November 4, 2005

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 05000289/2005005

Dear Mr. Crane:

On September 30, 2005, the Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Three Mile Island, Unit 1 (TMI) facility. The enclosed report documents the inspection findings that were discussed October 12, 2005, with Mr. Glen Chick 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.

The report documents two NRC-identified 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.

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 05000289/2005005
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
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Manager Licensing - TMI, AmerGen
Vice President - General Counsel and Secretary, AmerGen
T. O'Neill, Associate General Counsel, Exelon Generation Company
J. Fewell, Esq., Assistant General Counsel, Exelon Nuclear
Correspondence Control Desk - AmerGen
Chairman, Board of County Commissioners of Dauphin County
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R. Janati, Director, Bureau of Radiation Protection, State of PA
J. Johnsrud, National Energy Committee
E. Epstein, TMI-Alert (TMIA)
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J. Boska, NRR
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U.S. NUCLEAR REGULATORY COMMISSION

REGION 1

Docket No:	05000289
License No:	DPR-50
Report No:	050000289/2005005
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Three Mile Island Station, Unit 1
Location:	PO Box 480 Middletown, PA 17057
Dates:	July 1, 2005 - September 30, 2005
Inspectors:	David M. Kern, Senior Resident Inspector Javier M. Brand, Resident Inspector Jennifer A. Bobiak, Reactor Inspector, DRS Patrick W. Finney, Reactor Inspector, DRS James A. Krafty, Reactor Inspector, DRS Thomas A. Moslak, Health Physicist, DRS Andrew A. Rosebrook, Project Engineer, DRP
Approved by:	Ronald R. Bellamy, Chief Projects Branch 7 Division of Reactor Projects (DRP)

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000289/2005005; 7/01/2005 - 9/30/2005; AmerGen Energy Company, LLC; Three Mile Island, Unit 1; Heat Sink, Maintenance Implementation.

The report covers 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**. The inspectors identified a non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." Neither procedures or acceptance criteria were established, nor were visual inspection results documented, to support verification that the reactor building fan emergency cooling coils were sufficiently maintained to perform their intended safety function. Consequently, AmerGen did not fulfill their NRC Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment," commitment to perform visual inspections and trend the material condition of the air side of the cooling coils each refueling outage.

  This finding is greater than minor because it affected the reactor safety cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Additionally, if left uncorrected, the finding could become a more significant safety concern in that inspections of the reactor building emergency cooling coils were not performed and trended as committed to in the licensee's NRC GL 89-13 response. The finding was of very low safety significance because the reactor building emergency cooling coils remained capable of performing their safety function. This finding has been entered into the licensee's corrective action program as issue report 371356.

  A contributing cause of this finding is related to the cross-cutting area of human performance, because AmerGen did not develop sufficient resources, such as complete and accurate procedures, to ensure the visual heat exchanger inspections were performed and trended. The finding is also cross-cutting in the area of problem identification in that station personnel had completed several periodic reviews and self assessments of the GL 89-13 program, and did not identify that some of the required inspections had not been performed, had no procedures or acceptance criteria, and were not documented. (Section 1R07)

Cornerstone: Barrier Integrity

- **Green.** The inspectors identified a non-cited violation of 10 CFR 50.65(a)(2)/(a)(1) in that the licensee's demonstration of effective control of performance or condition of the control building and machine shop heating and ventilation system had become invalid, and although the licensee had a reasonable number of opportunities, the licensee did not place the system in (a)(1) status in a reasonable amount of time. Consequently, the licensee did not establish goals and monitor the performance or condition of the Control Building & Machine Shop Heating & Ventilation System as required by 10 CFR 50.65(a)(1) when the demonstration of effective control of performance or condition of the system through appropriate preventive maintenance as allowed by 10 CFR 50.65(a)(2) became invalid. The demonstration of effective control of performance or condition in (a)(2) status became invalid as a result of multiple maintenance preventable functional failures within a 3-year period, the most recent of which was a failure on June 28, 2005, that the licensee did not correctly recognize as a maintenance preventable functional failure.

  This finding is more than minor because it affects the Barrier Integrity Cornerstone and its design attribute of maintaining the functionality of the control room envelope. Additionally, if left uncorrected it could impact the licensee's ability to properly trend performance and establish goals to provide reasonable assurance that structures, systems, and components (SSCs) within the scope of the Maintenance Rule remain capable of fulfilling their intended functions. This finding was determined to be of very low safety significance because the incorrect functional failure assessment did not, by itself, result in an actual degradation of the radiological barrier function provided for the control room. The licensee has entered the issue into their corrective action program as issue report 349025.

  A contributing cause of this finding is a cross-cutting issue in the area of human performance, because maintenance personnel did not properly perform procedure 1400-F-1, to achieve the required AH-F-3B filter gasket compression. The inspectors determined this was also a cross-cutting issue in the area of problem resolution, because the maintenance rule functional failure review was too narrowly focused. It did not address why the bypassed filters' gaskets did not show signs of compression, did not address criteria for post event inspection of the failed filters, and did not consider potential procedural compliance aspects. As a result, corrective actions identified to prevent recurrence were too narrowly focused. (Section 1R12)

B. Licensee Identified Violations

A violation of very low significance regarding technical specification requirements was identified by the licensee. 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

AmerGen Energy Company, LLC (AmerGen), operated Three Mile Island, Unit 1 (TMI) at or near 100 percent rated thermal power throughout the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment (71111.04 - 3 samples)

a. Inspection Scope

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

- On July 6, the inspectors walked down portions of the makeup and purification system, and verified configuration alignment of makeup pumps MU-P-1B and MU-P-1C while maintenance personnel replaced the 4160V breaker on makeup pump MU-P-1A. The walkdown was performed in conjunction with ultrasonic testing weld preparations on the suction lines between the borated water storage tank and emergency core cooling systems.

- On August 8, the inspectors walked down portions of the reactor building spray system. The inspectors verified configuration alignment of building spray pumps BS-P-1A and BS-P-1B and the building spray valves between the sodium hydroxide tank and the decay heat pumps.

- On September 13, the inspectors walked down portions of the turbine driven emergency feedwater (EFW) pump (EF-P-1) and associated components, while both flow paths of EFW system to the 'B' steam generator were isolated due to scheduled surveillance testing of the 'B' motor driven EFW pump (EF-P-2B).

