July 27, 2001

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

SUBJECT: THREE MILE ISLAND STATION, UNIT 1 - NRC INSPECTION REPORT 50-289/01-04

Dear Mr. Warner:

On June 30, 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 July 13, 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). All three issues were determined to involve violations 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. 20555-001; with copies to the Regional Administrator, Region I; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555-001; and the NRC Resident Inspector at the Three Mile Island Unit 1 facility.

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/NRC/ADAMS/index.html (the Public Electronic Reading Room).

We appreciate your cooperation. Please contact me at (610) 337-5146 if you have any questions regarding this letter.

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 50-289/01-04
Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION 1

Docket No:	50-289
License No:	DPR-50
Report No:	50-289/01-04
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Three Mile Island Station, Unit 1
Location:	P.O. Box 480 Middletown, PA 17057
Dates:	May 13-June 30, 2001
Inspectors:	J. Daniel Orr, Senior Resident Inspector Craig W. Smith, Resident Inspector Neil S. Perry, Senior Project Engineer
Approved by:	John F. Rogge, Chief Projects Branch 7 Division of Reactor Projects

SUMMARY OF FINDINGS

Three Mile Island, Unit 1
NRC Inspection Report 50-289/01-04

IR 05000289-01-04, on 05/13 - 06/30/2001, AmerGen Energy Company, LLC, Three Mile Island Unit 1, resident inspector report, surveillance testing, operability evaluations, event follow-up.

The inspection was conducted by resident and region based inspectors. The inspection identified three Green findings, which were all 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. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. <u>Inspector Identified Findings</u>

Cornerstone: Initiating Events

- **Green**. Control room operators did not properly acknowledge a computer alarm indicating a leaking reactor coolant system safety relief valve (SRV) for about 100 minutes.

  The safety significance of the operator delay was very low (Green) because the SRV seat leakage did not have an immediate affect on continued plant operation. The seat leakage also did not alter the opening characteristics of the SRV. 10 CFR 50, Appendix B, Criterion V, "Procedures," requires, among other requirements, that activities affecting quality shall be accomplished in accordance with procedures. TMI emergency procedure 1202-29, "Pressurizer System Failure," listed two existing symptoms that required a more timely implementation for this instance. The control room operators' untimely implementation of TMI emergency procedure 1202-29, "Pressurizer System Failure," constituted a violation of 10 CFR 50, Appendix B, Criterion V, "Procedures."

Cornerstone: Mitigating Systems

- **Green**. AmerGen failed to establish new inservice testing reference values for the 'B' decay heat removal pump following pump modification in August 2000.

  The safety significance of the non-conservative reference values was very low (Green) because subsequent vibration measurements never exceeded a level requiring corrective action. Technical specification 4.2, Reactor Coolant System Inservice Inspection and Testing, requires inservice testing of the decay heat removal system to be conducted in accordance with the American Society of Mechanical Engineers (ASME) Code. The ASME Code requires that when a test reference value is affected by maintenance or repair, a new reference value shall be determined prior to declaring the pump operable. AmerGen's failure to establish new reference values for the 'B' decay heat removal pump prior to returning the pump to service following a modification constituted a violation of the technical specification requirement to conduct inservice testing in accordance with the ASME Code.

- **Green**. System engineers did not promptly initiate corrective actions to understand the cause of a reactor trip breaker failure to re-close following testing and to investigate the potential for a common mode failure that could have adversely affected the reactor protection system's capability to rapidly shut down the reactor.

  The safety significance of the delay in initiating corrective actions to understand the failure mechanism of the faulty breaker was very low (Green) because subsequent investigation revealed the breaker's capability to perform its safety related function to trip open was not compromised. The failed breaker was immediately replaced with a spare and was retested prior to returning the reactor protection system to service. 10 CFR 50, Appendix B, Criterion XVI, "Corrective Actions," requires, among other requirements, that conditions adverse to quality be promptly corrected. The system engineers' failure to promptly initiate corrective actions and have the faulty reactor trip breaker investigated constituted a violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action."

B. <u>Licensee Identified Violations</u>

- No violations were identified.

Report Details

Summary of Plant Status

AmerGen Energy Company, LLC (AmerGen) restarted Three Mile Island Unit 1 (TMI) on May 13, 2001, after performing a reactor shutdown on May 12, 2001, to reseat a leaking reactor coolant system safety relief valve. TMI resumed operation at about 80% power due to an abnormal gassing condition on the ‘A’ main transformer. On June 7, 2001, operators reduced reactor power to 2% and secured the main turbine to install a temporary transformer in place of the ‘A’ main transformer. TMI returned to 100% power on June 14, 2001.

