July 28, 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 INSPECTION REPORT 05000289/2005004

Dear Mr. Crane:

On June 30, 2005, the Nuclear Regulatory Commission (NRC) completed an inspection at your Three Mile Island, Unit 1 (TMI) facility. The enclosed report documents the inspection findings that were discussed July 20, 2005, 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.

The report documents one self-revealing finding and four inspector-identified findings of very low safety significance (Green). These findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating them as non-cited violations (NCVs) consistent with Section VI.A of the NRC Enforcement Policy. If you contest the NCVs, you should provide a response within 30 days of the date of this inspection report, 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 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.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, Chief
Reactor Projects Branch 7
Division of Reactor Projects

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

Enclosure: Inspection Report 05000289/2005004
w/Attachment: Supplemental Information

cc w/encl:
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Site Vice President - TMI Unit 1, AmerGen
Plant Manager - TMI, Unit 1, AmerGen
Regulatory Assurance Manager - TMI, Unit 1, AmerGen
Senior Vice President - Nuclear Services, AmerGen
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Vice President - Operations Support, AmerGen
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U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	05000289
License No:	DPR-50
Report No:	050000289/2005004
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Three Mile Island Station, Unit 1
Location:	PO Box 480 Middletown, PA 17057
Dates:	April 1, 2005 - June 30, 2005
Inspectors:	David M. Kern, Senior Resident Inspector Javier M. Brand, Resident Inspector Ram S. Bhatia, Reactor Inspector, DRS Patrick W. Finney, Reactor Inspector, DRS Christopher A. Hott, Health Physicist, DRS Nancy T. McNamara, Emergency Preparedness Inspector, DRS Ronald L. Nimitz, Senior Health Physicist, DRS
Approved by:	Ronald R. Bellamy, Chief Projects Branch 7 Division of Reactor Projects (DRP)

Enclosure

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SUMMARY OF FINDINGS

IR 05000289/2005004; 04/01/2005 - 06/30/2005; AmerGen Energy Company, LLC; Three Mile Island, Unit 1; Maintenance Implementation, Post Maintenance Testing, and Radioactive Material Processing and Transportation.

The report covers a 13-week period of inspection by resident inspectors and announced inspections by regional inspectors. Five Green non-cited violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a non-cited violation (NCV) of Technical Specification (TS) 6.8.1.a for deficient maintenance procedures on safety-related system expansion joints, and for not performing engineering evaluations when in-service safety-related expansion joints exceeded their recommended service life. The maintenance procedure and scheduling inadequacies resulted in station personnel being unaware of the age or condition of numerous expansion joints that had exceeded their recommended service life by an unknown period of time.

  This finding was more than minor because it affected the mitigating systems cornerstone and affected the reliability of two trains of a nuclear river water mitigating safety system. In all three systems that were reviewed, expansion joints would have continued to degrade if left uncorrected. Additionally, two expansion joints in the condensate system were degraded. The complete failure of these partially collapsed expansion joints would likely result in an initiating event. The finding is of very low safety significance since no equipment was rendered inoperable due to the aged expansion joints.

  A contributing cause of this finding is related to the cross-cutting area of human performance, because maintenance and testing procedures were insufficient to provide reasonable assurance that safety related and important-to-safety expansion joints would continue to remain capable of performing their design functions. Specifically, procedures did not address expansion joint service life, incorporate industry experience, or specify vendor recommended inspections be performed to support the continued use beyond the established service life. (Section 1R12)

- Green. A self revealing non-cited violation of TS 6.8.1.a was identified for not properly implementing maintenance procedures that affected the performance of the safety-related 'B' emergency diesel generator (EDG). Licensee staff did not properly apply lubricant and torque the exhaust manifold bolts to the EDG turbocharger. This caused an exhaust leak and degraded the EDG during a monthly surveillance run due to loose and missing bolts in an exhaust manifold extension. Maintenance personnel performed an extent-of-condition investigation and documented the occurrence in their corrective action program.

  This finding is more than minor because it affects the mitigating systems cornerstone objective of ensuring reliability of systems that respond to initiating events and is associated with the equipment performance reliability attribute. The finding is of very low safety significance since the missing bolt did not cause the EDG to become inoperable.

  A contributing cause of this finding is a cross-cutting issue in the area of human performance, because maintenance personnel did not follow work instructions to apply lubricant and torque the turbocharger exhaust manifold bolts, document final torque values, or document lubrication used in completed work orders. A second contributing cause affected the cross-cutting area of problem resolution, because the initial engineering evaluation was too narrowly focused. Engineers did not use technical calculations or modeling to support conclusions regarding the quantity of leaking exhaust and its associated impact on diesel loading capability and room design temperature until challenged by the inspectors. (Section 1R12)

- Green. The inspectors identified a non-cited violation of TS 6.8.1.a in that on March 29, 2005, operators did not properly implement procedural requirements for recharging the two-hour emergency air system, and mispositioned valve IA-V-1769. The mispositioned valve caused both air banks to partially depressurize and reduced the reliability of the supported mitigating systems (emergency feedwater (EFW) and main steam (MS)) to perform their decay heat removal function. Operators identified and repressurized the air banks, but did not recognize and correct the cause of the degraded condition until the inspectors identified the causes.

  The finding was more than minor because the degraded two-hour air system pressure affected the reliability of the EFW and MS systems to perform their accident mitigation functions in response to initiating events. The deficiency affected the configuration control, equipment performance, and human performance attributes of the mitigating system cornerstone. The finding is of very low safety significance because bank air pressure did not drop below the value required for operability and, therefore, the system remained capable of performing its safety function.

  A contributing cause of this finding is related to the cross-cutting area of human performance, because operators did not follow procedural instructions to open IA-V-1769 and procedure quality was deficient in that procedure usage category

3 (informational use only) was insufficient to ensure the procedure was properly followed step-by-step for this important safety-related activity. The finding is also cross-cutting in the area of problem resolution in that AmerGen's initial assessment of the event did not determine or correct the actual causes of the degraded air bank pressure. (Section 1R19)

- Green. The inspectors identified a non-cited violation of TS 6.8.1.a for deficient maintenance procedures that did not contain sufficient work instruction or acceptance criteria to ensure the safety related 'B' 125/250 volt battery was properly reassembled following replacement of battery cell #2. Additionally, workers did not properly follow the procedure instructions in that certain steps were performed out of order.

  This issue affected the mitigating systems cornerstone and was more than minor because it affected the reliability of the 'B' train of the 125/250 volt power system to perform its accident mitigation functions in response to initiating events. The deficiency affected the procedure quality and equipment performance attributes of the mitigating system cornerstone. The finding is of very low safety significance because the 'B' 125/250 volt battery bank was not inoperable for greater than the TS allowed outage time.

  A contributing cause of this finding is related to the cross-cutting area of human performance, because operators did not follow procedure 1420-DC-3 steps in the order specified and procedure quality was deficient because it did not provide instruction to perform intercell battery resistance checks or torque the battery rack connection bolts to verify seismic qualification prior to declaring the battery operable. Additionally, procedure usage level was insufficient based upon the potential impact of an error. (Section 1R19)

Cornerstone: Public Radiation Safety

- Green. The inspectors identified a non-cited violation of 10 CFR 20.1501 associated with failure to evaluate the adequacy of a change to the procedure for collecting samples of radioactive spent resin for analysis to support transfer of radioactive material to a waste processor for ultimate disposal. Specifically, in December 1998, AmerGen reduced the tank recycle requirements, prior to sample collection, from three tank volumes to 15 minutes, and did not evaluate the effect of this change on the representativeness of the sample. Consequently, the spent resin tank sample procedure instruction was not evaluated to ensure a representative sample, and therefore AmerGen could not validate that the total radionuclide activity was accurately determined and provided to the waste processor prior to the shipment in accordance with 10CFR20, Appendix G.

  The finding is greater than minor in that it affected the public radiation safety cornerstone objective. Specifically, the issue involved an occurrence in the radioactive material transportation program that was contrary to NRC or Department of Transportation regulations. Using the Public Radiation Safety

SDP flow chart, this finding is of very low safety significance, because it involved a radioactive material control issue, it did involve transportation, no radiation limit was exceeded, it did not involve a breach of packaging, it did not involve a Certificate of Compliance finding, it did not involve a low-level burial ground issue, and it did not involve a failure to make an emergency notification issue. AmerGen reviewed previous shipments and concluded that, due to the generally low radioactivity of the shipments made, there was no likelihood that a shipment was improperly packaged for shipment or would have been misclassified per 10 CFR 61. Consequently, no actual safety consequence was identified.

A contributing cause of this finding is related to the cross-cutting area of problem identification in that AmerGen did not identify this problem during routine self-assessments and audits of its radioactive waste transportation and disposal program. (Section 2PS2)

B. Licensee Identified Violations

None.

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, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01 - 1 sample)

a. Inspection Scope (Summer Readiness)

The inspectors reviewed the TMI design features and AmerGen's implementation of procedures to protect risk significant mitigating systems and components from adverse weather effects due to high temperatures. In addition, the inspectors reviewed AmerGen summer readiness preparation for TMI documented in memorandum 5500-2005-012, dated May 13, 2005. This letter discusses the actions taken or planned in accordance with WC-AA-107, "Seasonal Readiness," Rev. 1. The inspectors also reviewed the corrective action data base to verify that AmerGen personnel are identifying and resolving weather-related equipment problems.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04S - 1 sample; 71111.04Q - 3 samples)

a. Inspection Scope

Complete System Walkdown

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

- On June 2, 2005, the inspectors verified configuration alignment of the nuclear services system following a planned maintenance system outage on the NS-P-1C pump. The inspectors conducted a detailed review of the alignment and condition of the system using the applicable one-line diagram 302-610, "Nuclear Services Closed Cycle Cooling Water," Rev. 74 and procedure OP-TM-541-272, "Operating Mode Lineup Verification," Rev. 0. In addition, the inspectors reviewed and evaluated the corrective action program reports for impact on system operation and interviewed the system engineer.