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.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05 - 10 samples)

a. Inspection Scope

The inspectors performed 10 inspection samples. The inspectors conducted fire protection inspections for several plant fire zones, selected based on the presence of equipment important to safety within their boundaries. The inspectors conducted plant walkdowns and verified the areas were as described in the TMI Fire Hazard Analysis Report (FHAR), and that fire protection features were being properly controlled per surveillance procedure 1038, "Administrative Controls-Fire Protection Program," Rev. 63. The plant walkdowns were conducted throughout the inspection period and included assessment of transient combustible material control, fire detection and suppression equipment operability, and compensatory measures established for degraded fire protection equipment in accordance with procedure OP-MA-201-007, "Fire Protection System Impairment Control," Rev. 2. 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. Additional documents reviewed during the inspection are listed in the Attachment. Fire zones and areas inspected included:

- AB-FA-2, Auxiliary Building Elev. 261', 'B' Building Spray Pump Vault
- AB-FZ-6, Auxiliary Building Elev. 305', Demineralizer & 'A' Motor Control Center Area
- CB-FA-2D, Control Building Elev. 322', East Battery Charger Areas
- CB-FA-3A thru 3D, Control Building Elev. 338', 4160 1D, 1E Switchgear Rooms
- FH-FW-1, Fuel Handling Building Elev. 281'
- IB-FZ-2, Intermediate Building Elev. 295', Turbine Driven EFW Pump Area
- IB-FZ-3, Intermediate Building Elev. 295', Motor Driven Emergency Feedwater (EFW) Pump Area
- IB-FZ-4, Intermediate Building, Elev. 295'
- IB-FZ-6, Intermediate Building Elev. 322'
- SBO-FA-1, Station Blackout (SBO) Diesel Generator Building

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 1 sample)

Internal Flood Protection

a. Inspection Scope

On July 19 to 21, the inspectors reviewed and assessed protection measures for internal flooding events associated with the 'A' and 'B' emergency diesel generator (EDG) rooms. The inspectors reviewed the updated final safety analysis report

(UFSAR) and other selected design basis documents to identify those areas susceptible to internal flooding, and reviewed TMI Unit 1 Probabilistic Risk Assessment (Section 10: Internal Flood Analysis) and associated flood protection reports to identify risk significant flood areas and protective features. The inspectors performed visual inspections of flood barriers, system boundaries, and reviewed documents to identify potential water sources that could impact the EDG rooms. The inspectors evaluated whether appropriate flood protection controls and appropriate precautions to mitigate the risk from internal flooding events were being implemented. In addition, the inspectors assessed the potential hazards for common mode flooding and whether the floor drains were being maintained to prevent backflow of water into unwanted areas. The inspectors also interviewed the EDG system engineer and the TMI flood protection specialist. The documents used to support this inspection included:

- 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"

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07A - 2 samples)

a. Inspection Scope

The inspectors performed two inspection samples. The inspectors evaluated the heat removal capability of the 'A' decay heat closed river water heat exchanger. This component is a water-to-water heat exchanger with river water from the decay heat river water system on the tube side and decay heat closed cycle cooling on the shell side. The inspectors reviewed work order R1831792, which documented a clean-and-inspect activity performed on November 1, 2003. The inspectors verified that the results met the acceptance criteria. The responsible system engineer was interviewed to verify the inspection results were appropriately categorized against preestablished acceptance criteria, the frequency of inspection is sufficient, and that various bio-fouling treatment processes are utilized to ensure continued satisfactory heat exchanger performance.

The inspectors also reviewed the heat removal capability of the cooling coils (RR-1-A, B, C) associated with the reactor building cooling fans AH-E-1A, 1B, and 1C. These coils are air-to-water heat exchangers with river water from the reactor building emergency cooling water system (RR) on the tube side and the reactor building atmosphere on the air side. The inspectors also reviewed applicable documents to ensure the licensee was implementing the commitments made to meet the recommended actions to NRC Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment," regarding these coils. In addition, the inspectors reviewed TMI's Heat Exchanger System Health Report and interviewed the system engineers and their

supervisors. Additional documents reviewed during the inspection are listed in the Attachment. The following documents were also reviewed;

- Topical Report No. 119, "NRC GL 89-13 Program Description," Rev. 3, dated February 28, 2005
- TMI's response to NRC GL 89-13 documented in GPU Nuclear Letter 6710-96-2097, dated June 6, 1996
- Surveillance procedure 1303-11.9, "Reactor Building Emergency Cooling System," Rev. 63, completed on November 17, 2003 under work order R2010277
- Preventive maintenance procedure IC-250, "Reactor Building Cooling Fans-Air Flow Rate Measurements," Rev. 0, completed on November 18, 2003 under work order R2012007
- Work order R1834688, "AH-E-1C Cooling Coil River Side Inspection," completed on November 14, 2003

b. Findings

Introduction. The inspectors identified a Green NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the lack of procedures and acceptance criteria to ensure visual inspections of material condition of the air side of the safety-related reactor building emergency cooling coils were performed and documented each refueling outage. Consequently, AmerGen did not fulfill a commitment made to address recommendations in NRC GL 89-13.

Description. NRC GL 89-13 was issued because a number of nuclear industry events indicated that some service water systems were not properly designed or maintained to ensure the systems would perform their intended safety function. The NRC requested that licensees either complete the recommended actions to address the specific areas of concern or develop equally effective actions to ensure that latent failures do not remain unidentified. Specifically, the NRC recommended that licensees implement a routine inspection and maintenance program that assures corrosion, erosion, protective coating failure, silting, and bio-fouling do not degrade performance of the safety-related systems supplied by service water.

The licensee informed the NRC by letter, GPU Nuclear 6710-96-2097, dated June 6, 1996, of their plans to perform a combination of visual inspections and trending of the reactor building fan emergency cooling coils of both the air side and the water side during each refueling outage. The letter states in part, that "The inspection, trending, and chemistry control programs provide a reliable means to verify adequate heat transfer capability to satisfy the requirements of NRC GL 89-13, enclosure 2."

During this inspection, station personnel were unable to provide the inspectors with documentation of the inspections or trend analysis performed on the air side of the reactor building emergency cooling coils. AmerGen did not have any procedures or work instructions to perform the required visual inspections of the air side of the reactor

building emergency cooling coils. However, the system engineer was able to find photographs taken during the last two refueling outages (Fall of 2001 and 2003) which depicted that the cooling coils were in good condition. These photographs were taken by engineers for activities unrelated to NRC GL 89-13. For example, in 2003, inspections were performed as part of a follow-up action to assess the effects of a minor boric acid leak that existed inside the reactor building containment.