1 REACTOR SAFETY

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

R01 Adverse Weather Protection

a. Inspection Scope

The inspectors evaluated AmerGen’s preparations for hot weather by walking down TMI abnormal procedure 1203-34, “Control Building Ventilation System.” The control building houses electric power, instrument and control support systems for many risk significant mitigation systems. TMI abnormal procedure 1203-34, “Control Building Ventilation System,” provided compensatory measures to maintain control building ventilation with portable staged equipment upon a complete loss of all normal ventilation. The inspectors verified that adequate instructions existed in the procedure, that portable equipment was readily accessible, and that all equipment rooms were considered in the abnormal procedure.

b. Findings

No findings of significance were identified.

R04 Equipment Alignment

a. Inspection Scope

The inspectors conducted partial system walkdowns during planned maintenance on the emergency feedwater (EFW) system, and during concurrent maintenance on the ‘B’ nuclear services closed cooling water pump and the ‘B’ nuclear services river water pump. The systems were selected because all three systems perform risk significant maintenance rule functions. The inspectors performed the EFW partial walkdown after the ‘B’ motor driven EFW pump was removed from service. The inspectors verified the system alignment was in accordance with operating procedures 1303-11.57, “EFW Flowpath Check,” and 1300-3F, “IST of EF-P-2A/B and Valves,” and verified operating parameters were consistent with the plant operating condition. The inspectors used TMI operating procedures 1104-11, “Nuclear Service Closed Cooling Water System,” and 1104-30, “Nuclear River Water,” to verify that the nuclear services closed cooling water and river water systems were properly aligned during maintenance activities. All three systems were also partially walked down following the maintenance activities to verify a proper return to service.

b. Findings

No findings of significance were identified.

R05 Fire Protection

.1 Fire Protection Walkdowns

a. Inspection Scope

The inspectors conducted fire protection inspections for the following plant fire zones:

- auxiliary building ‘B’ engineered safeguards motor control center area
- auxiliary building 281' elevation penetration area
- intermediate building 305' elevation corridor
- intermediate building 322' elevation general areas

The rooms and areas were selected based on enclosing or proximity to risk significant equipment. 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.

b. Findings

No findings of significance were identified.

.2 Fire Drill and Fire Brigade Response

a. Inspection Scope

The inspectors observed the fire brigade response during portions of the emergency preparedness exercise on May 23, 2001. The inspectors verified that drill personnel were ready to fight a fire, that the drill scenario was followed, and that the drill objectives and acceptance criteria were met.

b. Findings

No findings of significance were identified.

R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed AmerGen's external flooding mitigation plans and equipment were consistent with the design requirements and risk analysis assumptions. The inspectors reviewed the final safety analysis report and related flood analysis documents and identified risk significant areas that can be affected by external floods. The inspectors walked down risk significant areas of the plant to verify flood control equipment and mitigating plans were in place and capable of performing their design functions. The inspectors reviewed emergency procedure 1202-32, "Flood," for areas where operator actions are credited for coping with flooding to verify the procedure could achieve the desired actions. The inspectors sampled AmerGen's corrective action data base to determine if problems concerning external flood control measures were being identified at an appropriate threshold and properly addressed for resolution.

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 four equipment performance issues:

- two main steam valve (MS-V-13A and MS-V-4A) stroke time issues
- 'A' nuclear service closed cooling water pump loose shaft coupling bolts
- 'A' motor driven emergency feedwater pump loose bearing housing cover bolts

The aspects of maintenance rule implementation inspected included safety significance classification, a(2) performance criteria or a(1) goals and corrective actions, and maintenance preventable functional failure determinations. The inspectors referenced TMI administrative procedure 1082, "NRC Maintenance Rule," and NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Plants."

b. Findings

No findings of significance were identified.

