UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

October 22, 2001

William T. Cottle, President and
Chief Executive Officer
STP Nuclear Operating Company
P.O. Box 289
Wadsworth, Texas 77483

SUBJECT: SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 50-498/01-05; 50-499/01-05

Dear Mr. Cottle:

On September 22, 2001, the NRC completed an inspection at your South Texas Project Electric Generating Station, Units 1 and 2, facility. The enclosed report documents the inspection findings which were discussed on September 25, 2001, with you 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. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC has identified two issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC has also determined that violations are associated with these issues. These violations are being treated as noncited violations (NCVs), consistent with Section VI.A of the Enforcement Policy. These NCVs are described in the subject inspection report. If you contest the violation or significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the South Texas Project Electric Generating Station, Units 1 and 2, facility.

Since September 11, 2001, South Texas Project 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 STP Nuclear Operating Company. 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, its enclosure, and your response will be made 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).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

C. E. Johnson, Chief
Project Branch A
Division of Reactor Projects

Dockets: 50-498
50-499
Licenses: NPF-76
NPF-80

Enclosure:
NRC Inspection Report
50-498/01-05; 50-499/01-05

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-498 50-499
Licenses:	NPF-76 NPF-80
Report Nos:	50-498/01-05 50-499/01-05
Licensee:	STP Nuclear Operating Company
Facility:	South Texas Project Electric Generating Station, Units 1 and 2
Location:	FM 521 - 8 miles west of Wadsworth Wadsworth, Texas 77483
Dates:	June 24 through September 22, 2001
Inspectors:	N. F. O'Keefe, Senior Resident Inspector G. L. Guerra, Resident Inspector T. O. McKernon, Senior Operations Engineer, Operations Branch P. C. Gage, Senior Operations Engineer, Operations Branch J. S. Dodson, Health Physicist D. R. Carter, Health Physicist
Approved By:	C. E. Johnson Chief, Project Branch A
Attachment:	Supplemental Information

SUMMARY OF FINDINGS

South Texas Project Electric Generating Station, Units 1 and 2
NRC Inspection Report 50-498/01-05; 50-499/01-05

IR 05000498-01-05; IR 05000499-01-05; on 06/24-09/22/2001; STP Nuclear Operating Company; South Texas Project Electric Generating Station; Units 1 & 2. Integrated Res/Reg Rpt; Maint risk assess, event fwp, licensed operator requal prog, rad monitoring inst, & ALARA plan & controls.

The inspection was conducted by resident inspectors, and region based operator licensing and plant support inspectors. The inspection identified two Green issues and two noncited violations. The significance of issues is indicated by their color (Green, White, Yellow, or Red) and was determined by the Significance Determination Process in Inspection Manual Chapter 0609. 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 licensee did not recognize that performing maintenance on the Train 1C engineered safety features (ESF) load sequencer rendered Auxiliary Feedwater (AFW) Pump 1C inoperable. A defective new part was not bench-tested, and caused a load shed and deenergized an ESF bus when the load sequencer was energized for testing. The bus had to be manually reenergized because the associated standby diesel generator was out of service. A noncited violation was identified for Work Order 212619, a procedure required by Technical Specification 6.8.1 and Regulatory Guide 1.33, which was inappropriate to the circumstances. This issue was in the licensee's corrective action program under Condition Report 01-14840.

  This issue had an actual impact on safety because auxiliary feedwater was unintentionally made inoperable and nonfunctional. The violation for the procedure inappropriate to the circumstances was more than minor because of this actual impact on safety. The finding was of very low safety significance (Green) because only one of four trains of AFW was affected, impacting only the mitigation system cornerstone (Section 1R13.2).

Cornerstone: Event Followup

- Green. Operators failed to recognize that two routine evolutions using the chemical and volume control system conflicted. When they attempted to add borated water to the system, only pure water was added, challenging operators to take action to avoid an unintended power increase above 100 percent. The inspectors concluded that the root cause of this event was a failure to recognize that the plant was not in the configuration required by the procedure in use, in part because of a culture that permitted a loose interpretation of what constituted the required system alignment. The licensee's corrective action program underclassified the significance of this event, and as a result did not adequately identify the cause. It was initially treated as minor because operators were able to negate the effect of the error. The inspectors concluded that this should have been treated as a reactivity management event as defined in the licensee's procedures. Failure to follow 0POP02-CV-0001, "Makeup to the Reactor Coolant System," Revision 17, was a violation of Technical Specification 6.8.1 and Regulatory Guide 1.33. This violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy (Reference Condition Reports 01-14307 and 01-14309).