Analysis. The lack of instructions or procedures and acceptance criteria to ensure required visual inspections of the reactor building emergency ventilation fan cooling coils were performed, constituted a performance deficiency. The inspectors determined that this finding was more than minor because the failure to ensure proper cooling coil condition has the potential to impact the equipment performance attribute of the Mitigating Systems cornerstone. In this case, the finding potentially affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences.

This finding was evaluated using NRC Manual Chapter 0609, "Significance Determination Process," Appendix A, Phase 1. Engineers were able to show that visual inspections were performed and photographs of the coils had been taken during the two previous refueling outages (Fall of 2001 and 2003). These inspections were performed by engineers as part of separate activities, unrelated to GL 89-13. For example, the 2003 inspections were due to a known minor boric acid leak that occurred during the 2002/2003 operating cycle. The inspectors noted that these visual inspections did not specifically evaluate or have acceptance criteria for corrosion, erosion, protective coating failure, or debris as recommended by NRC GL 89-13. The photographs of the cooling coils, both as-found and as-left, did not reveal any actual degradation or loss of safety function. The inspectors concluded that the reactor building fan emergency cooling coils remained capable of performing their design safety function and therefore this finding was considered to be of very low safety significance (Green).

A contributing cause of this finding is related to the cross-cutting area of human performance, because AmerGen did not develop sufficient resources, such as complete and accurate procedures, to ensure the visual heat exchanger inspections were performed and trended. The finding is also cross-cutting in the area of problem identification, because although station personnel had completed several periodic reviews and a self assessment specifically focused at NRC GL 89-13 program implementation, they did not identify that the reactor building emergency cooling coil air side visual inspection requirements were not met.

Enforcement. Technical Specification (TS) 6.8.1 and NRC Regulatory Guide 1.33 require written procedures for certain safety-related activities including inspections of the reactor building containment heat removal system. Title 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that instructions, procedures or drawings including appropriate quantitative or qualitative acceptance criteria be provided for determining that important activities have been satisfactorily accomplished. Contrary to the above, AmerGen did not provide instructions or

procedures with acceptance criteria to ensure visual inspections of material condition of the air side of the safety-related reactor building fan emergency cooling coils were performed and documented each refueling outage. Consequently, AmerGen did not fulfill a commitment made to address recommendations in NRC GL 89-13. Because this violation was of very low safety significance and was entered into the TMI corrective action program (IR 371356), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. **NCV 05000289/2005005-01, No Procedures or Acceptance Criteria to Ensure Visual Inspections of Reactor Building Fan Air Side Emergency Cooling Coils Were Performed and Documented.**

1R11 Licensed Operator Requalification (71111.11Q - 2 samples)

a. Inspection Scope

The inspectors performed two inspection samples. On September 12 and 19, the inspectors observed licensed operator requalification training at the control room simulator for the 'D' and 'B' operator crews, respectively. The inspectors reviewed the operators' ability to correctly evaluate the simulator training scenario and implement the emergency plan. The inspectors observed the operators' simulator drill performance and compared it to the criteria listed in simulator scenario No. 9, "Reactor Coolant Pump High Vibration and Sheared Shaft, Turbine Trip Failure, and RCS Leak Degrading to a Large Break Loss of Coolant Accident," Rev. 9. The inspectors observed supervisory oversight, command and control, communication practices, and crew assignments to ensure they were consistent with normal control room activities. The inspectors observed operator response during the simulator drill transients and verified the fidelity of the simulator to the actual plant. The inspectors evaluated training instructor effectiveness in recognizing and correcting individual and operating crew errors, including post-training remediation actions. The inspectors attended the post-drill critiques in order to evaluate the effectiveness of problem identification. The inspectors verified that emergency plan classification and notification training opportunities were tracked and evaluated for success in accordance with criteria established in Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," Rev. 2. Additional documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R12 Maintenance Implementation (71111.12 - 2 samples)

a. Inspection Scope

The inspectors performed two inspection samples. The inspectors evaluated Maintenance Rule (MR) implementation for the issues listed below. Specific attributes reviewed included MR scoping, characterization of failed structures, systems, and components (SSCs), MR risk categorization of SSCs, SSC performance criteria or

goals, and appropriateness of corrective actions. The inspectors verified that the issues were addressed as required by 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants;" NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Rev. 2; and AmerGen procedure ER-AA-310, "Implementation of the Maintenance Rule," Rev. 3. Additional documents reviewed during the inspection are listed in the Attachment.

- On July 5, the inspectors evaluated the licensee's evaluation of the dioctyl phthalate (DOP) test failure of the control building emergency ventilation high efficiency particulate filter AH-F-3B from a maintenance rule perspective. The inspectors verified that appropriate corrective actions were initiated and documented with IRs. Extent of condition follow-up, operability, and functional failure determinations were reviewed to verify they were appropriate for the filter failure.

- IR 308057 described a diagnostic test failure of the cooling water system inlet valve DC-V-2A that occurred on March 3, 2005. The inspectors reviewed the performance and maintenance history of this valve and evaluated the licensee's response from a maintenance rule perspective. Engineers determined that although the valve was found to have increased friction and did not meet the seat load criteria, the actuator spring force was sufficient for opening the valve against loads imposed by system flow, differential pressure across the disc, and the higher than expected friction loads. The evaluation also determined that this condition was not a maintenance rule functional failure because it did not affect operability of the valve. The inspectors verified that the operability of the valve was not affected and verified that an adequate extent-of-condition review was performed.

b. Findings

Introduction. The inspectors identified a Green non-cited violation of 10 CFR 50.65(a)(2)/(a)(1) when station personnel incorrectly concluded that the June 28, 2005, AH-F-3B failure was not maintenance preventable. Consequently, the licensee did not establish goals and properly monitor the performance or condition of the control building and machine shop heating & ventilation system as required by 10 CFR 50.65(a)(1) when the demonstration of effective control of performance or condition of the system through appropriate preventive maintenance as allowed by 10 CFR 50.65(a)(2) became invalid.

Description. On June 28, 2005, the high efficiency particulate air (HEPA) filter in the AH-F-3B Control Room Emergency Filtering System failed the penetration and system bypass DOP test portion of surveillance procedure 1303-11.13, "Control Room Filtering System Test," Rev. 16. The system was declared inoperable as required by TS 3.15.1, the filters were replaced and successfully tested, and the system was declared operable on June 30. The inspectors reviewed the maintenance and test documents and

interviewed the maintenance contractor who performed the work to better understand what had caused the failure.

