December 12, 2001

Mr. Mark E. Warner
Vice President, TMI Unit 1
AmerGen Energy Company, LLC
Three Mile Island Nuclear Station
PO Box 480
Middletown, PA 17057-0480

SUBJECT: THREE MILE ISLAND STATION, UNIT 1-NRC INSPECTION REPORT 50-289/01-07

Dear Mr. Warner:

On November 10, 2001, the NRC completed an inspection at your Three Mile Island Unit 1 facility. The enclosed report documents the inspection findings which were discussed on November 14, 2001, with Mr. George Gellrich and other members of your staff.

This 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.

Based on the results of this inspection, the inspectors identified three issues of very low safety significance (Green). These issues were determined to each involve a violation of NRC requirements. However, because of the very low safety significance and because the problems have been entered into your corrective action process, the NRC is treating these issues as non-cited violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these non-cited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Three Mile Island Unit 1 facility.

Since September 11, 2001, Three Mile Island Station Unit 1 has assumed a heightened level of security based on a series of threat advisories issued by the NRC. Although the NRC is not aware of any specific threat against nuclear facilities, the heightened level of security was recommended for all nuclear power plants and is being maintained due to the uncertainty about the possibility of additional terrorist attacks. The steps recommended by the NRC include increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with local law enforcement and military authorities, and limited access of personnel and vehicles to the site.

The NRC continues to interact with the Intelligence Community and to communicate information to AmerGen Energy Company, LLC. In addition, the NRC has monitored maintenance and other activities which could relate to the site's security posture.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm.html (the Public Electronic Reading Room).

Sincerely,

/RA/

John F. Rogge, Chief
Projects Branch 7
Division of Reactor Projects

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

Enclosure: NRC Inspection Report 50-289/01-07
Attachments: A - Circumferential Cracking of RPV Head Penetration Nozzles Reporting Requirements
B - Supplemental Information

cc w/encl:
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REGION 1

Docket No:	50-289
License No:	DPR-50
Report No:	50-289/01-07
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Three Mile Island Station, Unit 1
Location:	PO Box 480 Middletown, PA 17057
Dates:	September 30-November 10, 2001
Inspectors:	J. Daniel Orr, Senior Resident Inspector Craig W. Smith, Resident Inspector Thomas F. Burns, Reactor Inspector, DRS Joseph E. Carrasco, Reactor Inspector, DRS E. Harold Gray, Senior Reactor Inspector, DRS Jason C. Jang, Senior Health Physicist, DRS Ronald L. Nimitz, Senior Health Physicist, DRS David M. Silk, Senior Emergency Preparedness Inspector, DRS
Approved by:	John F. Rogge, Chief Projects Branch 7 Division of Reactor Projects

SUMMARY OF FINDINGS

IR 05000289-01-07, on 9/30 - 11/10/2001, AmerGen Energy Company, LLC, Three Mile Island Unit 1, integrated resident inspector report, personnel performance during non-routine plant evolutions, post-maintenance testing, surveillance testing, event follow-up.

The inspection was conducted by resident inspectors, two senior health physicists, and three region-based reactor inspectors. The inspection identified three Green findings, which were classified as non-cited violations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609 "Significance Determination Process" (SDP). Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation.

A. Inspector Identified Findings

Cornerstone: Initiating Events

- **Green**. Control room operators did not properly follow plant operating procedures for a reactor coolant system cooldown and draindown to a mid-loop condition. The procedure errors resulted in exceeding the reactor coolant system temperature limit and prolonging the time spent in the higher risk mid-loop condition. The same plant operating procedures were inadequate and did not establish steps to positively control the pressurizer cooldown rate at all times. Consequently, the pressurizer cooldown rate technical specification limit was nearly exceeded. The procedure problems increased the risk for a loss of reactor coolant system inventory control while the plant was drained down in the mid-loop condition.

  The safety significance of this finding was very low (Green) because redundant safety measures were not affected and remained in place to prevent an inadvertent loss of reactor coolant system inventory control. Technical specification 6.8, "Procedures and Programs," requires that written procedures be established, implemented, and maintained to control refueling operations. The control room operators' failure to follow operating procedure 1103-11, "Reactor Coolant System Inventory Control," was a violation of technical specification 6.8, "Procedures and Programs." (Section 1R14.1)

Cornerstone: Mitigating Systems

- **Green**. AmerGen failed to take adequate corrective actions to ensure the bearing housing cover bolts on the 'B' emergency feedwater (EFW) motor driven pump were properly installed following maintenance. In February 2001, system engineers found loosening of the cover bolts to be the root cause for an extended period of pump inoperability. Adequate corrective actions were not established to ensure the cover bolts were properly tightened following corrective maintenance activities.

  The safety significance of this finding was very low (Green) because AmerGen took immediate corrective actions to ensure proper cover bolt installation prior to returning the pump to service. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action," requires that for significant conditions adverse to quality corrective action shall be taken to preclude repetition. AmerGen's failure to assure the EFW motor driven pump outboard bearing cover housing was properly reassembled was a violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action." (Section 1R19.1)

- **Green**. Maintenance and test supervisors failed to properly implement surveillance procedure 1303-11.3, "Main Steam Safety Valves." The procedure error resulted in the setpoints for two safety valves being left outside the tolerance prescribed by the test procedure.

  The safety significance of this finding was very low (Green), because AmerGen took immediate corrective action to retest the two valves. Technical specification 6.8, "Procedures and Programs," requires that written procedures shall be established, implemented and maintained covering surveillance and test activities of equipment that affects nuclear safety. The supervisors' failure to implement the main steam safety valve test procedure as written was a violation of technical specification 6.8, "Procedures and Programs." (Section 1R22.1)

Cornerstone: Emergency Preparedness

- **To Be Determined**. Licensee sirens in Lancaster County were inoperable October 5 through October 9, 2001, due to the radio transmitter being de-energized at the county facility. The transmitter is part of the siren actuation system. This issue is unresolved pending further investigation into the lines of ownership and maintenance of the actuation system. (Section 4OA3.3)

B. Licensee Identified Violations

- No violations were identified.