The licensee calculated that, if left uncorrected, power could have increased to just over 100.3 percent, which would not have challenged any safety limits. An inadvertent dilution is an initiating event analyzed in the Updated Final Safety Analysis Report, Chapter 15, and this event was bounded by that analysis. However, this issue was determined to be more than minor because the violation suggested a programmatic problem in procedure adherence that could have a realistic potential safety or regulatory impact. If left uncorrected, this violation would become a more significant safety and regulatory concern, because understanding and properly adhering to approved procedures is a key element of human performance necessary to support reactor safety. This finding was determined to have very low safety significance (Green) because the operators were able to negate the effect of the error (Section 4OA3).

B. Licensee Identified Violations

Two violations of very low safety significance were identified by the licensee and reviewed by the inspectors. Corrective actions taken or planned by the licensee appear reasonable. These violations are listed in Section 4OA7 of this report.

Report Details

Plant Status

Unit 1 began this inspection period at full power. End-of-cycle coastdown operation began on September 16, 2001, with power being slowly lowered until it reached 94 percent by the end of the inspection.

Unit 2 began the inspection period at full power. On June 29, 2001, power was reduced to 83 percent to perform corrective maintenance on Turbine Governor Valve 4 and was returned to full power the same day. On August 5, power was reduced to 95 percent to perform repairs on the qualified data processing system and was restored to full power the same day. On August 12, power was again reduced to 80 percent to perform similar corrective maintenance on Turbine Governor Valve Number 4 and Steam Generator Feed Pump 23 and was returned to full power the same day. The unit remained at full power through the remainder of the inspection.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdown

a. Inspection Scope

On July 10, 2001, the inspectors performed a partial system walkdown of the Unit 2 Emergency Core Cooling System, Train A while Train B was removed from service for planned maintenance. The inspectors used Plant Operating Procedure 0POP02-SI-0002, "Safety Injection System Initial Lineup," Revision 13, and Piping and Instrument Diagram 5N129F05013 "Safety Injection System," to verify that the required standby and support systems were in a proper standby lineup. The inspectors also examined component material condition.

On July 18, 2001, the inspectors performed a partial system walkdown of the Unit 1 fuel handling building emergency exhaust system, Train A while Train B was removed from service for planned maintenance. The inspectors used Plant Operating Procedure 0POP02-HF-0001, Revision 17, "Fuel Handling Building Ventilation," to verify that the required standby and support systems were in a proper standby lineup. The inspectors also examined component material condition.

On September 8, 2001, the inspectors performed a partial system walkdown of the Unit 2 steam generator power operated relief valve (SG PORV) system while Train D was out of service. Inspectors used system alignments from Procedure 0POP02-MS-0001, "Main Steam System," Revision 17, and Piping and Instrument Diagrams 5S109Z40079 and 5S109F00016 to verify that the required trains were in a proper standby lineup. The inspectors also verified that each train had the proper setpoint entered in the qualified data processing system.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Fire Area Walkdowns

a. Inspection Scope

The inspectors used Inspection Procedure 71111.05 to evaluate the control of transient combustibles and ignition sources. The licensee's fire preplans and Fire Hazards Analysis Report were used to identify important plant equipment, fire loading, detection and suppression equipment locations, and planned actions to respond to a fire in each of the plant areas selected. The inspection included observing the material condition and operational lineup of fire protection systems and fire barriers used to prevent fire damage or propagation. The following plant areas were inspected:

- Unit 1 radiation protection office and changing area in the mechanical auxiliary building (Fire Zone Z147)

- Unit 1 refueling water storage tank and reactor makeup water storage tank rooms (Fire Zones Z103 and Z104)

- Unit 1 Train D main steam line and auxiliary feedwater rooms (Fire Zones Z400 and Z409)

- Unit 2 auxiliary shutdown panel room (Fire Zone Z071)

- Unit 2 Train B cable spreading area (Fire Zones Z047)

- Unit 2 Train D main steam line and auxiliary feedwater rooms (Fire Zones Z400 and Z409)

b. Findings

No findings of significance were identified.