Maintenance history indicated that all 40 filter cartridges in AH-F-3B had been replaced in April 2004. Procedure 1303-11.13 was performed successfully following their replacement and no maintenance was performed on the AH-F-3B HEPA filter since then. During testing on June 28-29, 2005, technicians determined that six of the 40 filters were leaking around the filter gasket that provides the seal between the filter cartridge and the filter frame. Since the six leaking filters were from the same manufacturer, all 18 filters on AH-F-3B from this manufacturer were replaced.

Engineers concluded that the AH-F-3B failure was not a maintenance preventable functional failure (MPFF), as documented in IR 349025. They attributed the test failure to the filters being from a specific manufacturer. No actual filter defect was identified or documented. Subsequent failure analysis testing by the filter manufacturer did not identify any deficiencies. The licensee's conclusion was based on 1) the April 2004 post maintenance test was successful, 2) no maintenance was done on AH-F-3B since the successful April 2004 test, and 3) the DOP test was successful following replacement of all 18 filters from the suspect manufacturer lot on June 30, 2005.

The inspectors noted that indentations from gasket compression are typically evident when filters have been properly installed in the filter housing. Some of the failed filters showed no indication of gasket compression. The maintenance contractor stated that although this particular manufacturer's filters met dimensional tolerances, they were on the small side of the specified tolerance and may not have overlapped the filter frame enough to provide sufficient surface area to seal against the gasket. The inspectors noted that procedure 1440-F-1, "Air Handling Filter Replacement," Rev. 13, step 8.9.4 required clamping and tightening filters until gasket compression was 75 to 95%. The contractor indicated that it was difficult to check gasket compression because the filters are close together and extend out a significant distance from the frame.

The inspectors determined that the licensee's functional failure assessment was deficient in that it did not address two important pertinent issues: 1) some of the failed filters show no signs of gasket compression when the filter replacement procedure (1440-F-1) required 75 to 95% gasket compression; 2) failure analysis performed on the filter cartridges found no problems. The inspectors determined that the bypass flow past the six filter cartridge gaskets and absence of gasket compression were evidence that the six filter cartridges had not been properly installed. The inspectors concluded that the licensee had improperly categorized the filter failure, and the AH-F-3B test failure was an MPFF, because filter cartridge installation and gasket compression were not performed in accordance with procedure 1440-F-1. The filter's MPFF, on June 28, 2005, was the fourth MPFF in a 3 year period for the Control Building & Machine Shop H&V System and caused it to exceed its maintenance rule performance criteria of <4 MPFFs in a rolling 3-year period.

In response to the inspectors' concerns, the Maintenance Rule Expert Panel directed that an expedited apparent cause evaluation be performed to support reevaluation of the functional failure classification (IR 349025, Action 5).

Analysis. Failure to properly characterize and account for one MPFF that caused the licensee's 10 CFR 50.65 (a)(2) demonstration of the Control Building & Machine Shop H&V System performance to become invalid is a performance deficiency. If left uncorrected, this maintenance rule knowledge deficiency could impact the licensee's ability to properly trend performance and establish goals to provide reasonable assurance that SSCs within the scope of the Maintenance Rule remain capable of fulfilling their intended functions.

This finding is more than minor because it affects the Barrier Integrity Cornerstone and its design attribute of maintaining the functionality of the control room envelope. Additionally, if left uncorrected it could impact the licensee's ability to properly trend performance and establish goals to provide reasonable assurance that structures, systems, and components (SSCs) within the scope of the Maintenance Rule remain capable of fulfilling their intended functions. The inspectors evaluated the issue using NRC Manual Chapter 0609, "Significance Determination Process," Appendix A, Phase 1. This finding was determined to be of very low safety significance (Green) because the incorrect functional failure assessment did not, by itself, result in an actual degradation of the radiological barrier function provided for the control room. Further, although the deficient filter replacement maintenance activity degraded the control room radiological barrier function, the system remained capable of providing the designed control room radiological barrier function.

A contributing cause of this finding is a cross-cutting issue in the area of problem resolution, because the maintenance rule functional failure review was too narrowly focused. It did not address why the bypassed filters' gaskets did not show signs of compression, did not address criteria for post event inspection of the failed filters, and did not consider potential procedural compliance aspects. As a result, corrective actions identified to prevent recurrence were too narrowly focused. The inspectors determined there was also a cross-cutting issue in the area of human performance, because maintenance personnel did not properly perform procedure 1400-F-1, step 8.9.4 to achieve required 75-95% AH-F-3B filter gasket compression.

Enforcement. Title 10 CFR 50.65 (a)(1) requires that licensees monitor performance or condition of in-scope SSCs against established goals to provide reasonable assurance that these SSCs are capable of fulfilling their intended function. Title 10 CFR 50.65 (a)(2) provides that monitoring as specified in paragraph (a)(1) is not required where it has been demonstrated that the performance or condition of an SSC is being effectively controlled through the performance of appropriate preventive maintenance, such that the SSC remains capable of performing its intended function. Contrary to the above, the licensee did not establish goals and monitor the performance or condition of the Control Building & Machine Shop Heating & Ventilation System as required by 10 CFR 50.65(a)(1) when the demonstration of effective control of performance or condition of the system through appropriate preventive maintenance as allowed by 10 CFR

50.65(a)(2) became invalid. The demonstration of effective control of performance or condition in (a)(2) status became invalid as a result of four MPFFs within a 3-year period, the most recent of which was a failure of filter AH-F-3B, on June 28, 2005, that the licensee did not correctly recognize as a MPFF. Because the finding was of very low safety significance and was entered into the licensee's corrective action program (IR 349025), the violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy: **NCV 05000289/2005005-02, Deficient AH-F-3B Functional Failure Assessment and Maintenance Rule Performance Monitoring and Goal Setting.**

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13 - 6 samples)

a. Inspection Scope

The inspectors performed six samples inspection. The inspectors reviewed the scheduling and control of maintenance activities in order to evaluate the effect on plant risk. This review was against criteria contained in AmerGen Administrative Procedure 1082.1, "TMI Risk Management Program," Rev. 4 and WC-AA-101, "On-Line Work Control Process," Rev. 10. The inspectors reviewed the routine planned maintenance, restoration actions, and/or emergent work for the following equipment removed from service. Additional documents reviewed during the inspection are listed in the Attachment.

