January 17, 2001

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

SUBJECT: THREE MILE ISLAND NUCLEAR STATION - NRC INSPECTION REPORT NO. 05000289/2000-010

Dear Mr. Warner:

On December 15, 2000, the NRC completed a team inspection at the Three Mile Island Unit 1 reactor facility. The enclosed report presents the results of that inspection. The preliminary results of this inspection were discussed with Mr. G. Gellrich and other members of your staff on December 15, 2000.

The inspection was an examination of activities conducted under your license as they relate to the identification and resolution of problems, and compliance with the Commission's rules and regulations, and with the conditions of your license at Three Mile Island Nuclear Station, Unit 1. Within these areas, the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

On the basis of the sample selected for review, the team concluded that, in general, problems were properly identified, evaluated, and resolved. The resulting evaluations or root cause analyses of the corrective action process (CAP) items reviewed by the team were acceptable and had the appropriate corrective actions prescribed. Nevertheless, the team identified one issue of very low safety significance (Green) associated with the failure to include two nuclear services closed cooling water system valves in the in-service testing program. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating this issue as a Non-Cited Violation, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny this non-cited violation, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Region I; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Three Mile Island Nuclear Station.

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 (PARS) 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).

Sincerely,

/RA J. C. Linville Acting for/

Wayne D. Lanning, Director
Division of Reactor Safety

Docket No. 05000289
License No. DPR-50

Enclosure: NRC Inspection Report No. 05000289/2000-010
Attachments (1) NRC's Revised Reactor Oversight Process
(2) List of Documents Reviewed
(3) Supplemental Information
(4) List of Acronyms Used

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NAME	J Yerokun/**M.Gray for**		D Lew/ **DCL**		J Rogge/ **JFR**		J Trapp/ **DCL**		W Lanning/ **JCL for**	
DATE	01/08/01		01/ 09/01		01/09/01		01/09/01		01/ 16/01	

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U.S. NUCLEAR REGULATORY COMMISSION

REGION 1

Docket No.	05000289
License No.	DPR-50
Report No.:	2000-010
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Three Mile Island Station, Unit 1
Location:	P.O. Box 480 Middletown, PA 17057
Dates:	November 27, 2000 to December 1, 2000 December 11, 2000 to December 15, 2000
Inspectors:	Jimi Yerokun, Senior Reactor Engineer, Team Leader Craig Smith, Resident Inspector Mel Gray, Reactor Engineer Thomas Moslak, Health Physicist
Approved by:	David C. Lew, Chief Performance Evaluation Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000289-00-010; on 11/27 - 12/15/2000; AmeriGen Energy Company, LLC; Three Mile Island (TMI) Unit 1; Annual baseline inspection of identification and resolution of problems. Findings identified in identification and resolution of issues.

The inspection was conducted by three region-based inspectors and one resident inspector. The inspection identified one Green issue of very low safety significance, which was classified as a non-cited violation. The significance of 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. (Attachment 1)

Identification and Resolution of Problems

The team concluded that, based on the samples reviewed, the implementation of the corrective action program at Three Mile Island was acceptable. In general, problems were identified and corrective actions implemented adequately for risk significant problems. The licensee was effective at identifying and tracking problems. The licensee's resolution of problems was adequate. Items entered into the corrective action process were properly classified and prioritized for resolution. The evaluations and root cause analyses reviewed were of good quality and depth. The prescribed corrective actions for the items reviewed appeared appropriate to correct the problems. The backlog of corrective actions was being managed well and the team did not identify any backlogged action that represented an adverse effect on plant risk. Issues identified in the Nuclear Oversight Audits and Self Assessment Reports reviewed had been properly entered into the corrective action process.