Report Details

Summary of Plant Status

AmerGen Energy Company, LLC (AmerGen) operated Three Mile Island, Unit 1 (TMI) at reduced power due to end-of-cycle coastdown effects until the unit was shut down October 9, 2001, for the Cycle 14 refueling outage (14R). Major outage activities included: refueling of the reactor core, vessel head penetration inspections and repairs, once-through steam generator inspections and repairs, main and auxiliary transformer replacements, and plant process computer replacement.

1 REACTOR SAFETY

Initiating Events/Mitigating Systems/Barrier Integrity [REACTOR - R]

R04 Equipment Alignment

a. Inspection Scope

The inspectors conducted a partial system walkdown on the decay heat removal system while the reactor coolant system (RCS) was maintained in a mid-loop configuration. The mid-loop configuration was a relatively higher shutdown risk configuration because the RCS was breached and limited alternate means of decay heat removal were available. The inspectors also conducted a partial system walkdown on the spent fuel pool cooling system after a full core off load was performed. The spent fuel pool cooling system was at times more vulnerable while only one of two fuel pool cooling pumps was available due to planned electrical bus outages. The inspectors verified the system alignments were in accordance with operating procedures 1104-4, “Decay Heat Removal System” and 1104-6, “Spent Fuel Cooling System,” and that operating parameters were consistent with the plant operating condition.

b. Findings

No findings of significance were identified.

R05 Fire Protection

a. Inspection Scope

The inspectors conducted fire protection inspections in the decay heat removal pump vaults during core reload and in the reactor building. The rooms and areas were selected based on enclosing equipment important to shutdown risk management in the case of the decay heat removal pump vaults and the reactor building being inaccessible for routine inspector activities while the plant is operating. The inspectors conducted plant walkdowns and verified the areas were as described in the fire hazard analysis report. The plant walkdowns included observations of combustible material control, fire detection and suppression equipment operability, and compensatory measures established for degraded fire protection equipment. The inspectors also observed portions of fire damper surveillance testing for the relay room in the control tower building.

b. Findings

No findings of significance were identified.

R07 Heat Sink Performance

a. Inspection Scope

The inspectors observed data collection for ‘A’ decay heat closed cooling heat exchanger performance monitoring. The inspectors verified that the performance of the ‘A’ decay heat closed cooling heat exchanger was accurately evaluated. The inspectors also observed as found conditions of the ‘A’ decay heat closed cooling heat exchanger and nuclear services closed cooling water heat exchangers after the heat exchangers were opened for outage activities. Both decay heat closed cooling and nuclear services closed cooling heat exchangers perform heat removal functions on systems important to preventing core damage.

Findings

b. No findings of significance were identified.

R08 Inservice Inspection Activities

a. Inspection Scope

The inspected areas included reactor coolant system (RCS) head penetration piping, reactor pressure vessel (RPV) nozzle to piping welds, the 10-year RPV ultrasonic testing examination, eddy current testing of steam generator tubes, corrective actions and significant non-code repairs.

The inspector reviewed AmerGen’s program, procedures and eddy current test (ECT) activities for monitoring degradation of once-through steam generator (OTSG) tubes, including selected licensee commitments in response to Generic Letters 95-03, 97-05 and 97-06, the TMI OTSG degradation assessment report for the current refuel outage (14R) and the inspection summary report of OTSG inspection performed during the previous refuel outage (13R) in 1999. This OTSG information included an evaluation of primary to secondary leakage, a comparison of the size and number of tube flaws and degradation mechanisms identified during the current outage against the previous outage and confirmation that tube ECT scope and expansion criteria met technical specification (TS) requirements. In addition, the inspector reviewed AmerGen’s in-situ pressure testing screening criteria, tube selection for pressure testing and test results of tubes tested during 13R for compliance with Electric Power Research Institute In-Situ Pressure Test Guidelines.

The inspector confirmed that OTSG tube repairs were performed when required by the established repair criteria limits including the criteria of “plug on detection” for indications where depth sizing techniques are unavailable. The tube plugging limit was verified to be in accordance with the TS. The inspector confirmed that ECT procedures being used were qualified in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Sections V and XI, 1989 Edition, No Addenda and 1995 Edition with 1996 Addenda including site specific techniques. Also, the inspector verified that data analyst personnel were certified in accordance with

American National Standards Institute/American Society for Nondestructive Testing (ANSI/ASNT) CP-189, 1991 edition and had successfully completed site specific training and examination. The inspector verified that the licensee was identifying OTSG inservice inspection (ISI) problems at an appropriate threshold and entering them in the corrective action program.

ECT identified damage to four tubes surrounding one separated but previously plugged tube. This condition was being evaluated by the TMI staff for significance and corrective actions. An NRC special inspection, 50-289/2001-012, was initiated in response to this degraded condition.

The inspector reviewed portions of the procedures for underwater in-vessel visual inspection (VT) of the RPV and computer based ultrasonic testing (UT) examination of the RPV and attached piping welds. The inspector also interviewed UT personnel and observed a hands on demonstration of the model representing the computer controlled remote manipulator and examined the transducers to be used in the UT scanning of the reactor The personnel qualification and certification of the individuals engaged in the UT examination of the reactor vessel and associated piping welds were reviewed. The inspector verified that performance demonstration initiative qualification rules and the requirements of ASME Code Section XI were met.

The inspector sampled ISI packages representing volumetric and surface examinations to verify that these examinations were performed in accordance with the pertinent approved procedures and followed the requirements established by the ASME Code Section XI. The inspector reviewed the resolution of bound spherical bearings identified on main steam piping snubber MS0291 that was found by visual examination during the 14R outage. The related corrective action process (CAP) document (No. T2001-0819) was reviewed.

The inspector reviewed Reactor Building IWE ASME Section XI Visual Examination of the containment liner performed during the 13R outage and the subsequent repairs performed on the liner in six areas and the moisture barrier affected areas. The team, as a follow-up reviewed the 14R VT activities to ensure the adequacy of the corrective action and the adherence to the requirements established for ASME Code Section XI repairs.

The inspector also reviewed a non-code repair performed on a six inch diameter, fire service piping weld FS-205 that is located in the screen house. The piping system was repaired in accordance with ANSI B31.7 Code and the post repair non-destructive examination was magnetic particle (MT) examined following MT guidelines of an approved procedure.