- On July 15, fire service pump FS-P-1 was removed from service for periodic surveillance testing using procedure 3303-M1, "Fire System Valve Lineup Verification," Rev. 54. (Risk document 516, Rev. 6)

- On August 10, contractors prepared a weld on a common line from the borated water storage tank to emergency core cooling systems for non-destructive examination (NDE). Inspectors discussed the NDE preparation with Operations, Maintenance, Work Management and Engineering personnel to verify risk had been properly assessed. (AR 361543, work order C2010672)

- On August 29, technicians performed electrical breaker maintenance and testing on the 'C' reactor building emergency cooling heat exchanger discharge valve, RR-V-4C. (Risk document 1143, Rev. 1)

- On August 30, operators performed procedure 1300-3K, "Inservice Testing of Reactor River Water Pumps and Valves," Interim Change 18711. (Risk document 565, Rev. 5)

- On September 23, technicians replaced a leaking air actuator trip valve for the 'A' main steam atmospheric dump valve MS-V-4A. (Risk documents 749 and 911)

- On September 23 to 26, emergent maintenance activities were performed to investigate and correct the cause of an unexpected low oil pressure trip of the 'A' control building chiller unit AH-C-4A (Risk document 1146).

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions (71111.14 - 2 samples)

a. Inspection Scope

The inspectors performed two inspection samples. The inspectors reviewed human performance during the following non-routine plant evolutions, to determine whether personnel performance caused unnecessary plant risk or challenges to reactor safety. The inspectors evaluated whether the evolutions were properly implemented according to the applicable procedures and TS limiting condition for operations (LCOs). Additional documents reviewed during this inspection are listed in the Attachment.

- On August 15, control room operators received a control rod drive motor fault alarm and the integrated control system (ICS) automatically transferred to "track," as designed. Operators transferred the Group 7 control rods to the auxiliary power supply in accordance with procedure 1202-8, "Control Rod Drive Equipment Failure," Rev. 56 and verified plant stability. Troubleshooting activities were performed from August 15 to September 1 to identify and correct the cause of the motor fault alarm using work orders A2121874 and C2011028. The Group 7 control rods were transferred between the normal power supply and the auxiliary power supply several times during troubleshooting activities. Technicians replaced several components, which corrected symptoms, but did not fully correct the problem. The troubleshooting plan was properly updated based on their results and technicians ultimately replaced the CRD-PC-4 programmer control assembly. The motor fault was corrected and Group 7 control rods were returned to the normal power supply on September 1.

- On August 16, operators observed an unplanned power increase from 100% reactor power while restoring the ICS to automatic mode, during recovery from a Group 7 control rod drive mechanism motor fault (IR 363336). An hour earlier, operators had placed the ICS in manual in response to the motor fault. The power increase was caused by failure of the steam generator/reactor master controller during ICS restoration. Operators responded promptly to limit the unplanned reactivity increase using procedure OP-TM-621-471, "ICS Manual Control," Rev. 1. The inspectors verified that the maximum power reached was 101.3% and that power was restored to below 100% within 7 minutes.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 4 samples)

a. Inspection Scope

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

- On July 25, engineers identified that the reactor building fan motor drain line check valves were not periodically tested and, as a result, the reactor building fans may not be environmentally qualified for potential water intrusion during a loss of coolant accident (IR 356528). The check valves (AH-V-20A, AH-V-20B, and AH-V-20C) were shut on July 27 as a compensatory measure. Operability evaluation OPE-05-013, "AH-V-68A, AH-V-68B, AH-V-68C," Rev. 0 concluded the reactor building fans remained operable.

- In August, Heat Sink Protection System (HSPS) selector switches were identified through surveillances to be failing at an unexpectedly high frequency. The inspectors interviewed operations personnel and the system engineer. The inspectors also reviewed operability evaluation OPE-05-015, "HSPS Selector Switch," Rev. 0. The evaluation concluded that there was reasonable assurance that the HSPS will perform its design function to actuate and control EFW flow and that the switches remain operable.

- On September 1, during a scheduled air operator test of 'A' decay closed cooling water system inlet valve DC-V-2A, technicians identified that the instrument air supply to the valve was directly connected to the positioner and then to the actuator, thereby bypassing the regulator. This condition resulted in greater than expected air pressure being supplied to these components (100 psi vs 80 psi). The inspectors reviewed IRs 369196, 369389, 308057, 3079919, and engineering calculation C-1101-900-E420-184, "ACE AOV Design Basis Capability Calculation For DC-V-2A," Rev. 0. The inspectors verified that valve DC-V-2A is a normally open valve designed to fail open on loss of instrument air and that its safety function was to open to provide cooling water flow to the cooler. The engineering evaluation concluded that both the valve positioner and the actuator were designed for air pressures greater than 100 psi and would not

be affected by the increased air pressure condition and the valve remained operable.

- On September 22, during performance of a scheduled in-service test (IST) surveillance, technicians identified that the "A" main steam atmospheric dump air operated valve (MS-V-4A) exceeded the established acceptable valve opening stroke time of 3.4 seconds (alert value). The first test stroke time was 3.9 seconds. A second stroke test showed at 5.21 seconds, which exceeded the maximum allowed opening stroke time of 5.0 seconds. The valve was declared inoperable and troubleshooting was performed. A leaking air actuator trip valve was replaced, and the valve stroked open at 3.5 seconds during a post maintenance test. Engineers did not believe the leaking air trip valve was the cause for the slower stroke times. However, a subsequent IST performed showed repetitive stroke tests within 3.5 seconds, which is well within the maximum allowable range. A diagnostic air test was also performed and the valve stem friction was found within the allowable range. The inspectors reviewed the engineering operability evaluation documented in IR 376878. The evaluation determined that although no cause for the increased stroke times was identified, the valve was operating reliably. The evaluation also determined that the valve safety function is to open with no specified time. Based on IST requirements, engineers increased the test frequency for this valve from quarterly to monthly.

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

- On July 6, station personnel performed testing of makeup pump MU-P-1A following replacement of its 4160V breaker. Testing was performed in accordance with procedure FTP 732.01, "4160V Breaker Replacement," Rev. 3.

- On July 22, station personnel performed testing of the station blackout (SBO) Diesel Generator following its scheduled outage. Testing was performed in accordance with procedure 1107-9, "SBO Diesel Generator," Rev. 50.

- On September 1, station personnel tested Group 7 control rod motion following replacement of the CRD-PC-4 programmer control assembly. Testing was

performed in accordance with work order C2011028, task 07 and procedure OP-TM-622-201, “Control Rod Motion,” Interim Change 18670.

- On September 1, station personnel performed diagnostic testing of decay closed cooling water valve DC-V-2A, following maintenance activities to rebuild the air actuator. Testing was performed in accordance with procedure MA-AA-743-310, “Diagnostic Testing And Evaluation Of Air Operated Valves,” Rev. 4.

