August 24, 2001

Mr. Mark E. Warner
Vice President - TMI Unit 1
AmerGen Energy Company, LLC
P.O. Box 480
Middletown, PA 17057

SUBJECT: THREE MILE ISLAND UNIT 1 - NRC INSPECTION REPORT 50-289/01-010

Dear Mr. Warner:

On July 13, 2001, the NRC completed a team inspection of your Three Mile Island facility. The enclosed report documents the inspection findings which were discussed on July 13, 2001, with Mr. G. 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 team identified two findings of very low safety significance (Green), both of which involved violations of NRC requirements. However, because of their very low safety significance and because they were entered into your corrective action program, the NRC is treating these issues as a 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 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, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Three Mile Island 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 Publically Available Records (PARS) component of the NRC's document management system (ADAMS). ADAMS is accessible from the NRC website at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Wayne D. Lanning, Director
Division of Reactor Safety

Docket No. 50-289
License No. DPR-50

Enclosure: Inspection Report 50-289/01-010
cc w/encl:
Amergen Energy Company - Correspondence Control Desk

E. Fuhrer, Regulatory Engineering
J. McElwain, Manager, Regulatory Assurance
G. Gellrich, Plant Manager
M. Gallagher, Director-Licensing
J.A. Benjamin, Licensing - Vice President, Exelon Corporation
TMI-Alert (TMIA)
D. Allard, PADER
M. Schoppman, Framatome

<u>Distribution</u> w/encl (**VIA E-MAIL**):
Region I Docket Room (with concurrences)
D. Orr, SRI - NRC Resident Inspector
H. Miller, RA
J. Wiggins, DRA
J. Rogge, DRP
N. Perry, DRP
T. Haverkamp, DRP
P. Hiland, OEDO
E. Adensam, NRR
T. Colburn, NRR
H. Pastis, NRR
W. Lanning, DRS
L. Doerflein, DRS
M. Modes, DRS
DRS File

DOCUMENT NAME: C:\Program Files\Adobe\Acrobat 4.0\PDF Output\tmi01-010rev2.wpd
After declaring this document “An Official Agency Record” it **will** be released to the Public.
To receive a copy of this document, indicate in the box: "**C**" = Copy without attachment/enclosure "**E**" = Copy with attachment/enclosure "**N**" = No copy

OFFICE	RI/DRS		RI/DRS		RI/DRP		RI/DRS		RI/DRS	
NAME	MModes		JTrapp		JRogge		LDoerflein		WLanning	
DATE	07/31/01		08/01/01		08/01/01		08/21/01		08/22/01	

OFFICIAL RECORD COPY

U.S. NUCLEAR REGULATORY COMMISSION

REGION I

Docket No:	50-289
License No:	DPR-50
Report No:	50-289/01-010
Licensee:	AmerGen Energy Company, LLC (AmerGen)
Facility:	Three Mile Island Station, Unit 1
location:	P.O. Box 480 Middletown, PA 17057
Dates:	June 25, 2001 to July 13, 2001
Inspectors:	M. Modes, Team Leader L. Prividy, Senior Reactor Inspector L. Cheung, Senior Reactor Inspector S. Pindale, Reactor Inspector M. Ferdas, Reactor Inspector K. Cornely, Student (Observer)
Approved by:	Lawrence T. Doerflein, Chief Systems Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000289/2001-010, on 6/25 thru 7/13/2001; AmerGen Energy Company, LLC (AmerGen), Three Mile Island Station, Unit 1, Safety System Design and Performance Capability.

The inspection was conducted by region-based inspectors. The inspection identified two Green findings, both of which were 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. Inspector Identified Findings

Cornerstone: Mitigating Systems

Green. The NRC found that design assumptions, in the form of administrative controls, were not correctly translated into plant procedures for cross connecting the safety related nuclear service river water system (NR) to the non-safety related secondary river water system (SR) in the event of a total loss of SR. The absence of the administrative controls represented a credible impact on safety in that the NR system pumps could have been operated in a runout condition and thus jeopardize the ability of the NR system to perform its safety related function. This issue affects the mitigating cornerstone since the NR system is used during engineered safety (ES) operations to provide cooling water to required heat loads. This issue was determined to be of very low safety significance using the SDP because there was no actual loss of safety function, the very low probability of operating the SR and NR system in a cross connected alignment, and the seismic qualifications of the non-safety related SR system. The failure to incorporate design basis assumptions into procedures for the NR system was considered a non-cited violation of 10 CFR 50, Appendix B. Criterion III, Design Control. (Section 1R21b)