Partial System Walkdowns

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

- On April 21, the inspectors verified configuration alignment of the three emergency feedwater (EFW) pumps and the 'A' train of the 2 hour backup instrument air supply to the EFW injection valves while maintenance personnel replaced valves IA-V-1617B and IA-V-1618B. These valves had been leaking and degraded the 'B' train 2 hour backup instrument air supply to the EFW injection valves.

- On April 26-27, the inspectors walked down the 'A' low pressure injection (LPI) train along with portions of the 'A' decay heat closed cooling water, 'A' decay heat river water, and 'A' makeup systems during a planned maintenance outage on the 'B' LPI train.

- On May 2, the inspectors walked down the 'B' LPI train along with portions of the 'B' building spray and 'B' nuclear services closed cooling water system during a planned maintenance outage on the 'A' LPI train.

The partial system walkdowns were conducted on the redundant and standby equipment to ensure that trains and equipment relied on to remain operable for accident mitigation were properly aligned and protected. Additional documents reviewed during the inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05A - 1 sample; 71111.05Q - 9 samples)

2. Annual Drill Observation

a. Inspection Scope

The inspectors performed one inspection sample. The inspectors observed an unannounced fire brigade drill on June 16 to evaluate the readiness of station personnel to respond to and combat fires. The drill demonstrated response to a fire in the Unit 1 Turbine Building South Switchgear Room (322 foot elevation). The inspectors observed fire brigade members' use of protective clothing and turnout gear, including self-contained breathing apparatus, and their methodology and communications in fighting the fire. The inspectors reviewed the drill scenario objectives, determined whether the objectives were met and observed the post-drill critique to verify that the licensee identified, discussed and entered adverse conditions and performance into the corrective action program.

b. Findings

No findings of significance were identified.

2. Area Walkdowns

a. Inspection Scope

The inspectors performed nine 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, 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 were controlled per procedure OP-MA-201-007, "Fire Protection System Impairment Control," Rev. 2. In addition, the inspectors verified that applicable clearances between fire doors and floor met the specified criteria per Attachment 1 of Engineering Technical Evaluation CC-AA-309-101, "Engineering Technical Evaluations," Rev. 7. Fire zones and areas inspected included:

- Fire Zone TB-FA-1, Turbine Building
- Fire Zone CB-FA-1, Control Building (305' elev.)
- Fire Zone DG-FA-1, 'A' Emergency Diesel Generator room
- Fire Area DG-FA-2, 'B' Emergency Diesel Generator room
- Fire Zone IB-FZ-2, Intermediate Building - Turbine Driven EFW Pump room
- Fire Area CB-FA-3D, Control Building - Relay room (338'6" elev.)
- Fire Area AB-FA-1, Auxiliary Building - Decay Heat Removal Pit 'A' (261' elev.)

- The inspectors reviewed issue report (IR) 338092 and action request (AR) A2066403, which evaluated a degraded starter direct current (DC) motor for one of the diesel driven fire pumps (FS-P-1). The evaluation determined that the starter continued to work properly and the operability of the fire pump is not affected. The inspectors interviewed the fire protection system engineer and electrical technicians, and verified that lubricating oil samples taken at the diesel crank case were normal. The inspectors also verified that actions were in place to replace the DC starter motor as soon as practical.

- The inspectors reviewed IR 340803 which documented that Air Intake Tunnel fire suppression system halon bottle pressures were degraded, but remained operable until they would be refilled during the next scheduled surveillance (September 2005). The inspectors questioned whether the surveillance procedure instructions were sufficient and whether the halon system would remain operable until September. Based upon discussion with the inspectors,

engineers initiated actions to recharge the halon bottles in July, to revise procedure 1303-12.11, “Halon System Tests,” Rev. 41, and to verify small nitrogen bottles were maintained available onsite for repressurizing the halon bottles.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06 - 1 sample)

a. Inspection Scope

The inspectors reviewed the site’s external flooding mitigation strategy including applicable sections of the Updated Final Safety Analysis Report (UFSAR) and historic IRs. The inspectors verified compensatory measures outlined in Emergency Procedure 1202-32, “Flood,” Rev. 61. The review included floor drains and sumps in the intermediate building (295' elev.), auxiliary building (281' elev.), heat exchanger vault (271' elev.), and flood gates/doors on the Unit 1 auxiliary building and air intake pagoda structure.

b. Findings

No findings of significance were identified.

1R12 Maintenance Implementation (71111.12Q - 4 samples)

a. Inspection Scope

The inspectors performed four 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. 2. Additional documents reviewed during the inspection are listed in the Attachment.

- Evaluated overall maintenance effectiveness of the EFW system by reviewing resolution of selected system material degradation issues reported in IRs during the period April 1, 2003 to April 1, 2005. Emergency feedwater system issues included elevated EF-P-1 bearing water outlet temperatures, and program

implementation of industry operational experience related to flow induced pipe wall erosion.

- Evaluated overall maintenance effectiveness of expansion joints in safety systems and systems important to safety. Expansion joint condition, replacement periodicity, inspection methods, and use of industry operational experience were evaluated. Systems reviewed included the nuclear river water, control building chill water, and condensate water systems.

- IR 286662 described a catastrophic failure of the 'A' reactor building emergency cooling fan AH-E-1A that occurred in December 30, 2004. The inspectors evaluated AmerGen's response to this failure from a maintenance rule perspective. The inspectors also verified that actions were initiated per IRs 337171 and 318327 to address a minor deficiency identified regarding the mis-identification of this failure as not a maintenance rule functional failure. The system engineer determined that this deficiency was caused by the lack of adequate guidance in the applicable maintenance rule procedures (ER-AA-310 and ER-AA-310-1004), regarding potential masking or shadowing of equipment failures in systems that have redundant components. Although the evaluation of the AH-E-1A functional failure for maintenance preventable applicability was still ongoing at the close of this inspection period, the inspectors verified that there has been no other AH-E-1A failures since 1993, and the identified deficiency would not impact the current Maintenance Rule (a)(2) categorization of this fan even if it is determined that this failure was a maintenance preventable functional failure.

- IR 344477 described an exhaust leak on the opposite-control side of the 'B' emergency diesel generator (EDG) that occurred on June 16. The inspectors reviewed the maintenance history on the EDG and evaluated the licensee's response to the event from a maintenance rule perspective. Engineers determined that this was not a maintenance rule functional failure. The inspectors verified that the exhaust leak did not affect the operability of the EDG and verified than an adequate extent-of-condition review was performed.

b. Findings

Deficient Maintenance Procedures Result in Undetected Expansion Joint Degradation and Safety-Related Expansion Joints Exceeding Service Life

Introduction. The inspectors identified that the recommended service life for safety-related and risk-important expansion joints was exceeded on several occasions without engineering justification or internal inspections. Expansion joint service life and installation dates were not tracked by station personnel, nor was vendor or industry guidance for inspections incorporated into station procedures. Not maintaining a replacement schedule and not performing adequate inspections of expansion joints in

safety-related systems which resulted in exceeding the recommended service life was determined to be of very low safety significance (Green) and an NCV of TS 6.8.1.a.

Description. The inspectors identified degraded material condition or excessive age for rubber expansion joints in safety-related systems and systems important to safety including: nuclear river water cooling (NR), control building chill water (CBCW), and the condensate system. Specifically, ages of only 5 of the 14 expansion joints inspected could be determined by the licensee. Of the 5 joints whose ages were identified, 4 were beyond the vendor recommended service life of 12 years, and 3 of these 4 joints had visible external degradation. According to station personnel, the joints whose ages were indeterminate may be the original joints installed during plant construction.

On April 1, 2005, the inspectors performed expansion joint walkdowns and identified that two of the three expansion joints in the safety-related NR system had surface degradation. The inspectors questioned the degradation and age of the joints (IRs 320086 and 320094). One of the two degraded joints had a date stamp indicating it was 19 years old. Station personnel could not determine the age of the other joint. In the CBCW system, 7 of the 8 expansion joints inspected were coated with an unidentified material and appeared to be well beyond 12 years old. Station personnel were also unable to determine the age of any CBCW system expansion joints. Station personnel had no justification or basis for exceeding the service life of the joints in these systems. On November 4, 2004, station personnel identified that two of three condensate pump suction expansion joints were partially collapsed. It was then identified through vendor interaction that they had been in service for 15 years. These joints were kept in service at that time on the basis that no surface cracking was observed. However, it was identified by the inspectors on April 4 that these joints had been coated with a rubber crack sealant several years earlier, making it improbable that existing surface cracks could be detected through visual inspection. A catastrophic collapse of a condensate pump suction expansion joint would result in a rapid loss of main condenser vacuum and a subsequent reactor trip.

Station personnel inspected expansion joints in the NR system and CBCW system every two years using repetitive work order A 1726908, “Inspect Group ‘A’ Expansion Joints.” However, the recurring external inspection did not incorporate vendor or industry guidance in that the external inspection procedure did not incorporate recommended acceptance criteria and did not inspect all recommended attributes. This inspection procedure was last performed on the NR system in November 2004, with no deficiencies or abnormalities noted.

Station personnel did not perform internal inspections of expansion joints. Expansion joint installation procedure 1410-Y-36, “Expansion Joint Replacement,“ Rev. 8, does not require internal inspections of expansion joints. The vendor manual and EPRI 1003189, “Expansion Joint Maintenance Guide,” recommend internal inspections by opportunity when the system is opened for maintenance. EPRI 1003189 also has inspection guidance for extending the service life of rubber expansion joints, which includes internal inspections, hydrostatic tests, destructive examination upon removal, and accelerated

aging tests. These tests are necessary because external inspections only provide limited data on joint health, and industry operational experience, also contained in EPRI 1003189, shows that one out of four expansion joint failures occurs via rupture instead of leak.