R13 Maintenance Risk Assessment and Emergent Work Control

a. Inspection Scope

The inspectors reviewed AmerGen's planning and risk assessment for three planned risk significant maintenance activities:

- 'B' motor driven EFW pump system outage
- 'A' nuclear river water pump and strainer system outage

- concurrent outages on the 'B' nuclear services closed cooling water pump and 'B' nuclear river water pump with surveillance testing on the 'B' emergency diesel generator

The inspectors reviewed the risk assessment of these maintenance activities with respect to 10 CFR 50.65(a)(4). The inspector reviewed the outage risk analyses to assure that concurrent work would not negatively impact the overall safety of the facility. The inspectors referenced TMI administrative procedure 1082.1, "TMI Risk Management Program," and NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Plants."

b. Findings

No findings of significance were identified.

R15 Operability Evaluations

.1 Control Rod Drive Circuit Breaker Failure

a. Inspection Scope

On June 4, 2001, during routine surveillance testing of the reactor protection system, one of the four redundant reactor trip breakers failed to re-close after testing. The inspectors reviewed AmerGen's corrective actions taken in response to this equipment deficiency.

b. Findings

The inspectors determined system engineers did not promptly investigate the failure of the reactor trip breaker to re-close following testing and did not initiate a corrective action process (CAP) form for more than 72 hours. The timeliness for understanding the cause of the breaker failure was important because the reactor trip breakers provide the means by which the reactor protection system rapidly shuts down the reactor. It was important to understand the cause of the breaker failure to assess any common mode failure mechanisms with the other reactor trip breakers or the reactor protection system. The safety significance of this finding was very low (Green) because the faulty breaker was immediately replaced with a spare and satisfactorily retested. Additionally, subsequent testing of the failed breaker did not reveal any generic concerns which would have prevented the breaker from performing its safety related function to open. The system engineers' failure to promptly identify the cause of the failed reactor trip breaker constituted a violation of 10 CFR 50, Appendix B, Criterion XVI, "Corrective Action."

The TMI reactor protection system consists of four redundant channels each providing a trip signal to one of four reactor trip breakers. When the reactor protection system detects a condition requiring the reactor to be rapidly shut down, each channel individually sends a signal to its respective reactor trip breaker and the breaker automatically opens. For testing and reliability considerations, a minimum of two, and no more than three of four reactor trip breakers need open to initiate an automatic reactor shutdown.

During routine testing on June 4, 2001, the reactor trip breaker associated with the ‘A’ reactor protection system channel successfully tripped open in response to test signals, but subsequently failed to re-close when the test signals were removed. The faulty breaker was immediately replaced with a spare and was satisfactorily retested prior to returning the ‘A’ reactor protection system channel to service. However, system engineers did not initiate prompt corrective actions to understand the cause of the breaker’s failure to re-close and the potential for common mode failure with the other reactor trip breakers. The initial operability evaluation did not appropriately consider the potential for the breaker failure mechanism to also affect the ability of the breaker to trip open in response to a signal from the reactor protection system. On June 7, 2001, system engineers initiated a CAP form (T2001-0571) and the failed breaker was investigated. System engineers subsequently determined the failure mechanism would not have affected the breaker’s ability to open and that there were no common mode failure concerns with the other reactor trip breakers.

This finding was more than minor because if left uncorrected an undetected failure mechanism remaining in service could degrade the reactor protection system’s capability to automatically shut down the reactor. The safety significance of the delay in initiating corrective actions to understand the reactor trip breaker failure mechanism was very low (Green) because subsequent investigation revealed the breaker’s capability to perform its safety related function to trip open was not compromised. 10 CFR 50, Appendix B, Criterion XVI, “Corrective Action,” requires that conditions adverse to quality be promptly identified and corrected. Contrary to this requirement, system engineers did not promptly initiate corrective actions to understand the cause of a reactor trip breaker failure to re-close following testing. However, because of the very low safety significance, and because AmerGen has entered this problem into its corrective action process (T2001-0632), the violation is being treated as a non-cited violation (**NCV 05000289/2001-004-01**).

.2 Additional Operability Evaluations

a. Inspection Scope

The inspectors reviewed two operability evaluations for degraded equipment issues: differences between the installation of the ‘A’ and ‘B’ emergency diesel generator governor linkages; and a one-time anomaly with the ‘B’ EFW system flow control valve (EF-V-30B) controller. The inspectors verified the degraded equipment issues 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

a. Inspection Scope

The inspectors reviewed post-maintenance tests performed by AmerGen in conjunction with the following maintenance activities:

- planned system outage on the ‘B’ motor driven EFW pump
- unplanned failure of a source range nuclear instrument and swap over of inputs to an alternate source range detector
- planned outage on the ‘A’ nuclear river water pump and strainer
- planned performance of river water system clam kill and post-performance mortality evaluation

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.