Green. The NRC identified that inadequate acceptance criteria were used in surveillance procedure 1300-3D, "IST of Decay Heat River Water Pumps and Valves," for testing the decay heat (DH) river water pumps after they were replaced in late 1999. The inadequate surveillance acceptance criteria occurred by not correctly translating the design basis parameters of the replacement pumps. As a result, the testing did not assure the pumps would deliver the design basis required flow of 6000 gallons per minute as specified in the Updated Final Safety Analysis Report (UFSAR). The lack of proper acceptance criteria in surveillance procedure 1300-3D had a credible impact on safety because the pumps could have unknowingly degraded below the real acceptable performance limit. Since there was no actual degradation of the DH river water system subsequent to the installation of the new pumps, this issue was determined to be of very low safety significance by the SDP. The failure to incorporate design assumptions into procedures for the DH river water pumps was considered a non-cited violation of 10 CFR 50, Appendix B. Criterion III, Design Control. (Section 1R21b)

Report Details

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R02 Evaluations of Changes, Tests, or Experiments (IP71111.02)

a. Inspection Scope

The team reviewed a sample of safety evaluations (SEs) required by 10 CFR 50.59 for changes to facility systems, structures, and components or procedures as described in the Three Mile Island Unit 1 Updated Final Safety Analysis Report (USFAR). The SEs were selected from a list of changes implemented during the last year. The review was conducted to verify that the changes to the facility or procedures as described in the UFSAR, and test and experiments not described in the UFSAR, were reviewed and documented by the licensee in accordance with 10 CFR 50.59. The team also verified that the changes, tests, and experiments did not require prior NRC approval or a license amendment.

The team also reviewed a sample of changes and tests for which Amergen determined that a safety evaluation was not required. This review was performed to verify that Amergen's threshold for performing safety evaluations was consistent with the requirements of 10 CFR 50.59. Lastly, the team verified that the problems identified with the implementation of the safety evaluation program were entered into the corrective action program.

b. Findings

No findings of significance were identified.

1R21 Safety System Design and Performance Capability (IP 71111.21)

a. Inspection Scope

The team reviewed the design and performance capability of nuclear services river water (NR) and other systems necessary to successfully mitigate selected transients and accident scenarios associated with a loss of NR. The inspection focused on NR, decay heat (DH), nuclear services closed cooling water (NSCCW), intermediate closed cooling water (ICCW), decay heat closed cooling water (DHCCW), and makeup (MU), and associated support systems based on system performance requirements.

The team reviewed the design and licensing basis documents for the NR and MU systems, including the Updated Final Safety Analysis Report (UFSAR), plant Technical Specifications (TS) and system design basis documents (SDBD) to determine the system and component functional requirements during normal and accident conditions.

The team reviewed selected electrical calculations and analyses, and instrument setpoint calculations, to verify that the assumptions were appropriate, that proper engineering methods and models were used, and there was an adequate technical basis to support the conclusions. The review was performed to determine if the design basis was in accordance with the licensing commitments and regulatory requirements; and the design output documents such as electrical drawings and design calculations were correct.

For selected mechanical calculations, the team verified that the assumptions were appropriate and agreed with the current plant configuration, that proper engineering methods and hydraulic models were used, and that there were adequate technical bases to support the conclusions. The team performed independent calculations to evaluate the document adequacy when appropriate. The team verified that the ability of the nuclear services and decay heat river water systems were not adversely affected by changes made as a result of plant modifications.

The team reviewed operator actions assumed following the identification of transient and accident initiation conditions during selected sequences associated with a loss of NR.
The team verified that normal, abnormal, and emergency procedures were consistent with the systems' design and licensing basis, risk, and operating assumptions. In addition, the team reviewed the system interfaces (instruments, controls, and alarms) and the alarm response procedures available to operators to support operator decision making.

The NRC review included implementing activities for licensee commitments made in response to Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment." Other documents reviewed included engineering analyses, operability and safety evaluations, plant modifications, piping and instrumentation drawings (P&IDs), electrical schematics, instrumentation and control drawings, logic diagrams, and instrumentation set points. Plant procedures including surveillance tests and maintenance activities were also reviewed to ensure that they supported the system design and licensing basis.