The inspector reviewed a sample of radiographs (RTs) performed as a pre-outage activity to determine if the RT technique, evaluation and documentation were appropriate to the application. The RTs were performed on orifice flow restrictors located in the make-up and letdown piping system. This RT examination was performed to ensure that the orifices were free from obstruction.

AmerGen’s activities performed in response to NRC Bulletin 2001-01, “Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles,” were inspected against the requirements of temporary instruction (TI) 2515/145. The description of the inspection scope and results is in section 4OA5 as specified by the TI.

The inspector verified the licensee had identified ISI problems at an appropriate threshold and entered them into the corrective action program for disposition. The type and scope of the corrective actions for a sample of several ISI related condition reports were reviewed.

b. Findings

No findings of significance were identified.

R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors verified AmerGen's implementation of the maintenance rule for an emergent repair to the 'A' motor driven emergency feedwater pump (EF-P-2A) that occurred on September 22, 2001. EF-P-2A was declared inoperable after a pump packing adjustment resulted in increased and excessive packing leakoff spraying onto the pump outboard bearing. Emergency feedwater has the fifth highest system importance to the TMI total core damage frequency. The aspects of maintenance rule implementation inspected included safety significance classification, a(2) performance monitoring or a(1) goal setting and corrective actions, and maintenance preventable functional failure determinations. The inspectors referenced: 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants;" NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Plants;" and AmerGen administrative procedure 1082, "NRC Maintenance Rule."

b. Findings

No findings of significance were identified.

R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed AmerGen's shutdown risk management for the following risk significant activities:

- Reactor coolant system mid-loop operation
- 1A Engineered-safeguards motor control center bus outage
- Containment closure contingencies during various plant configurations

The inspectors reviewed the risk assessment of these maintenance activities with respect to 10 CFR 50.65(a)(4). The inspectors reviewed AmerGen Topical Report 097, "TMI-1 Outage Fuel Protection Criteria," Revision 5 to assure that all aspects of shutdown risk management were considered and implemented.

b. Findings

No findings of significance were identified.

R14 Personnel Performance During Non-routine Plant Evolutions

.1 Procedure Errors during Reactor Coolant System Cooldown and Mid-loop Operation

a. Inspection Scope

The inspectors observed main control room operators perform an RCS cooldown and draindown to a mid-loop condition. The inspectors reviewed operating procedures, evolution plans, contingency plans and observed crew briefings. The inspectors observed higher risk portions of the evolutions.

b. Findings

The inspectors identified three procedure problems during the conduct of the cooldown and draindown evolutions. The problems increased the risk for a loss of RCS inventory control while the plant was drained down in the mid-loop condition. The safety significance of this finding was very low (Green) because redundant safety measures were not affected and remained in place to prevent an inadvertent loss of RCS inventory control. The three procedure problems were a violation of TS 6.8, "Procedures and Programs," and are being treated as a single non-cited violation.

The periods of reduced RCS inventory and mid-loop usually present the greatest risk during shutdown. The procedure errors occurred at the beginning of the refueling outage with high decay heat load. Control room operators calculated the time to boil following termination of forced decay heat removal flow to be 4 minutes, with the core becoming uncovered in less than 60 minutes, if decay heat flow could not be restored. These calculations assumed an expected RCS heatup rate based on the elapsed time since the reactor was shutdown and were conservative estimates.

The procedure problems are summarized below:

- Operators failed to control RCS temperature below the 140°F limit required by operating procedure 1103-11, "Reactor Coolant System Water Level Control." Prior to commencing the draindown, operators purposely reduced the decay heat removal system flow rate to provide additional margin against pump vortexing during transition to the mid-loop condition. Because of the reduced flow rate, the decay heat removal system capacity was reduced and the RCS temperature rose above 140°F to a new equilibrium valve at 159° F. The inspectors found that prior to beginning the draindown evolution, operators were aware that the 140°F limit would be exceeded, but made a conscious decision to not follow the procedure. An evaluation had been performed indicating reactor coolant temperatures as high as 165°F were anticipated, but the procedure was not changed to reflect a new upper temperature limit. Further, operating crews trained on the draindown evolution using the plant simulator and believed the procedure provided sufficient latitude to exceed 140°F for the planned evolution. Exceeding the procedure temperature limit reduced the margin to boiling and increased the risk for loss of RCS inventory control event at the higher risk mid-loop condition.

- Operators did not accurately follow operating procedure 1103-11 and failed to open the reactor coolant pump seal vent valves prior to draining down to mid-loop. Consequently, the operators had to maintain the plant at the mid-loop condition for an extended period of time until the valves were properly positioned. The delay in opening the vent valves prolonged the time at the higher risk mid-loop condition increasing the likelihood for loss of RCS inventory control event at the higher risk mid-loop condition.
- Operating procedure 1103-11 did not provide adequate guidance to control room operators to ensure the TS required pressurizer cooldown rate of 100°F per hour was not exceeded. After initiating decay heat removal system flow and cooling down the RCS to less than 140°F, operators were directed by the operating procedure to depressurize the system to atmosphere by opening the pressurizer vent valves. Opening the vent valves resulted in an insurge of relatively colder RCS inventory into the pressurizer. The operating procedure did not specify a limit on the differential temperature between the RCS and the pressurizer prior to commencing the depressurization to limit the pressurizer cooldown rate. After opening the pressurizer vent valves operators observed a rapid increase in pressurizer cooldown rate and immediately shut the valves. The maximum pressurizer cooldown rate was later calculated at 96°F over one hour, but occurred within the first five minutes after opening the vent valves. Failure to control the cooldown rate nearly exceeded TS limits established to protect the integrity of the RCS.

The procedure problems had a credible impact on safety. The inspectors evaluated the risk significance of this finding using the NRC's significance determination process for shutdown operations. The inspectors determined that although procedure problems resulted in an increased likelihood of a loss of RCS inventory control, the finding did not meet the threshold for consideration as an actual loss of inventory control. Therefore, the finding did not require a quantitative risk assessment and was screened as being of very low safety significance (Green). The initiating events cornerstone was applicable because each of the procedure problems increased the risk for a loss of RCS inventory control. The inspectors also determined that the procedure problems constituted a violation of TS 6.8, "Procedures and Programs," which requires written procedures be established, implemented, and maintained to control refueling operations. However, because of the very low safety significance of the finding and because AmerGen entered the items into its CAP (numbers 00078654, 00078657, and 00078494), these procedure problems are being treated as a non-cited violation (**NCV 50-289/01-07-01**).