Cornerstone: Mitigating Systems

- Green. A non-cited violation of Technical Specifications 4.2.2, was identified associated with the failure to include Nuclear Services Closed Cooling Water System (NSCCWS) valves NS-V-84 and NS-V-85 in the inservice testing program. The valves are required to be leak tight to ensure that the NSCCWS can perform its heat removal functions without excessive loss of inventory to the Reactor Building Emergency Cooling Water System during accident conditions. The risk associated with the failure to include the NSCCWS valves in the inservice testing program was determined to be very low safety significance because during the last refueling outage, the licensee had obtained reasonable indication that the valves had adequate leak tightness and would perform their function. (4OA2.2)

Report Details

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems (71152)

.1 Effectiveness of Problem Identification

a. Inspection Scope

The team reviewed items selected from various AmerGen processes and activities to determine if the licensee was properly characterizing and entering problems into the corrective action process for evaluation and resolution. The team's review included: logs, electronic task tracking system (ETTS) items, work requests and/or job orders, surveillance deficiency reports (SDR), temporary modifications, system health reports, and occupational radiation safety performance indicators. The team also performed plant walk-downs and conducted interviews with plant personnel to determine if risk significant problems were appropriately identified and entered into the corrective action process for evaluation and resolution.

In preparation for the inspection, the team obtained and reviewed the licensee procedures listed on Attachment 2 of this report, to understand the process for implementing the corrective action program at Three Mile Island.

b. Findings

The team noted that the corrective action process was the licensee's primary process for identifying and resolving problems. Issues were entered into this process as "CAPs". However, there were other processes that represented an element of the licensee's corrective action program for tracking issues that did not meet the threshold of being a CAP. The team found that problems and issues were promptly identified and entered into a TMI process. In general, issues identified through the other processes that met the threshold for CAPs, appropriately had CAPs generated. However, the team identified the following instances where issues did not have CAPs as required: (1) Operating Experience (OE) 10927 regarding operating experience on vacuum breaker check valves; (2) ETTS 31748 regarding operating experience on emergency feedwater pump turbine overspeed trip; (3) Operating Experience on Control Rod Drive Mechanism leakage; and (4) SDR on building spray pressure switch. The licensee subsequently generated CAPs T2000-0991, T2000-1035, T2000-1023 and T2000-1033, respectively, to address the issues. The issues were determined to be minor. None of the systems involved in the above issues was degraded and there was no safety consequence.

.2 Prioritization and Evaluation of Issues

a. Inspection Scope

The team reviewed items selected from the licensee's corrective action process covering all seven cornerstones to determine whether AmerGen was properly evaluating and resolving problems adverse to quality. The review included the appropriateness of the assigned significance, the timeliness of resolutions, and the scope and depth of the root cause evaluations (or apparent cause evaluation). The items were selected based on factors such as the plant risk insights derived from TMI's Individual Plant Evaluation (IPE), and the system's maintenance rule risk significance. The CAPs, NCVs and Licensee Event Reports (LERs) listed in Attachment 2 were reviewed for this purpose.

b. Findings

In general, the evaluations and root causes reviewed were of good quality and reflected proper consideration for common cause and extent of condition. Resolutions were completed and corrective actions developed in a timely manner.

However, the team identified an instance where the licensee had not effectively evaluated an issue in 1996 regarding two Nuclear Service Closed Cooling Water System (NSCCWS) valves. The licensee had not identified that these valves should be included in the Inservice Testing (IST) program. NSCCWS is connected to the Reactor Building Emergency Cooling Water System (RBECWS) through normally open ½ inch manual valves NS-V-84 and NS-V-85. This connection ensures that the RBECWS cooling coils are filled with corrosion inhibited water and provides for monitoring cooling coil integrity. During accident conditions, the RBECWS will operate at a lower pressure than the NSCCWS, consequently the NSCCWS will lose some inventory to the RBECWS via the crosstie valves and ultimately to the river discharge canal until operators isolate this crosstie by closing valves NS-V-84 and NS-V-85.

In reviewing an ETTS item regarding these valves, the inspector questioned whether the valves should be included in the IST program since the leak tightness of the valves would be required to maintain NSCCWS inventory during an accident. The licensee provided internal correspondence from 1996 regarding the testing of these valves. Following further review, the licensee concluded that these valves should have been included in the IST program. The licensee generated CAP T2000-1027 to address this issue.