- On September 26, station personnel performed post maintenance testing of the “A” control building chiller AH-C-4A following troubleshooting and replacement of the auxiliary lubricating oil pump pressure cut-off switch, a clogged eductor dryer filter, and an oil heater indicating a blown fuse. The PMT was completed satisfactorily per Work Order M2124364. The inspectors verified that an 8-hour confidence run was also completed satisfactorily.

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

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22 - 7 samples)

a. Inspection Scope

The inspectors performed seven inspection samples. The inspectors observed and reviewed the following operational surveillance tests, concentrating on verification of the adequacy of the test to demonstrate the operability of the required system or component safety function. Inspection activities included review of previous surveillance history to identify previous problems and trends, observation of pre-evolution briefings, and initiation/resolution of related IRs for selected surveillances. Additional documents reviewed during the inspection are listed in the Attachment.

- On July 7-8, procedure IC-57.1, “MS-V-3A-F Actuator Testing,” Rev. 0.

- On July 27, procedure 1301-1, “Shift and Daily Checks,” Rev. 150, Section B.3 (Reactor Building Air Temperature) during a period of elevated summer temperature.

- On August 12, operators performed interlock testing on spent fuel bridge FZ-A-3 in preparation for fuel moves and the upcoming refueling outage. The inspectors observed the performance of procedure 1303-11.4, “Refueling System Interlock Tests”, Rev. 42. The inspectors also reviewed IRs associated with the bridge and interviewed the operators and system engineer.

- On August 30, procedure 1303-4.19, “High Pressure Injection/Low Pressure Injection Analog Channel Test,” Rev. 25.

- On August 30, procedure 1300-3K, “In-service Testing of Reactor River Water Pumps and Valves,” Interim Change 18711. During preparation for this inspection, the inspectors determined that procedure revisions had not fully addressed test methodology deficiencies previously documented in NRC Inspection Report 05000389/2004004. In response to the inspector’s concerns, engineers implemented further corrective actions (IRs 244066, 336716, and 357423) to ensure the test methodology met the American Society of Mechanical Engineers Operation and Maintenance Code requirements.

- On September 13, procedure OP-TM-424-202, “In-service Testing for EF-P-2B,” Rev. 2. The inspectors verified that an operator remained in the area of specific valves isolated for the test as required by the procedure and TS Section 3.4.2.1. The inspectors also verified that system unavailability was properly tracked and documented.

- On September 22, technicians performed OP-TM-411-204, “Quarterly Test Of MS-V-4A And MS-V-4B Valves During Normal Plant Operations For IST,” Rev. 3. The “A” main steam atmospheric dump air operated valve (MS-V-4A) exceeded the maximum allowed opening stroke time of 5.0 seconds (stroke was 5.21 seconds). The valve was declared inoperable and troubleshooting was performed. A leaking air actuator trip valve was replaced. A subsequent IST showed repetitive stroke tests within 3.5 seconds, well within the maximum allowable range.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 1 sample)

a. Inspection Scope

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

- TM 05-00582, “Connect NI-12 Cable J1 To A4 Pre-amplifier,” Rev. 1. This modification was installed as a temporary solution to address significant signal degradation of the control room panel wide range nuclear power level indication, degradation of plant process computer wide range indication Point A1052, and spiking of the source range startup rate plant process computer Point A0578. This condition required entry into a 30-day LCO per TS 3.5.7, “Remote Shutdown System.”

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness [EP]

1EP6 Drill Evaluation (71114.06 - 1 sample)

a. Inspection Scope

The inspectors performed one inspection sample. The inspectors observed an emergency event training evolution conducted at the Unit 1 control room simulator to evaluate emergency procedure implementation, event classification, and event notification. The event scenario involved multiple safety-related component failures and plant conditions warranting a simulated General Emergency event declaration. The licensee counted this training evolution for evaluation of Emergency Preparedness Drill/Exercise Performance Indicators (PIs). The inspectors observed the drill critique to determine whether the licensee critically evaluated drill performance to identify deficiencies and weaknesses. Additionally, the inspectors verified the Drill/Exercise PIs were properly evaluated consistent with Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," Rev. 2.

- On July 13, the inspectors evaluated Emergency Preparedness Training Drill T03 from the control room simulator.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety

2PS3 Radiological Environmental Monitoring Program (71122.03 - 10 samples)

a. Inspection Scope

During the period July 18- 22, the inspectors conducted the following activities to verify that the licensee implemented the Radiological Environmental Monitoring Program (REMP) consistent with the site TSs and the Off-site Dose Calculation Manual (ODCM) to validate that radioactive effluent releases met the design objectives of Appendix I to 10 CFR 50. Additionally, the inspectors verified that radiological surveys and controls were adequate to prevent the inadvertent release of radioactive material into the public domain. Implementation of these controls was reviewed against the criteria contained in 10 CFR 20 & 50, relevant TSs, and the licensee's procedures. This inspection activity represents completion of ten samples relative to this inspection area.

Field Inspections

The inspectors evaluated the material condition of air samplers and observed the change out of particulate and iodine filter samples at five air sampling stations located at ODCM map designations E1-2, F1-3, H3-1, G2-1, and A3-1.

The inspectors evaluated the material condition of water compositors and observed the collection of water samples at six water sampling stations located at ODCM map designations J1-2, K1-1, G15-3, G15-2, A3-2, and Q9-1.

The inspectors evaluated the material condition of seven environmental dosimeters located at ODCM map locations J1-1, H1-1, K1-4, L1-1, E1-2, F1-1, and H3-1.

The inspectors observed the processing of three milk samples taken at ODCM map locations G2-1, D2-1, and E2-2.

The inspectors reviewed the calibration records and maintenance records, including quality control records, for the in-place water compositors and air sampling equipment.

The inspectors reviewed the sampling procedures implemented for collecting water, air, and milk samples.

The inspectors reviewed the results of the vender's inter-laboratory comparison program to verify the adequacy of the licensee's environmental sample analyses.

The inspectors reviewed the results of the annual (2004) land use census to determine if changes in sampling locations were warranted.

The inspectors verified that the meteorological instruments were operable, calibrated, and properly maintained. The inspectors compared local readouts of wind speed/direction and delta temperature at the meteorological tower with those in the control room to determine if there were any significant differences in data.

