February 11, 2008

Mr. Charles G. Pardee
Chief Nuclear Officer and Senior Vice President
Exelon Generation Company, LLC
200 Exelon Way
Kennett Square, PA 19348

SUBJECT: THREE MILE ISLAND STATION, UNIT 1 – NRC INTEGRATED INSPECTION REPORT 5000289/2007005

Dear Mr. Pardee:

On December 31, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Three Mile Island, Unit 1 (TMI) facility. The enclosed inspection report documents the inspection results, which were discussed January 22, 2008, with Mr. Rusty West and other members of your staff.

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

This report documents one Severity Level IV non-cited violation, three NRC-identified findings, and three self-revealing findings of very low safety significance (Green). All of the findings were determined to involve violations of NRC requirements. Additionally, licensee-identified violations which were determined to be of very low safety significance are listed in Section 4OA7 of this report. However, because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating these violations as non-cited violation (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis of your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, 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, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

We appreciate your cooperation. Please contact me at 610-337-5200 if you have any questions regarding this letter.

Sincerely,

**/RA/**

Ronald R. Bellamy, Ph.D., Chief
Projects Branch 6
Division of Reactor Projects

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

Enclosure: Inspection Report 05000289/2007005
w/Attachment: Supplemental Information

cc w/encl:
Chief Operating Officer, AmerGen
Site Vice President – TMI Unit 1, AmerGen
Plant Manager – TMI, Unit 1, AmerGen
Regulatory Assurance Manager – TMI, Unit 1 AmerGen
Senior Vice President – Nuclear Services, AmerGen
Vice President – Mid-Atlantic Operations, AmerGen
Vice President – Operations Support, AmerGen
Vice President – Licensing and Regulatory Affairs, AmerGen
Director Licensing – AmerGen
Manager Licensing – TMI, AmerGen
Vice President – General Counsel and Secretary, AmerGen
T. O'Neill, Associate General Counsel, Exelon Generation Company
J. Fewell, Esq., Assistant General Counsel, Exelon Nuclear
Correspondence Control Desk – AmerGen
Chairman, Board of County Commissioners of Dauphin County
Chairman, Board of Supervisors of Londonderry township
R. Janati, Director, Bureau of Radiation Protection, State of PA
J. Johnsrud, National Energy Committee
E. Epstein, TMI-Alert (TMIA)
D. Allard, PADEP

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Director Licensing – AmerGen
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Distribution w/encl: **(VIA EMAIL)**
S. Collins, RA
M. Dapas, DRA
D. Lew, DRP
J. Clifford, DRP
R. Bellamy, DRP
S. Barber, DRP
A. Rosebrook, DRP
D. Kern, DRP, Senior Resident Inspector
J. Brand, DRP, Resident Inspector
C. LaRegina, DRP, Resident OA
G. West, RI, OEDO
R. Summers, OE
H. Chernoff, NRR
J. Lubinski, NRR
P. Bamford, PM, NRR
E. Miller, NRR
ROPreports@nrc.gov
Region 1 Docket Room (with concurrences)

SUNSI Review Complete _AAR___(Reviewer's Initials)

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REGION 1

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| Docket No: | 05000289 |
| License No: | DPR-50 |
| Report No: | 05000289/2007005 |
| Licensee: | AmerGen Energy Company, LLC (AmerGen) |
| Facility: | Three Mile Island Station, Unit 1 |
| Location: | PO Box 480<br>Middletown, PA 17057 |
| Dates: | October 1 – December 31, 2007 |
| Inspectors: | David M. Kern, Senior Resident Inspector<br>Javier M. Brand, Resident Inspector<br>Ronald M. Nimitz, Senior Health Physicist<br>John Commiskey, Health Physicist<br>Jeffrey Bream, Reactor Engineer<br>E. Harold Gray, Senior Reactor Inspector<br>Thomas Burns, Reactor Inspector<br>Jeff Kulp, Reactor Inspector<br>Paul Frechette, Physical Security Inspector<br>Doug Tifft, Reactor Inspector |
| Approved by: | Ronald R. Bellamy, Ph.D., Chief<br>Projects Branch 7<br>Division of Reactor Projects (DRP) |

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## SUMMARY OF FINDINGS

IR 05000289/2007005; 10/01/2007 – 12/31/2007; AmerGen Energy Company, LLC; Three Mile Island, Unit 1; Operability Evaluations, Refueling and Other Outage Activities, Surveillance Testing, and Event Follow-up.

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

A. NRC-Identified and Self-Revealing Findings

**Cornerstone: Barrier Integrity**

- Green. The inspectors identified a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, for ineffective corrective actions to a previously identified NCV regarding a failure to maintain structural design clearances inside the reactor building. This violation involves several permanently installed structures inside the containment that did not meet the required separation distance to the containment liner. The inadequate structural clearance increased the likelihood of damage to the safety-related containment liner during a postulated seismic event. Corrective actions included evaluation of the specific conditions [Issue Reports (IR) 694026, 700592, and 700679] and initiation of actions to move the elevator support structure away from the containment liner during the 2009 refueling outage.

  This finding is more than minor because it impacted the configuration control attribute of the Barrier Integrity cornerstone objective to ensure the containment barrier protects the public from radionuclide releases. The containment design parameter for clearance between structures and the containment liner was not maintained. The finding is of very low safety significance because the issue did not involve an actual open pathway in the physical integrity of the containment. This finding has a cross-cutting aspect in the area of Problem Identification and Resolution, because engineering inspections performed as corrective actions to a previous NCV did not thoroughly evaluate the containment liner for additional clearance deficiencies [P.1(c)]. (Section 1R15)

- Green. A self-revealing NCV of Technical Specification (TS) 6.8.1.c was identified for failure to properly implement procedures to safely move control rod assemblies (CRAs) within the spent fuel pool (SFP). Fuel handling operators did not monitor the control mast load cell during CRA movement activities and did not verify the transit path (including the control mast) was clear of obstruction prior to bridge or trolley movement. These human performance deficiencies resulted in a damaged CRA and had the potential to damage the affected fuel assembly (FA) cladding fission product barrier. Corrective actions included verifying proper CRA handling equipment operation, increased personnel and supervisory oversight for all FA or CRA moves, event lesson learned briefings, and various procedure revisions to strengthen verification requirements.

The issue was more than minor because it affected the human performance attribute of the Barrier Integrity cornerstone objective to ensure the fuel cladding barrier protects the public from radionuclide release. The CRA was damaged and another CRA had to be selected for core reload. However, the inspectors determined the affected FA fuel clad barrier was not damaged and that containment controls were unaffected. Therefore a Phase 2 quantitative assessment was not required and the issue screened to Green (very low safety significance). The finding has a cross-cutting aspect in the area of Human Performance, because fuel handling operations personnel did not follow procedure requirements for safely moving CRAs in the SFP [H.4(b)]. (Section 4OA3.2)

- Green. A self-revealing NCV of TS 6.8.1.c was identified for failure to properly establish and implement procedures to safely reload fuel into the reactor vessel core. Fuel handling operators proceeded to insert a FA without properly verifying the fuel movement could be safely accomplished. The FA became hung up on a cable, was damaged, and required replacement and core redesign for cycle 18 operation. Corrective actions included core redesign, a stand-down and event briefing for all refueling personnel, procedure revisions, redesign of the shoehorn cables, a root cause evaluation of fuel handling errors, and additional cameras and viewing monitors to further improve visibility during core reload.

  The issue was more than minor because it affected the human performance attribute of the Barrier Integrity cornerstone objective to ensure the fuel cladding design barrier protects the public from radionuclide release. The FA was damaged and another FA was selected for core reload, requiring core redesign analysis. However, the inspectors determined the affected FA fuel clad barrier remained intact and that containment controls were unaffected. Therefore a Phase 2 quantitative assessment was not required and the issue screened to Green (very low safety significance). The finding has a cross-cutting aspect in the area of Human Performance, because fuel handling personnel proceeded ahead in the face of uncertainty, without stopping to restore proper visibility [H.4(a)]. (Section 4OA3.3)

**Cornerstone: Mitigating Systems**

- Severity Level IV. The inspectors identified a Severity Level IV NCV of TS 6.8.1.j for not properly implementing and maintaining procedures for controlling plant staff work hours of personnel performing safety-related activities. Procedure LS-AA-119, Overtime Controls, was deficient in that it permitted the plant manager to authorize work-hour deviations for routine refueling outage activities. Consequently, the plant manager authorized over 700 personnel to work greater than 72 hours and up to 84 hours per 7-day period for routine outage support activities during the TMI refueling outage (1R17), which exceeded the TS requirements. The affected workers included reactor operators, senior reactor operators, auxiliary operators, health physicists, key maintenance personnel, emergency response organization members, reactor engineers supporting reactivity manipulations and fuel handling, and engineering and professional personnel performing safety-related work. The licensee entered this issue into their corrective action program (IR 713257).

Changing implementing procedure, LS-AA-119, so that it no longer complies with the facility technical specification for controlling the plant staff overtime has the potential to impact the NRC's ability to perform its regulatory function. The violation affected the Mitigating Systems cornerstone and is more than minor because, if left uncorrected, the excessive work hours would increase the likelihood of human errors during refueling outage activities and response to plant events. This violation is characterized as a Severity Level IV in accordance with the NRC Enforcement Policy. This issue has a cross-cutting aspect in the area of Human Performance, because procedure LS-AA-119 did not provide adequate instructions to provide reasonable assurance that station management would properly control overtime for plant staff performing safety-related functions [H.2(c)]. (Section 1R20)

- Green. The inspectors identified an NCV of TS 6.8.1.a for failure to maintain the appropriate reactor coolant system (RCS) vent area required by station procedures during mid-loop operation. The reduced vent area degraded operators' ability to add water to the RCS in the event of a loss of decay heat removal (DHR) and caused reactor vessel level indication to be inaccurate. Corrective actions included operator crew briefings, communications between radiological protection and operations personnel, and initiation of IRs 698486 and 705000.

  This issue affected the configuration control attribute of the Mitigating Systems cornerstone and was more than minor because it affected the availability of water from the borated water storage tank (BWST) to the RCS in the event of a loss of decay heat removal and caused reactor vessel level indication to be inaccurate. The inspectors determined that although design margin was reduced, the RCS gravity feed and bleed from the BWST function remained operable. The inspectors also concluded that level indication remained sufficient to alert operators if a significant change occurred which would warrant operator actions. Therefore a Phase 2 quantitative assessment was not required and the issue had very low safety significance. The finding has a cross-cutting aspect in the area of Human Performance, because work control during removal of the High Efficiency Particulate Air (HEPA) fans and ventilation hoses from the once through steam generator (OTSG) handholes was deficient. Operators and Radiation Protection (RP) technicians did not appropriately coordinate work activities to ensure the required RCS vent area was maintained when technicians established radiological postings for the OTSG handhole area [H.3(b)]. (Section 1R20)

- Green: The inspectors identified an NCV of TS 4.2.2 for failure to test eight safety-related valves in accordance with American Society of Mechanical Engineers Operations & Maintenance (ASME OM) Code requirements. Procedure OP-TM-211-211 contained no procedural steps to verify and document local position indication for eight safety-related make-up system valves. Plant staff revised the procedure and successfully tested the valves prior to the completion of this inspection period.

  The finding is more than minor because it affected the procedure quality attribute of the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. This finding was determined to be of very low safety significance since the condition did not involve an actual loss of safety function. This finding has a cross-cutting aspect in the area of Human Performance, because TMI did not ensure complete and accurate procedures were available for testing eight safety-related make-up system valves [H.2(c)]. (Section 1R22)

- Green. A self-revealing NCV of TS 6.8.1.a was identified for failure to properly coordinate maintenance and operational activities associated with installing the OTSG primary lower manways during mid-loop operation. Installation of the OTSG lower manway cover, while temporary ventilation fans were exhausting air from the OTSG handhole RCS vent, caused an unexpected drop in reactor vessel level indication and declaration of an Unusual Event emergency. Corrective actions included removing the ventilation fans and ventilation hose from the OTSG handholes, restricting the use of the fans, and initiating IRs 698291, 698693, and 699314.

  This issue affected the configuration control attribute of the Mitigating Systems cornerstone and was more than minor because this equipment lineup error affected the accuracy of reactor vessel level instrument indication during mid-loop operations, a high risk evolution. The inspectors determined that although all four reactor vessel level instruments were affected, their collective level indications, trends, and alarms provided sufficient information to alert operators in the event of an actual loss of inventory. Therefore a Phase 2 quantitative assessment was not required and the issue was of very low safety significance. The finding has a cross-cutting aspect in the area of Human Performance, because the installation and removal of temporary OTSG ventilation during installation of the OTSG lower primary manway were not appropriately coordinated to ensure the operational impact on reactor vessel level indication while in mid-loop operation was understood. Consequently reactor vessel level indication was inaccurate and not understood by operations personnel while the plant was in an elevated shutdown risk condition [H.3 (b)]. (Section 4OA3.1)

B. Licensee Identified Violations

Violations of very low safety significance that were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective action tracking numbers are listed in Section 4OA7 of this report.

# REPORT DETAILS

Summary of Plant Status

Three Mile Island, Unit 1 (TMI) began the inspection period at 100 percent rated thermal power and gradually reduced power due to end-of-cycle fuel depletion. On October 21, operators began a plant shutdown and the turbine output breakers were opened on October 22, beginning the 17th refueling outage (1R17). Major work accomplished during this refueling outage included modification of the reactor building emergency core cooling sump, replacement of a vital bus inverter, replacement of an electrical containment penetration, inspection and mitigation of dissimilar metal welds on the pressurizer, reactor core refueling, and steam generator tube inspections. The 30 day refueling outage was completed on November 21 and the reactor achieved 100 percent rated thermal power on November 23. On November 30, operators performed an unplanned power reduction to 40 percent reactor power in response to a leak from the main condenser waterbox manway (section 4OA3.4). Operators returned the plant to full power on December 1, following replacement of all 12 main condenser waterbox manway gaskets.

## 1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01 – 1 site sample)

a. Inspection Scope

The inspectors walked down risk significant plant areas on December 6, 2007, and assessed AmerGen's protection for cold weather conditions. The inspectors evaluated outside instrument line conditions and the potential for unheated components and ventilation. The walkdown included the condensate storage tanks and safety-related river water system components located within the intake structure. The inspectors also reviewed implementation of procedures WC-AA-107, Seasonal Readiness, Rev. 4 and OP-AA-108-111-1001, Severe Weather Guidelines, Rev. 2 for cold weather conditions.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

Partial System Walkdowns (71111.04Q – 5 samples)

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

- On October 25 and November 13, 2007, the inspectors walked down portions of the reactor coolant and decay heat removal systems while the piping was at mid-loop (drain down) operation during IR17 (2 samples).

- On October 30, the inspectors walked down the ‘B’ emergency diesel generator (EDG) train, while the ‘A’ EDG train was unavailable for testing.

- On October 30, the inspectors walked down the ‘B’ low pressure injection (LPI) train, while the ‘A’ LPI train was running for shutdown cooling.

- On November 16, the inspectors walked down the ’A’ LPI train, while the ‘B’ LPI train was running for shutdown cooling.

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. Documents reviewed can be found in the attachment.