The operational readiness and material condition of the selected systems were assessed by reviewing appropriate documents, including operating procedures, component maintenance history records, preventive maintenance records, and system health reports. Specifically, the team reviewed maintenance activities on the following equipment: NR pumps, MU pumps, emergency diesel generator (EDG), and selected valves located in the NR and MU system. As part of this effort, the circuit breakers for the NR and MU system pumps and the EDGs were reviewed to determine the adequacy of circuit breaker maintenance. The team also interviewed responsible TMI personnel, including licensed and non-licensed operators, system engineers, and maintenance personnel, regarding the operation and performance of selected systems and components.

Plant walk-downs of the selected systems were performed to verify that the physical installation of the system and components was consistent with design basis document assumptions. During these walk-downs the team examined the design, equipment condition and physical line-up of major components, including pumps, pump motors, motor-operated valve motors, valves, piping, and heat exchangers. The team also verified that the appropriate procedures and equipment were staged at locations to assist operators in performing the appropriate manual actions when required by station procedures.

The team selected open and closed corrective actions taken on the NR, High Pressure Injection (HPI), and MU systems covering a period of approximately two years. The actions were reviewed to determine if the licensee was identifying design issues at an appropriate threshold, entering them in the corrective action program, and taking appropriate actions.

b. Findings

Cross Connecting of Nuclear Services River Water (NR) and Secondary River Water (SR)

The team found that design assumptions were not properly translated into plant procedures for cross connecting the safety related NR and non-safety related SR, on a total loss of SR. This finding was determined to be of very low safety significance (Green) and treated as a non-cited violation (NCV).

The team reviewed design basis document 5430-88-0030 which specified the appropriate administrative controls to prevent NR pump runout and to prevent insufficient cooling water flow to the ICCW and NSCCW systems when cross connecting the NR to the SR system upon a postulated loss of SR. The design basis document outlined five precautions that were necessary to preclude a loss of the NR system and minimize the increase in risk when operating in a crosstie configuration. The team noted that abnormal procedure 1203-19, Rev. 23, “River Water System Failure, Decay Heat and Secondary Services (DR/SR),” did not contain one of the precautions necessary to prevent NR pump runout and subsequent loss of the system. The precaution stated, “prior to cross-tying the systems, the reason for the loss of the SR system has to be determined, to assure that opening the crosstie valves will not cause a loss of the NR system or that loss of the NR system due to the same reason as the SR loss is not imminent.” Procedure 1203-19 attempts to achieve this precaution in step 3.8.3, which instructs the operator to slowly open SR-V-2 prior to cross connecting NR to SR. However, this valve is already open during normal operations. This step in the procedure would preclude the operators from achieving the precaution statement necessary to prevent NR pump runout and thus cause insufficient cooling water flow to the ICCW and NSCCW systems.

This issue was considered more then minor because the failure to translate design basis information into procedures had a credible impact on safety. The existing procedure did not ensure that cross connecting of the two systems would not cause a more severe transient, specifically the loss of the NR system. This issue affects the mitigating cornerstone since the NR system is used during engineered safety (ES) operations to provide cooling water to required heat loads. However, since there was no actual loss of safety function, there was a very low probability of operating in a cross connected configuration, and the SR and NR system had equivalent seismically qualified underground piping portions and in-line components, the issue was determined to be of very low safety significance (Green) and was screened out in Phase 1 of the significance determination process (SDP). The inadequate abnormal procedure was contrary to 10 CFR 50, Appendix B, Criterion III, "Design Control," which requires that design requirements be correctly translated into procedures. The licensee entered this issue into their corrective action program and issued condition report (CAP) T2001-0659, and plans to revise the procedure to ensure that the design requirements are appropriately translated to preclude the possibility of the NR pumps operating in a runout condition. Due to the overall very low safety significance, this violation of 10 CFR 50, Appendix B, Criterion III, was treated as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368). **(NCV 50-289; 50-289/2001-010-01)**.

<u>Inadequate Pump Surveillance Acceptance Standards</u>

The team found that the surveillance test for the DH river water pumps contained inadequate acceptance criteria. This finding was determined to be of very low safety significance (Green) and treated as an NCV.