In response to the inspectors' concerns, plant personnel evaluated the installed expansion joints. Corrective actions included an extent-of-condition review, system walkdowns, consultations with vendors, system calculations, and external inspection of all expansion joints. In addition, plant personnel are in the process of implementing a new program that will ensure replacement of expansion joints on a routine basis. Other corrective actions include replacing expansion joints with unknown in-service lives at the earliest opportunity and installing a temporary repair clamp on the 'B' condensate pump expansion joint. Engineers also determined that the currently installed expansion joints would remain functional until replaced.

Analysis. The numerous expansion joints that exceeded their recommended service life without engineering justification, with no tracking of service life data and not performing adequate inspections, constituted a performance deficiency.

This issue affected the mitigating systems cornerstone and was more than minor because numerous safety-related and important-to-safety expansion joints exceeded their service life without sufficient inspections to justify continued operability. This unevaluated condition affected the reliability of two trains of a nuclear river water mitigating safety system, in that degraded expansion joints are more likely to fail catastrophically during design basis events when greater than normal hydraulic and thermal stresses are placed on those systems. In all three systems that were reviewed, expansion joints would continue to degrade if left uncorrected. In the condensate system, the complete failure of the partially collapsed expansion joints would likely result in an initiating event.

This finding was evaluated using NRC Manual Chapter 0609, "Significance Determination Process," Appendix A, Phase 1, and was determined to be of very low significance since the condition did not involve any actual failure in the safety-related expansion joints. In addition, subsequent engineering analysis determined that NR system configuration does not subject the expansion joints to significant pressure transient forces. Therefore, the NR expansion joints are likely to remain intact and continue to support the NR system design function.

A contributing cause of this finding is a cross-cutting issue in the area of human performance, because maintenance and testing procedures did not address expansion joint service life, incorporate industry experience, or specify vendor recommended inspections to support continued use beyond established service life.

Enforcement. TS 6.8.1.a requires that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Rev. 2, February 1978. Regulatory Guide 1.33,

Appendix A, states that preventive maintenance procedures and schedules should exist to replace or inspect parts that have a specific lifetime. Contrary to this requirement, system engineers did not maintain service life data for routine replacement, nor did a procedure exist for performing internal inspections on expansion joints. In addition, external inspections did not incorporate industry or vendor recommendations. Because this violation was of very low safety significance and was entered into the TMI corrective action program (IRs 320086, 320094), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy.

NCV 05000289/2005004-01, Deficient Maintenance Procedures Result in Undetected Expansion Joint Degradation and Safety-Related Expansion Joints Exceeding Service Life.

Deficient Maintenance Procedures and Personnel Error Degrade Safety-Related Emergency Diesel Generator Exhaust Manifolds

Introduction. A Green self-revealing NCV was identified for failure to implement an adequate maintenance procedure in accordance with TS 6.8.1.a that resulted in degradation of the 'B' emergency diesel generator (EDG). An exhaust leak that developed on the EDG was attributed to inadequate torque of an exhaust bolt during planned maintenance.

Description. On June 16, the 'B' EDG developed an exhaust leak on its opposite-control side (OCS) during its monthly surveillance run. Operators investigated the leak during the run, but continued the EDG's operation until the surveillance was completed. After the EDG was secured, operators removed the local heat shield and discovered that a manifold extension was missing one bolt and its accompanying bolt was loose. The missing bolt was found in between the other exhaust extensions. Both bolts were quarantined and sent to an offsite laboratory for analysis.

The 'B' EDG had undergone an overhaul in October 2004. A review of that overhaul's work order (R2026344), showed that procedure 1301-8.2, "Diesel Generator Major Inspection (Mechanical)," Rev. 78 was used. Step 8.2.7.5 of that procedure directs reinstallation of the exhaust manifold using the bolt torque value specified in Figure 2 (55 ft-lb). Although a note on Figure 2 specifies that high temperature anti-seize compound be used on all exhaust manifold bolts, no step in the procedure directs lubricant be applied. No lubricant is listed on the bill of materials for this work order. Step 8.20 requires the torque wrench information and type of bolt lubricant to be recorded. The work order remarks contained no documentation of final torque values for the manifold extensions.

As part of the extent-of-condition review (IRs 344477, 344680, and 344684), the licensee performed torque checks on all accessible, non-lock-wired bolts on the 'B' EDG exhaust manifold and visually examined all exhaust manifold bolts on the 'A' EDG. A new gasket and two new bolts were installed on the OCS of the 'B' EDG under work order C2010706. This activity referred to a different procedure, 1405-3.2, "Diesel Engine Maintenance,"

Rev. 35, for the installation of the gasket and bolt torque of 55 ft-lbs. The inspectors again noted that the completed work order did not document the final torque values or the application of an acceptable lubricant. The inspectors also identified deficiencies in the procedure for torque and lubrication, 1410-Y-72, "Bolt/Nut Torquing and Sequences," Rev. 19 (IR 347137). The procedure is inconsistent in that some steps direct the application of lubricant while in others, including the step for the bolt of interest, no lubrication is explicitly identified.

The results of the laboratory analysis showed that the apparent cause of the gasket failure was insufficient pre-load due to lack of lubrication on the bolts. The post event laboratory report confirmed that no lubrication was present on the analyzed bolts. The licensee concluded that inadequate torque was the cause of the bolt withdrawing from the exhaust extension.

The inspectors determined that the initial engineering operability evaluation was overly qualitative and too narrowly focused. Specifically the calculation did not address the amount of exhaust gas leakage, the associated EDG load loss, and the EDG room temperature heatup that may occur during the EDG mission time run in this degraded condition. The TMI EDGs only have approximately three percent margin between design load requirements and the full load capability when no degradation is present. The calculation did not use available quantitative calculations or modeling to support the operability determination.

In response to the inspectors' concerns, AmerGen engineers performed an analysis to determine the postulated leakrate of exhaust gases that might leave the extension, the associated impact to the turbocharger and loss of EDG loading capability, and finally, the impact of exhaust on EDG room temperature limits. Engineers concluded the 'B' EDG remained operable and the 'A' EDG was unaffected. The inspectors concluded that the revised operational evaluation (OPE 05-012) properly addressed the inspector's concerns.

Analysis. Technicians did not install the EDG turbocharger exhaust manifold bolts with sufficient torque as specified in the work instructions. This is a performance deficiency. The finding is more than minor because the resulting degraded EDG turbocharger air flow adversely affected a mitigating system's (EDG) reliability and capability to respond to initiating events. Using NRC Manual Chapter 0609, Significance Determination Process, "Appendix A" Phase 1, this finding was determined to be of very low safety significance (Green) since the condition did not result in an actual failure or inoperability of the 'B' EDG.

A contributing cause of this finding is a cross-cutting issue in the area of human performance, because AmerGen craft personnel did not follow maintenance instructions for lubricating and torquing EDG exhaust manifold bolts and the maintenance instructions were deficient in that they instruction for bolt lubrication was unclear. The inspectors determined this was also a cross-cutting issue in the area of problem resolution, because

the potential for common cause (deficient workmanship) to have adversely effected the other seven manifolds was not sufficiently evaluated.

Enforcement. TS 6.8.1.a requires that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Rev. 2, February 1978. Regulatory Guide 1.33, Appendix A, recommends written procedures for maintenance that can affect the performance of safety-related equipment. Procedure 1301-8.2 requires all EDG exhaust manifold bolts be lubricated and torqued to 55 foot pounds. Contrary to this requirement, AmerGen maintenance technicians did not properly lubricate and torque exhaust extension bolts to values identified in procedures. Because this violation was of very low safety significance and was entered into the TMI corrective action program (IRs 344477, 322262, 347137, 352825), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy. **NCV 05000289/2005004-02, Deficient Maintenance Procedures and Personnel Error Degrade Safety-Related Emergency Diesel Generator.**

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

a. Inspection Scope

The inspectors selected three samples for review. 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. The inspectors reviewed the routine planned maintenance, restoration actions, and/or emergent work for the following equipment removed from service:

- On April 20, electricians replaced the #2 cell on the 'B' 125/250 volt station battery due to degraded cell voltage (Risk Document 884).

- On April 21, the technicians replaced IA-V-1617B and IA-V-1618B as corrective maintenance to address air leakage from the 'B' train of the EFW and main steam (MS) two-hour backup air supply system (Risk Document 1081).

- On April 26-27, station personnel performed a planned maintenance outage for the 'B' LPI train, which involved elevated (orange) on-line maintenance risk (Risk Document 831).

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.

- On April 18, without prior notification to station personnel, several vehicles and approximately 30 people gathered outside of the site north access gate to conduct an anti-war demonstration. Security personnel responded by increasing their presence at the north access gate, using remote surveillance to assess potential challenges to station security, contacting the Pennsylvania state police, and implementing applicable procedures from the TMI Nuclear Station Security Plan.

- On April 27, operators filled and vented the building spray (BS) and decay heat removal (DHR) systems following a planned maintenance outage on the 'B' low pressure injection header. Repairs during this outage required that the systems be drained, which resulted in an elevated Orange maintenance risk condition and a 72 hour TS limiting condition of plant operation. It was important that the fill and vent evolutions be performed in a timely and controlled manner to support prompt system restoration. The fill and vent evolution was performed using OP-TM-212-255, "Venting Portions of DHR System Following Activities for Engineered Safeguards Standby Mode," Rev. 1 and OP-TM-214-552, "Fill and Vent BS Train B," Rev. 1.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 - 6 samples)

a. Inspection Scope

The inspectors selected six samples for review. 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 justified, and no unrecognized increase in plant risk resulted from the equipment issues. The inspectors referenced 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 AmerGen's operability evaluations. Additional documents reviewed during the inspection are listed in the Attachment.

- Higher-than-expected valve actuator friction loads resulted in the actuator not meeting the manufacturer's minimum required seating torque for Reactor Building Purge Exhaust Line outboard containment isolation valve AH-V-1A during testing on March 9, 2005. Operators placed the valve in its design function position (closed) until engineering assessment, valve repair, and valve retest were complete.