Complete System Walkdown (71111-04S – 1 sample)

On December 15, 2007, the inspectors performed one complete system walkdown sample on the emergency feedwater system. The inspectors conducted a detailed review of the alignment and condition of the system using the applicable one-line diagram 302-082, Emergency Feedwater, Rev. 24 and procedure OP-TM-424-000, Emergency Feedwater System, Rev. 6. In addition, the inspectors reviewed and evaluated the corrective action program reports for impact on system operation and interviewed the system engineer and control room operators.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Annual Drill Observation (71111.05A – 1 sample)

a. Inspection Scope

The inspectors observed an announced fire brigade drill on November 29, 2007, to evaluate the readiness of station personnel to respond to and fight fires. The drill demonstrated response to a fire in the Unit 1 Control Building 1D 4160 volt switchgear room. The inspectors observed fire brigade member use of protective clothing and appropriate turnout gear, including self-contained breathing apparatus (SCBA), and their approach and methods to combat the fire as well as their interaction with the control room staff. The inspectors observed implementation of the fire fighting strategies by the fire brigade and communications among participants throughout the drill. The inspectors reviewed the drill scenario objectives, determined whether drill scenario objectives were met, and observed the post drill critique to verify that the licensee identified, discussed, and entered adverse conditions into the corrective action program. Minor performance deficiencies were documented in IR 704915. Additional documents reviewed during this inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

.2 <u>Area Walkdowns</u> (71111.05Q – 7 samples)

a. <u>Inspection Scope</u>

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. 69. The plant walkdowns were conducted throughout the inspection period and included assessment of transient combustible material control, fire detection and suppression equipment operability, and compensatory measures established for degraded fire protection equipment in accordance with procedure OP-MA-201-007, Fire Protection System Impairment Control, Rev. 5. In addition, the inspectors verified that applicable clearances between fire doors and floors met the criteria of Attachment 1 of Engineering Technical Evaluation CC-AA-309-101, Engineering Technical Evaluations, Rev. 9. Additional documents reviewed during this inspection are listed in the Attachment. Fire zones and areas inspected included:

- Fire Zone RB-FZ-1A, Reactor Building Elevation 281', Outside Secondary Shield Wall-North;
- Fire Zone RB-FZ-1B, Reactor Building Elevation 281', Outside Secondary Shield Wall-South East;
- Fire Zone RB-FZ-1C, Reactor Building Elevation 281', Outside Secondary Shield Wall-South West;
- Fire Zone RB-FZ-1D, Reactor Building Elevation 281', Outside Secondary Shield Wall-South East;
- Fire Zone RB-FZ-1E, Reactor Building Elevation 281', Inside Secondary Shield Wall-West;
- Fire Zone RB-FZ-2, Reactor Building Elevation 308', Outside Secondary Shield Wall;
- Fire Zone RB-FZ-3, Reactor Building Elevation 346', Reactor Building Operating Floor.

b. <u>Findings</u>

No findings of significance were identified.

1R06 <u>Flood Protection Measures</u> (71111.06 – 1 internal sample)

a. <u>Inspection Scope</u>

The inspectors reviewed the following IRs which documented multiple minor deficiencies identified in the auxiliary building where internal flooding could adversely affect safety-related systems needed for safe shutdown of the plant. In addition, the inspectors performed detailed visual inspections of flood barriers and floor drains in the 'A' building spray vault.

- IR 683687, WDL-V-720A FME Concerns in 'A' Building Spray Vault
- IR 683761, FME Removed from 'A' Spray Vault Floor Drain
- IR 714421, Low Water Level in 'A' Decay Heat Vault
- IR 714429, Water Level Below Required Level in Loop Seal for WDL-V-721C
- IR 7114427, Low Water Level in 'A' Makeup Pump Loop Seal for WDL-V-721A

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection (71111.08 – 1 sample)

a. Inspection Scope

The inspectors observed selected samples of nondestructive examination (NDE) activities in process and reviewed documentation of completed NDE and repair/replacement activities. The sample selection was based on the inspection procedure objectives and risk priority of those components and systems where degradation would result in a significant increase in risk of core damage. The observations and documentation review was performed to determine whether the activities were performed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements. Additional documents reviewed during this inspection are listed in the Attachment.

The inspectors reviewed a sample of inspection reports, IRs, and Action Requests (ARs) that were initiated as a result of problems identified during inservice inspection (ISI) examinations. Also, the inspectors evaluated effectiveness in the resolution and corrective action of problems identified during ISI activities for selected samples.

The inspectors observed the performance of three NDE activities in process and reviewed documentation and examination reports for an additional three NDE activities. The inspectors reviewed five samples of welding activities on a pressure boundary and reviewed two ASME repair packages for repair/replacements performed this operating cycle.

The inspectors observed manual ultrasonic testing (UT), eddy current testing (ECT), and magnetic particle testing (MT) and reviewed inspection documentation of liquid penetrant (PT) and visual testing (VT) activities to determine effectiveness of the process, examiner, and equipment in identifying degradation of risk significant systems, structures and components and to evaluate the activities for compliance with the requirements of ASME Section IX of the Boiler and Pressure Vessel Code.

The inspectors observed the UT testing of welds FW 251, 252, 254 and MT testing of welds FW 038 and 039 in the main steam (MS) system (System 411). For dissimilar metal welds the inspectors reviewed the UT procedure, data, and practices for the 82/182 filler metal 4" spray nozzle, SP-0021-BM, Weld PR-009-BM and HPI weld MU-908-BM. Also, the inspectors reviewed the radiographs of two pressurizer relief nozzle flange replacement welds R-1 and R-2.

The inspectors reviewed the NDE report of the PT of a control rod drive mechanism (CRDM) to flange weld in the control rod drive (CRD) system. In addition, the inspectors reviewed issue reports (IRs 552813-02 and 66890) and the 17R Inspection Report listing of boric acid deposits identified this outage. The inspectors noted that the predominate nature of the leaks reported were determined to be minor and from mechanical seals or failed valve packing. The inspectors reviewed the disposition of a sample of these reports to determine that the identification, characterization, and repair instructions were complete and captured in the corrective action program.

The inspectors evaluated implementation of the steam generator program by review of portions of the steam generator management plan for outage 1R17 and the condition monitoring and final operational assessment of outage 1R16 activities. The inspectors reviewed plant specific steam generator design information, tube inspection criteria, control and monitoring of foreign objects, integrity assessments, degradation modes and tube plugging criteria. The inspectors determined through examination of calibration documentation that the eddy current testing probes and related inspection equipment in use had been calibrated and qualified for the expected types of active tube degradation. The inspectors determined that the licensee had performed the required review of the equipment calibration documentation and had accepted the equipment for service. Personnel training and qualification documentation was reviewed by the inspectors to determine that data acquisition and resolution analysts had been trained and tested in the eddy current inspection process.

The inspectors observed AmerGen's performance of portions of the one hundred percent bobbin inspection of selected tubes for their entire length in both generators. The inspectors reviewed the eddy current examination plan to determine whether the plan met TS requirements, and EPRI Guidelines. The inspectors reviewed the steam generator inspection plan to determine whether the identified areas of potential tube degradation (based on site-specific and industry experience) were being inspected, with special attention to areas that are known to represent potential eddy current testing challenges.

The inspectors evaluated the implementation of the steam generator inspection program by conducting interviews with data management and acquisition personnel, data analysts and resolution analysts. The inspectors interviewed the licensee's independent qualified data analyst, and reviewed selected samples of eddy current data and data analysis of selected tubes within the A and B steam generators.

No tubes were repaired during the period the inspectors were on site. Although the licensee was equipped to perform tube repairs, the decision was made that any tube identified as failing the acceptance criteria would be removed from service by plugging. Several tubes had been identified for plugging at the conclusion of the tube inspection activity. The inspectors reviewed the eddy current test data for four tubes selected from steam generator "A" and two tubes selected from "B". The samples selected represented tubes which exhibited wall thinning in excess of the specified acceptance criteria of a maximum degradation of 40% thru wall (localized or area). The tubes selected from "A" steam generator were #21-row 18, #32-row 96, #59-row 45 and #77-row 15. The tubes selected from "B" steam generator were #37-row 146 and #71-row 52. These tubes were removed from service by plugging. No tubes were identified as candidates for in-situ pressure testing during the inspection period. The inspectors reviewed data which indicated that steam generator leakage of greater than three gallons per day had not occurred during this operating cycle and was not noted during the post-shutdown visual inspection of the tube sheet faces.

The inspectors reviewed the procedures used to perform visual examinations for indications of boric acid leaks from pressure retaining components and also reviewed the visual examination records for 9 CRDM penetrations above the reactor pressure vessel (RPV) head and 15 RPV lower head incore penetrations.

The inspectors reviewed a sample of IRs and ARs initiated as a result of the inspections performed in accordance with the licensee's boric acid control (BACC) program. The inspectors reviewed five ARs shown on Attachment 1 that identified evidence of both active and inactive leak locations which could result in degradation of safety significant components. The inspectors reviewed operability evaluations and corrective actions provided in the AR and determined that the actions specified were consistent with the requirements of the ASME Code and 10 CFR 50, Appendix B, Criterion XVI.

The inspectors reviewed the Weld Process Travelers and observed in-process welding for portions of the full structural weld overlays of welds PR-021BM, DH-001BM, and DH-498. Additionally, welding of pressurizer relief nozzle flange replacements was observed.

The inspectors performed a visual evaluation of selected portions of the TMI reactor building containment liner coating from the concrete floor slab (281 ft), entrance hatch (308 ft), and the operating floor (346 ft) elevations. The evaluation was made to determine compliance with the requirements of ASME Section XI, IWE (requirements for Class MC and Metallic Liners of Class CC components). During this examination, the inspectors noted that the liner coating (above the moisture barrier location) exhibited no signs of peeling, blistering, or corrosion.

However, the inspectors noted there was evidence of corrosion activity at the liner plate/concrete floor interface, which the licensee was evaluating. The licensee had removed the moisture barrier and caulking materials in this location and was in the process of cleaning the barrier area prior to performing a visual inspection (VT-3). The visual inspection was to assess the condition of the liner plate, determine the presence of moisture, and assess the extent of corrosion activity. Selected corroded locations which were estimated to be "worst case" by visual examination were cleaned and further examined to determine remaining wall thickness by UT and pit depth measurements using a pit gauge. The inspectors reviewed the wall thickness determinations and noted that the remaining wall thickness in "worst case" locations varied from a low of 0.239" to a high of 0.377". The licensee indicated the liner was originally fabricated using carbon steel plate thickness of 3/8" nominal thickness (walls and dome) and 3/4" nominal thickness (below the concrete floor slab). The inspectors compared the as found wall thicknesses in the moisture barrier corroded locations with the design minimum wall thickness requirement and found the remaining material to be within design requirements.

The licensee initiated IR 00694554, IS-T1R17-RB Liner Corrosion Indication #53 on Plate C1-32 to capture the liner corrosion issue in their corrective action program. The inspectors reviewed this IR to evaluate the licensee actions to identify the cause of the corrosion, planned action for engineering analysis of the condition and repair/rework considerations if necessary. Also, the inspectors discussed and reviewed the licensee planned actions to monitor this condition during the next and future operating cycles. Additional containment liner corrosion issues identified later during the outage are reviewed separately in Section IR15 (Operability Evaluations).

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

On December 21, 2007, the inspectors observed licensed operator requalification training at the control room simulator for the 'D' operator crew. The inspectors observed the operators' simulator drill performance and compared it to the criteria listed in TMI Operational Simulator Scenario Number 23, Loss of 1B Screen House MCC-FW Line Break on 'B' OTSG Inside Containment, Rev. 6. The inspectors reviewed the operators' ability to correctly evaluate the simulator training scenario and implement the emergency plan. The inspectors observed supervisory oversight, command and control, communication practices, and crew assignments to ensure they were consistent with normal control room activities. The inspectors observed operator response during the simulator drill transients. The inspectors evaluated training instructor effectiveness in recognizing and correcting individual and operating crew errors. The inspectors attended the post-drill critiques in order to evaluate the effectiveness of problem identification. The inspectors verified that emergency plan classification and notification training opportunities were tracked and evaluated for success in accordance with criteria established in Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, Rev. 5.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12Q – 1 sample)

a. Inspection Scope

The inspectors evaluated the listed sample for Maintenance Rule (MR) implementation by ensuring appropriate: 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. Additionally, extent of condition follow-up, operability, and functional failure determinations were reviewed to verify they were appropriate. 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", Nuclear Management and Resources Council (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. 5. The inspectors verified that appropriate corrective actions were initiated and documented in IRs, and that engineers properly categorized failures as maintenance rule functional failures and maintenance preventable functional failures, when applicable.

- IR 704832 describes a trip of one of the three reactor building emergency ventilation fans (AH-E-1B), while the Unit was operating at 100% power. The other two fans remained operating. Inspections and troubleshooting did not identify the cause of the trip, and engineers determined the fan remained operable.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed the scheduling, control, and restoration during the following maintenance activities to evaluate their effect on plant risk. This review was against criteria contained in AmerGen Administrative Procedure 1082.1, TMI Risk Management Program, Rev. 6 and WC-AA-101, On-Line Work Control Process, Rev. 14.

- On October 9, 2007, the intermediate closed cooling pump (IC-P-1B) was removed from service for scheduled maintenance activities. The condition elevated the online maintenance risk profile to yellow (Risk Document 1022, Rev. 4).

- On October 17, the DH-V-7A breaker was removed from service for scheduled maintenance activities. The condition elevated the online maintenance risk profile to yellow (Risk Document 1142, Rev. 1).

- On October 30, the 'A' engineered safeguards train was removed from service for scheduled emergency sequence and power transfer test, while the plant was shutdown for refueling outage. The condition elevated the shutdown risk profile to orange.

- On November 13, operators drained down the reactor coolant system to midloop (between 12 and 15 inches above the cold leg pipe centerline) after completion of reloading the fuel into the reactor vessel, to support removal of the steam generator cold leg dams and installation of the lower diaphragm manway covers. The condition elevated the shutdown risk profile to orange.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15 – 4 samples)

a. Inspection Scope

The inspectors verified that degraded conditions in question were properly characterized, operability of the affected systems was properly evaluated in relation to TS requirements, applicable extent of condition reviews were performed, and no unrecognized increase in plant risk resulted from the equipment issues. The inspectors referenced NRC IMC Part 9900, Operable/Operability-Ensuring the Functional Capability of a System Component and AmerGen procedure OP-AA-108-115, Operability Determinations, Rev. 3, to determine acceptability of the operability evaluations. Additional documents reviewed during this inspection are listed in the Attachment. The inspectors reviewed operability evaluations for the following degraded equipment issues:

- During the 1R17 refueling outage, AmerGen engineers inspected the containment liner plate to concrete interface around the complete reactor building perimeter. Multiple corrosion concerns were identified at various locations including, the incore instrumentation room, underneath the equipment hatch, and near the containment sump area (IRs 692798 and 693738). The inspectors performed independent visual inspections of these and other similar locations, interviewed the system engineer and contractors, and reviewed the licensee's evaluations and applicable corrective actions. AmerGen engineers concluded that the as-found wall thicknesses for the identified corrosion remained within design requirements. After additional remedial action, the inspectors verified that a complete repair of the moisture barrier had been performed and that the licensee had implemented long term actions for the continued monitoring and inspections of the areas in question. Additional containment liner corrosion issues that were identified earlier during outage are also reviewed in Section IR08 (Inservice Inspection)

- Fuel assembly guide tube growth beyond original analysis (see section 4OA3.5) raised concerns that control rod assemblies may not be capable of achieving the necessary shutdown margin and related rod insertion times. Based on assessing control rod assembly drag resistance and rod drop time test data, engineers determined that at the end of the upcoming operating cycle (cycle 17) two control rod assemblies may not fully insert into the core in response to a reactor trip (IR 699861). Engineers concluded that the control rods would meet their 75 percent insertion response time and therefore remained operable. Additionally, the overall core reactivity insertion requirements would continue to be met to ensure safe shutdown of the reactor.

- Operators heard unusual noises from emergency feedwater pump EF-P-1B during a quarterly surveillance test (IR 499576). Additional testing and monitoring were performed with no further anomalies (IR 502008). A test plan to monitor for long term degradation was developed under work order A2144784.

- The inspectors performed visual inspections of the structural integrity of the reactor building containment liner during the outage to ensure that required seismic clearances between the liner and permanently installed structures were maintained, to verify the liner surface was free of defects, and to assess the condition of the safety-related coatings inside containment. The inspectors also reviewed controls of transient equipment and other activities to protect the liner and the liner coatings from damage. In addition, the inspectors evaluated the licensee's corrective actions to address a previous NRC identified deficiency regarding permanently installed structures that were in direct contact or close proximity to the containment liner (refer to NCV 05000289/2003005-03, Failure To Maintain Structural Design Clearances Inside Reactor Building Containment).

b. Findings

Ineffective Corrective Actions for Failure to Maintain Structural Design Clearances Inside Reactor Building Containment.

Introduction. The inspectors identified a Green NCV of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, for ineffective corrective actions to a previously identified NCV regarding a failure to maintain structural design clearances inside the reactor building. The clearance is designed to prevent damage to the safety-related containment liner during a postulated seismic event.

<u>Description</u>. In November 2005, while the plant was shutdown for refueling outage 1R16, TMI engineers completed detailed inspections and walkdown of the clearance between the containment liner and permanently installed structures. These actions were performed as corrective actions to address deficiencies identified by the inspectors during the 2003 refueling outage (NCV 05000289/2003005-03). The engineering inspection results were documented and evaluated in TMI Technical Evaluation A2099765, As-Found Liner Clearances. This Technical Evaluation also determined that the one inch clearance requirements detailed in structural drawing 421054 were overly conservative and established revised minimum clearance requirements for the more critical elevations of the containment building.