.2 Fire Brigade Drills

a. Inspection Scope

The inspectors observed an announced fire brigade drill on August 21, 2001, to evaluate the readiness of the licensee's personnel to fight fires. The fire was simulated to be in the Unit 1 Emergency Bus 1L switchboard. Licensee performance was evaluated against criteria listed in Inspection Procedure 71111.05. The inspectors observed the predrill brief, and then observed the execution of the drill from various areas of the plant.

b. Findings

No findings of significance were identified.

1R06 Flood Protection (71111.06)

a. Inspection Scope

The inspectors performed inspections of the electrical auxiliary building in Unit 1 to verify that the licensee's flood mitigation plans and equipment were consistent with the licensee's design requirements and risk-analysis assumptions. The inspectors reviewed the Updated Final Safety Analysis Report, the Individual Plant Examination for External Events Report, and various historical license basis documents to evaluate the internal and external flooding design and how current station procedures support that design. The inspectors also walked down the reactor containment building tendon gallery as a potential source of internal flooding to the electrical auxiliary building. The inspectors inspected the perimeter of the main cooling reservoir embankment to observe its condition. The following additional documents were also reviewed:

- 0POP04-ZO-0002, "Natural or Destructive Phenomena Guidelines," Revision 15

- Condition Report Engineering Evaluation 97-9007-2, "Evaluation of Flooding Frequency"

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

.1 Biennial Review of Licensed Operator Requalification Program

a. Inspection Scope

Examination security measures and procedures were evaluated for compliance with 10 CFR 55.49. Maintenance of license conditions was evaluated for compliance with 10 CFR 55.53 by review of facility records, procedures, and tracking systems for licensed operator training, qualification, and watchstanding. Remedial training and examinations for examination failures were reviewed for compliance with facility procedures and responsiveness to address areas failed.

In addition, the inspectors (1) interviewed eight personnel (five reactor operators, two instructors/evaluators, and a training manager) regarding the policies and practices for administering examinations; (2) observed the administration of two dynamic simulator scenarios to three requalification crews by facility evaluators; (3) observed the unit operations division managers involvement in crew and individual evaluations; and, (4) observed three facility evaluators administer five job performance measures. Job performance measures were observed being performed by an average of three requalification candidates. The inspectors also reviewed the remediation process for one crew and one individual who failed the simulator portion of the operating examination.

b. Findings

No findings of significance were identified.

.2 Quarterly Review of Licensed Operator Requalification Training

a. Inspection Scope

The inspectors evaluated the performance of licensed operators during training at the licensee's simulator facility on August 8, 2001. The inspectors observed the performance of two operating crews, one from each unit, for clarity and formality of communications, correct use of procedures, high risk operator actions, and the oversight and direction provided by the shift supervisor. The inspectors verified the licensee's use of emergency action levels for proper emergency classification.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Functional Failure Review (71111.12)

a. Inspection Scope

The inspectors independently verified that the licensee properly implemented 10 CFR 50.65, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," for the following equipment performance problems:

- Unit 2 supply breaker to ESF Bus E2B failure (Work Authorization Number (WAN) 202561, Condition Report (CR) 01-4926)
- Unit 1 control power breaker found open on Essential Chiller 11C (WAN 209239, CR 01-10365)
- Unit 1 and 2 loss of integrated computer system data highway (WAN 209613, CR 01-10359)
- Unit 2 Essential Chiller 21B load cycling due to prerotation vane control problems (WAN 211235, CR 01-11834)
- Unit 2 balance of plant diesel generator breaker failed to shut (WAN 212461, CR 01-12649)
- Unit 1 Train C ESF load sequencer failure following corrective maintenance (WAN 212619, CR 01-12954, 01-14685)

- Unit 1 Train C component cooling water containment isolation valve CC-0198 local leak rate test failure (WAN 211874, CR 01-12043)

The inspectors focused the review on whether the structures, systems, or components (SSCs) that experienced problems were properly characterized in the scope of the program. The inspectors also reviewed whether the SSC failure or performance problem was properly characterized. The inspectors assessed the adequacy of the licensee's significance classification for the SSC. This included the appropriateness of the performance criteria established for the SSC (if applicable), and the adequacy of corrective actions for SSC's classified in accordance with 10 CFR 50.65 a(1) as applicable.