R22 Surveillance Testing

.1 Decay Heat Pump Surveillance Testing

a. Inspection Scope

On May 18, 2001, the inspectors observed in-service testing of the decay heat removal system pumps and valves conducted in accordance AmerGen surveillance procedure, 1300-3B, “In-Service Testing of ‘A’ and ‘B’ Decay Heat Removal Pumps and Valves.” The inspectors observed portions of the test and compared the test results against the acceptance criteria established in the test procedure. The inspectors reviewed system design documents and maintenance records to determine if the acceptance criteria were appropriately established and in accordance with Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requirements.

b. Findings

The inspectors determined that AmerGen failed to establish new reference values for pump vibration measurements following a modification to the ‘B’ decay heat removal pump (DH-P-1B) in August 2000. The safety significance of this finding was very low (Green) because subsequent pump vibration measurements never exceeded a level requiring corrective actions. AmerGen’s failure to establish new reference values constituted a violation of technical specifications, which require testing of the decay heat removal system to be conducted in accordance with the ASME Code. The ASME Code requires that new reference values be established, or the previous values reconfirmed, prior to declaring a pump operable when the reference values may have been affected by maintenance or repair.

TMI technical specifications required AmerGen to conduct quarterly testing of the decay heat removal system pumps and valves in accordance with Section XI of the ASME Boiler and Pressure Vessel Code. Pump vibration measurements are required to be recorded and analyzed each test. Historically, the ‘B’ decay heat removal pump experienced increased vibration measurements. System engineers determined that the increased vibration measurements were the result of the pump design and testing method. The increased vibration measurements were not indicative of degrading pump performance. In 1996, the NRC granted the licensee a relief from the ASME Code requirements and allowed the absolute alert range value for pump vibration measurements to be raised from 0.325 inches per second (ips) to 0.4 ips. For pump vibration measurements in excess of the alert range value, the ASME Code required testing twice as often until the cause was corrected.

While reviewing the May 18, 2001, test data, the inspectors identified that the pump vibration measurements were significantly below the reference values listed in the test procedure. The inspectors found that in August 2000, AmerGen installed a stiffener on the pump bearing support plate to reduce pump bearing vibration measurements. Post-modification testing indicated that the modification was successful as evidenced by the pump thrust bearing vibration measurements in the vertical direction decreasing from 0.280 ips in the test just prior to the stiffener being installed, to 0.110 ips in the post-modification testing. The system engineer and in-service test engineer were aware of the change in pump vibration measurements, but did not initiate a change to the test procedure and establish new reference values. The engineers stated that they wanted to observe pump performance over a full year of testing before establishing new reference values. Prior testing of the decay heat removal pumps indicated some variations in pump vibration measurements based on the temperature of the water being recirculated through the pump during testing. However, the ASME Code specifically stated new reference values shall be established prior to returning the pump to service. The ASME code did not allow for an evaluation period prior to establishing new reference values.

This finding is more than minor because the ASME Code is based on measuring degradation from a fixed baseline. AmerGen's failure to establish new reference values could have masked DH-P-1B degradation. The safety significance of AmerGen's failure to establish new reference values was very low (Green), because recorded vibration measurements, since the modification was installed, showed no variation outside the ASME Code acceptable range of values. Technical specification 4.2, Reactor Coolant System Inservice Inspection and Testing, requires inservice testing of the decay heat removal system to be conducted in accordance with the ASME Code. The ASME Code requires that when a test reference value is affected by maintenance or repair, a new reference value shall be determined prior to declaring the pump operable. Contrary to this requirement, AmerGen failed to establish new reference values for the 'B' decay heat removal pump prior to returning the pump to service following a modification to the pump bearing support in August 2000. However, because of the very low safety significance of this violation, and because AmerGen has entered this problem into its corrective action process (T2001-0602), this violation is being treated as a non-cited violation (**NCV 05000289/2001-004-02**).