- On April 11, while inspecting the river intake structure, the inspectors observed that the motor casings for nuclear service river water pump NR-P-1C and decay heat river water pump DR-P-1A were very hot (185E Fahrenheit) and questioned whether this was an indication of motor degradation. The inspectors reviewed vendor technical manuals, periodic thermography data, motor nameplate data, and discussed this issue with component monitoring engineers.

- The inspectors reviewed IR 286790, which evaluated elevated vibration readings identified on December 31, 2004, in the control building emergency ventilation return fan AH-E-19A. The inspectors interviewed the ventilation system engineer and the component maintenance optimization group supervisor to evaluate the resolution of the elevated vibration. The evaluation determined that although the vibration levels had reached the fault level, they lowered back to the alert range, the limits established were conservative, and the vibration level remained constant. The evaluation concluded that the condition did not indicate bearing defects or bearing stress issues based on evaluation of the vibration spectrum data. Engineers determined AH-E-19A fan operability was not affected and that the fan would be able to perform its safety function for its 30-day mission time.

- On May 16, engineers identified that the 'C' reactor coolant pump (RCP) motor lower oil reservoir level indication was degraded. The motor has a known oil leak, but the level indicator did not change. Therefore, the low level alarm annunciators may not function as designed to provide early indication of an oil leak, which is a precursor to motor damage and a potential fire hazard (IRs 335619 and 340553). Troubleshooting and compensatory measures were performed using these IRs.

- On June 5, an electrical ground (7 kilo ohm resistance) developed on the '1B' station battery bus (IR 341147). Operators implemented OP-TM-PPC-C4123, "Station Battery 1B Ground Resistance Low Alarm," Rev. 0 and initiated ground isolation procedures. Technicians determined the ground only effected one of three bus phases. Operators verified continued battery operability by monitoring battery voltage and discharge current. The ground was subsequently identified and isolated on June 6 (IR 341332).

- On June 24, the 'A' RCP high first stage seal leakoff alarm actuated shortly after technicians replaced the leak detection transmitter and pre-amplifier components. Indicated seal leakoff increased from 3.6 to 5 gallons per minute (gpm). Plant procedures require the RCP be tripped at 6 gpm and the reactor be tripped at

8 gpm. On June 27, the inspectors observed leakoff had increased further to 5.1 gpm and determined that operators had not yet validated the accuracy or reliability of the seal leakoff indication. The inspectors expressed concern that this degraded condition (or indication) increased the likelihood for a plant transient and questioned what operators had done to verify performance of the 'A' RCP seal (IRs 347496, 348015, and 348403). In response to the inspectors' concerns, an adverse condition monitoring plan was developed in accordance with OP-AA-108-111, "Adverse Condition Monitoring and Contingency Planning," Rev. 1 and technicians determined the new pre-amplifier was deficient. Operators determined that the RCP seal was not degraded and alternative parameter monitoring was established until the pre-amplifier was successfully replaced and tested on June 29.

b. Findings

No findings of significance were identified

1R16 Operator Work-Arounds (71111.16 - 2 samples)

a. Inspection Scope

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

Additionally, the inspectors reviewed the status of planned and ongoing efforts to reduce the number of open OWAs and challenges with the coordinator responsible for the program. The inspectors also toured the control room, and discussed items of particular concern with the responsible system engineers to ensure the items were being addressed on a schedule consistent with their relative safety significance. The inspectors specifically evaluated the following degraded conditions to determine whether it should be added to the station list of OWAs. The inspectors observed that quarterly OWA collective assessments, as well as additional attributes of the OWA program, were not being implemented in accordance with OP-AA-102-103. IR 346929 was initiated to address this observation. The following degraded material conditions were reviewed as potential OWAs:

- Due to an oil drainage issue, the control building chillers would not start without oil addition by electricians immediately prior to starting the chiller. This action was necessary to compensate for an oil drainage deficiency with was either due to poor design or bearing wear. Equipment was prestaged and shift staffing was revised to ensure an electrician was present, inside the protected area, 24 hours

per day. The inspectors questioned whether this degraded condition should be categorized as an OWA list and treated with associated station priority (IR 321027). In response to the inspector's concern, the issue was categorized as an OWA, pending completion of a plant modification to correct the deficiency.

- The safety-related two-hour backup instrument air system was degraded due to two leaking valves (IA-V-1617B and IA-V-1618B), which caused the 'B' train air bank to gradually depressurize. Several compensatory measures were established including issuance of a night order which emphasized a modified valve closure configuration, increased frequency for checking pressure and recharging air banks, and a work order to repair/replace the valves. The material deficiency was corrected on April 21, 2005.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications (71111.17 - 2 samples)

a. Inspection Scope

The inspectors performed two inspection samples. The modifications were reviewed to determine whether they were designed and/or implemented as required by CC-AA-102, "Design Input and Configuration Change Impact Screening," Rev. 9 and CC-AA-103, "Configuration Change Control," Rev. 8. The inspectors verified the modification supported plant operation as described in the UFSAR and complied with associated TS requirements. The inspectors reviewed the function of changed components, the change description and scope, and the associated 10 CFR 50.59 screening evaluations. Additional documents reviewed during this inspection are listed in the Attachment. The following permanent plant modifications were reviewed:

- Engineering Change Request 04-00208, "Incore Neutron Flux Monitoring Recorder Digital Upgrade: IM-NR-1 and IM-NR-2," Rev. 2.

- Engineering Change Request 04-00158 "Upgrade Radiation Monitors RM-G-26, 27," Rev. 2

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19 - 8 samples)

a. Inspection Scope

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

- On April 20, technicians replaced 'B' 125/250 volt battery cell #2 due to degraded individual cell voltage. The replacement was performed using procedure 1420-DC-3, " Station Battery Cell Replacement and Charging," Rev. 19. Post maintenance testing was performed in accordance with work order A2103534 and procedure 1301-5.8.2, "Station Battery 1B Quarterly," Rev. 3.

- On April 21, the technicians replaced IA-V-1617B and IA-V-1618B as corrective maintenance to address air leakage from the 'B' train of the EFW and MS two-hour backup air supply system. Post maintenance testing was performed in accordance with work order C2010255.

- On April 27, technicians performed testing of containment purge valve AH-V-1A following actuator overhaul. Testing was performed as specified in work order C2010016, "AH-V-1A Actuator Overhaul"; MA-AA-743-310, "Diagnostic Testing and Evaluation of Air Operated Valves," Rev. 3; and OP-TM-823-201, "Stroke Time Testing of AH-V-1A," Rev. 0.

- On April 27, PMT was performed following approximately 30 work activities completed during the 'B' LHI outage. Work activities included repair of numerous boric acid leaks; replacement and/or test of relief valves DH-V-57B and DH-V-13B; and lubricate, oil change, and repack pump DH-P-1B. PMT was performed in accordance with work orders C2009848, C2009853, C2009817, R1802469, C2010068, and C2009765.

- During the week of May 30, the Group 6 pressurizer heater breaker was replaced. During the review of PMT performance, the licensee identified a missing operator rod in the Group 1 breaker. The inspectors monitored the extent-of-condition review and interviewed the system manager on the potential impact of the condition.

- On June 9, the 'A' EDG (EG-Y-1A) was sampled for its monthly lube oil analysis and fan drive component lubrication/inspection. The PMT was completed satisfactorily per surveillance test procedure 1303-4.16, Emergency Power System, Rev.106.

- On June 14, the 'B' EFW pump (EF-P-1B) was taken out of service for a planned maintenance outage. The PMT was completed satisfactorily in accordance with procedures OP-TM-424-202 Rev.2 and OP-TM-424-252 Rev.0.

- On June 16, during a scheduled monthly surveillance test run of the 'B' EDG (EG-Y-1B), operators identified an exhaust leak on the opposite-control-side (OCS) of the engine. Under work order C2010706, the bolts and associated gasket were replaced. The inspectors reviewed data from the test run that followed the repairs and torque checks. Deficiencies noted are discussed in section 1R12.

b. Findings

Deficient Work Instructions and Procedure Error during 125/250 Volt Battery Cell Replacement

Introduction. The inspectors identified a Green NCV of TS 6.8.1.a in that maintenance procedures did not contain sufficient work instruction or acceptance criteria to ensure the safety related 'B' 125/250 volt battery was properly reassembled following replacement of cell #2. Additionally, workers did not properly follow the procedure instructions in that certain steps were performed out of order.

Description. Since October 2004, periodic surveillance tests revealed that 'B' 125/250 volt station battery cell #2 was degrading. On April 20, 2005 electricians replaced cell #2.

The inspectors observed the battery cell replacement and PMT. Although technicians performed the work carefully and with procedures in hand, the inspectors noted several deficiencies. Early during the cell replacement (procedure 1420-DC-3), the inspectors observed that technicians performed steps out of order. Technicians installed jumpers J1 and J2 (steps 8.6.12 and 8.6.13) which connected a 3-cell bypass battery bank to the 'B' battery bank prior to establishing the required bypass bank voltage (steps 8.6.9 to 8.6.11). The inspectors questioned why steps were performed out of order. Technicians acknowledged the error, removed the jumpers, put equipment in a safe condition, contacted supervision, and got resolution prior to continuing the maintenance. Additionally, the inspectors noted that the procedure usage level was incorrectly specified as "category 2" (reference use). The procedure should have been designated as "category M" for multi-level use, due to the designation of section 8.6 as a "category 1" (continuous use) activity. The inspectors also determined that the pre-evolution walkthrough and briefing were deficient in that the maintenance crew did not fully understand the procedure instructions. The procedure had been revised since the last time this procedure was used to replace a battery cell. The inspectors determined that momentary installation of the jumper out of sequence did not adversely affect safety.