Unrestricted Release of Material from the Radiologically Controlled Areas

The inspectors observed a technician performing source and system operability checks on the Small Article Monitors (SAMs) 9 and 11 located at the radiological control points in the turbine building and auxiliary building. The inspectors confirmed that the SAM's had the appropriate detection sensitivity and alarmed at a conservative set-point. SAM calibration records were reviewed to assure that the instruments were properly calibrated and maintained.

The inspectors reviewed the most current 10 CFR Part 61 sampling data of nuclide distribution, and the licensee's application of this data in determining that the radiation monitoring instruments, calibration sources, and alarm set-points were appropriate for the identified radiation types.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152 - 1 sample)

.1 Annual Sample - Station Battery Health Due to Cell Replacement and Recurrent Grounds

a. Inspection Scope

The inspectors selected one sample for review. The inspectors reviewed several IRs relating to station battery grounds and cell degraded voltage. The inspectors reviewed Exelon's responses to the IRs to ensure that the full extent of the issues were identified, appropriate evaluations were performed, and appropriate corrective actions were specified and prioritized. The inspectors walked down affected areas of the plant and interviewed relevant station personnel. The inspectors evaluated Exelon's responses to the IRs against the requirements of Exelon's corrective action program. Additional documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified. However, the inspectors identified that Exelon did not complete a common-cause analysis required for the identified trend of station battery grounds as specified in procedure LS-AA-125-1005, "Coding and Analysis Manual," Rev. 4. The lack of performing a common-cause analysis did not, in this case, adversely impact the licensee's corrective actions or system performance. The inspectors verified that the cause of the grounds was known and that corrective actions were appropriate. Therefore, this issue was considered minor and no findings of significance were identified. This issue was captured in IR 355344.

.2 REMP and Radioactive Material Control Program

a. Inspection Scope

The inspectors reviewed 17 IRs, a Nuclear Oversight Department Audit of the REMP, a REMP self-assessment, and vender audits to evaluate the licensee's threshold for identifying, evaluating, and resolving problems in implementing the REMP and radioactive material control program. This review was conducted against the criteria contained in 10 CFR 20, TSs, and the licensee's procedures.

b. Findings

No findings of significance were identified.

.3 Cross-References to PI&R Issues Reviewed Elsewhere

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

Section 1R07 describes a finding in which station personnel performed several periodic nuclear oversight audits and a self assessment, but did not identify an important deficiency. These audits and the self assessment specifically focused on implementation of the GL 89-13 safety-related heat exchanger program, but did not identify that required reactor building emergency cooling coil air side visual inspection requirements were not met.

Section 1R12 describes a finding in which the maintenance rule functional failure review was too narrowly focused. It did not address why the bypassed filters' gaskets did not show signs of compression, did not address criteria for post-event inspection of the failed filters, and did not consider potential procedural compliance aspects. Thus, engineers did not establish performance goals for the control building and machine shop heating and ventilation system as required by 10 CFR 50.65(a)(1)/(a)(2). As a result, corrective actions identified to prevent recurrence were too narrowly focused.

4OA3 Event Follow-up (71153)

(Closed) Licensee Event Report (LER) 05000289/2005001, Control Building Ventilation Fan Inoperable Due To Cracked Fan Hub

On February 3, 2005, maintenance personnel and engineers identified a cracked fan hub during replacement of the control building ventilation fan (AH-E-19B). The fan was being replaced due to a degraded elevated noise condition that existed since December 2001, and degraded further on August 20, 2003. Subsequent assessment concluded the cause of the cracked hub was age related degradation in combination with torque stresses. 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).

AmerGen documented its analysis of the event and its immediate and planned corrective actions in its corrective action process (IRs 258108, and 297543). The inspectors reviewed the LER and associated licensee corrective actions. The engineering evaluation determined that the failure caused a condition where the fan would not have been able to perform its 30-day mission time safety function between August 20, 2003 to September 29, 2004. The engineering evaluation also concluded that the root cause for the untimely investigation and repairs of the degraded fan was that Engineering, Maintenance, and Operations did not apply appropriate rigor to the

troubleshooting, operability determination, and decision-making regarding component health. No new performance issues were identified during this review. However, since this condition resulted in a condition prohibited by TS, this issue was determined to be a licensee identified finding and is documented in Section 4OA7 of this report. This LER is closed.

4OA4 Cross-Cutting Aspects of Findings

Human Performance

Section 1R07 describes a finding in which AmerGen did not develop sufficient resources, such as complete and accurate procedures, to ensure the visual heat exchanger inspections ensure required visual inspections of the reactor building fan emergency cooling coils were performed and trended. This resulted in the licensee's inability to fulfill a commitment made to adequately address recommendations in NRC GL 89-13.

Section 1R12 describes a finding in which maintenance personnel did not properly perform procedure 1400-F-1, to achieve required 75-95% AH-F-3B filter gasket compression. Consequently, the control room ventilation radiological barrier was degraded.

4OA5 Other

.1 TI 2515/163, Operational Readiness of Offsite Power - Follow-up Inspection Activities

Cornerstones: Initiating Events, Mitigating Systems

a. Inspection Scope

The inspectors performed additional follow-up inspection to supplement NRC assessment of Temporary Instruction 2515/163, "Operational Readiness of Offsite Power." The inspectors collected and reviewed licensee procedures and supporting information pertaining to the offsite power system specifically relating to the areas of offsite power operability, the maintenance rule (10 CFR 50.65), and the station blackout rule (10 CFR 50.63). The inspectors reviewed this data against the requirements of 10 CFR 50.63; 10 CFR 50.65; 10 CFR 50 Appendix A General Design Criterion 17, "Electric Power Systems"; and Plant TSs.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

Exit Meeting Summary

On October 12, 2005, the resident inspectors presented the inspection results to Mr. Glen Chick 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 confirmed that proprietary information was not provided or examined during the inspection.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as an NCV.