.2 <u>Additional Surveillance Testing</u>

a. <u>Inspection Scope</u>

The inspectors reviewed two additional surveillance testing activities: emergency safeguards actuation system testing with the 'A' nuclear service closed cooling water pump out of service and screen house silt level measurements. The surveillance activities were selected based on contribution to plant risk. The inspectors observed portions of the selected surveillance tests and verified, based on the test results, that the systems met technical specification and procedural requirements. The inspectors sampled AmerGen's corrective action process 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 a temporary plant modification to improve the 'A' main feed pump speed indication sensitivity. The increased instrument sensitivity supported troubleshooting activities for 'A' main feed pump speed oscillations. The inspectors also reviewed a temporary modification that supported troubleshooting a one-time anomaly with the 'B' EFW system flow control valve (EF-V-30B) controller. Both temporary modifications were associated with systems that provide risk significant maintenance rule functions. The inspectors reviewed the temporary modification documentation against AmerGen administrative procedure 1013, "Temporary Modifications and Bypass of Safety Functions;" walked down the temporary modification installation; and considered the impact on system operability.

b. Findings

No findings of significance were identified.

4 OTHER ACTIVITIES

OA1 Performance Indicator Verification

a. Inspection Scope

The inspectors verified data submitted by AmerGen for the high pressure safety injection system unavailability performance indicator. The inspectors reviewed operating logs, maintenance rule records, and the corrective action process database to verify the accuracy and completeness of the reported unavailability data. Records were reviewed for reported performance indicator data covering the last two quarters of 2000 and the first two quarters of 2001.

b. Findings

No findings of significance were identified.

OA3 Event Follow-up

.1 Reactor Coolant System Safety Relief Valve Seat Leakage

a. Inspection Scope

The inspectors responded to the main control room, walked down the main control room panels and interviewed control room operators shortly after a reactor coolant system safety relief valve began to leak past its valve seat. The inspectors reviewed the operators' response and emergency procedure implementation.

b. Findings

The inspectors determined that control room operators did not properly acknowledge a computer alarm indicating a leaking reactor coolant system safety relief valve (SRV) for about 100 minutes. The control room operators' failure to implement TMI emergency procedure 1202-29, "Pressurizer System Failure," in a timely manner constituted a violation of 10 CFR 50, Appendix B, Criterion V, "Procedures." The safety significance of this finding was very low (Green) because the SRV seat leakage did not have an immediate effect on continued plant operation.

One of two pressurizer SRVs began to leak past its valve seat to the reactor coolant drain tank (RCDT) at 4:10 a.m. on May 11, 2001. The leakage was indicated by a steady computer alarm that sensed a temperature increase in the SRV discharge line. The computer alarm was visually present on a control room monitor and initially audible until silenced. The control room operators did not recognize the unexpected alarm until shift turnover at about 5:50 a.m. TMI emergency procedure 1202-29, "Pressurizer System Failure" listed two existing symptoms that required implementation no later than 6:10 a.m. for this instance: relief valve discharge line temperature alarm for a period in excess of two hours and RCDT level increasing. At 6:40 a.m. the control room operators entered TMI emergency procedure 1202-29, "Pressurizer System Failure."

The control room operators quantified the SRV seat leakage at about 0.8 gallons per minute. AmerGen reseated the SRV on May 12, 2001, by shutting down the reactor and then reducing reactor pressure to about 350 psig below its normal operating condition.

The operator's inattentiveness to the SRV computer alarm and delay in entering TMI emergency procedure 1202-29, "Pressurizer System Failure" had an actual impact on safety because reactor coolant system leakage increased above its normal value and the increase was not evaluated in a timely manner. The inspectors used the initiating event cornerstone to evaluate the significance of the finding because a leaking SRV of sufficient magnitude is a primary system loss of coolant accident initiator. The inspectors conservatively bounded the significance determination of this finding by assuming that the initiating event was a stuck open SRV. Assuming that the initiating event was a stuck open relief valve was conservative in that the actual seat leakage did not alter the valve's opening characteristics. During this time, and even until the SRV was reseated on May 12, 2001, all mitigation systems were available. The Phase 2 SDP worksheets for a stuck open relief valve with all mitigation systems available evaluated this finding as very low safety significance (Green). The control room operators' failure to enter TMI emergency procedure 1202-29, "Pressurizer System Failure," in a timely manner constituted a violation of 10 CFR 50, Appendix B, Criterion V, "Procedures." However, because of the very low safety significance and because

AmerGen has entered this procedure problem into its corrective action process (T2001-0473), this violation is being treated as a non-cited violation (**NCV 05000289/2001-004-03**).