In late 1999, the DH river water pumps were replaced by plant modification MD-11355-001. The inspectors reviewed surveillance procedure 1300-3D, "IST of Decay Heat River Water Pumps and Valves," which was used to ensure these safety-related pumps do not degrade below the performance assumed in the accident analysis. The surveillance test acceptance criteria were such that a degradation of approximately 3% in the DH river water pumps' performance would have been acceptable. An alert value would not have been reached until the pumps degraded 5%. However, the team determined that the surveillance test did not take into consideration the effects of low river water level and the maximum allowable pressure drops across the trash rakes, traveling water screens, discharge strainers, and heat exchangers. The team determined that when considering the fouled system conditions with the allowed 3% degradation, the DH pumps would not have been able to deliver the 6000 gpm as specified in Table 6B-6 of the UFSAR.

The licensee placed this problem into the corrective action program by issuing CAP No. T2001-0673 on July 6, 2001. Based on preliminary calculations, the licensee concluded that the hydraulically weakest pump (DR-P-1A) could degrade 2 to 3% and deliver the required flow of 6000 gpm under design basis conditions while accounting for instrument error. Because the DH river water pumps were relatively new with no noted degradation, they were considered to be operable, pending the completion of a formal calculation to technically support this conclusion. The licensee completed calculation C-1101-533-E410-013, Rev. 0 dated July 11, 2001, and concluded that the pumps were operable. The team reviewed this calculation and observed the following errors:

- The calculation did not properly account for river water level instrument error.

- The calculation referred to calculation C-1101-533-E510-008, Rev. 2 for the basis for the flow error of 138 gpm associated with instruments DR-DPI-

1303A&B. This was incorrect since the flow error associated with these instruments was not included in calculation C-1101-533-E510-008, Rev. 2.

- The calculation was based on a river water temperature of 90 degrees Fahrenheit (°F). For correctness the calculation should be based on the design basis river water temperature of 95 °F.

The licensee revised calculation C-1101-533-E410-013 to resolve these errors and concluded that the pumps could be allowed to degrade 3% and still be considered operable. The team reviewed the revised calculation and had no further comments.

This issue was considered to be more than minor because the lack of proper acceptance criteria in surveillance procedure 1300-3D had a credible impact on safety. The pump flow rates may have degraded below the design flow rates which could affect the operability of the DH river water system. This issue affects the mitigating cornerstone since the DH system is used for low pressure injection and for long-term decay heat removal. The issue was determined to be of very low safety significance (Green) and was screened out in Phase 1 of the SDP because there was no actual degradation of the DH river water system subsequent to the installation of the new pumps. The failure to incorporate proper design assumptions into procedures for the DH river water pumps was contrary to 10 CFR 50, Appendix B. Criterion III, “Design Control,” which requires that design requirements be correctly translated into procedures. The licensee entered this issue into their corrective action program and issued condition report (CAP) T2001-0673 and plans to revise the procedure with the correct acceptance limit. Due to the overall very low safety significance, this violation of 10 CFR 50, Appendix B, Criterion III, was treated as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy, issued on May 1, 2000 (65FR25368).

4. OTHER ACTIVITIES (OA)

4OA6 Meetings

.1 Exit Meeting Summary

On July 13, 2001, the team presented the inspection results to Mr. G. Gellrich and other members of the licensee’s staff. During the inspection, the team reviewed one proprietary vendor study which was returned to the licensee. The team verified that the inspection report does not contain proprietary information.

KEY POINTS OF CONTACT

Licensee
G. Gellrich - Plant Manager
J. Schork - Regulatory Assurance
J. McElwain - Manager, Regulatory Assurance
S. Queen - Senior Manager, Plant Engineering
R. Masoero - System Engineering
A. Asenpota - TMI NOS
J. Bashista - Engineering
J. Robertson - Plant Operations Director
E. Cartwright - Work Management

NRC
L. Doerflein, Chief, Systems Branch
D. Orr, Senior Resident Inspector

LIST OF ITEMS OPEN, CLOSED, AND DISCUSSED

Opened/Closed

50-289/2001-010-01	NCV	Cross Connecting of Nuclear River Water and Secondary River Water
50-289/2001-010-02	NCV	Inadequate Pump Surveillance Acceptance Standards