During follow-up visual inspections of the containment liner to assess the licensee's corrective actions during 1R17, the inspectors noted several additional deficiencies that had not been identified or evaluated by the licensee. These deficiencies involved permanent structures that were in contact with the liner or closer than the newly established minimum clearances. These deficiencies included: 1) a heavy duty elevator metal shock absorber system (vertical and horizontal frames and heavy metal spring) installed at the bottom of the elevator shaft in contact or less than 1/8 inch away from the liner (IR 694026); 2) a 14 foot long sheet metal box structure bolted on top of the reactor building elevator shaft (IR 700592); and 3) a metal box for penetration 317 (IR 700679) also installed less than 1/8 inch away from the liner. Subsequent engineering evaluation of these deficiencies determined that the maximum relative seismic displacement between the containment building shell and the structures was small and no damage to the containment liner would have resulted in case of impact during a seismic event, due to the robust liner (3/8 inch thick carbon steel plate) and the thickness of the concrete behind it.

The inspectors also observed multiple examples where outage related transient metal components came in direct contact with the containment liner during the 1R17 refueling outage activities. In some cases, contact between the containment liner and the components resulted in minor damage (scratches and gouges) to the liner. The inspectors determined that these issues also indicated ineffective corrective actions to similar issues identified by the inspectors during the 2003 refueling outage. The identified components included several scaffold structures and poles, one heavily loaded four-wheeled cart, and several large metal pieces being installed in the new reactor building sump. Engineers evaluated these issues under IR 691297 and determined that no significant damage to the containment liner occurred.

<u>Analysis</u>. Failure to ensure that permanent structures located inside containment were properly installed per the applicable structural drawing 421054 and TMI technical Evaluation A2099765 constitutes a performance deficiency.

This finding is more than minor because it impacted the configuration control attribute of the Barrier Integrity cornerstone objective to ensure the containment barrier protects the public from radionuclide releases. The containment design parameter for clearance between structures and the containment liner was not maintained. Using NRC Manual Chapter 0609, "Significance Determination Process," Appendix A, Phase 1, this finding was determined to be of very low significance since the issue did not involve an actual open pathway in the physical integrity of the containment boundary.

This finding has a cross-cutting aspect in the area problem identification and resolution, because engineering inspections did not thoroughly evaluate a similar issue previously identified, such that the extent-of-condition assessment did not identify three clearance deficiencies to the containment liner during implementation of their corrective actions [P.1(c)].

Enforcement. 10 CFR 50, Appendix B, Criterion XVI, Corrective Action, requires, in part, that conditions adverse to quality such as deficiencies, deviations, and nonconformances are promptly identified and corrected. Contrary to this requirement, station personnel failed to identify several liner clearance deficiencies during implementation of corrective actions to address a previously identified NRC violation. As a result, the licensee did not ensure that structural components located inside containment were properly installed per design drawing 421054 and Engineering Technical Evaluation A2099765. Because the violation was of very low safety significance and TMI entered this issue into its corrective action program (IRs 694026, 700592, 700679, and 691297), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: **NCV 05000289/2007005-01, Ineffective Corrective Actions for Failure to Maintain Structural Design Clearances Inside Reactor Building Containment.**

1R19 Post Maintenance Testing (71111.19 – 5 samples)

a. Inspection Scope

The inspectors reviewed and/or observed the following post-maintenance test (PMT) activities 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. Additional documents reviewed during this inspection are listed in the Attachment.

- On October 22, 2007, instrument air compressor IA-P-4 was successfully post maintenance tested using work order C2016098 following replacement of the low and high pressure elements.

- On November 8, operators performed post maintenance testing of feedwater check valve FW-V-12B in accordance with procedure 1410-V-31, Crane Tilting Check Valve Inspection, Rev. 29, following complete valve overhaul and replacement of valve internal components.

- On November 8, operators performed filling and venting of the 'A' decay heat pump DH-P-1A in accordance with procedure OP-TM-21-253, Venting DH Train in DHR Standby Mode, Rev. 6, following a scheduled maintenance outage.

- On December 6, operators performed testing in accordance with procedure OP-TM-541-208 , IST Of NS-P-1A/B/C, Rev. 4, following scheduled inspection and corrective maintenance to replace the disc for check valve NS-V-10A.

- On December 20, operators performed testing in accordance with procedure 1300-3EA, IST Of 'A' Spent Fuel Pump and Valves, Rev. 1, following a scheduled inspection and corrective maintenance outage.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

Station personnel conducted refueling outage 1R17 from October 22 to November 21, 2007. The inspectors reviewed selected reactor shutdown, refueling, outage maintenance, and reactor startup activities to determine whether shutdown safety functions (i.e., reactor decay heat removal, reactivity control, electrical power availability, reactor coolant inventory, spent fuel cooling, and containment integrity) were properly maintained as required by TSs and TMI-2006-010, TMI-1 Outage Fuel Protection Criteria, Rev. 2. Specific attributes evaluated included configuration management, communications, instrumentation accuracy, and identification and resolution of problems. The inspectors closely evaluated configuration and inventory control during periods of reduced RCS inventory due to the associated increase in shutdown risk. The inspectors also performed inspections of accessible areas inside containment, interviewed applicable engineers, supervisors, and plant operators, and consulted with NRC specialists. Additional documents reviewed during the inspection are listed in the Attachment. Specific activities evaluated included:

- Safety Shutdown Review Board review of the TMI-1 1R17 Outage Risk Profile for plant conditions 10 and 11 conducted on November 7, 2007. This review assessed plant equipment contingencies and approved recharacterization of plant risk from red to orange for the period that the spent fuel pool cooling function was not operable.
- Plant cooldown
- Reactor vessel head removal and lift
- RCS drain to mid-loop
- Fuel offload and reload,
- Engineered safeguards train 'A' emergency sequence and power transfer test
- Reactor building emergency core cooling system sump as-found inspection
- Reactor building walkdown to inspect for indication of RCS leakage and boric acid corrosion
- Reactor coolant system draindown to mid-loop operation in accordance with procedure 1103-11 on October 25-26 and November 12-13
- Core flood flow testing
- Plant heatup
- Restoration of containment integrity.
- Plant Startup and Power ascension
- Unit Restart Review, following completion of 1R17

b. Findings

.1 Deficient Control of Plant Staff Overtime

Introduction. The inspectors identified a Severity Level IV non-cited violation of TS 6.8.1.j for not properly implementing and maintaining procedures for controlling plant staff work hours of personnel performing safety-related activities. Procedure LS-AA-119, Overtime Controls, Rev. 4 was deficient in that it permitted the plant manager to authorize work-hour deviations for routine refueling outage activities. Consequently, the plant manager authorized over 700 personnel to work greater than 72 hours and up to 84 hours per 7-day period for routine outage support activities during the TMI refueling outage (1R17), which exceeded the TS requirements.

Description. TS 6.8.1.j requires overtime for staff performing safety-related functions to be limited in accordance with NRC Generic Letter (GL) 82-12. This GL, that addresses Nuclear Power Plant Staff Working Hours, specifies, in part, that during extended periods of shutdown for refueling individuals should not be permitted to work more than 72 hours in any 7-day period. NRC GL 82-12 only permits deviation from this work-hour limit for “very unusual circumstances.” NRC GL 82-12 differentiates between work hour controls for “refueling outages” and for “very unusual circumstances.”

The inspectors determined that LS-AA-119 does not limit authorization of deviations from the work-hour limits to “very unusual circumstances” in a manner consistent with GL 82-12. LS-AA-119, Sections 2.6.1 and 4.2.1 state, “in addition to those “very unusual circumstances” described above, deviation from above overtime guidance may be considered for refueling outage and forced outages activities, and other significant emergent activities as deemed necessary by station management.” The inspectors observed that these controls are contrary to the guidance of NRC GL 82-12, as required by the TS. Sections 2.6.1 and 4.2.1 were first added to procedure LS-AA-119 in Rev. 3, which became effective at TMI on May, 27, 2005. Therefore, during 1R17, overtime for TMI workers performing safety-related activities was not properly controlled as required by TS 6.8.1.j.

AmerGen planned and scheduled TMI 1R17 based on station employees and contractor labor working 12 hours/day, 7 days/week through the duration of the outage. This 84 hour/week work schedule was offered to and accepted by most AmerGen personnel and contractors working on 1R17 activities. The plant manager or his designated deputy approved LS-AA-119, Attachment 1, Overtime Guideline Deviation Authorization forms for over 340 AmerGen employees and over 390 contract employees to perform routine refueling outage support activities. The affected workers included reactor operators, senior reactor operators, auxiliary operators, health physicists, key maintenance personnel, emergency response organization members, reactor engineers supporting reactivity manipulations and fuel handling, and engineering and professional personnel performing safety-related work.

Based on record reviews, personnel interviews, and in-plant walkthroughs, the inspectors determined that the majority of the authorized individuals worked greater than 72 hours and up to 84 hours per 7-day period during 1R17. Many of these workers performed safety-related work. The inspectors expressed concern that plant staff overtime was not being controlled as required by TS 6.8.1.j. Issue Report 713257 was initiated to evaluate the inspectors' concern and any related impact to Spring 2008 refueling outages among the 10 nuclear sites affected by this Exelon corporate procedure.

Analysis. The inspectors determined that failure to properly maintain and implement procedures to limit work-hours for plant staff performing safety-related functions in accordance with TS 6.8.1.j was a performance deficiency. Rev. 3 to LS-AA-119, which permitted deviation from the work-hour limits for refueling outages, was a change to the TS-required process for controlling plant staff overtime, and provided new implementing requirements that were contrary to the TS 6.8.1.j requirements. This change to the plant staff overtime controls has the potential to impact the NRC's ability to perform its regulatory function and is addressed through traditional enforcement.

In order to characterize the severity level of this violation, the NRC Enforcement Policy Supplement 1, "Reactor Operations", examples were reviewed. The violation was considered to be consistent with Example D.5 of Supplement 1. Therefore, the violation was characterized as a Severity Level IV violation. In addition, the violation was evaluated using the SDP. The violation affected the Mitigating Systems cornerstone and is more than minor because, if left uncorrected, the excessive work hours would increase the likelihood of human errors during refueling outage activities and response to plant events. The finding has been reviewed by NRC management in accordance with IMC 0609, Appendix M, Significance Determination Process Using Qualitative Criteria. The resulting increased likelihood of human error would adversely affect the station's defense-in-depth. However, the violation was determined to be of very low significance, because no significant events or human performance issues were directly linked to personnel fatigue as a result of the hours worked.

This issue has a cross-cutting aspect in the area of Human Performance, because procedure LS-AA-119 did not provide adequate instructions to provide reasonable assurance that station management would properly control overtime for plant staff performing safety-related functions to assure nuclear safety as required by TS 6.8.1.j. [H.2(c)].

Enforcement. Technical specification 6.8.1.j requires procedures be established, implemented, and maintained covering the control of Plant Staff Overtime, to limit the hours worked by staff performing safety-related functions in accordance with the NRC Policy Statement on working hours (NRC GL 82-12). NRC GL 82-12, Nuclear Power Plant Staff Working Hours, dated June 15, 1982, specifies, in part, that during extended periods of shutdown for refueling, guidelines shall be followed that limit individuals to working no more than 72 hours in any 7-day period. Recognizing that very unusual circumstances may arise, requiring deviation from this guideline, such deviation shall be authorized by the plant manager or his deputy, or higher levels of management.

Contrary to the above, procedures for the control of Plant Staff Overtime were not established, implemented, and maintained to limit work hours in accordance with TS 6.8.1.j. Specifically, sections 2.6.1 and 4.2.1 of procedure LS-AA-119, Overtime Controls, Rev. 4, permit the plant manager or designated manager to authorize deviations from the GL 82-12 work hour guidelines during refueling outage activities. Periodic refueling outages do not qualify as "very unusual circumstances" for which work hour deviations may be authorized. Consequently, during various time periods between October 15 and November 18, 2007, while in a refueling outage not qualifying as a "very unusual circumstance," the plant manager or designated manager authorized over 340 licensee employees (including reactor operators, senior reactor operators, auxiliary operators, engineers, work planners, health physicists, key maintenance personnel, and the emergency response organization members) and over 390 contractors to work up to 84 hours in a 7-day period to perform routine refueling outage support activities. The majority of these individuals worked more than 72 hours during a 7-day period. Many of these workers performed safety-related work and none of these workers were restricted from performing safety-related activities. Because this violation was of very low safety significance, was not repetitive or willful, and it was entered into the licensee's corrective action program (IR 713257), this violation is being treated as an NCV, consistent with section VI.A.1 of the NRC Enforcement Policy. **NCV 05000289/2007005-02, Deficient Control of Plant Staff Overtime**.

.2 Deficient Control of Reactor Coolant System Vent Area during Mid-Loop Operation

Introduction. A self-revealing Green NCV of TS 6.8.1.a was identified for failure to maintain appropriate RCS vent area required by station procedures during mid-loop operation. The reduced vent area degraded operators' ability to add water to the RCS in the event of a loss of DHR and caused reactor vessel level indication to be inaccurate.

Description. At 11:03 a.m. on November 13, 2007, operators recommenced filling the RCS, following termination of an Unusual Event (UE). The event had been declared earlier that morning based on inaccurate reactor vessel level indications which resulted from failure to maintain appropriate configuration control of the RCS vent area and ventilation flow path while in mid-loop operation (see Section 4OA3). The RCS was at mid-loop operation at 31" level (approximately 6 feet above top of fuel) with the reactor vessel head in place, but not tensioned. The plant shutdown risk condition was elevated (Orange), because the RCS would begin boiling in 35 minutes if the DHR system was lost.

The inspectors entered the reactor building containment to verify plant conditions, including system lineups, were properly maintained to support RCS refill as required by procedure 1103-11. This procedure requires a vent area of 28.8 square inches be maintained to prevent RCS pressurization during RCS gravity fill. The inspectors found radiological posting signs almost completely covering the two handhole openings. Only 6 square inches of area remained uncovered. Rope and metal support bars held the signs in place. The inspectors promptly informed operators of this discrepancy. Operators promptly stopped the RCS refill evolution and the radiological posting signs were repositioned, thereby restoring the required RCS vent area. Based on data from the plant process computer, the inspectors determined that reactor vessel level instrumentation had indicated slightly (2") higher than actual level during the RCS refill activity, due to the deficient vent area.

Based on interviews, the inspectors determined RP technicians had installed the postings shortly after operators removed the ventilation hoses, and were unaware of the importance of maintaining adequate RCS ventilation area. Communication between RP technicians and operators was ineffective. Similar to the cause of the UE earlier the same morning, configuration controls to maintain the minimum required RCS ventilation area were deficient. Immediate corrective actions included operator crew briefings, communications between RP and operations personnel, and initiation of IRs 698486 and 705000.

The inspectors determined station personnel's failure to maintain the minimum required RCS ventilation area per procedure 1103-11, while in mid-loop operation was a performance deficiency. Operators and RP technicians did not sufficiently communicate and understand the impact of their actions on mid-loop operation.

Analysis. This issue affected the configuration control attribute of the Mitigating Systems cornerstone and was more than minor because this configuration control error affected the availability of the BWST as a source to add water to the RCS in the event of a loss of decay heat removal and caused reactor vessel level indication to be inaccurate. The inspectors performed a Phase 1 SDP evaluation in accordance with IMC 0609, Appendix G, Attachment 1, Checklist 3, Pressurized Water Reactor Cold Shutdown and Refueling Operation with RCS Open and Refueling Cavity Level < 23'. The inspectors discussed the issue with station engineers, reviewed design calculations, and determined that although design margin was reduced, the RCS gravity feed and bleed from the BWST function remained operable. The inspectors also concluded that although the issue caused all four reactor vessel level indications to indicate higher than actual level, the level indication function remained sufficient to alert operators if a significant change occurred which would warrant operator actions. Therefore a Phase 2 quantitative assessment was not required and the issue screened to Green (very low safety significance).

The finding has a cross-cutting issue in the area of human performance, because work control during removal of the HEPA fans and ventilation hoses from the OTSG handholes was deficient. Operators and RP technicians did not appropriately coordinate work activities to ensure the required RCS vent area was maintained when technicians established radiological postings for the OTSG handhole area. Consequently there was an operational impact on core refill capability and reactor vessel level indication. [H.3.(b)].