The failure to include NSCCWS valves NS-V-84 and NS-V-85 in the IST Program is a violation of TMI Unit 1 Technical Specifications (TS). Section 4.2.2 of the Technical Specifications requires the inservice testing of valves that are within the scope of the American Society of Boiler and Pressure Vessel Code, Section XI. This issue is more than minor since periodic testing of the leak tightness of these valves is required to ensure that the NSCCWS can perform its heat removal function during accident conditions. The issue affects the mitigating systems cornerstone. In considering the risk significance of this issue, the licensee indicated that during the last refueling outage, during a period while the RBECWS was drained, the NSCCWS surge tank level was not noted to decrease, indicating that the NS-V-84 and NS-V-85 valves would likely have performed their function. Therefore, since these valves would most likely have performed their function, there is a very low risk significance associated with this violation. The team reviewed the licensee's determination of the low safety significance associated with this issue and found it reasonable. In accordance with the NRC Enforcement Policy and the NRC Significance Determination Process, this issue is considered a Non-Cited Violation (Green) **(NCV 050000289/2000-010-001)**

Also, the team found an evaluation regarding a 1999 reactor cavity drain down event which had been identified as a non-cited violation (CAP T1999-1220), where the broadness and the extent of condition reviews were weak. The issues associated with a similar event, a 1997 reactor cavity overfill event that had been identified as a violation of NRC requirements, were not considered. The licensee subsequently generated CAP T2000-1024 to address the matter.

.3 Effectiveness of Corrective Actions

a. Inspection Scope

The team selected a sample of CAPs from those listed in Attachment 2 of this report and reviewed the corrective actions specified or implemented to determine whether the actions were commensurate with the problems, and were implemented or scheduled to be implemented in a timely fashion. The team also reviewed the backlog of corrective actions to determine if there were any items that individually or collectively represented an adverse effect on plant risk significance or an adverse trend in the implementation of the corrective action program.

b. Findings

There were no findings identified during this inspection. The prescribed corrective actions for the CAPs reviewed appeared appropriate to correct the problems. The backlog appeared to be appropriately managed. The team did not identify any items in the backlog reviewed that represented an adverse effect on plant risk.

.4 Assessment of Licensee Audits and Assessments

a. Inspection Scope

The team reviewed 5 Nuclear Safety Oversight audits and 22 departmental self-assessments to determine if: (1) problems and issues identified in the audits and assessments were properly entered into the CAP when required, (2) the licensee's assessment of performance in the CAP area reflected that they understood the problems that existed with the CAP, and (3) the licensee's assessment of performance in the CAP area was comparable to the NRC's assessment results.

The team attended Management Review Team meetings to observe the licensee's discussion of CAP issues including the assignment of significance and priorities for the CAPs. In addition, the team reviewed the meeting minutes from several Nuclear Review Board and Plant Review Group (PRG) meetings to determine if issues identified by the oversight committees were entered into the corrective action process, when necessary.

b. Findings

There were no findings identified during this inspection. The assessments reflected that the licensee was aware of the issues with the corrective action process such as the ones identified by the team. Issues identified in the various audits and assessments reviewed had been properly entered in the corrective action process. However, in one instance, the team observed that the licensee's actions taken for issues identified in Assessment 95928-PA-00-002, which addressed instances where required CAPs were not generated, could have been broader. The licensee responded to each of the instances separately, and did not consider the extent of condition concerning CAP threshold and why issues were not entered into the corrective action process.

.5 Assessment of Safety-Conscious Work Environment

a. Inspection Scope

The team interviewed plant personnel to determine if personnel were hesitant to identify safety issues.

b. Findings

There were no findings identified during this inspection.

4OA6 Meetings, Including Exit

.1 Exit Meeting Summary

The team presented the inspection results to Mr. George Gellrich and other members of the AmerGen staff during an exit meeting on December 15, 2000. The licensee acknowledged the findings presented. No information examined or reviewed during the inspection was considered to be proprietary.