The inspectors determined that the instructions for battery cell replacement and PMT (work order A2103534 and procedures 1420-DC-3 and 1301-5.8.2) were deficient in that they did not provide instruction to perform intercell battery resistance checks or torque the battery rack connection bolts prior to declaring the battery operable. The inspectors

expressed concern that without performing these actions, excessive high resistance at a terminal connection point or a non seismically qualified rack configuration could exist and be undetected. Institute of Electrical and Electronics Engineers (IEEE) Standard 484-2002, “Installation of Vented Lead-Acid Batteries for Stationary Applications and IEEE 450-2002, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications” recommend intercell connection resistance be verified when battery connections are cleaned and reassembled, including for cell replacement. After evaluating the inspectors’ concerns, station personnel promptly performed these additional work activities and revised procedures to require them for future battery cell replacements. The as-found intercell resistance values were sufficient for operability. The as-left battery rack bolt torque values were sufficient for seismic integrity.

Analysis. The inspectors determined that performing procedure steps out of order was a performance deficiency. In addition, procedure instructions were deficient in that they did not include instruction to perform intercell battery resistance checks or torque the battery rack connection bolts prior to declaring the battery operable.

This issue affected the mitigating systems cornerstone and was more than minor because it affected the reliability of the ‘B’ train of the 125/250 volt direct current power system to perform its accident mitigation functions in response to initiating events. The deficiency affected the procedure quality and equipment performance attributes of the mitigating system cornerstone. The inspectors performed a Phase 1 SDP evaluation in accordance with Inspection Manual Chapter 0609, Appendix A. The finding screened to Green (very low safety significance) because the ‘B’ 125/250 volt battery bank was not inoperable for greater than the TS allowed outage time.

This finding is also a cross-cutting issue in the area of human performance, because operators did not follow procedure 1420-DC-3 steps in the order specified and procedure quality was deficient because it did not provide instruction to perform intercell battery resistance checks or torque the battery rack connection bolts to verify seismic qualification prior to declaring the battery operable. Additionally, procedure usage level was incorrectly specified as “category 2" (reference use). The procedure should have been designated as “category M” for multi-level use, due to the designation of section 8.6 as a “category 1" (continuous use) activity.

Enforcement. TS 6.8.1.a requires, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Rev. 2, February 1978. Regulatory Guide 1.33, Appendix A, recommends procedures for performing maintenance that can affect the performance of safety-related equipment. IEEE Standards 484-2002 and 450-2002 recommend intercell connection resistance be verified when battery connections are cleaned and reassembled, including for cell replacement. Vendor technical manual 0021, “Stationary Battery Installation and Operating Instructions” specifies rack braces and rails be torqued to 30 foot pounds and 20 foot pounds, respectively. Procedure 1420-DC-3 and work order A2103534 specify work instructions for replacement and PMT

of a 125/250 volt battery cell. Contrary to the above, on April 20, 2005, technicians did not implement procedure 1420-DC-3 as written when they performed steps out of order. Technicians installed jumpers J1 and J2 (steps 8.6.12 and 8.6.13) which connected a 3-cell bypass battery bank to the 'B' battery bank prior to establishing the required bypass bank voltage (steps 8.6.9 to 8.6.11). Additionally, work order A2103534 and procedure 1420-DC-3 were not properly maintained in that they did not provide instruction to perform intercell battery resistance checks or torque the battery rack connection bolts prior to declaring the battery operable. Because this violation was of very low safety significance and was entered into the TMI corrective action program (IRs 326794 and 355900), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: **NCV 05000289/2005004-03, Deficient Procedure and Personnel Error While Replacing 'B' 125/250 Volt Battery Cell**

Mispositioned Valve IA-V-1769 and Degraded 2-Hour Backup Air Supply to EFW and MS

Introduction. The inspectors identified a Green NCV of TS 6.8.1.a in that operators mispositioned valve IA-V-1769, while attempting to recharge the two-hour backup air supply banks. These air banks provide an air supply to several valves which provide decay heat removal. The mispositioned valve caused the air banks to partially depressurize and reduce the reliability of the supported mitigating systems (EFW and MS). Operators identified and repressurized the air banks, but did not recognize and correct the cause of the degraded condition until the inspectors identified the causes.

Description. On March 20, AmerGen operators and engineers identified that the 'B' train of the two-hour backup instrument air (IA) system was depressurizing at 17 pounds per square inch gauge (psig) per hour due to leakage from IA-V-1617B and IA-V-1618B (IRs 315038 and 315043). The two-hour backup IA system supplies air to operate valves for (1) the steam supply to the turbine driven EFW pump, (2) the EFW header injection, and (3) steam generator atmospheric relief valves. On March 29, the inspectors and operators observed that the 'B' train of the two-hour backup air supply indicated 1460 psig, which was below the minimum of 1700 psig specified in operator logs. Operators recharged both air banks to 2200 psig and increased the frequency of monitoring bank air pressure to 4-hour intervals instead of 12 hour intervals (IR 318363).

The inspectors discussed the event with the shift manager (SM) and system engineer to evaluate operability and problem resolution. The inspectors reviewed calculation C-1101-852-5360-004, "2 Hour Back-up Air System Capacity," Rev. 1 and verified the 'B' air bank pressure had remained above the value required for operability. The inspectors also identified several deficiencies which were not identified or corrected during the licensee's initial assessment of the event:

- The SM was not aware the air banks had been recharged the previous evening, which was before the low pressure condition was identified. Both the 'A' and 'B' air banks began depressurizing following the air charge.

- The operations activity to recharge air banks was not routinely logged or communicated during shift turnovers.
- The SM initially declared the system operable without appropriate supporting basis from station calculations.
- Late on March 28 or early on March 29, operators left valve IA-V-1769 closed at the end of recharging the two-hour air banks instead of open as required by procedure 1104-25, “Instrument and Control Air System,” Rev. 127 (IRs 319499 and 330965). The open position is required to ensure proper seating of bank isolation valves and train separation. The procedurally required position (open) was a corrective action to address similar air bank leakage identified in August 2004 (IR 241706).
- The usage level for procedure 1104-25 for charging the air banks was inappropriate. Usage level was specified as “category 3" which is for informational use only. The appropriate level should have been “category 2" (reference use) or “category 1" (continuous use) to ensure the procedure was properly followed step-by-step for this important safety-related activity, considering that the system was already degraded (leakage from IA-V-1617B and IA-V-1618B).

The inspectors informed station personnel of these additional causal factors. A prompt investigation report was immediately initiated, which confirmed the inspectors’ findings.

Analysis. The inspectors determined that mispositioning IA-V-1769 was a performance deficiency. In addition, the work instruction was deficient in that the specified procedure usage level was not sufficient to provide reasonable assurance that the air charge would be performed successfully.

This issue affected the mitigating systems cornerstone and was more than minor because the degraded two-hour air system pressure affected the reliability of the EFW and MS systems to perform their accident mitigation functions in response to initiating events. The deficiency affected the configuration control, human performance, equipment performance, and procedure quality attributes of the mitigating system cornerstone. The inspectors performed a Phase 1 SDP evaluation in accordance with Inspection Manual Chapter 0609, Appendix A. The finding screened to Green (very low safety significance) because bank air pressure did not drop below the value required for operability and, therefore, the system remained capable of performing its safety function.

This finding is also a cross-cutting issue in the area of human performance, because operators did not follow procedure 1104-25 instruction to open IA-V-1769 and procedure quality was deficient in that usage category 3 was inappropriate for this activity. The finding is also cross-cutting in the area of problem resolution in that (1) station personnel did not determine the actual causes of the degraded air bank pressure during their initial event investigation independent of the inspectors, and (2) corrective action in response to a similar 2004 event was ineffective, because the deficient procedure use category was not identified.

Enforcement. TS 6.8.1.a requires, in part, that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Rev. 2, February 1978. Regulatory Guide 1.33, Appendix A, recommends procedures for operation of safety-related equipment including IA, MS, and EFW systems. Step 3.10.9 of Procedure 1104-25 requires valve IA-V-1769 to be opened to ensure proper seating of two-hour backup IA system bank isolation valves and train separation. Procedure HU-AA-104-101, "Procedure Use and Adherence," Rev. 0, requires station personnel to follow procedures exactly as written. Additionally, HU-AA-104-101 states that procedure use category will be designated based upon the probability of making an error and the impact of an error. Contrary to the above requirements, on March 29, 2005, station personnel did not perform procedure 1104-25 requirements as written when recharging the two-hour emergency air system and consequently mispositioned IA-V-1769. This degraded the reliability of the EFW and MS systems. Additionally, the procedure usage category was incorrectly specified as "category 3" which is for informational use only. Because this violation was of very low safety significance and was entered into the TMI corrective action program (IRs 315038, 315043, 318363, 319499, 320108, and 330965), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: **NCV 05000289/2005004-04, Deficient Procedure and Operator Error Degrade Two-Hour Emergency Air Supply to Emergency Feedwater and Main Steam Systems.**

1R22 Surveillance Testing (71111.22 - 6 samples)

a. Inspection Scope

The inspectors performed six inspection samples. The inspectors observed and reviewed the following operational surveillance tests, and compared test data with established acceptance criteria to verify the systems demonstrated the capability to perform the intended safety function(s). The inspectors also verified that the systems and components maintained operational readiness, met applicable TS requirements, and were capable of performing the design basis functions.

- OP-TM-212-202, "Inservice Testing of DH-P-1B and Valves from Engineered Safeguards Standby Mode," Rev. 5, performed on April 27, 2005.

- On June 1, operators observed a decrease in makeup tank level and received an alarm for RM-A-2 which indicated elevated reactor building airborne particulate radiation levels (IRs 340130 and 340272). These were indications of a potential increase in RCS leakage. The inspectors reviewed performance of OP-TM-220-251, "RCS Leak Rate Determination Using the Plant Process Computer," Rev. 1. Unidentified RCS leakage remained below the TS limits and returned to typical values of 0.02 gpm on June 3.