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 1.22, Appendix A, specifies procedures for the operation of safety-related systems including the RCS. Procedure 1103-11, step 2.2.2 requires a minimum RCS vent area be maintained when reactor vessel water level is <50". Procedure HU-AA-104-101, Procedure Use and Adherence, Rev. 1, requires station personnel to follow procedures exactly as written. Contrary to the above requirement, on November 13, 2007, station personnel did not maintain the minimum required RCS vent area when posting radiological conditions after removing HEPA fans and ventilation hoses from the OTSG handhole vents. Because this violation was of very low safety significance and was entered into the TMI corrective action program (IRs 698486 and 705000), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: **NCV 05000289/2007005-03, Deficient Control of Reactor Coolant System Vent Area During Mid-Loop Operation.**

1R22 Surveillance Testing (71111.22)

a. Inspection Scope (2 IST Samples and 3 Routine Surveillance Samples)

The inspectors observed and/or reviewed the following operational surveillance tests to verify adequacy of the test to demonstrate the operability of the required system or component safety function. Inspection activities included review of previous surveillance history to identify previous problems and trends, observation of pre-evolution briefings, and initiation/resolution of related IRs for selected surveillances.

- On October 11, procedure 1303-11.3, Surveillance Test and Set Main Steam Safety Valves, Rev. 31
- On October 17, procedure OP-TM-211-203, IST of MU-V-14A/B and DH-V-7A/B, Rev. 0
- On October 18, procedure 1302-6.14, PORV and Code Safety D/P Monitors, Rev. 14
- On October 18, procedure OP-TM-211-211, High Pressure Injection Test, Rev. 4 and interim change 23913
- On November 6, procedure 1410-V-31, Crane Tilting Disc Check Valve Inspection, Rev. 29

b. <u>Findings</u>

<u>Deficient Procedure Causes Failure to Perform Surveillance Testing of Valves in Accordance with ASME OM Code</u>

<u>Introduction</u>: The inspectors identified that AmerGen failed to test multiple safety-related valves in accordance with ASME Code for the Operation and Maintenance of Nuclear Power Plants. MU-V-14A/B, MU-V-16A/B/C/D, MU-V-36, and MU-V-37 had not been observed locally to verify that remote position indication was accurately indicated as required by the ASME Code. This finding was determined to be of very low safety significance (Green) and was characterized as a Non Cited Violation of Technical Specification 4.2.2.

<u>Description</u>: On October 17, 2007 NRC inspectors observed the quarterly in-service testing (IST) of stroke time for MU-V-14A/B and DH-V-7A/B in accordance with TMI procedure OP-TM-211-203. The inspectors asked AmerGen personnel when and with what procedure the IST position verification testing would be performed for MU-V-14A/B. The inspectors were informed that the position verification testing was scheduled to be performed during the upcoming refueling outage and in accordance with procedure OP-TM-211-211.

Inspectors reviewed TMI procedure OP-TM-211-211 and noted that the acceptance criteria included the statement: “Local indication agreed with remote position indication for the following valves: MU-V-14A/B, MU-V-16A/B/C/D, and MU-V-36/37.” However, the inspectors identified that there were no procedural steps to verify and document local position indication for the subject valves. Inspectors interviewed AmerGen personnel and determined that since the procedure did not explicitly direct performance of the position indication verification and document the results, it can not be concluded that local position for the subject valves was ever verified. When interviewed, no operators could recall actually performing the local to remote valve position verification.

AmerGen personnel initiated IR 00688282 and implemented a procedure change before the scheduled performance of OP-TM-211-211. On October 26, 2007, OP-TM-211-211 was performed and MU-V-14A/B, MU-V-16A/B/C/D, MU-V-36, and MU-V-37 were tested satisfactorily.

Analysis: The inadequate procedure, which led to a failure to perform in-service testing in accordance with ASME Code requirements, was a performance deficiency.

The finding is more than minor because it affected the Procedure Quality aspect of the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The failure to test the safety-related valves in accordance with ASME Code requirements reduced the reliability of the valves to perform their safety-related function. The finding was evaluated using NRC Manual Chapter 0609, Appendix A, “Determining the Significance of Reactor Inspection Findings for At-Power Situations,” Phase 1, and was determined to be of very low significance (Green) since the condition did not involve an actual loss of safety function.

This finding has a cross-cutting aspect in the area of Human Performance, because AmerGen personnel failed to ensure that complete and accurate procedures were available for the testing of safety-related valves. (H.2(c))

Enforcement: Technical Specification 4.2.2 requires that, “IST of ASME Code Class 1, Class 2, and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(f).” ASME Code-2001, Section ISTC-3700 “Position Verification Testing” requires, “valves with remote position indicators shall be observed locally at least once every 2 years to verify that the valve operation is accurately indicated.”

Contrary to the above, local position observation of valves MU-V-14A/B, MU-V-16A/B/C/D, MU-V-36, and MU-V-37 was not performed. Because this violation was of very low safety significance and TMI entered this finding into their corrective action program (IR 688282), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: **05000289/2007005-04, Deficient Procedure Causes Failure to Perform Surveillance Testing of Valves in Accordance with ASME OM Code**.

1EP6 Drill Evaluation (71114.06 - 1 sample)

a. Inspection Scope

The inspectors performed one inspection sample. The inspectors observed an emergency event training evolution conducted on December 21, 2007, at the Unit 1 control room simulator to evaluate emergency procedure implementation, event classification, and event notification. The event scenario involved multiple safety-related component failures and plant conditions warranting a simulated Alert emergency event declaration. The inspectors observed the drill critique to determine whether the licensee critically evaluated drill performance to identify deficiencies and weaknesses. Minor deficiencies involving an incorrect event declaration were documented in IR 714876.

b. Findings

No findings of significance were identified.

## 2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Controls (71121.01 – 16 samples)

a. Inspection Scope

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

Inspection Planning - Performance Indicators

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

Plant Walkdowns, RWP Reviews, Job Reviews

The inspectors walked down selected radiological controlled areas and reviewed housekeeping, material conditions, posting, barricading, and access controls to radiological areas. The inspectors reviewed exposure significant work areas to determine if radiological controls were acceptable and conducted selective independent radiation surveys. The inspectors toured areas of the containment building, the auxiliary building, outdoor areas, and the radwaste shipping area.

The inspectors selectively reviewed the radiological controls for radiologically significant outage work activities including fuel movement and spent fuel pool work, steam generator tube examinations, pressurizer welding and inspection, scaffolding and insulation activities, reactor head inspections, and cleaning, inspection and modifications to the reactor building sump. The reviews included evaluation of the adequacy of all applied radiological controls including radiation work permits (RWP), procedure adherence, radiological surveys, job coverage, system breach surveys, airborne radioactivity sampling and controls, and contamination controls. The reviews included barrier integrity and engineering controls for potential airborne radioactivity areas and radioactive source term, and radiation levels present.

The inspectors discussed controls for radiation dose rate gradients (e.g., steam generator work activities), to verify that AmerGen had applied appropriate radiological controls including use of multiple dosimeters or repositioning of dosimetry to accurately measure radiation doses. The inspectors discussed radiological conditions with workers to evaluate worker knowledge of conditions and actions for electronic dosimetry alarms.

The inspectors reviewed applicable RWPs and electronic personnel dosimetry alarm setpoints (both integrated dose and dose rate) to verify that the set-points were commensurate with ambient/expected conditions, plant policy, and were appropriate for the conditions. In evaluating RWPs, the inspectors reviewed electronic dosimeter dose/dose rate alarm reports to determine if the set points were consistent with the survey indications and plant policy. The inspectors verified that workers were knowledgeable of the actions to be taken when a dosimeter alarms or malfunctions.

The inspectors observed various radiological controls briefings. The inspectors performed independent surveys of selected areas including the auxiliary, and containment buildings to confirm the accuracy of survey maps. During the tours, the inspectors verified the adequacy of radiological boundaries and postings, engineering controls were in place, air samplers were properly located, and that TS locked High Radiation Areas (HRAs) were properly secured and posted.

The inspectors reviewed and discussed inter-comparison of electronic dosimeter and thermoluminescent dosimeter results to identify anomalies and licensee actions. A review was performed of individuals that received > 100 mrem for the year for the past four calendar quarters including the outage.

The inspectors reviewed and discussed internal dose assessments for 2007 to identify any apparent actual occupational internal doses greater than 50 millirem committed effective dose equivalent (CEDE). The review also included the adequacy of evaluation of selected dose assessments, and included evaluation of potential intakes associated with hard-to-detect radionuclides (e.g., airborne transuranics). The inspectors reviewed 2007 whole body counter logs and data. The inspectors reviewed Personnel Contamination Reports (PCR) and whole body counting results to evaluate the assessment methods and adequacy.

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

<u>Problem Identification and Resolution</u>

The inspectors selectively reviewed self-assessments and audits to determine if identified problems were entered into the corrective action program for resolution. The inspectors evaluated the database for repetitive deficiencies or significant individual deficiencies to determine if self-assessment activities were identifying and addressing the deficiencies.

The review also included evaluation of data to determine if any problems involved Performance Indicator (PI) events with dose rates greater that 25 R/hr at 30 centimeters, greater than 500 R/hr at 1 meter or unintended exposures greater than 100 millirem total effective dose equivalent (TEDE), 5 rem shallow dose equivalent (SDE), or 1.5 rem lens dose equivalent (LDE). The inspectors also reviewed the corrective action database for non-PI radiological incidents to determine if follow-up activities were being conducted in an effective and timely manner consistent with radiological risk.

The review also included problem reports since the last inspection which involved potential radiation worker or radiation protection personnel errors to determine if there was an observable pattern traceable to a similar cause. The review included an evaluation of corrective actions, as appropriate. (Section 4OA2)

<u>High Risk Significant, High Dose Rate HRA and VHRA Controls</u>

The inspectors discussed procedure changes for HRA Access controls since the previous inspection, with the Radiation Protection Manager and selected supervisors to determine if changes resulted in a reduction in the effectiveness and level of worker protection.

The inspectors conducted a selective review of HRA controls (e.g., adequate posting and locking of entrances). The inspectors verified that locked HRAs were properly secured and posted and that surrounding area dose rates met regulatory criteria. The inspectors discussed controls for High and Very High Radiation Areas (VHRAs) with radiation protection technicians. The inspectors reviewed and observed controls used for ongoing work such as steam generator tube examinations and reactor head inspections and examinations. The inspectors reviewed the access key inventory for HRA and VHRA areas and conducted a key inventory. The inspectors verified procedure adherence by observing and questioning radiation protection personnel and workers.

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

The inspectors evaluated radiation protection technician performance and proficiency relative to control of hazards and work activities. In addition, the inspectors reviewed problem reports for worker or radiation protection technician performance. The inspectors questioned both radiation workers and radiation protection personnel regarding on-going activities and knowledge of controls and conditions. The inspectors reviewed condition reports to identify repetitive performance issues associated with workers or radiation protection personnel.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02 – 8 samples)

a. Inspection Scope

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

Inspection Planning, Radiological Work Planning

The inspectors reviewed pertinent information since the previous inspection regarding plant collective exposure history, current exposure trends, and ongoing and planned activities. The inspectors determined the plant's current 3-year rolling average collective exposure. The inspectors evaluated site specific trends in collective exposures (using NUREG-0713 and plant historical data). The inspectors discussed proposed occupational radiation exposure estimates for 2008.

The inspectors selected work activities likely to result in the highest personnel collective exposures and selectively reviewed the planning and preparation for those work activities. The inspectors evaluated the level of detail associated with projected dose estimation. The work activities reviewed included steam generator inspections, welding activities, reactor sump modifications, and scaffolding. The inspectors reviewed the integration and implementation of ALARA requirements into procedures and RWP documents. The inspectors compared results achieved with the intended dose established for the work tasks reviewed.

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

Job Site Inspections and ALARA Controls

The inspectors observed selected radiological controls briefing. The inspectors observed ongoing work activities (steam generator work, refueling work, welding, job coverage activities, sump work) to evaluate implementation of ALARA controls for the activities. The inspectors evaluated use of engineering controls to reduce dose (shielding, decontamination).

Verification of Dose Estimates and Exposure Tracking

The inspectors reviewed AmerGen's method for adjusting exposure estimates or replanning work when unexpected changes in scope, radiation levels, or emergent work were encountered, to determine if the adjustments were based on sound radiation protection and ALARA principles. The inspectors also reviewed the frequency of these adjustments to evaluate the original ALARA planning process.

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

Source-Term Reduction and Control

The inspectors reviewed and discussed AmerGen's understanding of the plant source-term, including knowledge of input mechanisms to reduce the source term and the source-term control strategy in place. The inspectors selectively reviewed and discussed AmerGen's cobalt reduction strategy designed to minimize the source-term external to the core. Also reviewed were fluid clean-up methods used to remove radioactivity. The inspectors reviewed reactor coolant chemistry data to evaluate the effectiveness of post shutdown source term reduction efforts, including strategies employed such as system flushes, installation of temporary shielding, and chemistry controls. The inspectors evaluated dose reduction results achieved against priorities since the last refueling cycle. The inspectors discussed the TMI five year source term reduction plan. The inspectors also reviewed selected Station ALARA Council Meeting Minutes for 2007.

Declared Pregnant Workers

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

Radiation Worker/Radiation Protection Technician Performance

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

Problem Identification and Resolution

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

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

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation and Protective Equipment (71121.03 – 4 samples)

a. Inspection Scope

The inspectors selectively reviewed radiation monitoring/measurement instrumentation. The review was against criteria contained in applicable TS and station procedures.

Inspection Planning/Identification of Additional Radiation Monitoring Equipment

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

Verification of Instrument Calibration, Operability, and Alarm Setpoint Verification

The inspectors selectively reviewed calibration and operability check records for a variety of radiological survey instrumentation in use for radiological job coverage and area monitoring. The instrumentation included portable survey meters, scaler-counters, and portable area radiation monitors. The inspectors evaluated the adequacy of calibration sources used relative to the in-plant source term. The following instruments were reviewed: Ludlum 375 (130427, 125348, 191878, 117673//125358); RM -G-9 (Unit 1 Fuel Pool Bridge).

The inspectors also reviewed calibration and field checks of instruments located in the field: (AMS-3 #930, AMS-3A #197, AMS-3A #254, AMS-3A #375, AMS-3A #789, AMS-3A #1008, AMS-4 #6087, Aptek w/DP11A #9812-076A, LoVol #1324, RM-14 #075961, RO20 #07112, RO2A #75826, RO20 #77704, RO20 #78108, Telepole #6607/023, Telepole #6607/036).

Problem Identification and Resolution

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

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

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

a. Inspection Scope

The inspectors selectively reviewed the packaging and shipment preparation of a non-exempt radioactive material shipment (RS-07-154), including packaging and vehicle radiation dose rates; placarding of vehicle; completion of applicable shipping papers; qualification of personnel overseeing and processing shipment; general truck and trailer condition; package shoring, closure and use requirements.

The inspectors also selectively reviewed training provided for station personnel relative to 49 CFR 172, and NRC Bulletin 79 -19.

b. Findings

No findings of significance were identified.

## 4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151 – 11 samples)

a. Inspection Scope

Cornerstone: Mitigating Systems (6 samples)

The inspectors reviewed the Performance Indicator (PI) assessment for safety system functional failures (SSFFs). The inspectors verified accuracy of the reported data through review of selected station operating logs, system health reports, SSFF databases, and Licensee Event Reports for the period October 2006 through September 2007. The inspectors also reviewed the licensee assessment of mitigating systems performance indicators (MSPIs). Verification included the review of selected definitions, data reporting elements, calculation methods, definition of terms, use of clarifying notes, Consolidated Data Entry MSPI Derivation Reports for unavailability and unreliability, monitored component demands, demand failure data, operator logs, maintenance rule database entries, and corrective action program documents for the period October 2006 through September 2007. Reviews were performed to determine whether associated PI data had been accurately reported to the NRC in accordance with NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Rev. 5. Additional documents reviewed are listed in the Attachment. The following PIs were evaluated:

- Safety System Functional Failures
- MSPI: High pressure safety injection system
- MSPI: Emergency feedwater system
- MSPI: Emergency AC power system
- MSPI: Residual heat removal system
- MSPI: Support cooling water system

Cornerstone: Occupational Exposure Control Effectiveness (1 sample)

The implementation of the Occupational Exposure Control Effectiveness PI Program was reviewed for the period October 2006 to September 2007. The inspectors reviewed corrective action program records for occurrences involving HRAs, VHRAs, and unplanned personnel radiation exposures since the last inspection in this area. The inspectors reviewed individual radiation exposure results and selectively reviewed exposure records and associated radiation work permits. The review was against the applicable criteria specified in NEI 99-02, Rev. 5. The purpose of this review was to verify that occurrences that met NEI criteria were recognized and identified as PI occurrences.

Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual (RETS/ODCM) – Radiological Effluent Occurrences (1 sample)

The implementation of the RETS/ODCM PI was reviewed. The inspectors reviewed corrective action program records and projected monthly and quarterly dose assessment results due to radioactive liquid and gaseous effluent releases for the previous four quarters (October 2006 to September 2007). The inspectors selectively reviewed the 2006 Annual Effluent Release Report. The inspectors also reviewed and discussed potential abnormal releases via groundwater or effluents. The review was against the applicable criteria specified in NEI 99-02, Rev. 5. The purpose of this review was to verify that occurrences that met NEI criteria were recognized and identified as PI occurrences.

Cornerstone: Physical Protection (3 samples)

Security PIs were inspected during the annual security baseline inspection and the documentation was inadvertently omitted from the security baseline inspection report issued on March 20, 2007. The inspectors performed a review of PI data submitted by the licensee for the Physical Protection Cornerstone. The review was conducted of the licensee's programs for gathering, processing, evaluating, and submitting data for the Fitness-for-Duty, Personnel Screening, and Protected Area Security Equipment PIs. The inspectors verified that the PIs had been properly reported as specified in NEI 99-02, Rev. 4. The review included the licensee's tracking and trending reports, personnel interviews and security event reports for the PI data for the period January to December 2006. The inspectors noted from the licensee's submittal that there were no reported failures to properly implement the requirements of 10 CFR 73 and 10 CFR 26 during the reporting period. This inspection activity represents the completion of three samples relative to this inspection area.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152 – 5 Samples)

.1 Review of Issue Reports and Cross-References to Problem Identification and Resolution Issues Reviewed Elsewhere

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 IRs, reviewing selected IRs, attending daily screening meetings, and accessing the licensee's computerized corrective action program database.

Section 1R15 documents an NCV for deficient corrective actions to ensure structural design clearances inside the reactor building containment were maintained. The inspectors determined that extent-of-condition reviews, for a previous NRC identified NCV concerning deficient containment structural clearances, were too narrowly focused.

Section 4OA3 documents two NCVs for deficient procedure implementation when moving fuel assemblies and control rod assemblies within the reactor core and the spent fuel pool, respectively. In each case, fuel handling operators failed to identify obstructions in the fuel and control rod assembly movement pathway and consequently damaged a fuel assembly and a control rod assembly.

Section 4OA7 documents an NCV for untimely identification that the plant had met entry conditions for declaring an Unusual Emergency event while the RCS was drained down to mid-loop.

.2 Semi-Annual Review to Identify Trends

a. Inspection Scope (1 sample)

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

b. Findings

No findings of significance were identified. The inspectors noted that corrective actions to address multiple degraded plant material conditions were effective. At the close of this inspection period, online maintenance corrective action backlogs were low (2 corrective maintenance and 144 elective maintenance) and only one adverse condition monitoring plan remained in effect. Overall, with a few exceptions, AmerGen planned and implemented the 1R17 refueling outage safely. Control room deficiencies, corrective action backlog, plant material deficiencies, and generic industry safety issues (i.e., containment sump blockage, fuel assembly guide tube elongation, and dissimilar metal RCS butt welds) were properly corrected or evaluated within the outage scope. Plant management decision-making demonstrated appropriate safety perspective and risk insights.

Notwithstanding good overall corrective action program (CAP) performance, the inspectors noted several emerging performance deficiency themes. Repetitive problems occurred and included plant equipment configuration control deficiencies, plant security control errors (access control, control of weapons, and security post manning), and Emergency Action Level declarations being untimely or based on erroneous criteria during training and actual events. Additionally, some event critiques or investigations did not identify important performance deficiencies (i.e. Biennial Emergency Preparedness drill critique, annual unannounced fire brigade drill critique, security access control issue investigation, deficient ALARA radiological practices for containment penetration replacement). Each individual performance deficiency was entered into the CAP.

.3 Annual Sample: Water Accumulated Under the ECCS Sump Lining (IR 692103)

a. Inspection Scope (1 Sample)

The inspectors reviewed IR 692103 that identified that approximately 20 gallons of water was found in the containment emergency core cooling system (ECCS) sump lining during a drilling evolution being conducted for installation of the new ECCS sump. The new ECCS sump was installed during this refueling outage as part of TMI's corrective actions for NRC GL 2004-02, Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors (PWRs). The inspectors reviewed AmerGen's response and evaluation of the condition to ensure that the full extent of the condition was identified, appropriate evaluations were performed, and appropriate corrective actions were specified and prioritized. The inspectors walked down the ECCS sump and interviewed the system engineer and operators. Applicable documents and safety evaluations were also reviewed. Based on water chemistry analysis (pH and radioactivity) and review of past history and design documents, engineers determined that the water found under the ECCS sump lining was not due to ground water leakage which would indicate a possible containment liner breach. The evaluation concluded the water accumulated between the ECCS sump lining and the concrete base that sets on top of the containment liner. The source of the water was residual water from previous outages due to leakage past the reactor cavity seal into the incore instrumentation room. The inspectors also verified that no water leaked past the reactor cavity into the incore instrumentation area during this refueling outage, and that engineers had developed enhanced long term actions to inspect and trend the containment liner for corrosion. Specific documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.4 Annual Sample: Nuclear Service Relief Valves Failed As Found Pressure Test

a. Inspection Scope (1 sample)

The inspectors reviewed IRs 636707, 638509, and 696649, which documented that several nuclear service (NS) system thermal relief valves (NS-V-37B, 38, and 211) failed the as-found lift bench test. Two valves (NS-V-37B and 38) also failed to re-seat after lifting. The inspectors reviewed AmerGen's response and evaluation of the condition to ensure that appropriate evaluations were performed, and appropriate corrective actions were specified and prioritized. The failed valves were replaced. The inspectors verified that adequate flow margin existed and that operability of the NS system was not affected by this condition.

b. Findings

No findings of significance were identified.

.5 Annual Sample: Pre-staging of Materials In Containment

a. Inspection Scope (1 Sample)

The inspectors reviewed AmerGen's corrective actions (IR 388006) to address a previously NRC identified violation (NCV 05000289/2005009-01) regarding deficient controls for loading materials inside the containment building while at power prior to the 1R-16 refueling outage. In addition, on September 6, 2007, the inspectors interviewed the system engineer and applicable engineering managers, operators, and the reactor building coordinator to understand the applicable corrective actions and the plans for loading materials in containment prior to the upcoming refueling outage 1R17.

b. Findings

No findings of significance were identified. The inspectors noted that AmerGen had implemented multiple corrective actions to enhance the controls of materials being loaded, including training, procedures, and assigning a new reactor building coordinator position for complete oversight. However, the inspectors identified that multiple minor deficiencies existed in the revised program that could impact the reliability of the containment sump and containment fire loading requirements. Specifically, TMI intended to load large quantities of scaffolding materials (some with unqualified paint/coatings), large quantities of electrical cables and extension cords, multiple temporary lights (including stringer lights), and multiple miscellaneous materials including rolls of tack cloths, rolls of duct tape, and rolls of nylon rope. Although some materials would have been loaded inside qualified metal boxes, most of the materials were to be piled in designated areas inside containment.

The inspectors were concerned that reliability of containment and ECCS systems components could be affected since AmerGen had not fully evaluated the aggregate effects of the materials in regards to 1) hydrogen generation, 2) debris generation and transport and reactor building ECCS sump blockage (chemical effect/sludge generation concerns), 3) fire loading due to piling materials in designated areas without assessing the effect on near-by safety components, and 4) high energy line break (HELB) zone of influence due to a postulated design basis accident and containment spray. An engineering evaluation was performed (IR 676002) and the licensee implemented significant changes to their pre-staging plans. Specifically, the licensee eliminated all aluminum scaffolding kick plates, unqualified coatings and painted components, changed designated lay-down areas away from HELB zones of influence and containment spray, and developed enhanced training and oversight to ensure these new guidelines were properly implemented.

.6 Annual Sample: Controls of Fuel Assemblies in the Spent Fuel Pool and Core Component Movement

a. Inspection Scope (1 Sample)

The inspectors previously identified concerns with licensee processes for controlling storage of spent fuel assemblies and other materials in the SFP, related to thermal management requirements. The issues were further discussed during a November 29, 2006, teleconference between AmerGen and NRC personnel. The inspectors reviewed AmerGen's responses to the IRs generated to ensure that the full extent of the condition was identified, appropriate evaluations were performed, and appropriate corrective actions were specified and prioritized. Additionally, the inspectors reviewed IRs which addressed SFP storage configuration challenges. The inspectors observed SFP material storage activities, verified selected SFP storage locations and contents, and interviewed relevant station personnel.

b. Findings

No findings of significance were identified. Process deficiencies and/or fuel storage concerns were properly documented in the IRs. Corrective actions included revision of the corporate procedures which established criteria for component storage within the SFP at all Exelon sites. Appropriate procedure revisions were implemented in a timely manner. AmerGen personnel properly addressed SFP Thermal Management Requirements throughout 2007, including the subject IRs.

.7 Problem Identification and Resolution for ISI/NDE Activities

a. Inspection Scope

The extent of oversight of In-service Inspection/Non Destructive Examination (ISI/NDE) activities including the topics of current quality assurance surveillance was reviewed. The inspectors reviewed a sample of IRs listed in the Attachment to confirm that identified problems were being documented for evaluation and proper resolution.

c. Findings

No findings of significance were identified.

.8 Problem Identification and Resolution for Radiation Safety

a. Inspection Scope

The inspectors selectively reviewed IRs and self-assessments to determine if identified problems were entered into the corrective action program for resolution. The inspectors selectively reviewed the reports to evaluate AmerGen's threshold for identifying, evaluating, and resolving problems (Self-Assessments: 560136-04, 560141-04; NOS Audit 07-2Q, 07-06). The review included a check of possible repetitive issues, such as worker or technician errors (IRs 643140, 627739, 664323, 665133, 665245, 666076, 666077, 666174, 692005, 692625, 691330, 692334, 692625, 691752, 693239, 705968, 701300, 706687, 665252).

b. Findings

No findings of significance were identified.

4OA3 Event Followup (71153 – 6 samples)

.1 Unexpected Reactor Vessel Level Indication Drop During Mid-loop Operation Leads to Declaration of Unusual Event

a. Inspection Scope

On November 13, 2007, at 6:32 a.m. operators declared an Unusual Event (UE) due to an unexpected indication of loss of RCS inventory while in mid-loop operating conditions. Operators received the RCS Drain-Down Level Low alarm and observed an instantaneous 3" drop in reactor vessel level indication, while maintenance personnel were reinstalling the OTSG primary side lower manway covers. The plant was in an elevated shutdown risk condition (Orange risk), because the RCS would begin boiling in a short period of time (26 minutes) if the DHR system was lost.

The inspectors were in the control room at the time of the event. The inspectors monitored control room indications and operator response to verify operators properly implemented associated procedures to maintain safe plant conditions and minimize the potential for radioactive release to the environment. NRC personnel entered the monitoring mode in the Region I Incident Response Center and remained in constant communications with the control room staff to assess the licensee's response to this event. The licensee terminated the Unusual Event at 9:29 a.m., following verification that there had been no actual loss of RCS inventory and that the plant was stable, and at 10:01 a.m, the NRC exited the Monitoring mode. Station personnel determined use of temporary fans to draw air out of the OTSG while the lower manways were installed caused a false low level indication. The inspectors conducted follow-up interviews, record reviews, and plant walkdowns to identify any related licensee performance deficiencies and verify the causes of this event were understood and corrected. Additional documents reviewed during this inspection are listed in the Attachment.

b. Findings

Deficient Control of Temporary Ventilation during Mid-Loop Operation Leads to Unusual Event

Introduction. A self-revealing Green NCV of TS 6.8.1.a was identified for failure to properly coordinate maintenance and operational activities associated with installing the

OTSG primary lower manways during mid-loop operation as required by station procedures. Installation of the OTSG lower manway cover while temporary ventilation fans were exhausting air from the OTSG handhole RCS vent, caused an unexpected drop in reactor vessel level indication and declaration of an Unusual Event emergency action level.

Description. The plant was in refueling mode on November 13, 2007, and RCS level indication was 13 inches (approximately 4 1/2 feet above top of active fuel), with the reactor vessel head in place but not tensioned. At 5:58 a.m., station personnel were in the process of reinstalling OTSG primary lower manway covers when control room operators received the RCS Drain-Down Low Level alarm. All four level reactor vessel level indications dropped to 10" and remained stable. Operators responded properly by reducing DHR flow to preclude the potential for pump suction vortexing and loss of DHR. No indications of actual loss of RCS inventory were visible in containment or evident from monitoring adjacent systems. At 6:32 a.m., the shift manager declared the UE based on Emergency Action Level entry condition MU-9, Unplanned Loss of RCS Inventory below Flange Level with Irradiated Fuel in the Reactor Vessel Lasting Greater than 15 Minutes. The declaration was recognized by the licensee to be untimely. The untimely EAL declaration is a separate licensee identified NCV and is discussed in Section 4OA7.

Radiological technicians secured the temporary HEPA filter ventilation fans on the OTSG primary handholes prior to exiting containment, upon declaration of the UE. Reactor coolant system level indication immediately returned to the 13" pre-event level, with no additional operator action following the HEPA filter ventilation fan shutdown. Station personnel determined that there had been no actual loss of RCS inventory. The changes in reactor vessel level indication were the consequences of changes to the temporary OTSG ventilation fans and ventilation pathways during the drain-down period. Upon installing the OTSG primary lower manway cover with the HEPA fan running, a slight negative pressure was created by the fan suction, causing reactor vessel level indication to drop 3". Upon securing the fan, pressure across the handhole vent equalized and RCS level returned to the accurate pre-event level.

The inspectors reviewed station procedures. Procedure 1103-11, step 2.3.2 notes that reactor vessel level indications provide accurate indication of water level when the overpressure is at atmospheric pressure. Procedure 1103-11, step 3.2.2.3.4 requires additional supervisory emphasis and evaluation to address coordination of maintenance and operational activities if reactor vessel will be drained to <50" with fuel in the reactor vessel. Procedure 1103-11, step 3.2.2.3.10 directs that the HEPA filter ventilation fans be restarted after achieving the desired mid-loop level, but does not address the possible effect on RCS overpressure and associated reactor vessel level instrument indication. The procedures for installation and removal of the OTSG primary lower manways require the Operations Director (or designee) to verify RCS conditions have been established to support removal or installation of the manway. The inspectors determined that maintenance activities (installation of the OTSG primary lower manway cover and operation of the temporary HEPA ventilation filter fans) and mid-loop operations were not properly coordinated to ensure reactor vessel level indication remained accurate and that changes were understood by the operating crew. Corrective actions included removing the HEPA ventilation filter fans and ventilation hose from the OTSG handholes, restricting the use of the fans, briefing the operating crews on this event, and initiating IRs 698291, 698693, and 699314.

The inspectors determined that station personnel's failure to properly coordinate maintenance and operations activities while in mid-loop operations as required by station procedures was a performance deficiency. Operations supervision authorized maintenance personnel to install the OTSG lower primary manway covers, while HEPA fans purged the RCS from the OTSG handholes, without properly assessing the impact on a critical operational parameter (RCS level indication). Consequently, reactor vessel level indication was inaccurate and not understood by operations personnel while the plant was in an elevated shutdown risk condition.

Analysis. This issue affected the configuration control attribute of the Mitigating Systems cornerstone and was more than minor because this configuration control error affected the accuracy of reactor vessel level instrument indication during mid-loop operations, a high risk evolution. The inspectors performed a Phase 1 SDP evaluation in accordance with IMC 0609, Appendix G, Attachment 1, Checklist 3, Pressurized Water Reactor Cold Shutdown and Refueling Operation with RCS Open and Refueling Cavity Level < 23'. The inspectors reviewed station drawings and records of reactor vessel level indication during the event. The inspectors determined that although all four reactor vessel level instruments were affected, their collective level indications, trends, and alarms provided sufficient information to alert operators in the event of an actual loss of inventory. Therefore, a Phase 2 quantitative assessment was not required and the issue screened to Green (very low safety significance).

The finding is also a cross-cutting issue in the area of human performance, because work control for installation and removal of temporary OTSG ventilation during installation of the OTSG lower primary manway cover was deficient. Operators and RP technicians did not appropriately coordinate work activities to ensure the operational impact on reactor vessel level indication while in mid-loop operation was understood [H.3(b)].