.2 Other Non-routine Plant Evolutions

a. Inspection Scope

The inspectors observed from the main control room the reactor shutdown, RCS inventory control at various vessel levels consistent for maintenance activities, and core offload and reload. The inspectors also observed portions of the core offload and reload from the spent fuel pool area and the reactor building refuel floor.

b. Findings

No findings of significance were identified.

R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed operability evaluations for the following degraded equipment issues affecting risk significant systems or components:

- Nuclear service river water pump 'A' inservice testing results in the required action range for differential pressure
- Fuel rod bowing from operating cycle irradiation
- Pressurizer cooldown rate above TS basis assumed step change value

The inspectors verified the degraded conditions were properly characterized, the operability of the affected systems was properly justified, and no unrecognized increase in plant risk resulted from the equipment issues.

b. Findings

No findings of significance were identified.

R19 Post-Maintenance Testing

.1 Inadequate Corrective Actions for Emergency Feedwater Pump Maintenance

a. Inspection Scope

The inspectors reviewed post-maintenance tests performed by AmerGen following a 'B' emergency feedwater (EFW) motor driven pump outboard bearing replacement. The inspectors reviewed work activities performed in conjunction with the bearing replacement to verify the post-maintenance test procedures were adequate and assured operability prior to returning the pump to service.

b. Findings

In reviewing the work activities performed to replace the ‘B’ EFW motor driven pump outboard bearing, the inspectors identified inadequate corrective actions by AmerGen to ensure the bearing housing cover bolts were properly installed. The safety significance of this finding was very low (Green) because AmerGen took immediate corrective actions to ensure proper installation of the bolts prior to returning the pump to service. AmerGen’s failure to take adequate corrective actions to ensure the EFW motor driven pump was properly reassembled was a violation of 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” and is being treated as a non-cited violation.

The EFW system provides a risk significant function to remove decay heat from the RCS when the main feedwater system is unavailable. The TMI EFW system consists of two motor driven and one turbine driven pumps, any two of which are required to meet the most limiting flow requirements for decay heat removal.

In February 2001, AmerGen conducted an evaluation to determine the root cause of an oil leak on the ‘A’ motor driven EFW pump that resulted in the pump being declared inoperable for a period 39 days. AmerGen found that the outboard bearing housing cover bolts had loosened during operation creating an oil leak of sufficient magnitude for the pump to be unable to perform its required function. AmerGen entered several items into its CAP (T2001-0173 and T2001-0305) to prevent recurrence. This event resulted in an NRC finding of low to moderate safety significance (White), and was documented in NRC inspection report 05000289/2001-002, dated May 9, 2001.

One of the corrective actions assigned was to create a preventive maintenance task to periodically verify the tightness of the bearing housing cover bolts on all three EFW pumps. Through consultation with the pump vendor, a torque value of 20 foot-pounds was established as an acceptable tightness to verify the bolts had not loosened during operation. However, no corrective action was established to ensure the proper torque value was included in the maintenance instructions for reassembling the pump following maintenance. In reviewing the work package for the ‘B’ EFW motor driven pump, the inspectors identified that the maintenance work package only required the bearing housing cover bolts to be “tightened” and did not specify a required torque value. The inspectors interviewed the technicians who performed the maintenance and found the cover bolts had only been installed “wrench tight.” AmerGen took immediate corrective actions to add an activity to the work package to require the bolts to be torqued to 20 foot-pounds.

This finding had an actual credible impact on safety. AmerGen failed to provide adequate guidance to control the reassembly of the ‘B’ EFW motor driven pump. Failure to properly tighten the bearing housing cover bolts could result in the cover coming loose during operation affecting the reliability of the EFW system.

This finding had a credible impact on safety; however, since only the mitigating system cornerstone is affected and because AmerGen took immediate corrective actions to ensure the cover bolts were properly torqued prior to returning the pump to service, the finding is considered to be of very low safety significance (Green). The inspectors also determined AmerGen’s failure to take adequate corrective actions following the February 2001 event to ensure future EFW pump maintenance activities were properly controlled constituted a violation of 10 CFR 50, Part B, Criterion XVI, "Corrective Actions." However, because of the very low safety significance and because AmerGen has entered this finding into its CAP (CR 00081396), this corrective action violation is being treated as a non-cited violation (**NCV 50-289/01-07-02**).

.2 Additional Post-Maintenance Testing

a. Inspection Scope

The inspectors reviewed post-maintenance tests performed by AmerGen in conjunction with the following outage work activities on risk significant equipment:

- 'A' emergency feedwater motor driven pump overhaul
- 'A' decay heat removal pump mechanical seal repair

The inspectors verified that the post-maintenance test procedures and test activities were adequate to verify operability and functional capability prior to the affected systems being returned to service.

b. Findings

No findings of significance were identified.

R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors reviewed several outage activities to verify that adequate shutdown safety measures were being maintained in accordance with technical specifications and AmerGen Topical Report 097, "TMI-1 Outage Fuel Protection Criteria," Revision 5. The inspectors also observed personnel performance during these activities to verify that operators and technicians were adequately prepared and properly performed the infrequent operations that could impact shutdown safety. Those outage activities included:

- Reactor plant shutdown and cooldown
- RCS inventory control and draindown to mid-loop configuration
- Containment closure controls and contingency planning
- Core offload and reload
- Spent fuel pool cooling management
- Reactor coolant pump tagging clearances

b. Findings

No findings of significance were identified.

R22 Surveillance Testing

.1 Procedure Errors during Main Steam Safety Valve Surveillance Testing

a. Inspection Scope

On October 2, 2001 the inspectors observed surveillance testing of the main steam safety valves (MSSVs) conducted in accordance with AmerGen surveillance procedure 1303-11.3, “Main Steam Safety Valves.” The surveillance is conducted each refueling interval to verify the pressure relief setpoints of the MSSVs. The inspectors observed portions of the test and compared the test results against the acceptance criteria established in the surveillance procedure. The inspectors reviewed the design basis documents for the pressure relief setpoints to determine if the acceptance criteria were appropriately established.

b. Findings

The inspectors identified that the maintenance and test supervisors failed to implement the MSSV test procedure as written which resulted in the setpoints for two safety valves being left outside the tolerance prescribed by the test procedure. The safety significance of the this finding was very low (Green), because AmerGen took immediate corrective action to retest the two valves. The supervisors’ failure to implement the test procedure as written was a violation of TS 6.8, “Procedures and Programs,” and is being treated as a non-cited violation.