ATTACHMENT 1
NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety	**Radiation Safety**	**Safeguards**
• Initiating Events • Mitigating Systems • Barrier Integrity • Emergency Preparedness	• Occupational • Public	• Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.
More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.
ATTACHMENT 2

LIST OF DOCUMENTS REVIEWED

Procedures

1097, Corrective Action Process, Revision 2
1000-ADM-7216.02, Root Cause Evaluation Procedure, Revision 2
1000-Adm-1291.03, Self Assessment, Revision 3
1001J, Technical Specification Surveillance Testing Program
1002, Rules for the Protection of Employees Working on Electrical and Mechanical Apparatus
1100-Adm-7218.01, Nuclear Safety Assessment Audit Program, Revision 6
1104-4, Decay Heat Removal System
1106-6, Emergency Feedwater System
1407-18, Work Request Processing, Revision 2
1407-19, Job Order Processing, Revision 3
1082, NRC Maintenance Rule, Revision 6
1086, Industry Operating Experience Review Process, Revision 2
EI-AA-101, Employee Concerns, Revision 0
EP-046T, Engineering Evaluations, Revision 5
Surveillance Procedure 1303-11.10, ES Emergency Sequence and Power Transfer Test
Surveillance Procedure 1303-4.11, HPI/LPI Logic and Analog Channel Test

CAPs (Problem Identification and Resolution Documents) - Numbers Only

SPR (Significant Problem Response) CAPs

T1998-1142	T1999-0399		
T2000-0018	T2000-0099	T2000-0158	T2000-0385
T2000-0398	T2000-0426	T2000-0464	T2000-0534
T2000-0542	T2000-0633	T2000-0673	T2000-0751

PR (Problem Response) CAPs

T1997-0800			
T1999-0271	T1999-0302	T1999-0442	T1999-0566
T1999-1134	T1999-1220	T1999-1255	
T2000-0052	T2000-0115	T2000-0120	T2000-0168
T2000-0202	T2000-0217	T2000-0246	T2000-0258
T2000-0292	T2000-0313	T2000-0314	T2000-0346
T2000-0394	T2000-0410	T2000-0445	T2000-0459
T2000-0462	T2000-0482	T2000-0491	T2000-0501
T2000-0532	T2000-0536	T2000-0562	T2000-0628
T2000-0631	T2000-0645	T2000-0669	T2000-0671
T2000-0732	T2000-0742	T2000-0809	T2000-0816
T2000-0826	T2000-0850	T2000-0853	T2000-0854
T2000-0867	T2000-0886	T2000-0893	T2000-0896
T2000-0900	T2000-0903	T2000-0924	T2000-0984
T2000-0991	T2000-0966	T2000-1007	T2000-1009
T2000-1027			

Material Nonconformance Report, Directed Action or MR(a)(1) CAPS

T2000-0345	T2000-0631	T2000-1142	T2000-0249
T2000-0636	T2000-0664	T2000-0711	T2000-0726
T2000-0748	T2000-0782	T2000-0878	T2000-0938
T2000-0950	T2000-0981		

Electronic Task Tracking System (ETTS)

7094	33996	37166	17757
15285	36133	12308	19059
28398	37156	13346	23275
31748	37158	13680	33976
			37530

Work Requests/Job Orders

WR 785558, DHR Flow Transmitter
WR 786332, Letdown Tank Valve
WR 792367, DH Suction from Borated Water Storage Tank Valve Operator
WR 801497, Emergency Lighting
WR 801628, Emergency Lighting Battery Leak
WR 801707, PORV Temperature Interlock Transmitter Setpoint
WR 801781, Nuclear Instrumentation (CAP T2000-0025)

WR 121300, DH-C-1A boron leak from vent line
WR 798612, NS-V-108A Chiller Setpoint Control
JO 178532, Auto Vent Leaking By, NR-V-0076B

ATTACHMENT 2
LIST OF DOCUMENTS REVIEWED (Cont.)

Nuclear Safety Oversight Audit Reports

S-TMI-99-13, Corrective Actions
S-TMI-99-09, TMI Radiological Controls
S-TMI-00-07, Corrective Actions
S-TMI-00-08, TMI Radiological Controls
S-TMI-00-15, Radwaste Managemement Audit Checklist