.2 (Closed) Licensee Event Report (LER) 50-289/2000-004-01, Supplement: Discovery of a Condition Outside the Plant Design Basis for the Small Break Loss of Coolant Accident Analysis of Record for the Core Flood Tank (CFT) Line Break Case. This LER provides supplemental information supporting TMI's original root cause investigation and actions taken to mitigate a CFT line break. The inspectors performed an onsite review of the LER, including verification of corrective action, and determined that the supplemental information was consistent with TMI's original corrective actions.

.3 (Closed) Licensee Event Report 50-289/2001-001-01: Emergency Feedwater Pump 2A Inoperable Greater than the Technical Specification Allowable Outage Time Due to an Incorrect Operability Determination. This LER supplement updates AmerGen's original submittal concerning the circumstances surrounding the 'A' motor driven emergency feedwater pump (EF-P-2A) being inoperable for longer than the technical specification allowed outage time. The root cause of the inoperable condition was loose cover bolts on the pump bearing that resulted in an excessive oil leak and abnormal pump bearing vibration. The LER supplement revised AmerGen's original determination that EF-P-2A was inoperable for 14 days, and documented AmerGen's conclusion that the pump was inoperable for a period of 39 days. The LER supplement listed long term corrective actions taken or planned by AmerGen in response to this event that were not included in the original LER.

NRC Inspection Report 05000289/2001-002, dated May 9, 2001, provided a detailed description of this event and AmerGen's performance deficiencies. Inspection report 05000289/2001-003, dated May 31, 2001, closed the original LER for this event and concluded AmerGen's root cause evaluation was consistent with the NRC's observations. The new information provided in this LER supplement does not change that conclusion. In a letter to AmerGen dated July, 5, 2001, the NRC provided its final significance determination for EF-P-2A being inoperable for 39 days as low to moderate risk significance (White). As a result of the White finding, AmerGen's corrective actions in response to this event will be subject to a supplemental inspection in accordance with the NRC Reactor Oversight Process.

OA6 Management Meetings

Exit Meeting Summary

On July 13, 2001, the resident inspectors presented the inspection results to members of AmerGen management led by Mr. George Gellrich. AmerGen acknowledged the findings presented. AmerGen did not indicate that any of the information presented at the exit meeting was proprietary.

Annual Assessment Letter Public Meeting

On June 27, 2001, the NRC met with the AmerGen staff to present the conclusions associated with the NRC's Annual Assessment of Three Mile Island issued in a letter dated May 31, 2001. The meeting was open for public observation. NRC presentation slides were placed into ADAMS under ML011800397 and were made available for public access.

ATTACHMENT

SUPPLEMENTAL INFORMATION

a. Key Points of Contact

D. Atherholt, Shift Operations Superintendent
G. Gellrich, Plant Manager
O. Limpias, Director - Site Engineering
J. McElwain, Manager, Regulatory Assurance
B. Merryman, Director, Maintenance
S. Queen, Senior Manager, Plant Engineering
J. Robertson, Plant Operations Director
J. Telfer, Director, Radiation Health & Safety
M. Warner, Vice President, TMI Unit I

b. Items Opened, Closed, and Discussed

Opened and Closed

05000289/2001-004-01	NCV	System engineers did not promptly initiate corrective actions to understand the cause of a reactor trip breaker failure to re-close following testing
05000289/2001-004-02	NCV	AmerGen failed to establish new reference values for the 'B' decay heat removal pump prior to returning the pump to service following a modification to the pump bearing
05000289/2001-004-03	NCV	Control room operators did not properly acknowledge a computer alarm indicating a leaking reactor coolant system safety relief valve

Closed

05000289/2000-004-01	LER	Supplement: Discovery of a Condition Outside the Plant Design Basis for the Small Break Loss of Coolant Accident Analysis of Record for the Core Flood Tank (CFT) Line Break Case
05000289/2001-001-01	LER	Emergency Feedwater Pump 2A Inoperable Greater than the Technical Specification Allowable Outage Time Due to an Incorrect Operability Determination

c. Acronyms

ADAMS	Agencywide Documents and Management System
AmerGen	AmerGen Energy Company, LLC
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Process
CFR	Code of Federal Regulations
CFT	Core Flood Tank
EFW	Emergency Feedwater
ips	Inches per Second
IR	Inspection Report
LER	Licensee Event Report
NCV	Non-cited Violation
NRC	Nuclear Regulatory Commission
PSIG	Pounds per Square Inch
RCDT	Reactor Coolant Drain Tank
SDP	Significance Determination Process
SRV	Safety Relief Valve
TMI	Three Mile Island, Unit 1