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 1.22, Appendix A, recommends procedures for the operation safety-related systems including the RCS. Procedure 1103-11, (1) requires a minimum RCS vent area be maintained when reactor vessel water level is <50", (2) states that reactor vessel level indications provide accurate indication of water level when the overpressure is at atmospheric pressure, (3) directs that the HEPA filter ventilation fans be restarted after achieving the desired mid-loop level, but does not address the possible effect on RCS overpressure and associated reactor vessel level instrument indication, and (4) requires additional supervisory emphasis and evaluation to address coordination of maintenance and operational activities if reactor vessel will be drained to <50" with fuel in the reactor vessel. The procedures for installation and removal of the OTSG primary lower manways require the Operations Director (or designee) to verify RCS conditions have been established to support removal or installation of the manway. Procedure HU-AA-104-101 requires station personnel to follow procedures exactly as written.

Contrary to the above, on November 13, 2007, station personnel did not maintain the minimum required RCS vent area when in mid-loop operation. Additionally, operations supervision was deficient, because maintenance activities (installation of the OTSG primary lower manway cover and operation of the temporary HEPA ventilation filter fans) and mid-loop operations were not properly coordinated to ensure reactor vessel level indication remained accurate and changes were understood by the operating crew. Because this violation was of very low safety significance and was entered into the TMI corrective action program (IRs 698291, 698693, and 699314), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: **NCV 05000289/2007005-05, Deficient Control of Temporary Ventilation During Mid-Loop Operation Leads to Unusual Event.**

.2 Damaged Control Rod Assembly Due to Deficient Fuel Handling Practices

a. Inspection Scope

On November 4, 2007, a CRA was damaged while operators attempted to move it from one fuel assembly (FA) to another FA stored within the SFP. The reactor vessel core had already been fully offloaded into the SFP as part of the planned refueling. Upon noticing the damaged CRA suspended from the refueling bridge control rod mast, the fuel handling operators notified outage control center management that the CRA was partially withdrawn from the FA, was bent, and was stuck. Fuel handling operations were halted, a recovery plan was developed, and a prompt investigation was initiated (IR 694289). Health physicists verified radiation levels in the SFP vicinity were normal, indicating that the fuel cladding barrier from the affected FA was not damaged. The inspectors conducted visual inspections, interviews, and document reviews to assess plant conditions, personnel and equipment performance associated with the damaged CRA, and licensee corrective actions.

c. Findings

Damaged Control Rod Assembly Due to Deficient Fuel Handling Practices

Introduction. A self-revealing Green NCV of TS 6.8.1.c was identified for failure to properly implement procedures to safely move FAs and CRAs within the SFP. Several procedure adherence errors and deficient verification practices resulted in a damaged CRA and had the potential to damage the affected FA cladding fission product barrier.

Description. On November 4, 2007, operators attempted to grapple a CRA from one FA for transfer to another FA. After two attempts, the grapple still did not indicate engaged. This CRA was slightly lower in the SFP than was typical, which can cause difficulty grappling the CRA. Operators had experienced this problem with several other CRAs during the previous week. The fuel handling supervisor (FHS) directed the fueling bridge operator to disengage the grapple, raise the control assembly grapple and mast, and prepare to move the fueling bridge to another location in the SFP in anticipation of relocating a different CRA. The bridge operator performed this action, without continuously monitoring the mast load cell as required by step 5.1.4 of procedure 1507-5, Spent Fuel Handling Bridge Operating Instructions, Rev. 34. Unrecognized by the bridge operator, the grapple had grasped the CRA and the CRA was partially withdrawn from the FA. The grapple did not indicate "grapple engaged" due to abnormal grappler clearances associated with the lower CRA pick-up location. The FHS transferred responsibility to an oncoming FHS without monitoring the bridge operator action.

Procedure 1505-1, Fuel and Control Component Shuffles, Rev. 48 requires visual verification that fuel movement can be accomplished. Step 5.2.9 of procedure 1507-5 requires that prior to moving the fuel bridge or trolley for any reason, operators ensure that nothing will be hit or run over. Fuel handling supervisors are trained to visually verify the end of the mast is clear of obstructions prior to moving the fuel bridge. Neither the bridge operator nor the FHS visually verified the mast was clear and the movement path was unobstructed prior to moving the fuel bridge to the next location. Upon reaching the next FA/CRA location, the on coming FHS observed that a bent and twisted CRA was extended between the mast and the previous FA location. The follow-up investigation determined procedures were not followed, verification practices were deficient, the FHSs did not communicate sufficiently to ensure the attempted CRA fuel handling move was complete prior to performing shift turnover, and that the oncoming FHS had no previous experience moving CRAs.

Station personnel developed two new procedures to support recovery of the damaged CRA. Operators were unable to reinsert the damaged CRA into its host FA, but were able to remove the CRA without further damaging the host FA. The CRA damage was extensive and the CRA could not be reused, and an alternate CRA was identified for reuse in the reactor core. Additional corrective actions included verifying proper CRA handling equipment operation, additional licensed Senior Reactor Operator oversight for all FA or CRA moves, briefing all refueling bridge personnel regarding lessons learned from this event and the importance of verifying critical actions, requiring additional spotters to confirm the fuel or control assembly mast clear of obstructions prior to horizontal movement, and various procedure revisions to strengthen verification requirements.

Failure to monitor the control mast load cell during CRA movement activities and to verify that the transit path (including the control mast) was clear of obstruction prior to bridge or trolley movement, as required by procedures, was a performance deficiency. The issue was more than minor because it affected the human performance attribute and the Barrier Integrity cornerstone objective to ensure the fuel cladding design barrier protects the public from radionuclide release.

Analysis. The finding was evaluated using NRC Manual Chapter 0609, Appendix G, Attachment 1 Checklist 4, Refueling Operation. The CRA was damaged and another CRA had to be selected for core reload. However, the inspectors determined the affected FA fuel clad barrier remained intact and containment controls were unaffected. Therefore, a Phase 2 quantitative assessment was not required and the issue screened to Green (very low safety significance). The finding is also a cross-cutting issue in the area of human performance, because fuel handling operations personnel did not follow procedure requirements for safely moving CRAs in the SFP [H.4(b)].

Enforcement. TS 6.8.1.c states, in part, that written procedures shall be properly established and implemented for refueling operation activities. Procedure 1507-5 requires the load cell to be monitored continuously during all operations when control assemblies are being raised or lowered. If indicated load on the load cell deviates from the acceptable range, then operations shall be stopped and the FHS notified. Additionally, prior to moving the fueling bridge or trolley for any reason, operators shall ensure that nothing will be hit or run over. Procedure 1505-1, Fuel and Control Component Shuffles, Rev. 48 requires visual verification that fuel movement can be accomplished. Contrary to the above, on November 4, 2007, fuel handling operators did not continuously monitor the mast load cell while attempting to grapple and raise a CRA. Fuel handling operations were not stopped when the load cell indicated a CRA loading problem. Additionally, neither the bridge operator nor the FHS verified the mast was clear and the movement path was unobstructed prior to moving the fuel bridge to the next location. Because this violation was of very low safety significance and was entered into the TMI corrective action program (IR 694289), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: **NCV 05000289/2007005-06, Damaged Control Rod Assembly Due to Deficient Fuel Handling Practices.**

.3 Damaged Fuel Assembly Due to Deficient Fuel Handling Practices

a. Inspection Scope

On November 7, 2007, a fuel assembly was damaged when operators reloaded it from the SFP to the reactor vessel. Fourteen of 177 FAs had previously been loaded into the reactor vessel core prior to this event. Operators had attempted unsuccessfully to insert the 15$^{th}$ FA for approximately 4 hours. The vertical “ZZ” tape indicated that the FA was not fully inserted. The FHS directed the FA to be withdrawn from the core to perform a visual inspection. An underwater camera inspection of the FA identified damage to the lower grid strap and one fuel pin. Fuel handling operations were promptly suspended and an event investigation initiated. There were no adverse radiological consequences from the event. The damaged FA was subsequently returned to the SFP and the cycle 18 core design was revised to accept an alternate FA in place of the damaged FA. The inspectors conducted visual inspections, interviews, and document reviews to assess plant conditions, personnel and equipment performance associated with the damaged FA, and licensee corrective actions.

c. Findings

Damaged Fuel Assembly Due to Deficient Fuel Handling Practices

Introduction. A self-revealing Green NCV of TS 6.8.1.c was identified for failure to properly establish and implement procedures to safely reload fuel into the reactor vessel core. Fuel handling operators inserted an FA without properly verifying the fuel movement could be safely accomplished. This damaged the FA, requiring FA replacement and core redesign for cycle 18 operation.

Description. Pyramid-shaped devices called shoehorns are used to guide a FA lower end fitting into the core during insertion for certain core locations. The shoehorn is moved to a location adjacent to the FA being lowered into the core using a cable suspended from the auxiliary bridge. The cable is attached to an eye-hook at the top of the shoehorn pyramid. After positioning the shoehorn, the cable must be slacked and laid down across the shoehorn face opposite from the FA insertion location to prevent cable interference with the FA insertion. The camera inspection revealed that the cable had inadvertently been slacked to the incoming FA side of the shoehorn. As the FA was lowered, a fuel pin and the lower grid strap became caught on the cable, thereby preventing full insertion into the core.

During the post-event critique, fuel handling personnel stated that reactor core water clarity, lighting, and visibility were good, with the exception of a partial shadow at the core location being loaded. The shadow resulted from light and camera placement in relation to a FA already loaded into an adjacent core location. The shadow impaired the view of the shoehorn and cable location from both the main and auxiliary fuel bridges. Despite the degraded view, fuel handling personnel did not halt fuel movements in order to relocate cameras and lights to improve visibility.

The inspectors observed that procedure 1505-1, Fuel and Control Component Shuffles, Interim Change 23978, requires visual verification that fuel movement can be accomplished. No procedures specifically address verifying that shoehorn cables were properly positioned prior to FA insertion. However, fuel handling operators are trained to lay the cable on the opposite face of the pyramid, facing away from the FA being lowered into the core. In this case, work practices were deficient in that refueling personnel proceeded without stopping to ensure proper visibility.

Corrective actions included core redesign to accept a replacement FA, a stand-down and event briefing for all refueling personnel, procedure revisions requiring independent underwater camera verification that shoehorn and other cables are clear of the FA insertion location, redesign of the shoehorn cables, a root cause evaluation of fuel handling errors during 1R17, and continuation of enhanced supervisory oversight which had begun on November 4 (see section 4OA3.2). Additionally, more cameras and viewing monitors were installed to further improve visibility during core reload. The inspectors directly monitored core reload activities following the event. Operators implemented lessons learned from this event and safely loaded the remaining 162 FAs in accordance with procedures.

Failure to visually verify a clear FA insertion path so that safe fuel movement could be accomplished, as required by procedures, was a performance deficiency. The issue was more than minor because it affected the human performance attribute and the Barrier Integrity cornerstone objective to ensure the fuel cladding design barrier protects the public from radionuclide release.

Analysis. The finding was evaluated using NRC Manual Chapter 0609, Appendix G, Attachment 1 Checklist 4, Refueling Operation. The FA was damaged and another FA was selected for core reload, requiring core redesign analysis. However, the inspectors determined the affected FA fuel clad barrier remained intact and that containment controls were unaffected. Therefore a Phase 2 quantitative assessment was not required and the issue screened to Green (very low safety significance). The finding is also a cross-cutting issue in the area of Human Performance, because fuel handling personnel proceeded in the face of uncertainty, without stopping to ensure proper visibility [H.4(a)].

Enforcement. TS 6.8.1.c states, in part, that written procedures shall be properly established and implemented for refueling operation activities. Procedure 1505-1 requires visual verification that fuel movement can be accomplished. Contrary to the above, on November 7, 2007, fuel handling operators did not visually verify that core location R7 was unobstructed so that the planned fuel movement could be safely accomplished. Consequently, FA NJ12MG was damaged when it became hung up on a shoehorn support cable. Because this violation was of very low safety significance and was entered into the TMI corrective action program (IRs 696075 and 697120), this violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy: **NCV 05000289/2007005-07, Damaged Fuel Assembly Due to Deficient Fuel Handling Practices.**

.4 <u>'A' Condenser Man-way Leak</u>

a. <u>Inspection Scope</u>

At 5:40 p.m. on November 30, 2007, a main condenser manway gasket failed, causing several hundred gallons per minute of circulating water to leak into the turbine building basement. Operators performed a rapid (3 percent per minute) power reduction from 100 to 40 percent power in accordance with procedure 1102-4, Power Operation, Rev. 114. Condenser vacuum remained stable during the event. Repairs required the lower power level to support isolation of portions of the circulating water system. The apparent cause of the gasket failure was a change in manufacturing process, which made the gaskets more sensitive to being cut when hold-down torque was applied during installation. The gasket replacement was considered a like-for-like replacement. The inspectors determined the difference in manufacturing process for this non-safety related gasket and the associated increased the likelihood of leakage when installed using the existing installation instructions. This was not reasonably within the licensee's ability to foresee and prevent. Therefore the inspectors concluded the failure to prevent the waterbox leak was not a performance deficiency warranting SDP review.

Procurement engineers obtained replacement gaskets of the original manufactured design for all 12 main condenser manways. The gaskets were replaced and the plant achieved 100 percent reactor power on December 1. The inspectors inspected equipment areas affected by the condenser leak and monitored the rapid power reduction from the control room. The inspectors monitored operator actions, maintenance repairs, and plant configuration during the event to verify the plant transient was managed safely and did not unnecessarily challenge plant safety systems with a plant trip initiating event.

b. <u>Findings</u>

No findings of significance were identified.

.5 <u>B12 Type Fuel Assembly Growth</u>

a. <u>Inspection Scope</u>

In April 2007, engineers identified greater than expected guide tube growth for twenty Mark B12 fuel assemblies located in the SFP. These assemblies had already been used in the core for two operating cycles. The measurements were taken to validate growth characteristics of zircaloy alloy M5 material, used in B12 fuel assembly guide tubes and fuel pin cladding, relative to exposure (burnup) in the core. Longitudinal growth exceeded the design limits approved for this fuel type. This led to a concern that guide tubes may become compressed against the fuel assembly upper end fitting during plant cooldown and cause permanent deformation or fuel damage in high burnup fuel assemblies, further affecting core shutdown margin or core coolable geometry. Operability Evaluation OPE-07-003, Mark B12 Fuel Assemblies, Rev. 2 concluded that the fuel assemblies and control rod assemblies in the core would remain operable through the end of operating cycle 16 (ended October 22, 2007). The inspectors also reviewed a technical evaluation (IR 620760-11) that concluded all B12 type fuel assemblies remained structurally capable of handling the loads associated with being moved using normal fuel handling equipment during 1R17 refueling activities. Station personnel developed an extensive fuel assembly inspection plan (measure assembly length, control rod assembly drag resistance, and visual inspections) to determine whether the B-12 fuel assemblies were safe for reuse during the next operating cycle. Contingency plans were prepared, including replacement of fuel assembly upper end fittings as necessary, to restore B12 fuel assembly design margins. Inspections identified excessive growth, excessive drag resistance, and broken upper end fitting leaf springs on some of the B12 fuel assemblies. Corrective actions included upper end fitting replacement on 92 B12 fuel assemblies to restore design clearance margins through the end of the next operating cycle. Additionally, two B12 fuel assemblies were removed from service due to excessive control rod binding.

The inspectors observed inspections and repairs, conducted interviews, and reviewed design documents, inspection data, measurement and repair techniques to verify reasonable assurance that all B12 fuel assemblies reloaded into the core for cycle 17 operation would remain within their approved design performance characteristics. The inspectors also verified that the issue would not prevent control rods from providing sufficient negative reactivity to shut down the reactor or would affect coolable core geometry. Additional documents reviewed during this inspection are listed in the Attachment.

b. Findings

No findings of significance were identified.

.6 (Closed) Licensee Event Report (LER) 05000289/2006-003-00, Automatic Reactor Trip Due to a Design Application Deficiency within the Reactor Coolant Pump Power Monitors Initiated by an Off-Site Grid Disturbance

On December 13, 2006, the reactor automatically tripped from 100 percent reactor power following a short duration power disturbance on a 230 KV transmission line located approximately 4 miles from TMI-1. The power disturbance was due to a single phase ground fault which resulted from an age-related failure of a cable splice in the 230 KV line. The ground fault affected power supplied to the reactor coolant pump motors. The reactor coolant pump power monitors (RCPPM) sensed inadequate power to the reactor coolant pumps and caused an unplanned reactor protection system actuation. Station personnel determined the cause of the reactor trip was a design application deficiency within the RCPPM, in that the design focused on fuel protection and specified a maximum time delay, but did not consider the effects of a grid transient of short duration. The event was previously documented in NRC Inspection Report 05000289/2006006. The inspectors determined the LER accurately described the event, and corrective actions have been implemented to address the cause. No findings of significance were identified. This LER is closed.