The MSSVs have several risk significant functions. The primary purpose of the MSSVs is to provide over pressure protection to the once-through steam generators (OTSGs) during a loss of secondary load. The MSSVs also provide a means of primary system decay heat removal in the event of a loss of the main condenser. Operators also mitigate the consequences of a steam generator tube rupture by maintaining affected OTSG pressure below the MSSV lift setpoints. Plant technical specifications require setpoint testing of the MSSVs such that at least 50 percent of the valves are tested each refueling cycle. The test is performed in-situ with the plant operating at power. A test device is attached to the valve and hydraulic pressure is applied to the valve causing a momentary lift. The hydraulic pressure is then converted to an equivalent steam pressure and the valve lift setpoint is determined.

During MSSV testing on October 2, 2001, the inspector identified that the lift setpoints recorded in the testing record did not account for the height difference between the lift test device and the test gauge used to measure the hydraulic pressure. The test device was physically located approximately 20 feet above the hydraulic pressure gage. Surveillance procedure 1303-11.3, “Main Steam Safety Valves,” required that the hydraulic pressure be corrected for this height difference. The affect of not correcting for the height difference was that the actual lift setpoint was lower than the value recorded in the test record. AmerGen re-calculated the setpoints for the valves that had been tested to account for the pressure correction that had been omitted and found two MSSVs with as-left setpoints outside the procedure allowed tolerance. The two valves were retested and adjusted to within the required as-left tolerance. The procedure as-left tolerance is set at +/- 1 percent to provide assurance the setpoint remains within the design basis tolerance of +/- 3 percent until next tested.

This finding had a credible impact on safety because assurance could not be provided that the MSSVs would remain within their assumed setpoint tolerance throughout the operating cycle. However, since only the mitigating system cornerstone is affected and because AmerGen took immediate corrective actions to retest the two MSSVs with setpoints outside the procedure required as-left tolerance, the finding is considered to be of very low safety significance (Green). The inspectors also determined that the test supervisor's failure to implement the MSSV test procedure as written was a violation of TS 6.8, "Procedures and Programs." However, because of the very low safety significance and because AmerGen has entered this finding into the CAP (CR 00077573), this procedure violation is being treated as a non-cited violation (**NCV 50-289/01-07-03**).

.2 Additional Surveillance Testing

a. Inspection Scope

The inspectors reviewed the following additional surveillance activities. The surveillances were selected based on contribution to plant risk.

- Main storage battery discharge tests
- Decay heat removal pump full flow testing
- Core flood and decay heat removal cold shutdown check valve in-service testing

The inspectors observed portions of the selected surveillance tests and verified, based on the test results, that the systems met TS and procedural requirements. The inspectors reviewed AmerGen's CAP for problems identified during previous performances of the tests to determine if problems involving surveillance testing were being identified and resolved at an appropriate threshold.

b. Findings

No findings of significance were identified.

R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed two temporary modifications: a fire hose connection to support the control building chillers during a nuclear service water system outage and a lifted lead modification that disabled an incorrect alarm for the station blackout diesel generator feed breaker to the '1D' safety-related 4Kv bus. The inspectors verified that the installation of the modifications was consistent with the written documentation and that there were no adverse affects on system operability.

b. Findings

No findings of significance were identified.

2 RADIATION SAFETY

Occupational Radiation Safety [OS]

OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope

The inspector conducted the following activities and reviewed the following documents based on radiological risk significance to determine the effectiveness of access controls to radiologically significant areas:

- The inspector toured TMI and selectively reviewed ongoing radiologically work activities including reactor vessel head penetration work (control rod drive mechanism [CRDM] and thermocouple work); primary and secondary side steam generator work; reactor cavity seal replacement; reactor fuel inspection work; core support assembly removal and inspection; reactor sump cleaning, and reactor coolant pump work. The inspector selectively reviewed the specified work with respect to radiological data, implementation of radiation work permits and prescribed controls, alarming dosimetry setpoints, and airborne radioactivity sampling. The inspector reviewed radiological surveys, aggregate radiation dose, selective personnel contamination reports, and personnel whole body count data. The inspector reviewed controls and surveys of material removed from the spent fuel pool and flooded reactor cavity. The inspector made radiological survey measurements to evaluate the accuracy of documented surveys. Also, the control and monitoring of personnel working in potential radiation dose gradients were evaluated.
- The inspector reviewed radiation worker performance to determine if workers were aware of radiation work permit requirements, radiological hazards, and implemented radiation work permit requirements.
- The inspector reviewed radiation protection technician proficiency to determine if technicians implemented adequate radiological controls.
- The inspector challenged locked High Radiation Area access points to determine if access controls were sufficient to preclude unauthorized entry. The inspector also inspected access points to two Very High Radiation Areas. The adequacy of posting and barricading of High Radiation Areas was also evaluated.
- The inspector reviewed a selection of self-assessments and licensee identified problems to determine if problems were properly entered into the corrective action program, the problems were evaluated, and corrective actions were initiated (CRs 78311, 78749, 78815, 79428, 79608, 80067, 80086).

The review in the above areas was against applicable licensee procedures, 10 CFR 20, and applicable technical specifications.

b. Findings

No findings of significance were identified.

OS2 ALARA Planning and Controls

a. Inspection Scope

The inspector evaluated the implementation of as low as reasonably achievable (ALARA) plans for radiological significant work tasks conducted during the current refueling outage (14R). The tasks reviewed included reactor vessel head penetration work, steam generator work activities, radiation shielding, reactor refueling, scaffolding construction, and core support assembly removal and inspection. The inspector reviewed work-in-progress reviews and aggregate radiation dose relative to initial dose estimates. The inspector also attended a Station ALARA Council Meeting on October 18, 2001. The inspector reviewed licensee response to elevated contamination levels encountered in the 'A' steam generator.