- TS 3.15.1 requires that two trains of control room emergency ventilation system be operable. Further, with one train inoperable, unit operation can continue for 7 days. If the inoperable train is not returned to operable status within 7 days, the unit is to be shutdown within 48 hours. With both trains inoperable, unit shutdown is required within 48 hours. Contrary to these requirements, engineers determined that one train of control room emergency ventilation (AH-E-19B) was inoperable from August 20, 2003 to September 29, 2004, due to a cracked fan hub and the reactor was not shut down. Additionally, the licensee identified that on two occasions between August 20, 2003 and September 29, 2004, the second control room emergency ventilation train was inoperable due to scheduled maintenance for greater than a 48-hour period and the plant was not shutdown as required. The inspectors determined this issue was of very low safety significance (Green), because this finding represented only a degradation of the radiological barrier function provided for the control room. This issue was determined to be of very low safety significance (Green) using Manual Chapter 0609, Appendix A, Attachment 1, screening worksheet for the Barrier Integrity cornerstone, since there was no actual safety consequences, and this event did not adversely affect health and safety of the public. AmerGen documented its analysis of the event and its immediate and planned corrective actions in its corrective action process (CRs-258108, and 297543). See Section 4OA3.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel
S. Acker, ODCM Specialist/REMP Coordinator
S. Baker, Radiation Protection Manager
G. Chick, Plant Manager
A. Miller, Regulatory Assurance
C. Smith, Regulatory Assurance Manager
R. Walton, Chemistry Manager
C. Wend, Radiation Protection Manager
R. West, Vice President, TMI Unit 1

Others
M. Murphy, Pennsylvania Department BRP

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

2005005-01	NCV	No Procedures or Acceptance Criteria to Ensure Visual Inspections of Reactor Building Fan Air Side Emergency Cooling Coils Were Performed and Documented (Section 1R07)
2005005-02	NCV	Deficient AH-F-3B Functional Failure Assessment and Deficient Maintenance Rule Performance Monitoring and Goal Setting (Section 1R12)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment
Drawing 302-082, "Emergency Feedwater Flow Diagram," Rev. 22
Drawing 302-660, "Makeup and Purification Flow Diagram," Rev. 41
Drawing 302-661, "Makeup and Purification Flow Diagram," Rev. 54
Drawing 302-712, "Reactor Building Spray Flow Diagram," Rev. 46
OP-TM-424-202, "IST of EF-P-2B," Rev. 2

Section 1R05: Fire Protection
IR 361073, FS-PI-515E Reading 40 psig Higher than Expected

Section 1R11: Licensed Operator Requalification

Procedures
1102-4, “Power Operations, “ Rev.
OP-TM-226 Series, “Reactor Coolant Pumps,” Rev.
OP-TM-EOP-001, “Reactor Trip,” Rev. 6
OP-TM-EOP-002, “Loss of 25 F Subcooled Margin,” Rev. 4
OP-TM-EOP-005, “OTSG Tube Leak,” Rev. 2
OP-TM-EOP-006, “LOCA Cooldown,” Rev. 4
OP-TM-EOP-010, “Emergency Procedures Rules, Guides And Graphs,” Rev. 4
EP-AA-111, “Emergency Classification And Protective Action Recommendations,” Rev. 10
EP-AA-1009, Table TMI3-1: Emergency Action Level (EAL) Matrix,” Rev. 3

Section 1R12: Maintenance Implementation

IR 349025, AH-F-3B HEPA Filters Failed DOP Test
IR 349018, Control Building DOP Testing LCO Duration Mis-Communication
IR 349266, RB Purge Terminated Due to High Exhaust Temperature
IR 349038, Notes from the Inspection AH-F-3B
IR 357805, Filters from Suspect Lot in CB Emergency Filter AH-F-3A.

Section 1R13: Maintenance Risk Assessments and Emergent Work Evaluation

IR 362379, NDE Prep WO Activities do not Invoke Procedure Requets

Section 1R14: Personnel Performance During Non-routine Plant Evolutions

Prompt Investigation Report for IR 363336
OP-AA-300-1540, “Reactivity Management Administration,” Rev. 1
OP-TM-622-451, “Transferring Rods to Auxiliary Power Supply,” Rev. 0
OP-TM-622-201, “Control Rod Movement,” Interim Change 18670
Work Order M2121461

Section 1R22: Surveillance Testing

IR 362483, Improper FW-A-3 Status Light Indication
IR 363697, Spent Fuel Bridge Air Receiver Blowdown Valve Stuck Close

Section 4OA2: Problem Identification and Resolution

Action Requests

00201213	00201954	00218283	00226918	00234567	00235642
00239258	00240401	00240651	00241793	00246228	00248039
00248372	00249355	00254310	00254313	00261494	00261800
00263393	00272037	00276800	00294914	00294928	00318402
00330490	00334591	00335381	00341147	00350068	A2072262

Completed Surveillances
1303-11.11, “Station Battery Load Test,” Rev. 30, Completed 11/05/03 and 11/11/03

Miscellaneous
Battery Ground Data Graph 07/01/04 - 07/18/05
DC System Monitoring Plan, 02/08/05
EED-B-1A Pilot Cell Voltage Data Graph, 06/30/05
EED-B-1B Pilot Cell Voltage Data Graph, 07/14/05
FSAR Section 8.2.2.6
OPE-04-016, "EED-B-1B Operability Evaluation," Rev. 0
System Health Report, 250/125 VDC System, June 2005
VM-TM-0021, "C&D Power Systems Station Batteries Vendor Manual," Rev. 14

Operating Experience Documents
OE-19078
OE-19134
OE-19481

Procedures
1107-2C, "Vital DC Electrical System," Rev. 3
1420-DC-5, "Locate Grounds on Station Battery Distribution System," Rev. 3
ER-AA-310, "Implementation of the Maintenance Rule," Rev. 3
LS-AA-125, "Corrective Action Program (CAP) Procedure," Rev. 8
LS-AA-125-1005, "Coding and Analysis Manual," Rev. 4
MAP A, "Main Annunciator Panel A," Rev. 40
OP-TM-PPC-C4123, "Station Battery 1A(B) Ground Resistance LO Alarm," Rev. 0

LIST OF ACRONYMS

ADAMS	Agencywide Documents and Management System
AmerGen	AmerGen Energy Company, LLC
CFR	Code of Federal Regulations
DOP	Dioctyl Phthalate
EDG	Emergency Diesel Generator
EFW	Emergency Feedwater
GL	Generic Letter
HSPS	Heat Sink Protection System
ICS	Integrated Control System
IMC	Inspection Manual Chapter
IR	Issue Report
IST	In-service Test
LCO	Limiting Condition for Operation
MPFF	Maintenance Preventable Functional Failure
MR	Maintenance Rule
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
ODCM	Off-site Dose Calculation Manual

PARS	Publicly Available Records
PI	Performance Indicator
PMT	Post Maintenance Test
psi	pounds per square inch
REMP	Radiological Environmental Monitoring Program
RR	Reactor Building Emergency Cooling Water System
SAM	Small Article Monitor
SBO	Station Blackout
SDP	Significance Determination Process
SSCs	Structures, Systems and Components
TM	Temporary Modification
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report