operations Shift Manager (Section 4OA5) |
| 05000289/2006006-02 | NCV | Failure To Follow FFD Procedures (Section 4OA5) |

Closed

| | | |
|---|---|---|
| 05000289/2005001-01 | LER | Control Building Ventilation Fan Inoperable Due To Cracked Fan Hub |
| 05000289/2006001 | LER | Design Change Error for the Decay Heat Valves Connecting the BWST and the RB Sump Negatively Impacted the Fire Mitigation Strategy for an Auxiliary Building Fire Area (Section 4OA3.4) |

## LIST OF DOCUMENTS REVIEWED

**Section 1R04: Equipment Alignment**

Drawings:
302-640, “Decay Heat Removal,” Rev. 79
302-641, “Decay Heat Pumps 1A/1B Aux. Systems,” Rev. 6
302-660, “Makeup and Purification,” Rev. 41
302-712, “Reactor Building Spray,” Rev. 48

Procedures:
1082.1, “TMI Risk Management Program,” Rev. 6
OP-TM-214-000, “Building Spray System,” Rev. 6
OP-TM-212-000, Decay heat removal System, Rev. 6
WC-AA-101, “On-Line Work Control Process,” Rev. 13

Other:
Equipment Tagging Clearance No. 06501506

**Section 1R06: Flood Protection Measures**
Other Documents:
Updated Final Safety Analysis Report (UFSAR) Section 2.6.4, “Flood Studies”
TMI Fire Hazard Analysis Report, Section 6.0, “Protection Against Water Spray to Conform with 10 CFR 50, Appendix R”
Section 10, “Internal Flooding Analysis”, from TMI Unit-1 Probabilistic Risk Assessment (Level 1) Update

**Section 1R11: Licensed Operator Requalification Program**
Procedures:
EP-AA-1009, “Radiological Emergency Response Plan - TMI Station Annex,” Rev. 7
EP-AA-112-100-F-01, “Shift Emergency Director Checklist,” Rev. E
EP-MA-114-100-F-01, “State/Local Event Notification Form,” Rev. D
OP-TM-EOP-001, “Reactor Trip,” Rev. 7
OP-TM-EOP-002, “Loss of 25F Subcooling Margin,” Rev. 5
OP-TM-EOP-006, “LOCA Cooldown,” Rev. 5
OP-TM-EOP-010, “Emergency Procedure Rules, Guides and Graphs,” Rev. 6

Issue Reports:
562920

**Section 1R15: Operability Evaluations**
Drawings:
302-831, “Reactor, Auxiliary and Fuel Handling Bldgs”, Rev. 54

Procedures:
ER-AA-380, “Primary Containment Leakrate Testing Program”, Rev. 4
M-58, “AH-V-1A/1B/1C/1D bearing, Seat and Yoke Maintenance”, Rev. 27
OP-TM-823-252, “Local Leak Rate Testing of Purge Supply Penetration Valves”, Rev. 1

Other Documents:
AR 462228
6610-PLN-4200.01, “Offsite Dose Calculation Manual (ODCM)”, Rev. 25

**Section 1R17: Permanent Plant Modifications**

ECR 06-00297, Rev. 0, "Install Vent Provision BS-V-83"
A/R A2139423, "Install Vent Provision BS-V-83"

**Section 1R22: Surveillance Testing**

Issue Reports:

| | |
|---|---|
| 562514 | 563398 |

**Section 20S1: Access controls**
**Section 20S2: ALARA Planning and Controls**
**Section 20S3: Radiation Monitoring Instrumentation and Protective Equipment**

Benchmarking Report (Recent)
Self-Assessment - 299322
Nuclear Oversight (NOS) Quarterly Report - July 2006
NOS Rapid Trending Report - October 2006
Plant source term analysis data
Various radiation monitor calibration and operability check data
Various radiological survey records for completed work activities including records
Various radiation work permits for completed work activities and associated ALARA plans.
Various personnel whole body count data results
Radiological Controls Contamination Logs
Five Year Source Term Reduction Plan
TMI 2006-1010 Exposure Reduction Plan
TMI 2006 Department Exposure Reduction Plans
TMI 2006 Department Dose Summaries
various 2006 Station ALARA Committee Meeting Minutes
RP-TM-460-1002, Rev.1, Access Control for Locked High Radiation Areas
RP-AA-460, Rev.11, Controls for High and Very High Radiation Areas
RP-TM-460-1008, Rev 0, Locked High Radiation Area Key Controls

**Section 4OA2: Identification and Resolution of Problems**

Calculations:
C-1101-826-5360-016, "Control Building Appendix R - Loss of HVAC," Rev. 2
DR5375339A-2, "TMI Unit 1 Control Tower Loss of Ventilation Study," Rev. 1
DR5375339B-2, "TMI Unit 1 Control Tower Loss of Chiller Study," Rev. 0

Issue Reports: (*NRC Identified During Inspection)

| | | |
|---|---|---|
| 226640 | 290431 | 268512 |
| 271790 | 273568 | 541610 |

Drawings:
302-842, "Control Building and Machine Shop Ventilation," Rev. 7
302-847, "Control Building Chilled Water," Rev. 21

Procedures:
1104-19, "Control Building Ventilation System," Rev. 69
OP-TM-AOP-034, "Loss of Control Building Cooling," Rev. 5

Safety Evaluations:
SE-000827-001, “DCN-Change Quality Class of Chilled Water System,” Rev. 0
SE-412384-017, “Appendix R Evaluation of Control Building Ventilation,” Rev. 1