The evaluation of licensee performance in this area was against criteria contained in applicable procedures, 10 CFR 20, and applicable technical specifications.

b. Findings

No findings of significance were identified.

OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspector selectively reviewed elements of the radiation monitoring and instrumentation calibration program. The inspector reviewed the calibration and checking of selected radiation monitoring instruments used by radiological controls personnel during job coverage surveys. The inspector also reviewed calibration of selected process radiation monitors. Calibration records for the following instruments were reviewed to evaluate the adequacy of calibration and conformance with applicable calibration procedures and programs:

Portable:
- RSO-50E (Sn. B311v)
- SAC-4 (Sn. 394)
- RM-14 (Sn. 2155)
- Telescan (Sn. 42530)
- R02A (Sn. 715)

Process:
- RM-G-9, spent fuel bridge monitor
- RM-G-6, auxiliary fuel handling bridge monitor
- RM-G-18, RCS sample area monitor
- RM-A-13G, spent fuel storage gas monitor
- RM-G-26, 27, A & B OTSG monitor
- RM-G-11, makeup demineralizer area monitor
- RM-A-12G, radio-chemistry laboratory sample gas monitor

The review was against criteria contained in applicable AmerGen procedures, 10 CFR 20, applicable technical specifications, and industry standards.

b. Findings

No findings of significance were identified.

Public Radiation Safety [PS]

PS1 Radiological Environmental Monitoring Program

a. Inspection Scope

The inspector reviewed the following documents to evaluate the effectiveness of AmerGen's Radiological Environmental Monitoring Program (REMP) at TMI. The requirements of the REMP are specified in the TS/Offsite Dose Calculation Manual (ODCM).

- the 1999/2000 Annual REMP Reports, including the evaluation of onsite well water (20 well water stations)
- the most recent ODCM, Revision 12, January 3, 2001, and technical justifications for ODCM changes, including sampling locations
- the most recent calibration results of the meteorological monitoring instruments for wind direction, speed, and temperature (performed on April 18, 2001)
- calibration and replacement of multiple meteorological instrument components (Surveillance Deficiency Report Numbers 1-8, November 29, 2000)
- review of the 2000 meteorological monitoring data recovery statistics
- the most recent calibration results for all REMP air samplers (calibrated in June, September, October, and November 2001)
- implementation of the interlaboratory and intralaboratory comparisons
- implementation of the environmental thermoluminescent dosimeter (TLD) program
- self-assessment (No. SA-2001-1033)
- CAP No. T2001-0192 and corrective actions
- Contractor Laboratory (Midwest Laboratory) Audit by the Nuclear Utilities Procurement Issues Committee Joint Quality Assurance (QA) Program Audit (Exelon Audit Report No. SR-2001-341)
- 2000 QA Audit (Audit No. S-TMI-00-14) for the REMP/ODCM and Meteorological Monitoring Program implementations and corrective actions
- 2001 QA Surveillance Audits in the areas of REMP, ODCM, and Meteorology
- Land Use Census procedure and the 2000/2001 results
- associated REMP procedures, including vendor analytical procedures

The inspector toured and observed the following activities to evaluate the effectiveness of the licensee's REMP:

- observation of the operability of meteorological monitoring instruments at the tower
- observation of air iodine/particulate and water sampling techniques

- walk-down to determine whether all air samplers, milk farms, and 25 percent of TLDs were located as described in the ODCM (including control and indicator stations) and for determining the equipment material condition

b. Findings

No findings of significance were identified.

PS3 Radioactive Material Control Program

a. Inspection Scope

The inspector reviewed the following documents to ensure that AmerGen met the requirements specified in its program for unrestricted release of material from the Radiologically Controlled Area (RCA). The review was against criteria contained in 10 CFR 20, NRC Circular 81-07, NRC Information Notice 85-92, NUREG/CR-5569, Health Position Data Base (Positions 221 and 250), and AmerGen's procedures.

- the most recent calibration results for the radiation monitoring instrumentation small article monitors (SAM-9 and SAM-11), including the alarm setting, response to the alarm, the lower sensitivity, and failure rate at the alarm setting
- AmerGen's criteria for the survey and release of potentially contaminated material using a gamma spectroscopy (calibration efficiency for bulk sample analyses)
- TMI-OOB-1 5933-2001-0013, Measurement Results and Release of Bulk Soil
- the methods used for control, survey, and release from the RCA
- associated procedures and records to verify the lower limits of detection for bulk sample analyses

The inspector also reviewed the circumstance and AmerGen's evaluations associated with its identification on February 9, 2001, of a nuclear service river water underground line leak. The review was against the requirements specified in 10CFR50.75(g)(1). The following documents were reviewed:

- CAP T2001-0153, Nuclear Service River Water Underground Line Repair
- TMI-OOB-1, 10CFR50.75 (g) Area East of Heat Exchanger Vault

b. Findings

No findings of significance were identified.

4 OTHER ACTIVITIES

OA1 Performance Indicator Verification

.1 Initiating Events Cornerstone

a. Inspection Scope

The inspectors reviewed AmerGen's performance indicator (PI) data submitted for the initiating events cornerstone for the last four quarters. The initiating events cornerstone PIs included unplanned scrams, scrams with loss of normal heat removal, and unplanned power changes. The inspectors reviewed plant operator logs, licensee event reports, and monthly operating reports.

b. Findings

No findings of significance were identified.

.2 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspector reviewed implementation of AmerGen's Occupational Exposure Control Effectiveness PI Program. Specifically, the inspector reviewed corrective action program records for occurrences involving high radiation areas, very high radiation areas, and unplanned personnel exposures since the last inspection in this area against the applicable criteria specified in National Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline, Revision 1, to verify that all occurrences, that met the NEI criteria, were recognized and identified as PI inputs. In addition, the inspector reviewed instances where personnel exceeded 100 millirem during the current refuel outage (as of October 19, 2001) to ascertain if any radiation work permit limit had been exceeded and if personnel had potentially received unplanned radiation exposures.

b. Findings

No findings of significance were identified.