LIST OF ACRONYMS

CAP	Corrective Action Program
DH	Decay Heat
DHCCW	Decay Heat Closed Cooling Water
EDG	Emergency Diesel Generator
HPI	High Pressure Injection
ICCW	Intermediate Closed Cooling Water
NCV	Non-Cited Violation
NR	Nuclear Services River Water System
NSCCW	Nuclear Services Closed Cooling Water
P&ID	Piping and Instrumentation Drawing
SDP	Significance Determination Process
SDBD	System Design Basis Documents
SE	Safety Evaluations
SR	Secondary River Water System
TS	Technical Specifications
UFSAR	Updated Final Safety Evaluation Report

DOCUMENTATION REVIEWED

50.59 Documents

SE-113202-868	Peerless River Water Pump Tube Stabilizer
SE-000224-025	EER JO (00170802): Outage 13R SG Inspection
SE-000224-026	OTSG Kinetic Expansion ET Indication Disposition Methods
SE-000224-007	USE of I-600 and I-690 Thermally Treated Plugs
SE-000224-014	Titanium Injection
SE-000224-019	In-situ Pressure Test of OTSG tubes
SE-000210-005	USFAR Section 901 and 905 Revision
SE-000531-010	NR-P-1A/1B/1C Replacement
SE-000531-015	Nuclear Services River Water Leak
SE-000861-001	EDG Maintenance Inspection Frequency Extension
SE-000542-009	Intermediate Closed Cooling Water System Flow Change
SE-000542-012	IC Pump Trip Alarm

Design Drawings

302-082	Emergency Feedwater
302-202	Nuclear Services River Water System Flow Diagram
302-221	Secondary Services Cooling System Closed Cycle
302-610	Nuclear Services Closed Cycle Cooling Water
302-611	Reactor Bldg Normal & Emergency Cooling Water System
302-620	Intermediate Cooling
302-640	Decay Heat Removal
302-645	Decay Heat Closed Cycle Cooling Water
302-660	Makeup and Purification Flow Diagram
302-661	Makeup and Purification Flow Diagram
302-662	Makeup and Purification MU Pump Auxiliary System Flow Diagram
SS-209-742	Makeup Pump MU-P-1A (1B,1C) Low Lube Oil Auxiliary Relays
E-208-169	ES Bus 1E UV and Potential Circuits
E-209-022	Electrical Elementary Diagram for Letdown block valve MU-V-3
E-208-214	Electrical Elementary Diagram, 4160V Switchgear, Makeup Pump MU-P-1C
E-208-216	Electrical Elementary Diagram, 4160V Switchgear, Makeup Pump MU-P-1B
E-208-562	Electrical Elementary Diagram, Makeup Pump C Main Oil Pump, MU-P-3C
E-206-022	One Line and Relay Diagram, 4160V Emergency Safeguard Switchgear
E-206-051	One Line Diagram 250/125V dc and 120V ac Vital Instrumentation

Engineering Calculations

C-1101-531-5310-010,	Nuclear River Water System Performance
C-1101-700-E510-010	TMI-1 AC Voltage Regulation Study and Appendix 8.4 Determination of Degraded Voltage Relay Tolerances
C-1101-531-E510-016	TMI-1 Nuclear Service River Water Pump IST - Pump Head Flow and Cooler Inlet Pressure Instrument Error
TDR 1064	Technical Data Report, TMI-1 Voltage and Frequency Response Study
C-1101-533-E540-004	Decay Heat River Water System Performance
C-1101-533-E510-008	Decay Heat River Water System Pump IST - Pressure and Flow Instrument Error
C-1101-543-E410-015	Decay Heat Closed Cooling Water Heat Exchanger (DC-C-2B) 13R Test Evaluation

C-1101-533-E410-013	Decay Heat River Water System Hydraulic Performance Using Field Test data
C-1101-211-E220-097	Interim Torque Setpoints for MU-V-16D
C-1101-531-E310-015	Low Intake Level With Silt Accumulation
C-1101-531-5310-010	Nuclear River Water System Performance
C-1101-531-E540-011	Analysis of Nuclear Services River Closed Cooling Water Heat Exchangers
C-1101-541-5360-020	Effect of 95 F River Water on Nuclear Services Closed Cooling Water System
C-1101-531-E510-016	Nuclear Services River Water Pump IST - Pump Head-Flow and cooler Inlet pressure Instrument Error