For the containment isolation valve test failure, the inspectors reviewed the test history of all containment isolation check valves of this design to determine the failure history due to questions about the licensee's having multiple test methods for these valves. The inspectors checked to see if all failures were properly recorded in equipment material history and were evaluated by the licensee's Maintenance Rule program.

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessment and Emergent Work Evaluation (71111.13)

.1 Multiple Activities Reviewed

a. Inspection Scope

The inspectors reviewed selected activities regarding risk evaluations and overall plant configuration control. The inspectors discussed emergent work issues with work control personnel and reviewed the potential risk impact of these activities to verify that the work was adequately planned, controlled, and executed. The activities reviewed were associated with:

- (Unit 1) Main Steam Isolation Valve 1D sealant injection repair
- (Unit 1) Extended allowed outage time for Essential Cooling Water Pump 1C
- (Unit 2) Train A electrical auxiliary building ventilation fan preventive maintenance
- (Unit 2) Train B safety injection modification and freeze seal controls
- (Unit 2) Main Turbine Governor Valve 4 repair work (on June 6, 2001, and again on August 12)
- (Common) Maintenance performed on two diesel fire pumps

b. Findings

No findings of significance were identified.

.2 Impact of Work on a Load Sequencer Not Properly Identified (71111.13, 71153)

a. Inspection Scope

The inspectors assessed the licensee's planning of scheduled maintenance on the Unit 1 Train C ESF load sequencer on September 18, 2001. The work resulted in an unexpected load shed and loss of power to a bus. Operators involved in the event, work schedulers involved in scheduling the load sequencer maintenance, and personnel from the risk and reliability group were interviewed. The inspectors reviewed the maintenance risk projections and actual results. The Maintenance Rule program coordinator was interviewed to identify how the licensee's program implemented 10 CFR 50.65 (a)(4). The following documents were reviewed:

- Work package for WAN 212619 for the Train C ESF load sequencer
- 0POP01-ZO-0006, "SDG, ECW Extended Allowed Outage Time," Revision 7
- 0PGP03-ZO-ECO1, "Equipment Clearance Orders," Revision 11
- 0POP04-AE-0001, "Loss of Any 13.8 KV or 4.16 KV Bus," Revision 19
- 0PGP03-ZA-0091, "Configuration Risk Management Program," Revision 2

The inspectors also reviewed the guidance provided for using the licensee's online risk calculator (RASCAL), "RASCAL System Guidelines," Revision 2.

b. Findings

The licensee did not recognize that performing maintenance on the Train 1C ESF load sequencer rendered Auxiliary Feedwater Pump 1C inoperable. A Green noncited violation was identified for an inadequate procedure required by Technical Specification 6.8.1 and Regulatory Guide 1.33.

Basic System Description

Each ESF train was capable of being powered from either offsite power or the standby diesel generator (SDG). In the event of a loss of offsite power or a condition requiring a safety injection actuation, the load sequencer would strip the loads from the bus (loss of offsite power scenario only), start the SDG, shut its output breaker if there was not already power on the bus, and start loads in a predetermined timing sequence. The automatic sequencing of loads was part of the safety function of each load.

Sequence of Events

On September 18, 2001, the licensee deenergized the Unit 1 Train C ESF load sequencer for corrective maintenance. After a module was replaced, the sequencer was reenergized with the intent of performing postmaintenance testing. Immediately upon the restoration of power, the sequencer unexpectedly initiated a load shed.

Standby Diesel Generator 13 did not start and loads did not sequence onto the bus because SDG 13 was out of service for planned maintenance.

Operators responded in accordance with procedure 0POP04-AE-0001, “Loss of Any 13.8 KV or 4.16 KV Bus,” Revision 19. However, the procedure assumed the deenergized ESF buses would be restored from the associated SDG, and did not have steps for reenergizing the bus from the normal offsite source. Failure to have a procedure to energize an emergency AC bus from its normal source of power was a violation of Technical Specification 6.8.1 and Regulatory Guide 1.33. This licensee identified violation will not be cited in accordance with Section VI.A.1 of the NRC enforcement policy and is listed in Section 4OA7 (NCV 498/2001005-03).