.3 Radiological Effluent TS/ODCM Radiological Effluent Occurrences

a. Inspection Scope

The inspector reviewed the following documents to ensure that AmerGen met all requirements of the Radiological Effluent TS/ODCM Radiological Effluent Occurrences performance indicator from the third quarter 2000 to the third quarter 2001:

- monthly projected dose assessment results due to radioactive liquid and gaseous effluent releases
- quarterly projected dose assessment results due to radioactive liquid and gaseous effluent releases
- associated procedures

The information contained in these records was compared against the criteria contained in NEI 99-02, Revision 1, "Regulatory Assessment Performance Indicator Guideline," to verify that all conditions that met the NEI criteria were recognized, identified, and reported within the PI.

Findings

b. No findings of significance were identified.

OA3 Event Follow-up

.1 Reactor Pressure Vessel Head Penetration Cracking

The inspectors reviewed 10 CFR 50.72 reports made to the NRC documenting Reactor Pressure Vessel (RPV) head penetrations that were identified to have through-wall cracks. 10 CFR 50.72 reports were made on October 12, 22, and 24, 2001. The RPV head penetration cracking was discovered during the 14R outage in response to NRC Bulletin 2001-01. AmerGen's activities in response to NRC Bulletin 2001-01 were reviewed by inspectors using TI 2515/145 and are described in Section 4OA5 of this report.

.2 Seriously Degraded Once-Through Steam Generator Tube

The inspectors reviewed a 10 CFR 50.72 report made to the NRC on October 20, 2001, documenting a seriously degraded tube in the 'B' OTSG. The degraded tube was discovered during routine eddy current inspections in the 14R refuel outage. The tube was mechanically damaged by an adjacent plugged tube that had severed in apparently the most recent operating cycle. The resident inspectors reviewed the circumstances with a senior reactor analyst and NRC management. Inspection Procedure 71153, "Event Followup" and NRC Management Directive 8.3, "NRC Incident Investigation Program" were used and a special inspection team was assembled to evaluate the circumstances of the OTSG severed tube. The results of the special inspection will be documented in NRC Inspection Report 50-289/2001-012.

.3 Determination of Responsibility for Operability of Siren Actuation System

The Lancaster County sirens were reported by the licensee to be inoperable from October 5 until October 9, 2001, due to a deactivated transmitter. (The transmitter sends an activation signal to the sirens.) Based upon telephone interviews with licensee personnel, the preliminary cause was determined to be an inadvertent deactivation of the transmitter which is located in the Lancaster County 911 center. The problem was discovered by the licensee contractor conducting the weekly silent test on October 9, 2001. The contractor notified the county who reactivated the transmitter and thus returned the sirens to an operable status. The licensee conducted an apparent cause investigation that will be finalized during the next inspection period.

The sirens are the method for notification of the public during a radiological emergency at TMI. The sirens are necessary to meet the requirement of planning standard 10 CFR 50.47(b)(5) which pertains to public notification. This planning standard is considered to be a risk significant planning standard because it directly impacts the health and safety of the public. The lines of ownership and responsibility for the maintenance of the siren actuation system are unclear. Therefore, pending the NRC's review of the licensee's apparent cause report to provide more details about the system and components related to this issue and to determine culpability, this issue will be unresolved. (**URI 50-289/01-07-04)**

OA5 TI 2515/145 - Circumferential Cracking of RPV Head Penetration Nozzles

a. Inspection Scope

The inspector reviewed AmerGen's activities to detect circumferential cracking of RPV head penetration nozzles in response to NRC Bulletin 2001-01 as required by TI 2515/145. This included interviews with analyst personnel, reviews of qualification records and procedures, and observations of selected video tape records of the RPV head visual examination. The inspector independently viewed a sample set of 34 out of the total 77 penetrations examined by the plant staff. In accordance with TI 2515/145, inspectors verified that deficiencies and discrepancies associated with the RCS structures and the examination process were identified and that they were placed in the licensee's CAP.

b. Findings

No findings of significance were identified.

The specific reporting requirements of TI 2515/145 are documented in Attachment A.

OA6 Management Meetings

Exit Meeting Summary

On November 14, 2001, the resident inspectors presented the inspection results to members of AmerGen management led by Mr. George Gellrich. The inservice inspection activities and occupational and public radiation safety inspection results were previously presented to members of AmerGen management. AmerGen acknowledged the findings presented. AmerGen did not indicate that any of the information presented at the exit meetings was proprietary.

ATTACHMENT A

TI 2515/145 - Circumferential Cracking of RPV Head Penetration Nozzles Reporting Requirements

a.1. The examination was performed by qualified and knowledgeable personnel with certification to the American Society of Mechanical Engineers Level III criteria for visual examiners. The visual examination performed was to determine leakage. Training was provided on the unacceptable conditions found at other plants.

a.2. The visual examination was in accordance with approved and adequate procedures.

a.3. The examination was adequate to identify, disposition and resolve deficiencies.

a.4. The examination performed was capable of identifying the primary water stress corrosion cracking phenomenon described in the Bulletin.

b. The general condition of the reactor vessel head was clean bare metal with some localized staining and grit like debris which appears to be a mixture of corrosion products, dry boron flakes and dirt. The inspection was well planned and coordinated with a map of the control rod drive mechanism (CRDM) and thermocouple locations. A detailed written record of the observations was prepared during the video observations. The CRDMs were initially all video visually examined in the as found condition. Debris was removed by vacuum cleaning under video observation from 45 of the CRDMs. Of these 45, deposits with the potential of being an indicator of leakage remained on 12 CRDMs. Of these 12, two were clearly visually classified as leakers. The remaining were scheduled for penetrant examination of the CRDM to head weld and ultrasonic examination of the CRDM housing weld.

c. Small boron deposits, as described in Bulletin 2001-01, could be identified and characterized by the visual examination technique used.

d. Two material deficiencies, visual indication of leakage associated with concerns in Bulletin 2001-01 were found. An additional set of 10 were submitted to ultrasonic examination. Of these 10, 2 of the CRDM to head welds had dye penetrant indications. All 8 of the thermocouple head penetrations were noted as leakers.

e. The as low as reasonably achievable radiation exposure controls for the visual examination process were effective. Penetrant examinations of the CRDMs to head welds are a high radiation exposure job. The development of a faster application with easier clean up would have the potential of reducing radiation dose.

TI 2515/145, Section 04.04 c, requires that inspectors report lower-level issues concerning data collection and analysis, and issues deemed to be significant to the phenomenon described in Bulletin 2001-01. The lower-level issue identified by the inspector is reported below.