Design Basis Documents

FSAR Section 6	Engineered Safeguards
FSAR Section 9.1	Makeup and Purification
FSAR Section 9.5	Decay Heat Removal System
FSAR Section 10.6	Emergency Feedwater System
SDBD-T1-531	System Design Basis Document for Nuclear Services River Water, Nuclear Services Closed Cooling Water, and Intermediate Closed Cooling Water
SDBD-T1-211	System Design Basis Document for Makeup and Purification
TDR 1183	Single Failure Analysis of Nuclear Services Closed Water and Nuclear River Water Systems
TMI Memorandum 5430-88-0030	Response to Task Request No. BT5017

Condition Reports

T2001-0659	1203-19 has Technical Inadequacy
T2000-0991	NR and SR Pump Vacuum Breaker PM
T2000-0391	NR-V-1B Found Out of Position
T2000-0164	Operability Evaluation MU-V-10
T2000-0858	Air Pressure to Closing Side of MU-V-3 Found Low
T2000-0065	Tube Chatter in SSCCW and NSCCW HX
T2000-0667	SR-P-0001C Experienced Increase Noise and Vibration
T2001-0552	NS-P-1A Motor Made Loud Noise
T2000-0886	NR-S-1C Partially Plugged
T2001-0279	Air Pressure to MU-V-3/SV1 is Above Design Value
T2001-0347	NS-P-1A Coupling Bolts Not Torqued to Proper Value
T2001-0577	NR-P-1A Breaker Found Degraded During PM
T2001-0673	Decay Heat River Water Pump Acceptance Criteria
T1999-0066	DR-P-1A Inoperable Due to High Strainer Pressure Drop
T2000-0603	Dead Clams Found in NS-C-1D Heat Exchanger
T2000-0564	MOV Calculation Revision Needed for MU-V-16D
T2000-0391	NR-V-1B Found Out of Position
T2000-0149	MU-P-1B Declared Inoperable during Surveillance Testing
T2000-0065	Tube Chatter in Secondary and Nuclear Services Closed Cooling Water Heat Exchangers
T2001-0028	Probable Leak Due to Micro Biological Influenced Corrosion on NS-C-1A Backwash Line
T2000-0886	NR-S-1C Partially Plugged

<u>Procedures</u>

1097	Corrective Action Process
1001A.2	Procedure Performance Improvement System
1041	IST Program Requirements
1202-38	Nuclear Services River Water Failure
1202-17	Loss of Intermediate Cooling System
1203-34	Control Building Ventilation System
1203-19	River Water System Failure, Decay Heat and Secondary Services (DR/SR)
1203-15	Loss of RC Makeup/Seal Injection
1203-20	Nuclear Services Closed Cooling System Failure
MAP B	Main Annunciator Panel B
MAP C	Main Annunciator Panel C
MAP D	Main Annunciator Panel D
MAP F	Main Annunciator Panel F
1210-1	Reactor Trip
1210-10	Abnormal Transients Rules, Guides, and Graphs
1210-6	Small Break LOCA Cooldown
1300-4C	IST of NSRW Pumps and Valves During Refuelings
1300-3D	IST of Decay Heat River Water Pumps and Valves
1300-4C	IST of Nuclear Services River Water Pumps and Valves during Refuelings
1301-9.7	Intake Pump House Floor, Silt Accumulation
1301-6.7	Monitoring of Silt Buildup in River Water Screen House
1104-32	Decay Heat River Water System
1104-30	Nuclear River Water System
1104-8	Intermediate Cooling System
1104-30	Nuclear River Water
1104-33	Screen House Equipment
1104-31	Secondary Service River Water System
1104-2	Makeup and purification System
1202-44	Low Water Level in the Intake Structure Pump House
M-144	Heat Exchanger Inspections and Cleanings
1410-V-18	Check Valve Maintenance
1303-5.2	Emergency Loading Sequence and HPI Logic Channel/Component Test
1302-5.31A	4160V D and E Degraded Grid Under-voltage Relay System Calibration
202/3	Makeup and Purification System Functional Test (Functional Testing of MU-V-18)

<u>Modification Packages</u>

MD-H440-001	ES Trip of the Non-ES Selected NR Pump
MD-G977-001	River Water System Strainer DP Switch Upgrade