**Section 4OA3: Event Follow-Up**
Procedures:
IC-192, “Calibration of Condenser Hood Pressure Tansmitters,” Interim Change 16173
IC-192, “Calibration of Condenser Hood Pressure Tansmitters,” Rev. 3
MA-AA-716-004, “Complex Troubleshooting,” Rev. 0
OP-AA-108-107, “Switchyard Control,” Rev. 2
OP-AA-108-107-1001, “Station Response to Grid Capacity Conditions,” Rev. 2
OP-AA-108-108, “Unit Restart Review,” Rev. 6
OP-TM-108-108-1008, “TMI-1 Supplement to OP-AA-108-108,” Rev. 4
OP-AA-108-114, “Post Transient Review,” Rev. 2
OP-TM-EOP-001, “Reactor Trip,” Rev. 7
OP-TM-EOP-010, “Emergency Procedure Rules, Guides, and Graphs,” Rev. 6

Issue Reports:

| | | | | | |
|---|---|---|---|---|---|
| 552589 | 552591 | 552598 | 552623 | 552633 | 552703 |
| 552188 | 553202 | 553664 | 553754 | 553858 | 553905 |
| 554026 | 554854 | 554864 | 554865 | 557450 | 557455 |
| 557462 | 557465 | 569086 | 569118 | | |

Other Documents:
On-Line Station Risk Evaluation Document 619, “Digital Turbine Control System,” Rev. 3
TMI-1 Updated Final Safety Analysis Report, Section 10.3.2, “Turbine Bypass,” Rev. 15
TMI-1 Updated Final Safety Analysis Report, Section 10.7.4, “Overpressure Protection,” Rev. 15
Plant Operations Review Committee Meeting 2006-34 Meeting Review Materials
Drawing SS-209-654, “Reactor Coolant Pump Power Monitor System Separation Requirements,” Rev. 1
Drawing SS-209-661, “Reactor Coolant Pump ‘B’ Power Monitor 1,” Rev. 8
Drawing SS-209-756, “Emergency Feedwater Auto-Start Actuation,” Rev. 13
NRC Generic Letter 2006-02, “Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power,” dated February 1, 2006

**Section 4OA5: Other**
Procedures:
1303-11.39A, “HSPS - EFW Auto Initiation,” Rev. 35
OP-TM-211-203, “IST of MU-V-14A/B and KH-V-7A/B,” Rev. 0
OP-TM-211-204, “IST of MU-V-36 and MU-V-37,” Rev. 0
OP-TM-211-205, “IST of MU-P-2A,” Rev. 2
OP-TM-211-206, “IST of MU-P-2B,” Rev. 2
OP-TM-211-449, “Aligning MU-P-1B to 1D 4160 volt Bus,” Rev. 2
OP-TM-424-201, “IST of EF-P-2A,” Rev. 2
OP-TM-424-202, “IST of EF-P-2B,” Rev. 2

Other Documents:
NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Rev. 4
Plant Operating Review Committee Meeting 2006-29 Minutes dated October 17, 2006
Reactor Oversight Program MSPI Basis Document - TMI Nuclear Station, 2nd Quarter 2006
Selected Operator Logs for January 2002 through September 2006

## LIST OF ACRONYMS

| | |
|---|---|
| ADAMS | Agencywide Documents and Management System |
| ALARA | As Low As is Reasonably Achievable |
| AmerGen | AmerGen Energy Company, LLC |
| BS | Building Spray |
| BWST | Borated Water Storage Tank |
| CFR | Code of Federal Regulations |
| CEDE | Committee Effective Dose Equivalent |
| DHR | Decay Heat Removal |
| DRP | Division of Reactor Projects |
| DTCS | Digital Turbine Control System |
| ED | Electronic Dosimeter |
| EFW | Emergency Feedwater |
| FFD | Fitness For Duty |
| FHAR | Fire Hazards Analysis Report |
| HRA | High Radiation Area |
| IMC | Inspection Manual Chapter |
| IR | Issue Report |
| IST | Inservice Testing |
| KV | Kilovolt |
| LDE | Lens Dose Equivalent |
| LER | Licensee Event Report |
| MR | Maintenance Rule |
| MSPI | Mitigating Systems Performance Index |
| NCV | Non-Cited Violation |
| NEI | Nuclear Energy Institute |
| NOV | Notice of Violation |
| NRC | Nuclear Regulatory Commission |
| NUMARC | Nuclear Management and Resources Council |
| ODCM | Offsite Dose Calculation Manual |
| OE | Operational Experience |
| OI | Office of Investigation |
| PARS | Publicly Available Records |
| PI | Performance Indicator |
| PMT | Post-Maintenance Test |
| RB | Reactor Building |
| RCA | Radiologically Controlled Area |
| RCP | Reactor Coolant Pumps |
| RETS | Radiological Effluent Technical Specifications |

| | |
|---|---|
| RHR | Residual Heat Removal |
| RIS | Regulatory Issue Summary |
| RO | Reactor Operator |
| RPS | Reactor Protection System |
| RWP | Radiation Work Permit |
| SDE | Shallow Dose Equivalent |
| SDP | Significance Determination Process |
| SRO | Senior Reactor Operator |
| SSC | Structures, Systems and Components |
| SSFF | Safety System Functional Failures |
| SBO | Station Blackout |
| TEDE | Total Effective Dose Equivalent |
| TMI | Three Mile Island, Unit 1 |
| TS | Technical Specifications |
| UFSAR | Updated Final Safety Analysis Report |
| UT | Ultrasonic Testing |
| VHRA | Very High Radiation Area |