1. The inspectors noted that although the video tapes were subtitled to identify each CRDM and the documentation of the examinations appeared to be complete, there was

no voice narration on portions of the video tape. This is in conflict with paragraph 4.14.2 of procedure ES-NDE-07T, Rev 0. This issue, although it did not detract from the validity of the examination records, was documented in Condition Report 00079108.

ATTACHMENT B

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

D. Atherholt, Shift Operations Superintendent
G. Gellrich, Plant Manager
O. Limpias, Director, Site Engineering
D. McDermott, Director, Maintenance
J. McElwain, Manager, Regulatory Assurance
S. Queen, Senior Manager, Plant Engineering
J. Robertson, Plant Operations Director
M. Warner, Vice President, TMI Unit I

b. Items Opened, Closed, and Discussed

OPENED

50-289/01-07-04	URI	Determination of Responsibility for Operability of Siren Actuation System

OPENED AND CLOSED

50-289/01-07-01	NCV	Procedure Errors During Reactor Coolant System Cooldown and Mid-Loop Operation
50-289/01-07-02	NCV	Inadequate Corrective Actions for Emergency Feedwater Pump Maintenance
50-289/01-07-03	NCV	Procedure Errors During Main Steam Safety Valve Surveillance Testing

c. List of Documents Reviewed

54-ISI-400-11, Rev 8/27/00	Multifrequency Eddy Current Examination of Tubing
1249, Rev 0	13R ECT and Visual Examination of Installed Plugs
TR107569-V1, Rev 5	PWR Steam Generator Examination Guideline
TR107620, Rev 1	Steam Generator In-Situ Pressure Test Guidelines
TR107621, Rev 1	Steam Generator Integrity Assessment Guidelines
ER-AP-420-0051, Rev 7/16/01	Conduct of Steam Generator Management Program Activities
ER-AP-335-040, Rev 0	Evaluation of Eddy Current Data for Steam Generator Tubing
ER-TM-335-1005, Rev 0	Analysis of OTSG Eddy Current Data at TMI
TMI,1, 14R	Outage OTSG Planned Work Scope

51-5005406, Rev 1	Qualified ET Exam Techniques for TMI 14R
TMI,1, 14R	Degradation Assessment Report
TDR 1249, Rev 0	Technical Data Report 13R Examination Results
TR 135, Rev 0	Topical Report on 13R Eddy Current Examination TMI OTSG Tubing (90 Day Report 13R)

Framatome Technologies Inc. Procedures:

54-ISI-240-39, Rev. 39	Nondestructive Examination Procedure Visible Solvent Removable Liquid Penetration Examination
54-ISI-836-03, Rev. 3	Inservice Inspection Procedure for the Ultrasonic Examination of Austenitic Piping Welds
54-ISI-106-09, Rev. 3/14/01	Remote Ultrasonic Examination of Reactors Vessel and Associated Piping Welds Using Remote Manipulators and Accusonex Acquisition and Analysis System
54-ISI-800-03, Rev. 7/11/96	Remote Ultrasonic Examination of Reactor Vessel Welds in Accordance with ASME Section XI, Appendix VIII, Supplements 4 and 6
54-ISI-364-00, Rev. 8/22/00	Remote Underwater In-Vessel Visual Inspection of Reactor Pressure Vessel, Vessel Internals, and Components in Pressurized Water Reactors

Framatome ANP UT Calibration Data Sheet No. TMI 005

Work Order Activity Numbers:
C2000099 02, PT on weld DH-028 pen. To 10" piping
C2000099 03 UT Examination on DH-028 pen. To 10" piping
C2000099 04 UT Examination on EF-031 pen. To 6" piping
C2000099 05 MT Examination on EF-031 pen. To 6" piping
C2000066 02 VT Containment Liner, CAP 0869
C2000090 for RPV head examinations

Make-up System Flow Element Radiographs:
MUFE240A, MU1FE , MUFE240B, MUFE240C

Visual Data Sheet for Reactor Building Liner Plate Interior Surface Indication No. 29, 30, 31, 32, 33 and the area at elevation 281'-0" Moisture Barrier.
VT Procedure ES-NDE-07T, Rev. 0 for CRDM and Thermocouple Vessel Head Penetrations
CR# 00079108
Exelon Document RS-01-182, dated 8/31/01. TMI Response to NRC Bulletin 2001-01.
NDE Examiner Qualification records for the head penetration VT inspectors
CRDM and Thermocouple vessel head penetration video tapes from the as found and vacuum cleaned conditions
NDE Procedure 54-PT-6-07, Rev. date 8/3/00
NDE Procedure 54-ISI-100-06 for UT of CRDM nozzles

AmerGen Document 5928-01-20214, dated 8/09/01, on ASME Relief Requests for RPV Head Repair

Corrective Actions:

CAP T1999-0802	Indications in Rolled Plugs
CAP T1999-0881	Weld Quality Issue at B&W Plants (Operating Experience)
CAP T1999-0931	Failed Plug Removal
CAP T2001-0827	Failed Plug Removal-broken head

d. Acronyms

14R	Cycle 14 Refueling Outage
13R	Cycle 13 Refueling Outage
ADAMS	Agencywide Documents and Management System
ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASNT	American Society for Nondestructive Testing
AmerGen	AmerGen Energy Company, LLC
CAP	Corrective Action Process
CFR	Code of Federal Regulations
CR	Condition Report
CRDM	Control Rod Drive Mechanism
DRS	Division of Reactor Safety
ECT	Eddy Current Test
EFW	Emergency Feedwater
IR	Inspection Report
ISI	Inservice Inspection
MSSV	Main Steam Safety Valve
MT	Magnetic Particle
NCV	Non-cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
OTSG	Once-through Steam Generator
PI	Performance Indicator
QA	Quality Assurance
RCA	Radiologically Controlled Area
RCS	Reactor Coolant System
REMP	Radiological Environmental Monitoring Program
RPV	Reactor Pressure Vessel
RT	Radiographs
SAM	Small Article Monitor
SDP	Significance Determination Process
TI	Temporary Instruction
TLD	Thermoluminescent Dosimeters
TMI	Three Mile Island, Unit 1
TS	Technical Specification

URI	Unresolved Item
UT	Ultrasonic Testing
VT	Visual Inspection