Departmental Self Assessments

95911-PA-00-001, Emergency Response Organization Augmentation
95911-PA-00-002, Emergency Action Level Revision Review
95911-PA-00-004, EP Drills, Exercises and Actual Event Evaluation
95928-PA-00-002, Corrective Action
95930-PA-00-05, Switching and Tagging Process
95934-PA-00-001, Annual ALARA Assessment
95934-PA-00-003, Evaluate Skin/Clothing Contaminations
95940-PA-00-004, Use of Equipment Trouble Tags
1920-OB-99-10, CAP Operability and Reportability Determinations
3630-OB-99-002, Primary Chemistry Lab Practices
95928-OB-00-003, Accountable Manager Quality Review of Responses and Corrective Actions
95934-OB-00-006, Radiological Instrument Program (71121.03)
95935-OB-00-024, Rad Material Control in Turbine & Intermediate Bldgs
95935-OB-00-025, Contamination Controls
95935-OB-00-026, Contamination Control Effectiveness
95935-OB-00-027, Worker Response to ED Alarms
95940-OB-00-014, Observation of TSS and SDR Process per CAP T1999-134 Corrective Action

SA-2000-1014, Radioactive Source Storage & Handling
SA-2000-1020, Annual ALARA Assessment
SA-2000-1042, Switching and Tagging Compliance
SA-2000-1044, Radiological Instrumentation Calibration
SA-2000-1046, NRC Inspection Procedures
SA-2000-1058, Procedure Adherence
SA-2000-1059, Security Use of Temporary Lighting

E210-OB-99-001, Component Reliability
E410-OB-99-004, Configuration Control

Non-Cited Violations

1999-009-01, Failure to Implement TS Required Procedure - Decay Heat System Pump.
2000-004-01, Failure to establish an adequate testing procedure for letdown line modification.

ATTACHMENT 2
LIST OF DOCUMENTS REVIEWED (Cont.)

Licensee Event Reports

1999-004-00, Emergency Feedwater Pump Inoperable.
2000-001-00, Automatic Start of EDG 1B - Failure of Fault Pressure Relay Circuit Card.
2000-002-00, Condition Outside Fire Hazards Design Basis for Alternate Shutdown Facility.
2000-003-00, Condition Outside the design Basis For the Control Building Envelope.
2000-004-00, Condition Outside Plant Design Basis for Small Break Loss of Coolant Accident.

Others

Action Item Number 99025, TSC Dose Assessment
Action Item Number 2000042, Siren Testing
EP Drill Critique Report - November 8, 2000
Nuclear Oversight Radiological Monitoring Reports 2000-012, 2000-017, 2000-024 and 2000-033
Operating Experience (OE) Report #10927, Service Water Pump Expansion Joint Water Hammer
Nuclear Review Board meeting minutes for January 6, 2000; May 4, 2000; and September 25, 2000.
Plant Review Group meeting minutes 2000-001 through 061.
TMI internal electronic correspondence, “Keeping NS-V84 Closed,” dated September 25, 1996.

ATTACHMENT 3

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

D. Atherholt	Operations
J. Cohen	Work Management
H. Crawford	Engineering
D. Ethridge	Maintenance
E. Frederick	Regulatory Assurance
E. Fuhrer	Regulatory Assurance
G. Gellrich	Plant Manager
D. Hoskings	Nuclear Oversight
W. Lopkoff	Regulatory Assurance
W. McSorley	Operations Support
B. Merryman	Maintenance
A. Miller	Regulatory Assurance
R. Munz	Nuclear Oversight
S. Queen	Engineering
J. Schork	Plant Review Group
B. Shumaker	Engineering
J. Telfer	Radiological Protection
S. Wilkerson	Engineering

Other

M. Murphy	Pennsylvania Department of Environmental Protection

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

NCV 050000289/2000-010-001	Failure to include two NSCCWS valves in the IST Program per TS 4.2.2 (Section 4OA2.2)

ATTACHMENT 4

LIST OF ACRONYMS USED

AmerGen	AmerGen Energy Company, LLC
CAP	Corrective Action Process document
ESAS	Engineered Safeguards Actuation System
ETTS	Electronic Task Tracking System
IPE	Individual Plant Evaluation
IST	Inservice Testing
LER	Licensee Event Report
NCV	Non-cited violation
NRC	Nuclear Regulatory Commission
NSCCWS	Nuclear Services Closed Cooling Water System
OE	Operating Experience
SDP	Significance Determination Process
SDR	surveillance deficiency reports
TS	Technical Specification
TMI	Three Mile Island Unit 1