- On June 3, the inspectors observed the quarterly emergency loading sequence and HPI logic channel/component test performed per procedure 1303-5.2, "Emergency

Loading Sequence and HPI Logic Channel/Component Test," Rev. 83 on the 'B' channel. The inspectors reviewed the completed surveillance and compared the results with historical data.

- On June 6, the inspectors observed the quarterly sampling of the 30,000 gallon diesel fuel tank and reviewed the chemistry analysis results. The inspectors also reviewed the site's commitments to regulatory guides and industry standards for diesel fuel oil systems.

- On June 9, the inspectors observed the quarterly HSPS-OTSG level and pressure channel III tests performed per procedure 1303-11.37C, "HSPS-OTSG Level and Pressure Channel III Tests," Rev. 22. The inspectors also reviewed the completed surveillance and compared the results with historical data.

- On June 16, during a scheduled monthly surveillance test run of the 'B' EDG (EG-Y-1B), operators identified an exhaust leak on the opposite-control-side of the engine. The operators continued the surveillance and completed the 1-hour run. The inspectors reviewed the completed surveillance 1303-4.16, "Emergency Power System," Rev. 106. Deficiencies noted are discussed in section 1R12 above.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23 - 2 samples)

a. Inspection Scope

The inspectors selected two samples for review. The inspectors reviewed the following temporary modifications (TM) and associated implementing documents to verify the plant design basis and the system or component operability were 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 04-00359, "Pressure Gages, Ball Valve & Adjustable Bypass for Control Building Chiller AH-C-4A," Rev. 1. This TM was installed as a temporary solution to the oil drainage issue described in section 1R16. This TM reduced the amount of time needed to start the chiller and supported further data collection to identify a permanent solution to the oil drainage issue.

- TM 05-00310, "Leak Repair and New Pipe Support for EX-V-36A," Rev. 0. This TM was installed to perform a sealant injection repair of steam leak on extraction steam valve EX-V-36A. The modification included installation of additional structural support for the sealant injection enclosure box.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness [EP]

1EP4 Emergency Action Level and Emergency Plan (E-Plan) Changes (IP 71114.04 - 1 sample)

a. Inspection Scope

The inspectors performed one inspection sample. During the period of April 1 - June 23, 2005, the NRC has received and acknowledge the changes made to TMI's E-Plan in accordance with 10 CFR 50.54(q), which AmerGen had determined resulted in no decrease in effectiveness to the E-Plan and continue to meet the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR 50. The inspectors conducted a sampling review of the E-Plan changes which could potentially result in a decrease in effectiveness. This review does not constitute an approval of the changes and, as such, the changes are subject to future NRC inspection. The inspection was conducted in accordance with NRC Inspection Procedure 71114, Attachment 4, and the applicable requirements in 10 CFR 50.54(q) were used as reference criteria.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS1 Access Control To Radiologically Significant Areas (71121.01 - 1 sample)

a. Inspection Scope

The inspectors performed one inspection sample. The inspectors toured areas controlled as High Radiation Areas (HRAs) and reviewed the effectiveness of access controls to these areas. The inspectors physically inspected and challenged three locked HRA access points to determine if access controls were sufficient to preclude unauthorized entry.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety (PS)

2PS2 Radioactive Material Processing and Transportation (71122.02 - 5 samples)

1. Inspection Planning/In-Office Inspection

a. Inspection Scope

The inspectors performed one inspection sample. The inspectors reviewed the solid waste system description in the UFSAR and recent radiological effluent release reports for information on the types and amounts of radioactive waste.

b. Findings

No findings of significance were identified.

2. System Walkdown

a. Inspection Scope

The inspectors performed one inspection sample. The inspectors walked down accessible portions of the station's radioactive liquid and solid waste collection, processing, and storage systems and locations to determine if systems and facilities were consistent with descriptions provided in the UFSAR, to evaluate their general material conditions, and to identify changes made to systems. Areas visually inspected included the Auxiliary Building, waste storage areas, and the material handling facility. Inspection records and previous surveys were also reviewed. The inspectors reviewed the following matters:

- the status of any non-operational or abandoned radioactive waste process equipment and the adequacy of administrative and physical controls for those systems;
- changes made to radioactive waste processing systems and potential radiological impact, including conduct of safety evaluations of the changes;
- current processes for transferring radioactive waste resin and sludge to shipping containers and mixing and sampling of the waste;
- radioactive waste and material storage and handling practices;
- sources of radioactive waste at the station, processing and handling of the waste; and
- the general condition of facilities and equipment.

The review was against criteria contained in the station’s UFSAR, 10 CFR Part 20, 10 CFR 61, the Process Control Program (PCP), and applicable station procedures.

b. Findings

No findings of significance were identified.

3. Waste Characterization and Classification

a. Inspection Scope

The inspectors performed one inspection sample. The inspectors reviewed the following matters:

- radio-chemical sample analysis results for radioactive waste streams;
- the development of scaling factors for hard-to-detect and measure radionuclides including radionuclide concentration determination for irradiated hardware;
- methods and practices to detect changes in waste streams;
- classification and characterization of waste relative to 10 CFR 61.55 and 10 CFR 61.56;
- implementation of applicable NRC Branch Technical Positions on waste classification, concentration averaging, waste stream determination, and sampling frequency;
- current waste streams and their processing relative to descriptions contained in the UFSAR and the station's approved PCP;
- current processes for transferring radioactive waste resin and sludge discharges into shipping/disposal containers to determine adequacy of sampling; and
- revisions of the PCP and the UFSAR to reflect changes.

The review was against criteria contained in 10 CFR 20, 10 CFR 61, 10 CFR 71, the UFSAR, the PCP, applicable NRC Branch Technical Positions, and licensee procedures.

b. Findings

Introduction. The NRC identified a Green NCV associated with failure to conduct a radiological evaluation to demonstrate that samples of radioactive spent resin, collected from the spent resin tank, were representative of the bulk waste, for purposes of quantification of total radionuclide activity in shipments of the spent resin transferred to a waste processor, in accordance with 10 CFR 20, Appendix G.

Description. 10 CFR 20.1501 requires, in part, that the licensee make or cause to be made, reasonable surveys that may be necessary to comply with the regulations in Part 20 to evaluate concentrations and quantities of radioactive materials. 10 CFR 20 defines a survey as an evaluation of the radiological conditions and potential hazards incident to, among other matters, the transfer and disposal of licensed radioactive materials. When appropriate, such an evaluation is to include measurements or calculations of concentrations or quantities of radioactive material present.

On April 13, 2005, the inspectors identified that AmerGen had listed primary spent resin as a waste stream for purposes of waste classification. The spent resin is collected in a spent resin storage tank and consists of make-up and cation demineralizer resin. When the spent resin tank is full, AmerGen collects a sample of the spent resin, then transfers this waste to an approved container for shipping purposes. The resin waste sample collected is analyzed and the results used to determine total radionuclide activity of the

shipment for purposes of transfer to an authorized waste processor in accordance with 10 CFR 20, Appendix G. The inspectors determined that prior to December 1998, AmerGen required that the spent resin tank be recirculated the equivalent of three tank volumes, before collection of a sample, to provide for a representative sample to be analyzed. In December 1998, AmerGen modified the procedure to require recirculation of the tank contents for 15 minutes (reference procedure 1104-29N, "Spent Resin and Used Pre-coat Tank Operations," Rev. 35), before collection of a sample. The inspectors evaluated the pump capacity and the tank volume and concluded the tank recirculation time was not sufficient to provide for a fully recirculated tank, and thus it was not apparent that the limited recirculation time would ensure a representative sample for purposes of quantification of total radionuclide activity. Further, the sample results are an input to the dose-to-curie method of activity determination. The inspectors subsequently determined that AmerGen could not produce an evaluation to demonstrate that the 15 minute recirculation time was adequate to fully recirculate the bulk waste within the tank to provide for a representative sample. AmerGen was unable to identify a technical basis for the run time or its adequacy.

Analysis. Amergen did not conduct a radiological evaluation, in accordance with 10 CFR 20.1501, to verify the adequacy of its revised spent resin sampling procedure to ensure samples of spent resin waste collected for analysis could adequately support waste transfer in accordance with 10 CFR 20, Appendix G. This is a performance deficiency in that a requirement was not met which was reasonably within AmerGen's ability to detect and correct.

The finding was greater than minor in that it is associated with the Public Radiation Safety cornerstone and did affect the program and process attribute of the cornerstone in that the issue involved an occurrence in the radioactive material transportation program that was contrary to NRC or Department of Transportation regulations. Specifically, AmerGen did not ensure that samples of spent resin were representative for purposes of determination of total radionuclide activity when transferring licensed radioactive materials to a licensed waste processor. Using the Public Radiation Safety SDP flow chart, this finding is of very low safety significance (Green) in that it did involve a radioactive material control issue, it did involve transportation, no radiation limit was exceeded, it did not involve a breach of packaging, it did not involve a Certificate of Compliance finding, it did not involve a low-level burial ground issue, and it did not involve a failure to make an emergency notification issue. AmerGen reviewed shipments, since the procedure change, and concluded that, due to the generally low radioactivity of the shipments made, there was no likelihood that a shipment was improperly packaged for shipment. In addition, and also due to the generally low relative radioactivity content of the shipments, the waste would not have been misclassified from a 10 CFR 61 perspective, for purposes of ultimate disposal. Consequently no actual safety consequence was identified.

Additionally, the failure to conduct a radiological evaluation, for purposes of waste transfer and disposal, is a cross-cutting issue in the area of problem identification and resolution. Specifically, AmerGen's focused audits and self-assessments of this area did

not provide for effective identification and corrective action for this issue. (Reference: 10 CFR Part 61 Assessment, August 2004; Check-in Self-Assessment, October 2004; Issue Report (IR) - 26262490, October 2004). AmerGen placed this issue into its corrective action program (IR - 325106).