4OA5 Other Activities

.1 Review of World Association of Nuclear Operators Plant Assessment

The World Association of Nuclear Operators performed a TMI plant assessment during the period March 5-16, 2007. The final Institute of Nuclear Power Operations assessment report was issued in September 2007. The inspectors reviewed the interim and final plant assessment reports. Problems identified in the reports were consistent with NRC findings and no new safety issues were identified.

.2 Temporary Instruction 2515/166 – Pressurized Water Reactor Containment Sump Blockage

a. Inspection Scope

The inspectors performed an inspection in accordance with Temporary Instruction (TI) 2515/166. The TI was developed to support the NRC review of licensee activities in response to NRC Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors." Specifically, the inspectors verified the implementation of the modifications and procedure changes were consistent with the proposed actions committed to in the GL response. The inspectors reviewed a sample of the licensing and design documents to verify that they were either updated or in the process of being updated to reflect the effects of the modifications and the new requirements for the containment sump and debris generation sources. This included a sample of material specifications, testing and surveillance procedures, and calculations. The inspectors performed a walkdown of the strainer installation to verify it was performed in accordance with the approved design change package. Finally, the inspectors verified that there were no choke-points not accounted for by the licensee's calculations that could prevent water from reaching the recirculation sump during a design basis accident. Additional documents reviewed during this inspection are listed in the Attachment.

b. Evaluation of Inspection Requirements

The TI requires the inspectors to evaluate and answer the following questions:

1. Did the licensee implement the plant modifications and procedure changes committed to in their GL 2004-02 response?

   The inspectors verified that actions implemented by the licensee as described in response to GL 2004-02 were complete as they related to the installation of the sump screen and evaluation of potential debris sources. Additionally, the inspectors found that procedures to programmatically control potential debris generation sources were updated. The inspectors noted that the sump surface area that was installed had a smaller surface area than was discussed in the GL response; however, updated calculations supported the smaller size. AmerGen updated the Three Mile Island GL 2004-02 response to reflect these changes on December 28, 2007. The inspectors noted that AmerGen had not completed the downstream effects evaluation or the effects of chemical precipitants on the strainer head loss at the time of the inspection.

2. Has the licensee updated its licensing basis to reflect the corrective actions taken in response to GL 2004-02?

   The inspectors verified that changes to the facility or procedures as described in the UFSAR that were identified in the licensee's GL 2004-02 response were reviewed and documented in accordance with 10 CFR 50.59. Finally, the inspectors verified that AmerGen intends to update the Three Mile Island Unit 1 UFSAR and TS bases to reflect the final modification and associated procedure changes made in response to GL 2004-02.

The TI will remain open to allow for the review of portions of the GL response that have not been completed. Specifically, AmerGen had not completed their downstream effects analysis or chemical precipitant analysis. The results of these analyses have the potential to impact the final size of the strainer, licensing basis and programmatic procedures.

c. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On January 22, 2008, the resident inspectors presented the inspection results to Mr. Rusty West and other members of the TMI staff who acknowledged the findings. Dr. Ronald Bellamy, Chief, NRC Region I Reactor Projects Branch 6 attended the exit meeting. The regional specialist inspection results were previously presented to members of AmerGen management. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee Identified Violations

The following violations of very low safety significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

- TS 6.8.1.f requires written procedures be established, implemented, and maintained covering Emergency Plan Implementation. AmerGen procedure EP-AA-111 states, “Once indication of an abnormal condition is available, classification declaration must be made within 15 minutes.” Contrary to this requirement, on November 13, 2007 at 5:58 a.m., control room operators observed an abrupt drop of RCS level by 3” during mid-loop operation. At 6:32 a.m. the Shift Manager declared an Unusual Event for an unplanned loss of inventory in the RCS with irradiated fuel in the core. The Unusual Event declaration was untimely by 19 minutes. The licensee recognized that the declaration was untimely shortly after the event declaration was made and promptly informed the NRC. This issue was placed in AmerGen’s corrective action program as IR 698693. This finding was evaluated using NRC Manual Chapter 0609, Appendix B, “Emergency Preparedness Significance Determination Process,” Sheet 2, and was determined to be of very low significance (Green) due to the issue being an actual event implementation problem of a Notice of Unusual Event.

- TS 6.8.1.a requires written procedures to be established, implemented, and maintained covering applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Rev. 2, February 1978. Regulatory Guide 1.33, Appendix A, recommends procedures for safe operation and shutdown of safety-related systems, including the emergency feedwater (EFW) system. Section 4.8 of AmerGen procedure OP-TM-424-101 requires the handwheels for the steam supply valves to the turbine driven emergency feedwater (TDEFW) pump EF-P-1 (MS-V-13A and B) to be in the "Full Out" (open) position. Contrary to this requirement, on November 16, 2007, station operators identified that both steam supply valves MS-V-13A and B were blocked closed, rendering the TDEFW pump inoperable for approximately 38 hours. This issue was placed in AmerGen's corrective action program as IR 700788. This finding was evaluated using NRC Manual Chapter 0609, "Significance Determination Process," Appendix A, Phase 1. The finding was determined to be of very low significance (Green) since the TDEFW pump was inoperable for less than the 72 hour allowed time period specified in the limiting condition for operation in TS Section 3.4.1.1.a(2), and at least two independent motor driven pumps powered from redundant emergency diesel trains remained operable.

ATTACHMENT: SUPPLEMENTAL INFORMATION

## SUPPLEMENTAL INFORMATION

## KEY POINTS OF CONTACT

Licensee Personnel

| | |
|---|---|
| C. Baker | Manager, Chemistry |
| B. Carsky | Director, Operations |
| T. Dougherty | Plant Manager |
| E. Eilola | Director, Site Engineering |
| R. Godwin | Training |
| J. Heischman | Director, maintenance |
| A. Miller | Regulatory Assurance |
| D. Mohre | Manager, Security |
| D. Neff | Manager, Emergency Preparedness |
| T. Roberts | Radiation Protection |
| C. Smith | Manager, Regulatory Assurance |
| D. Trostle | Operations Security Analyst |
| L. Weir | Manager, Nuclear Oversight Services |
| C. Wend | Manager, Radiation Protection |
| R. West | Vice President, TMI Unit 1 |
| H. Yeldell | Work Management |

## LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened & Closed

| | | |
|---|---|---|
| 05000289/2007005-01 | NCV | Ineffective Corrective Actions For Failure to Maintain Structural Design Clearances Inside Reactor Building Containment (Section 1R15) |
| 05000289/2007005-02 | NCV | Deficient Control of Plant Staff Overtime (Section 1R20) |
| 05000289/2007005-03 | NCV | Deficient Control of Reactor Coolant System Vent Area During Mid-Loop Operation (Section 1R20) |
| 05000289/2007005-04 | NCV | Deficient Procedure Causes Failure to Perform Surveillance Testing of Valves in Accordance with ASME OM Code (Section 1R22) |
| 05000289/2007005-05 | NCV | Deficient Control of Temporary Ventilation During Mid-Loop Operation Leads to Unusual Event (4OA3.1) |
| 05000289/2007005-06 | NCV | Damaged Control Rod Assembly Due to Deficient Fuel Handling Practices (Section 4OA3.2) |
| 05000289/2007005-07 | NCV | Damaged Fuel Assembly Due to Deficient Fuel Handling Practices (Section 4OA3.3) |

Closed

| | | |
|---|---|---|
| 005000289/2006-003-00 | LER | Automatic Reactor Trip Due to a Design Application Deficiency within the Reactor Coolant Pump Power Monitors Initiated by an Off-Site Grid Disturbance (Section 4OA3.6) |

Discussed

| | |
|---|---|
| Temporary Instruction 2515/166 | Pressurized Water Reactor Containment Sump Blockage (Section 4OA5.2) |

## LIST OF DOCUMENTS REVIEWED

**Section 1R04: Equipment Alignment**
OP-TM-212-101, Shifting DHR Trains A and B from ES Standby to DHR Standby, Rev. 3
OP-TM-212-112, Shifting DHR Train B from DHR Standby to DHR Operating Mode, Rev. 5
OP-TM-642-231, ES Train A Emergency Sequence and Power Transfer Test, Rev. 10
Procedure 1103-11, RCS Water Level Control, Rev. 64

**Section 1R05: Fire Protection**
Procedures:
OP-AA-201-003, Fire Drill Performance, Rev. 9
OP-TM-201-501-1001, Fire Brigade Response, Rev. 1
OP-TM-AOP-001, Fire, Rev. 1
OP-TM-AOP-001-C3A, Fire in 1D ES 4160V Switchgear Room, Rev. 1
TQ-AA-127, Fire Brigade Training Program, Rev. 4
Alarm response procedure HVB 3-10, CB 338' D 4160V Room Damper Trouble: Fire-Smoke, Rev. 15

Other Documents:
Fire Drill Scenario No TMI-1/008, TMI-1 Control building 338' Elevation, 1D 4160V Switchgear Room – Electrical Fire, Rev. 0
Maintenance Department Watch List Fire Brigade Assignments dated November 29, 2007
TMI-1 Fire Pre-Plan Strategies and Smoke Removal Plan for Zone CB-FA-3A, Control Building Elevation 338'6" 4160V Switchgear 1D Room

**Section 1R08: Inservice Inspection**
Issue Reports
00523976
00607832
00690585
00691875
00692325
00694554
AR 552813-02,"Boron crystals on Make-Up MU-V-180, System 211"
AR 668290,"Boric acid deposit on underside of bolted flange of turbine flowmeter"

NDE Examination Test Reports
14132, "Radiographic Inspection Report, Weld R2, Pressurizer Relief Nozzle"
14133, "Radiographic Inspection Report, Weld R3, Pressurizer Relief Nozzle"
ISI-071, "Liquid Penetrant Exam, RPV CRDM Adapter to Flange Welds"

ISI-059/60, “UT Calibration/Examination Data Sheet for weld Main Steam (MS) 0254, 45 and 60 degree angle”
ISI-064&65, “UT Calibration/Examination Data Sheet for weld MS 0252 45 degree angle”
ISI-053&54, “Magnetic Particle Examination Data Sheet for weld FW-038 and 039, Main Steam”
PR-021-NDE-A20-00, “Liquid Penetrant Exam, RCS, 10” Surge Nozzle SE Weld”
17R BACC IR, “17R Inspection Report, BACC Lead Identification from Walkdowns”

NDE Procedures and Programs
54-ISI-835-11, “Ultrasonic Examination of Ferritic Piping Welds (Includes PDI-UT-1)”
54-ISI-829-08, “Manual UT of Dissimilar Metal Piping Welds”, Rev. 8/7/07
ER-AA-335-003 R3, “Magnetic Particle Examination” Rev 3
ER-AA-335-002 R4, “Liquid Penetrant Examination” Rev 4
54-PT-200 R7, ”Color Contrast Solvent Removable liquid Penetrant Examination” Rev 7
54-5033662 R1, “Magnetic Particle Test” Rev 1
ER-AA-335-1008 R1, ”Code Acceptance and Recording Criteria for NDE Surface Examination” Rev 1
ER-AP-335-1012 R3, “Bare metal Visual Examination of PWR Vessel Penetrations and Nozzle Safe Ends” Rev 3
ER-AP-331 R3, “Boric Acid Corrosion Control (BACC) Program” Rev 3
ER-AP-331-1001 R2, “Boric Acid Corrosion Control (BACC) Inspection Locations, Implementation and Inspection Guidelines” Rev 2
ER-AP-331-1002 R3, “Boric Acid Corrosion Control Program Identification, Screening, and Evaluation” Rev 3
ER-TM-335-1005 R5, “Steam Generator Eddy Current Data Analysis Guidelines for TMI Unit 1” Rev 5
ER-TM-335-1006 R2, “OTSG Site Specific Performance Demonstration and Training Program” Rev 2
ER-AP-420 R6, “Steam Generator Management Program” Rev 6
54-ISI-400-15, “ETSS#1 TMI1R17 Examination Technique Specification Sheet (Bobbin)” Rev 0
54-ISI-400-15, “ETSS#2 TMI1R17 Examination Technique Specification Sheet (Rotate)” Rev 0
Steam Generator Degradation Assessment, Rev. 0, for TMI Unit 1, Outage R16
Personnel and Equipment Qualification and Calibration Certifications (Eddy Current Exam)
ETSS#1-TMI 1R17 Examination Technique Specification Sheet, 54-ISI-400-15, Bobbin Standard ASME Code Examination for Unsleeved Parent Tubing
ETSS#2-TMI 1R17 Examination Technique Specification Sheet, 54-ISI-400-15, Rotating Probe (.115/+point/.080HF) Kinetic Expansion, Lane & Wedge, Dent/Ding, Crevice Region Lower Tube Ends and SI from Bobbin
Condition Monitoring and Operational Assessment evaluation of Steam Generator Tubing at TMI, Unit 1 for RFO 16
TMI Steam Generator Tube Inspection Degradation Assessment for Outage R17
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TMI1-0012 R1, “IWE Component Rollout Inside Containment Liner 180-0 AZ” Rev 1
IE-153-02-003, “General Arrangement-Reactor Building, Elevation 331’0” “ Rev 0

Welding Procedures
55-WP1/8/43/F43OLTBSCa3-002, “Machine Gas Tungsten Arc Welding (Overlay) of P1/P8/P43 Materials”
55-WP1/8/F6Ca3-000, “Machine Gas Tungsten Arc Welding of P1/P8 Materials”

Alloy 600, Weld Materials 82/182 and MRP-139 Related Items
ECR TM 07-00369 001 Mitigate Alloy 600 at Pressurizer Surge Nozzle and DH Drop Line
ECR TM 05-00286 006 Mitigate Alloy 600 Upper Pressurizer Nozzles
Process Traveler #50-9052838, TMI Unit 1, Pressurizer Surge Nozzle Weld Overlay
Process Traveler #50-5062814-005, TMI Unit 1, Pressurizer Relief Flange
50.59 Review, Applicability Review and Screening of ECR 05-00286 Alloy 600 Mitigation
50.59 Review and Screening of ECR 07-00369, Pressurizer surge line and decay heat drop line
Letter 5928-07-20187, TMI to NRC, Pressurizer Dissimilar Metal Weld Overlays, Date 8/13/07
ECR TM-07-00369-001, Rev. 01. Mitigate Alloy 600-Pressurizer Surge Nozzle and Decay Heat Line
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Work Orders WO# C2013993, Task 15; C2013972, Task 18; and C2012883, Task 04 (for UT)

Miscellaneous
Focused Area Self Assessment Report, 2006 TMI BACC Program
Calculation no. DC-536910-00014-01-SE, Rev. 00. RB Liner Corrosion, dated 4/6/00
Design Analysis No. 0-1101-153-E410-041, Rev. 00. Minimum allowable Thickness for the Reactor Building Liner Plate, dated 11/9/07
EC/ECR No. AR A214529/ECR 07-00878
EN-MA-501 R4 Controlled Materials and Hazard Communication Program
T1R17 Oversight (for NDE) Log ER-AA-335-025

**Section 1R15: Operability Evaluations**
Other Documents:
TMI-1 TS 3.5.2, Control Rod group and Power Distribution Limits
TMI-1 TS 4.7.1, Control rod Drive System Functional Tests
Engineering document 51-9065949-000, Operability Assessment of TMI-1 Mark B-12 Fuel Assemblies in Cycle 17
Engineering document 51-9066143-000, TMI-1 EOC 16 and BOC 17 Control Rod Drag Assessment
TMI-1 UFSAR 14.1.2.7, Stuck-Out, Stuck-In, Or Dropped Control Rod Accident

**Section 1R20: Refueling and Other Outage Activities**
Procedures:
1101-3, Containment Integrity and Access Limits, Rev. 85
1102-1, Plant Heatup to 525 Degrees F, Rev. 166
1102-2, Plant Startup, Rev. 146
1102-4, Power Operation, Rev. 114
1102-11, Plant Cooldown, Rev. 137
1103-11, RCS Water Level Control, Rev. 64
1303-11.4, Refueling System Interlocks Tests, Rev. 44
1504-9, Closure Head Gasket Replacement, Rev. 21
1505-1, Fuel and Control Component Shuffles, interim change 23978
1505-3, Fuel Handling Problems, Rev. 19
1507-3, Main Fuel Handling Bridge Operating Instructions, interim change 23993
1507-5, Spent Fuel Handling Bridge Operating Instructions, interim change 23919
1507-12, Visual Inspections, Rev. 16
OP-AA-108-108, Unit Restart Review, Rev. 8,
OP-TM-213-203, Core Flood Train ‘A’ Flow Test, Rev. 3
OP-TM-213-204, Core Flood Train ‘B’ Flow Test, Rev. 3

OP-TM-642-231, Engineered Safeguards train ‘A’ Emergency Sequence and Power Transfer Test, Rev. 1
OP-TM-EOP-030, Loss of Decay Heat Removal, Rev. 2
MA-TM-134-903, Reactor Vessel Disassembly, Rev. 5
MA-TM-134-904, Reactor Vessel Reassembly, Rev. 3
MA-AA-716-008, Foreign Material Exclusion Program, Rev. 3
MA-AA-716-010-1008, Reactor Services Refuel Floor FME Plan, Rev. 0
MA-AA-716-022, Control of Heavy Loads Program, Rev. 2
E-14A, Periodic Reactor Building Polar Crane Inspection, Rev. 12

Other Documents:
NRC Inspection and Enforcement Bulletin 80-12, Decay Heat Removal System Operability
GPU Nuclear Calculation C-1101-212-5360-021, Steam Venting Through the CRD’s Openings During RCS boiling Due to Loss of DHR, Rev. 0
GPU Nuclear Calculation C-1101-212-5360-020, BWST Gravity Feed During Loss of DHR, Rev. 0
NRC Bulletin 96-02: Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment.
NRC Regulatory Issue Summary 2005-25: Clarification of NRC Guidelines for Control of Heavy Loads.
NRC Regulatory Issue Summary 2005-25 Supplement 1: Clarification of NRC Guidelines for Control of Heavy Loads.
Evaluation of Heavy Load Handling Operations at TMI-1; Volume II – Reactor Building

**Sections 2OS1, 2OS2 and 2OS3**
Procedures:
RP-AA-220, Bioassay Program
RP-AA-250, External Dose Assessment from Contamination
RP-AA350, Personnel Contamination Monitoring, Decontamination and Reporting
RP-AA-376, Radiological Postings
RP-AA-401, Operational ALARA Planning and Controls
RP-AA-460, High Radiation Area Controls
RP-AA-460-1001, Additional Controls for work in >1500 mrem/hr Fields
RP-TM-401-1002, Three Mile Island Outage ALARA Planning and Controls
RP-AA-403, Administration of the Radiation Work Permit Program
RP-TM-300-1006, Monitoring of Protective Clothing
6610-IMP-3282.01, Installation of Temporary Shielding
6610-ADM-4246.01, Operation, Calibration & Quality Assurance of the Canberra Whole Body Counting System

Documents:
TMI Radiological Protection T1R16 Refueling Outage Report 2005
TMI T1R17 Elevated Dose Rate Contingency Action Plan
2007 Temporary Shielding Log
RP-AA-220, Attachment 3, Annual Bioassay Program Review, Dated 9/26/07
Three Mile Island Fastscan Whole Body Counter Calibration Report, Dated 1/23/06
TMI RCS Loop Dose Rate Averages
1R17 Average D-Ring Dose Rates
1R16 to 1R17 Shutdown RCS SRMP Dose Rate Comparison
TMI 1R17 October 2007 Forced Oxidation Co-58 Cleanup
Plant source term analysis data

Various radiation monitor calibration and operability check data
Various radiological survey records for completed work activities including records
Various radiation work permits for completed work activities and associated ALARA plans.
Various personnel whole body count data results
Radiological Controls Contamination Logs
Five Year Source Term Reduction Plan
Various 2007 Station ALARA Committee Meeting Minutes
DOT Training Modules

**Section 4OA1: Performance Indicator Verification**
Procedures:
ER-AA-2020, Equipment Performance and Information Exchange (EPIX) and Mitigating Systems Performance Index (MSPI) Failure Determination Evaluation, Rev 3
1302-5.31C, 4160V 1D Bus Loss of Voltage/Degraded Grid Timing Relay Calibration and Logic Check, Rev 15
ER-AA-1047, Mitigating Systems Performance Index Basis Document, Rev 1
TMI-2006-004, MSPI Basis Document, Rev1

Other Documents:
Event Number 43498
LER 2006-002-00
LER 2006-003-00

Issue Reports:

| | | | | | |
|---|---|---|---|---|---|
| 544634 | 544751 | 578700 | 584916 | 618773 | 632990 |
| 633130 | 633208 | 673584 | | | |

**Section 4OA2.3: Identification and Resolution of Problems**
Drawings:
521006, Reactor Building Steel Liner Bottom Plate Anchors, Rev. 2
521030, Reactor Building Steel Liner, Rev. 10

Issue Reports:
629604
660989
678392

Other Documents:
TMI-1/FSAR Section 5.0, Containment Systems And Other Special features
ECR TM 07-00479, Reactor Building Approved Caulking Material, Outside D-Rings
ASME Section XI, Article IWE-2000, Examination And Inspection

**Section 4OA3: Event Followup**
Procedures:
1102-10, Plant Shutdown, Rev. 94
1303-11.4, Refueling System Interlock Tests, Rev. 44
1401-4.4, OTSG Manway Covers Removal / Replacement, Rev. 39
1401-4.4A, Remove/Install OTSG Primary Upper Manway, Rev. 1
1401-4.4B, Remove/Install OTSG Primary Lower Manway, Rev. 1
1401-4.4F, Remove/Install Secondary Manway and Handhole Covers, Rev. 1
EP-AA-1009, Radiological Emergency Plan Annex for TMI Station, Rev. 9

OP-TM-212-000, Attachment 7.2, Minimum Height of Water Required to Avoid Vortex Formation vs. Decay Heat Flow, Rev. 10

Other Documents:
NSRB presentation TMI Fuel Growth Recovery Initiative, dated July 25, 2007
Engineering document 12-9065768-001, Causal Analysis for Mark B12 Broken Hold Down Springs – TMI End of Cycle 16 Outage
Engineering document 32-9047380-001, RMI-1 Cycle 17 Hydraulic Lift Analysis
PORC Meeting Minutes dated November 13, 2007
Engineering document 32-9051083-000, Fuel Damage Assessment of Mark B12 Fuel with Measured High Growth Rates for TMI-1 Cycle 16
Engineering document 32-9056966-01, TMI-16 Mark-B12 Upper End Fitting Handling FEA
Engineering document 51-9065949-001, Operability Assessment of TMI-1 Mark B-12 Fuel Assemblies in Cycle 17
Engineering document 51-9066143-000, TMI-1 EOC 16 and BOC 17 Control Rod Drag Assessment
TMI-1 Cycle 16 Final Core Loading Plan
TMI-1 T1R17 Fuel Action Plan
TMI-1 UFSAR 14.1.2.9, Steam Line Break
Work order 2081705, Install Lower Primary Manway Cover

IRs

| | | | | | |
|---|---|---|---|---|---|
| 620760 | 651242 | 669026 | 690825 | 691781 | 692610 |
| 694362 | 694656 | 694965 | 695417 | 695440 | 698076 |
| 698984 | | | | | |

**Section 4OA5: Other**

Calculations:
ALION-CAL-EXEL2737-001, TMI-1 Reactor Building LOCA Debris Generation Calculation, Rev. 1
ALION-CAL-EXEL2737-002, TMI-1 Reactor Building LOCA Debris Transport Calculation, Rev. 1
C-1101-572-5320-005, RB Sump Minimum Level Setpoints, Rev. 2
C-1101-210-E610-011, LPI and BS Pump NPSH Margin Available from the RB Sump Following a LBLOCA, Rev. 6

Condition Reports (*indicates CR resulting from this inspection)

| | | | | | |
|---|---|---|---|---|---|
| 452286 | 689744 | 691255 | 694296 | 695985 | 697370* |
| 687751 | 690290 | 691288 | 694497 | 695986 | 697381* |
| 688644 | 690416 | 691698 | 695209 | 697335 | |
| 689736 | 690834 | 692103 | 695617 | 697329* | |
| 689743 | 690883 | 693764 | 694702 | 697352* | |

Procedures:
1101-3, Containment Integrity and Access Limits, Rev. 85
1104-40, Plant Sump and Drainage System, Rev. 51
1105-11, Auxiliary Instrumentation and Control Systems, Rev. 37
1105-21, Main Annunciator Panel Beta Control System, Rev. 12
1301-4.1, Weekly Surveillance Checks, Rev. 83
1302-5.25, Reactor Building Sump Level, Rev. 21
1302-5.25A, Calibration of WDL-LT-804 RB normal Sump Level Channel, Rev. IC-23909

1302-5.25D, Calibration of DH-LT-801 RB ECCS Sump Level Channel, Rev. 0
1302-5.25E, Calibration of DH-LT-811 RB ECCS Sump Level Channel, Rev. 0
CC-AA-102, Design Input and Configuration Impact Screening, Rev. 14
OP-TM-212-571, Draining the RB ECCS Sump, Rev. 0
OP-TM-EOP-30, Loss of Decay Heat Removal, Rev. 2
OP-TM-MAP-B0207, RB Sump Level HiLo, Rev. 4
OP-TM-PRF1-0404, RB ECCS Sump Lvl Hi, Rev. 0
OP-TM-PRF1-0405, RB Sump level Hi, Rev. 2
OP-TM-PRF1-0406, RB Flood Level Hi, Rev. 2
OS-24, Conduct of Operations During Abnormal and Emergency Events, Rev. 15

Drawings:
EXLNTM009-C101, Reactor Building Sump Strainer 83" Top Hat Assembly, Rev. 2
EXLNTM009-C102, Reactor Building Sump Strainer 48: Top Hat Assembly, Rev. 2
EXLNTM009-C103, TMI Strainer – Unit 1 Filter Element, Rev. 2
EXLNTM009-C201, Reactor Building Sump Strainer 3D Model 7 notes, Rev. 1
EXLNTM009-C202, Reactor Building Sump Strainer Frame Assembly Plan, Section, & Details, Rev. 1
EXLNTM009-C203, Reactor Building Sump Strainer Frame Assembly Sections & Details, Rev. 1
EXLNTM009-C204, Reactor Building Sump Strainer Frame Assembly Sections & Details, Rev. 1
EXLNTM009-C205, Reactor building Sump Strainer Frame Assembly Plan, Sections, & Details, Rev. 1
EXLNTM009-C206, Reactor Building Sump Strainer Typical Sealing Details, Rev. 1
EXLNTM009-C301, Reactor Building Sump Strainer Trash Rack General Notes, Rev. 2
EXLNTM009-C302, Reactor Building Sump Strainer Trash Rack Section & Details, Rev. 2
EXLNTM009-C303, Reactor Building Sump Strainer Trash Rack Section & Details, Rev. 2
EXLNTM009-C304, Reactor Building Sump Strainer Trash Rack Section & Details, Rev. 2
EXLNTM009-C305, Reactor Building Sump Strainer Trash Rack Stairs, Rev. 2
EXLNTM009-C306, Reactor Building Sump Strainer Trash Rack Grating Location & Details, Rev. 2
EXLNTM009-C-401, Reactor Building Normal Sump General Notes, Rev. 3
EXLNTM009-C-402, Reactor Building Normal Sump Plan & Sections, Rev. 2
EXLNTM009-C-403, Reactor Building Normal Sump Sections & Details, Rev. 2
EXLNTM009-C-404, Reactor Building Normal Sump Sections & Details, Rev. 0
EXLNTM009-C-501, Reactor Building – Sump Strainer Temporary Screen, Rev. 0

Miscellaneous:
06-00205, Engineering Change Request: Reactor Building Sump Modifications Installation of Modified Strainers and Trash Rack, Rev. 1
06-00205, 50.59 Screen: Reactor Building Sump Modifications, Rev. 0
06-00206, 50.59 Screen: RB Sump level Instrumentation Modifications, Rev. 3
06-00256, Replace DH-V-19A/B Internals, Rev. 0
11.2.01.550, Training: 1R17 Outage Modifications, Rev. 0
5928-05-20076, Response to Request for Additional Information Regarding NRC Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," dated 3/7/05
5928-05-20196, Exelon/AmerGen Response to NRC General Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors," dated 7/27/05

5928-05-20249, Exelon/AmerGen Response to NRC Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors,” Dated 9/1/05
ECT# TM-06-00206, RB Sump Level Instrumentation, Rev. 3
ER-TM-TSC-0010, TMI-1 Severe Accident Management Guidelines, Rev. 1
EXP-NMP 508.02-2, Task Specific Job Hazard Analysis Work Sheet, Rev. 5
RG 1.97, Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions during and following and Accident, Rev. 3
TQ-TM-104-240, Containment Systems, Rev. 1

## LIST OF ACRONYMS

| | |
|---|---|
| ACM | Adverse Condition Monitoring |
| ADAMS | Agencywide Documents and Management System |
| ALARA | As Low As is Reasonably Achievable |
| AmerGen | AmerGen Energy Company, LLC |
| AR | Action Request |
| ASME | American Society of Mechanical Engineers |
| ASME OM | American Society of Mechanical Engineers Operations & Maintenance |
| BACC | Boric Acid Control |
| BTP | Branch Technical Positions |
| BWST | Borated Water Storage Tank |
| CAP | Corrective Action Program |
| CFR | Code of Federal Regulations |
| CEDE | Committed Effective Dose Equivalent |
| CRA | Control Rod Assembly |
| CRD | Control Rod Drive |
| CRDM | Control Rod Drive Mechanism |
| DHR | Decay Heat Removal |
| DOT | Department of Transportation |
| DRP | Division of Reactor Projects |
| ECCS | Emergency Core Cooling System |
| ECR | Engineering Change Request |
| ECT | Eddy Current Testing |
| EDG | Emergency Diesel Generator |
| EFW | Emergency Feedwater |
| EPRI | Electronic Power Research Institute |
| ESAS | Engineered Safeguards and Actuation System |
| FA | Fuel Assembly |
| FHS | Fuel Handling Supervisor |
| GL | Generic Letter |
| HELB | High Energy Line Break |
| HEPA | High Efficiency Particulate Air |
| HP | Health Physics |
| HRA | High Radiation Area |
| IMC | Inspection Manual Chapter |
| INPO | Institute of Nuclear Power Operations |
| IR | Issue Report |
| ISI | Inservice Inspection |
| IST | Inservice Testing |
| LDE | Lens Dose Equivalent |
| LER | Licensee Event Report |
| LHRA | Locked High Radiation Area |
| LOCA | Loss of Coolant Accident |
| LPI | Low Pressure Injection |
| MIC | Microbiologically Induced Corrosion |
| MR | Maintenance Rule |
| MS | Main Steam |
| MSPI | Mitigating System Performance Indicator |
| MT | Magnetic Particle Testing |

| | |
|---|---|
| NCV | Non-Cited Violation |
| NDE | Nondestructive Examination |
| NEI | Nuclear Energy Institute |
| NRC | Nuclear Regulatory Commission |
| NS | Nuclear Service |
| NUMARC | Nuclear Management and Resources Council |
| OTSG | Once Through Steam Generator |
| OWA | Operator Work-Around |
| PADEP | Pennsylvania Department of Environmental Protection |
| PARS | Publicly Available Records |
| PCP | Process Control Program |
| PCR | Personnel Contamination Reports |
| PI | Performance Indicator |
| PI&R | Problem Identification and Resolution |
| PM | Preventative Maintenance |
| PMT | Post-Maintenance Test |
| PPC | Plant Process Computer |
| psig | pounds per square inch |
| PT | Penetrant Testing |
| PWR | Pressurized Water Reactor |
| RCPPM | Reactor Coolant Pump Power Monitor |
| RCS | Reactor Coolant System |
| RETS/ODCM | Radiological Effluent Technical Specifications/Offsite Dose Calculation Manual |
| RP | Radiation Protection |
| RPV | Reactor Pressure Vessel |
| RWP | Radiation Work Permit |
| Rx | Reactor |
| SCBA | Self Contained Breathing Apparatus |
| SDE | Shallow Dose Equivalent |
| SDP | Significance Determination Process |
| SFP | Spent Fuel Pool |
| SSC | Structures, Systems, and Components |
| SSFF | Safety System Functional Failures |
| TDEFW | Turbine Driven Emergency Feedwater |
| TEDE | Total Effective Dose Equivalent |
| TI | Temporary Instruction |
| TM | Temporary Modification |
| TMI | Three Mile Island, Unit 1 |
| TS | Technical Specifications |
| UE | Unusual Event |
| UFSAR | Updated Final Safety Analysis Report |
| UT | Ultrasonic Testing |
| VHRA | Very High Radiation Area |
| VT | Visual Testing |
| 1R16 | Unit One Refueling Outage Number 16 |
| 1R17 | Unit One Refueling Outage Number 17 |