Enforcement. 10 CFR 20.1501 requires, in part, that the licensee make or cause to be made, reasonable surveys that may be necessary to comply with the regulations in this part to evaluate concentrations and quantities of radioactive materials. 10 CFR 20 defines a survey as an evaluation of the radiological conditions and potential hazards incident to, among other matters, the transfer and disposal of radioactive materials. 10 CFR 20, Appendix G requires that the shipper of radioactive waste for ultimate disposal provide the total radionuclide activity in the shipment. Contrary to this requirement, procedure 1104-29N instructions regarding spent resin tank pre-survey recirculation time were not evaluated to assure a representative sample was obtained. Consequently, AmerGen could not ensure the waste processor with an accurate total shipment radionuclide activity value.

AmerGen documented this issue in its corrective action program (IR 325106), and suspended further shipments of this type pending further evaluation. Since this violation is of very low safety significance (Green) and Amergen entered the finding into its corrective action program, this violation is being treated as a NCV consistent with Section VI.A of the NRC Enforcement Policy. **NCV 05000289/2005004-05, Failure to Conduct Radiological Evaluation to Support Waste Transfer.**

4. Shipment Preparation

a. Inspection Scope

The inspectors performed one inspection sample. The inspectors selectively reviewed the training and qualification program for personnel handling, packaging, and shipping radioactive materials. The review was against criteria contained in NRC Bulletin 79-19 and 49 CFR 172 Subpart H.

b. Findings

No findings of significance were identified.

5. Shipment Records and Documentation

a. Inspection Scope

The inspectors performed one inspection sample. The inspectors selected and reviewed the records associated with five non-excepted shipments of radioactive material made since the previous inspection in this area (Shipment Nos. RS-05-029-1, RS-04-16-1, RS-04-031-I, RS-04-113, and RS-04-055). The following aspects of the radioactive waste,

radioactive material packaging, and radioactive material shipping activities were reviewed.

- implementation of applicable shipping requirements including completion of waste manifests;
- implementation of the specifications in applicable Certificates of Compliance for the approved shipping casks, including limits on package contents;
- classification and characterization of waste relative to 10 CFR 61.55 and 61.56
- implementation of recent NRC and DOT shipping requirements rule changes;
- implementation of 10 CFR 20 Appendix G;
- implementation of specific radioactive material shipping requirements;
- packaging of shipments;
- labeling of shipping containers;
- placarding of transport vehicles;
- conduct of vehicle checks;
- provision of driver emergency instructions;
- completion of shipping paper/disposal manifest;
- evaluation of package against package performance standards, as appropriate;
- conformance with procedures for cask loading, closure and use requirements including consistency with cask vendor approved procedures;
- use of latest revision documents.

The review was against criteria contained in 10 CFR 20; 10 CFR 61; 10 CFR71; applicable Department of Transportation requirements, as contained in 49 CFR 170-189; station procedures; disposal facility licenses; and Certificates of Compliance or vendor procedures for various shipping casks.

The inspectors also reviewed the year 2003 and 2004 (draft) TMI Annual Radioactive Effluent Release Report, relative to types and quantities of radioactive waste shipped offsite and relative to changes to the PCP.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems (71152)

1. Annual Sample-Reactor Building Emergency Cooling Fan AH-E-1A Failure (1 sample)

a. Inspection Scope

The inspectors performed 1 inspection sample. The inspectors reviewed IR 286662, which evaluated the December 30, 2004, catastrophic failure of the 'A' reactor building emergency cooling fan AH-E-1A. This event was selected based on its potential for

impacting the reactor building Barrier Integrity cornerstone. The inspectors also interviewed the system engineer and the corrective maintenance organization specialists, reviewed the vendor manual, and applicable UFSAR and TS documents.

The reactor building emergency cooling system consists of three independent cooling fans (AH-E-1A, B, and C). Failure of one fan does not render any of the two reactor building cooling system trains inoperable. The three fans are normally run continuously, to maintain the reactor building temperature below the TS limits. The system also performs a safety function to maintain the reactor building integrity by removing heat and reduce pressure following a design basis loss of cooling accident (LOCA). Laboratory analysis and inspections determined that the outboard motor bearing (opposite Drive End) had failed due to the ball retaining cage failing. The cause of failure was indeterminate. The inspectors verified that the failed motor was replaced and that an adequate extent-of-condition review was performed, including the conservative removal and refurbishment of AH-E-1C in March 2005 due to slightly elevated vibration readings. In addition, the inspectors verified that actions are planned to relocate the permanent vibration probes to better perform vibration monitoring of these fans.

b. Findings

No findings of significance were identified

2. 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 issues reports, by reviewing hard copies of selected issue reports, attending daily screening meetings, and accessing the licensee's computerized database.

Section 1R12 describes a finding for failure to implement procedures with regard to the torquing of bolts on the emergency diesel generator exhaust manifold. This finding is a cross-cutting issue in the area of problem resolution, because AmerGen's initial engineering evaluation was too narrowly focused in not using technical calculations or modeling to support conclusions regarding the quantity of leaking exhaust and its associated impact on diesel loading capability and room design temperature.

Section 1R19 describes a finding in which operators mispositioned IA-V-1769, which partially depressurized the 2-hour emergency air supply to the EFW and MS valves. This degraded the reliability of these mitigating systems to perform their safety related decay heat removal function. This finding is a cross-cutting issue in the area of problem resolution, because AmerGen's initial assessment of the event did not determine or correct the actual causes of the degraded air bank pressure independent of the inspectors.

Section 2PS2 describes a finding for failure to evaluate the adequacy of a change to the procedure for collecting samples of radioactive spent resin for analysis to support transfer of radioactive material to a waste processor for ultimate disposal. This finding is a cross-cutting issue in the area of problem identification in that AmerGen did not identify this problem during routine self-assessments and audits of its radioactive waste transportation and disposal program.

3. Radioactive Material Processing and Transportation

a. Inspection Scope

The inspectors reviewed audits and assessments of the radioactive waste handling, processing, storage, and shipping programs including the PCP. The inspectors also reviewed selected corrective action documents written since the previous inspection. Documents reviewed during the inspection are listed in the Attachment.

The review was against criteria contained in 10 CFR 20 Appendix G, 10 CFR 71.101, and applicable station audit and surveillance procedures.

b. Findings

One NRC identified finding, in the cross-cutting area of problem identification and resolution, is discussed in Section 2PS2.

4. Semi-Annual Review of 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.

The inspectors noted a trend of procedure quality and usage deficiencies. Examples included (1) deficient procedure and violations of station scaffold procedures (IR 334412); (2) violation of fire barrier procedures (IR 334407); (3) violation of EFW/MS 2-hour backup IA procedures (IR 319499); (4) deficient work instruction and noncompliance with 'B' EDG corrective maintenance work instructions; (5) deficient procedure and violation of station 125/250 volt DC battery cell replacement procedure (IRs 326794 and 355900); (6) deficient procedure for periodic surveillance inspection of the air intake tunnel halon fire suppression system (IR 348405); (7) noncompliance with station procedures for identification and evaluation of operator work-arounds (IR 346929); (8) deficient procedure for testing decay heat system valves with an

opportunity to precondition a valve prior to performing the stroke time test; and (9) deficient procedure for battery testing with acceptance criteria for individual cell voltage less conservative than industry standards and standardized TS. Several of these issues were documented as NRC findings.

Additionally, the inspectors noted several deficiencies in the area of problem resolution. AmerGen evaluation of several degraded equipment conditions was either not timely or was too narrowly focused. Examples included the 'B' EDG exhaust leak (IR 344477), degraded expansion joints (IR 320086), degraded air intake tunnel halon fire suppression system (IR 348405), and degraded emergency backup instrument air pressure (IR 318363). Several IRs were closed out as complete without the corrective actions being completed (i.e. IR 294515, 'A' and 'B' makeup pump cubicle fire doors found open). In each case new IRs were written to complete the corrective action at a later date. This resulted in corrective actions being repeatedly postponed, despite the issue appearing to be resolved. The inspectors also observed corrective action implementation being delayed without appropriate management approval (i.e. IR 260697, south access bridge security barrier configuration modification postponed). The inspectors discussed these trends with station management (IRs 325952 and 352825).

b. Findings

No findings of significance were identified.

4OA4 Cross-Cutting Aspects of Findings

Section 1R12 describes a finding in which the inspectors identified several installed safety related and important-to-safety expansion joints beyond their service life, with no basis for operability documented. This finding is a cross-cutting issue in the area of human performance, because maintenance and testing procedures were insufficient to provide reasonable assurance that safety related and important-to-safety expansion joints would continue to remain capable to perform their design functions.

Section 1R12 describes a finding in which the 'B' emergency diesel generator developed an exhaust leak due to a backed-out bolt. This finding is a cross-cutting issue in the area of human performance, because craft personnel did not follow work instructions to torque the turbocharger flange bolts, document final torque values, or document lubrication used in the completed work orders.

Section 1R19 describes a finding in which operators mispositioned IA-V-1769, which partially depressurized the 2-hour emergency air supply to the EFW and MS valves. This degraded the reliability of these mitigating systems to perform their safety related decay heat removal function. This finding is a cross-cutting issue in the area of human performance, because operators did not follow procedure 1104-25 instruction to open IA-V-1769 and procedure quality was deficient in that usage "category 3" (informational use only) was insufficient to ensure the procedure was properly followed step-by-step for this important safety-related activity.

Section 1R19 describes a finding in which maintenance procedures did not contain sufficient work instruction or acceptance criteria to ensure the safety related 'B' 125/250 volt battery was properly reassembled following replacement of cell #2. This finding is a cross-cutting issue in the area of human performance, because technicians did not follow procedure 1420-DC-3 work instructions in that certain steps were performed out of order and procedure quality for this safety related maintenance was deficient. Additionally, the procedure usage level was incorrectly specified as "category 2" (reference use).

Section 4OA5.2 refers to a finding documented in NRC Inspection Report 05000289/2005006, regarding expired emergency response organization (ERO) qualifications due to untimely training. This finding is a cross-cutting issue in the area of human performance, because the TMI emergency preparedness department staff did not follow applicable requirements, specified in the TMI Annex Emergency Plan, when scheduling ERO training. Additionally, AmerGen corporate emergency preparedness supervision did not ensure the required ERO training periodicity was properly understood and implemented.

4OA5 Other

1. TI 2515/163, Operational Readiness of Offsite Power

Cornerstones: Initiating Events, Mitigating Systems

a. Inspection Scope

The inspectors performed 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. This information was forwarded to the US NRC Office of Nuclear Reactor Regulation for further review.

b. Findings

No findings of significance were identified.

2. Additional NRC Assessment of NRC Finding 05000289/2005006-01, Emergency Response Organization Qualifications Expired Due to Untimely Training

NRC Inspection Report 05000289/2005006 dated June 30, 2005, documented a preliminary White finding associated with "Emergency Response Organization (ERO) Qualifications Expired Due to Untimely Training." During this inspection period the inspectors discussed the cause of the untimely training with both licensee staff and the NRC emergency preparedness specialists who identified the finding. The inspectors

determined that a contributing cause of the finding is in the area of human performance, because the TMI emergency preparedness department staff did not demonstrate sufficient knowledge of the TMI Annex Emergency Plan and did not apply sufficient level of detail when scheduling ERO training to ensure the training would be completed within the required periodicity. A secondary contributing cause within the cross-cutting area of human performance was that AmerGen emergency preparedness supervision was deficient. Specifically, AmerGen corporate emergency preparedness supervision did not ensure the required ERO training periodicity was properly understood and implemented.

3. Administrative Correction

One unresolved item (URI) was inadvertently assigned two tracking numbers in NRC Inspection Report 2005002. URI 2005002-02 was a duplicate item and is closed.

4OA6 Management Meetings

Exit Meeting Summary

On July 20, 2005, the resident inspectors presented the inspection results to Mr. Rusty West and other members of his 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.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel
S. Baker, Radiation Protection Manager
C. Arnone, Plant Operations Director
K. Coughlin, Control Room Shift Manager
D. DeBoer, Operations Support Manager
G. Chick, Plant Manager
E. Eilola, Director, Site Engineering
E. Fuhrer, Regulatory Assurance
J. Heischman, Director, Maintenance
T. Knisely, Security Manager
J. Marsden, Maintenance Rule Program Manager
A. Miller, Regulatory Assurance
D. Mohre, Nuclear Oversight Manager
C. Smith, Regulatory Assurance Manager
R. West, Vice President, TMI Unit 1
S. Wilkerson, Engineering Response Team Manager

Others
M. Murphy, PA Department of Environmental Protection

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

2005004-01	NCV	Deficient Maintenance Procedures Result in Undetected Expansion Joint Degradation and Safety-Related Expansion Joints Exceeding Service Life (Section 1R12)
2005004-02	NCV	Deficient Maintenance Procedures and Personnel Error Degrade Safety-Related Emergency Diesel Generator (Section 1R12)
2005004-03	NCV	Deficient Procedure and Personnel Error While Replacing ‘B’ 125/250 Volt Battery Cell (Section 1R19)
2005004-04	NCV	Deficient Procedure and Operator Error Degrade Two-Hour Emergency Air Supply to Emergency Feedwater and Main Steam Systems (Section 1R19)

2005004-05	NCV	Failure to Conduct Radiological Evaluation to Support Waste Transfer (Section 2PS2)

Closed

2005002-02	URI	

Updated

2005006-01	AV	ERO Qualifications Expired Due to Untimely Training (Section 4OA5.2)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Drawing 302-82, "Emergency Feedwater," Rev. 22
Drawing 302-202, "Nuclear Services River Water," Rev. 68
Drawing 302-273, " Emergency Feedwater and Main Steam Valve Two-Hour Backup Supply Air," Rev. 20
Drawing 302-610, "Nuclear Services Closed Cycle Cooling Water," Rev. 73
Drawing 302-640, "Decay Heat Removal," Rev. 79
Drawing 302-645, "Decay Heat Closed Cycle Cooling Water," Rev. 37
Drawing 302-661, "Makeup Purification," Rev. 54
Drawing 302-712, "Reactor Building Spray," Rev. 45

Section 1R12: Maintenance Effectiveness

IR 273060	Expansion Joint Replacements Needed on CW System
IR 320086	NR-P-1A Expansion Joint Condition Questioned
IR 320094	NR-P-1B Expansion Joint Condition Questioned
IR 203113	Condenser Expansion Joint Vulnerability Identified In OE17748
IR 268525	Condensate Pump 1A Inlet Expansion Joint Partially Collapsed
IR 268530	Condensate Pump 1C Inlet Expansion Joint Partially Collapsed
IR 250605	OE 17454, 18596, 17654 Indicate That TMI Should Inspect EFW
IR 172426	During 1300-3G Received Alarm A0331
IR 344477	EG-Y-1B Exhaust Leak on OCS
IR 344680	Extent of Condition for EG-Y-1A Loose Bolts
IR 344684	Extent of Condition for EG-Y-4 Loose Bolts
VM-TM-2795	Vendor Manual Mercer Rubber Expansion Joints
1410-Y-36	Expansion Joint Replacement Procedure, Rev. 8
A1726908	Expansion Joint Inspection General Activity Procedure
EPRI 1003189	Expansion Joint Maintenance Guide
LS-AA-115	"Operating Experience Procedure," Rev. 4

Section 1R15: Operability Evaluations

Westinghouse Instruction Leaflet 2951-OL, "Medium AC Motors Vertical, High Thrust, P-Base"
Westinghouse Letter RRS/DSE(99)-403, "DR-P-1A/B Motor Study Report"

MA-AA-716-230-1003	Thermography Program Guide, Rev. 1
MA-AA-716-012	Post Maintenance Testing, Rev. 4
IR 310670	AH-V-1A Test Results Indicate Valve Issues
OPE-05-008	AH-V-1A Operability Evaluation
VM-TM-2734	AOV Program Design Data - Vendor Manual Pratt 48 Inch Model NR1A Valve
C-1101-900-E420-186 ACE	AOV Design Basis Capability Calculation For AH-V-1A
OP-TM-MAP-F0103	"RCP Seal #1 Leak-off Flow," Rev. 1
CR 297543	AH-E-19B Was Inoperable During Past Period
AR-A2105561	Control Building Vent Fan AH-E-19A High Vibration Alarm
OP-AA-106-101-1006	"Operational And Technical Decision Making Process," Rev 2, (for control building ventilation fan AH-E-19A)

Section 1R17: Permanent Plant Modifications

UFSAR Section 7.3.3, "Incore Monitoring System," Rev. 17
TS Section 3.5.2, "Control Rod Group and Power Distribution Limits"
TS Table 3.5-3, "Post Accident Monitoring Instrumentation"
TS Table 4.1-1, Item 34, "Incore Neutron Detectors"

Section 1EP4: Emergency Action Level and Emergency Plan Changes

TMI Consolidated Emergency Plan
TMI Annex Emergency Plan and Implementing Procedures

Section 4OA2.3: Radioactive Material Processing and Transportation

Chemistry, Radwaste, and Process Control Audit, dated May 19, 2004 (AR 214020)
Check In Self-Assessment, 262490, dated October 1, 2004
Chemistry, Radwaste, and Process Control Program Supplemental Report, dated August 19, 2004 (AR 214020)
Receipt and Shipment of Radioactive Materials, Audit NOSPA-TM-05-1Q, dated March 8, 2005
Solid Radwaste Program Controls, Audit NOSPA-TM-03-4Q, dated December 16, 2003
Three Mile Island Assessment, 10 CFR Part 61 Database, dated August 2004
Nuclear Oversight Audit No. AR -00160620, dated June 2, 2000
Action/Issue Requests (195373, 202728, 207753, 213194, 220869, 224304, 233274, 236914, 242228, 261674, 296724, 242425, 242434)

Section 4OA5: Other

1107-11	"TMI Grid Operations," Rev. 13
1082.1	"TMI Risk Management Program," Rev. 5
1303-4.16	"TMI-Unit 1-Emergency Power System," Rev. 106
LS-AA-1010	"Exelon Reportability Reference Manual," Rev. 11
LS-AA-1400	"Exelon Reportability Reference Manual," Rev. 1
OP-AA-108-107-1001	"Station Response to Grid Capacity Conditions," Rev. 0

OP-TM-AOP-020	"Loss of Station Power," Rev. 7
OP-TM-864-901	"SBO Diesel Generator (EG-Y-4) Operations," Rev. 4
WC-AA-101	"On-line Work Control Process," Rev. 10

LIST OF ACRONYMS

ADAMS	Agencywide Documents and Management System
AmerGen	AmerGen Energy Company, LLC
AR	Action Request
BS	Building Spray
CBCW	Control Building Chill Water
CFR	Code of Federal Regulations
DC	Direct Current
DH	Decay Heat
DHR	Decay Heat Removal
DR	Decay River Water
E-Plan	Emergency Plan
EDG	Emergency Diesel Generator
EFW	Emergency Feedwater
EP	Emergency Preparedness
ERO	Emergency Response Organization
gpm	Gallons per Minute
HRA	High Radiation Area
IA	Instrument Air
IEEE	Institute of Electrical and Electronics Engineers
IMC	Inspection Manual Chapter
IR	Issue Report
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LPI	Low Pressure Injection
M&TE	Measuring and Test Equipment
MR	Maintenance Rule
MS	Main Steam
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NR	Nuclear River Water Cooling
NRR	Nuclear Reactor Regulation
OWA	Operator Work-Around
PCP	Process Control Program
PMT	Post Maintenance Test
RCP	Reactor Coolant Pump
SDP	Significance Determination Process
SM	Shift Manager
SSCs	Structures, Systems and Components
TI	Temporary Instruction

TM	Temporary Modification
TMI	Three Mile Island, Unit 1
TS	Technical Specifications
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item