Operators restored power to the bus from offsite power after first placing all associated load control hand switches in the pull-to-lock position and steam generator power operated relief valves in manual. These actions disable any automatic starting. Power restoration was performed at the direction of the shift supervisor under the administrative controls of 0PGP03-ZA-0010, “Performing and Verifying Station Activities,” Revision 25

Issues

Prior to the work, maintenance planners and control room operators recognized only part of the impact of removing the sequencer from service. Disabling of the loss of offsite power function was recognized and all associated equipment was properly declared inoperable. However, the fact that the safety injection function of the sequencer was also disabled was not recognized or accounted for. The Technical Specification Bases specified that the licensee declare the associated loads inoperable and enter the related Technical Specification action statements when the sequencer was deenergized. While most of those loads were already inoperable due to ongoing maintenance, auxiliary feedwater was incorrectly considered operable at the time the sequencer work was being performed. Similarly, the train was considered nonfunctional because operators did not identify, brief, or assign actions to individuals which would have permitted them to consider the system functional based on manual actions in lieu of automatic actions for Maintenance Rule purposes.

The inspectors determined that the work package for performing the sequencer maintenance had inconsistencies and did not adequately identify the impact of the work. The work package cover sheet incorrectly stated that no limiting condition for operation was impacted by the work, but a scope statement remarked that the sequencer would be unreliable during the performance of the work. Also, the inspectors determined that the load sequencer had no means of testing off-line, blocking the output, or determining the logic state prior to restoring power. This meant that the licensee could not avoid an actuation if the system malfunctioned due to the maintenance when power was restored for postmaintenance testing. In this case, the new part had a manufacturing defect which caused the sequencer to initiate a load shed (an ESF actuation and loss of power to safety-related mitigation equipment). Train C AFW was inoperable for 2 days until the sequencer was repaired. The inspectors concluded that even though the system design did not provide a feature to protect against an inadvertent actuation, the licensee had the capability to detect the problem with the new part by performing shop testing. The prejob brief recognized the potential for an ESF actuation, but no protective measures were taken to avoid one.

Based on the above, Work Package 212619 was considered to be a procedure that was inappropriate to the circumstances, which was a violation of Technical Specification 6.8.1 and Regulatory Guide 1.33. This nonrepetitive violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy and is in the licensee's corrective action program as Condition Report 01-14699 (NCV 498/2001005-01).

This issue had an actual impact on safety because auxiliary feedwater was unintentionally rendered inoperable and nonfunctional. The violation for the procedure inappropriate to the circumstances was more than minor because of this actual impact on safety. The finding was of very low safety significance (Green) because only one of four trains of AFW was affected, impacting only the mitigation system cornerstone.

1R14 Personnel Performance During Nonroutine Plant Evolutions (71111.14)

.1 Internal Inspection of Refueling Water Storage Tank

a. Inspection Scope

The inspectors observed the internal inspection of the Unit 1 refueling water storage tank by a remotely operated submersible camera. The tank inspection was conducted to complete action on a commitment to the NRC in response to a very small crack in the tank bottom plating. The purpose of the inspection was to verify that the crack size, location and growth rate were consistent with a licensee analysis which had concluded that the operability of the tank was not affected. The inspectors reviewed the Work Package (WAN 208885). The history of previous external inspections and engineering evaluations were documented in CRs 97-14680 and 97-16031. The inspectors reviewed the licensee's administrative controls for this infrequent evolution, including the basis for maintaining the tank operable during the inspection.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors used Inspection Procedure 71111, Attachment 15, to review selected operability evaluations conducted by the licensee during the report period involving risk-significant systems or components. The inspectors evaluated the technical basis for the licensee's operability determination, verified that appropriate compensatory measures were implemented, and verified that the licensee considered other preexisting conditions, as applicable. Additionally, the inspectors evaluated the adequacy of the licensee's problem identification and resolution program as it applied to operability evaluations. Specific operability evaluations reviewed are listed below:

- (Unit 2) Safety Injection Motor Operated Valve (MOV-16B) test data outside required values (WAN 122332, Condition Report Engineering Evaluation 01-10693-03)
- (Unit 1) Digital rod position indication power supply inadvertent shutdown (CR 01-12132)
- (Unit 2) Steam Generator PORV 2D hydraulic leak (CR 01-11436)
- (Common) Diesel Fire Pump 1 surveillance test failure (CR 01-11084)
- (Unit 2) Safety injection accumulator pressure transmitter (CR 01-12400)
- (Unit 1) Pipe break with single failure could cause auxiliary feedwater storage tank to be drained (CR 01-12051)

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors witnessed or reviewed the results of postmaintenance testing for the following maintenance activities:

- (Unit 1) Train B fuel handling building ventilation exhaust fan maintenance (WAN 129247, 181229, 209182)
- (Unit 2) Auxiliary Feedwater Pump 24 maintenance (WAN 180138)
- (Unit 2) Essential Chiller 22C maintenance (WAN 211919, 203249)
- (Unit 2) Auxiliary Feedwater Pump 22 motor, pump, and valve preventive maintenance (WAN 181041, 181052, 181053)
- (Unit 1) ESF load sequencer corrective maintenance (WAN 212619)

In each case, the associated work orders and test procedures were reviewed to determine the scope of the maintenance activity and determine if the test adequately tested components affected by the maintenance. The Updated Final Safety Analysis Report, Technical Specifications, and Design-Basis Documents, were also reviewed to determine the adequacy of the acceptance criteria listed in the test procedures.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

a. Inspection Scope

The inspectors observed portions of the new fuel receipt activities for the Unit 1 Outage 1RE10. These included shipping cask and fuel inspections and storage into the spent fuel pool. Inspection Procedure 71111.20 was used as a guide. Performance was compared against the requirements in the licensee’s fuel receipt, fuel handling, contamination control, and foreign material exclusion procedures. The inspectors reviewed the licensee’s corrective actions in response to identifying a damaged tamper seal (CR 01-14456), some foreign material (CR 01-14116), and an indentation in a grid strap (CR 01-14444).

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors evaluated the adequacy of periodic testing of the following important nuclear plant equipment. This included aspects such as preconditioning, the impacts of testing during plant operations, the adequacy of acceptance criteria including test frequency and test equipment accuracy, range and calibration, procedure adherence, record keeping, the restoration of standby equipment, and the effectiveness of the licensee’s problem identification and correction program. The inspectors observed or reviewed the following tests:

- (Unit 1) 0PSP10-ZG-0004, “End of Life Moderator Temperature Coefficient Measurement,” Revision 5
- (Unit 1) 0PSP03-AF-0007, “Auxiliary Feedwater Pump 14 Inservice Test,” Revision 17
- (Common) 0PTP03-FP-0106, “Fire Protection Water System Functional Test,” Revision 5, on Diesel Fire Pump 1
- (Unit 2) 0PSP03-RS- 0001, “Monthly Control Rod Operability,” Revision 11
- (Unit 1) 0PSP03-FW-0517, “Steam Generator Narrow Range Level Loop Calibration,” Revision 6
- (Unit 1) 0PSP11-CC-0003, “LLRT: M-27 RCFC 1B/2B Supply,” Revision 9

- (Unit 1) 0PSP11-CC-0006, “LLRT: M-24 RCFC 1C/2C Supply,” Revision 7

Following observations of testing one containment penetration and discussions of the results on another penetration with the test engineer, the inspectors reviewed aspects of the licensee’s test methodology for containment cooling water containment isolation valve local leak rate testing (LLRT) with subject matter experts in NRR. The results of Condition Report Engineering Evaluation 01-13456-1 were also reviewed. The following testing program documents were also reviewed:

- 0PEP07-ZE-0002, “LLRT Rig Operation,” Revision 5
- 0PEP07-ZE-0008, “Non-Intrusive Check Valve Testing,” Revision 4
- 0PGP03-ZE-0021, “Inservice Testing Program for Valves,” Revision 15
- 0PSP11-ZA-0005, “Local Leakage Rate Test Calculations, Guidelines, and Program,” Revision 13

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed Temporary Modification T1-01-4045-20, “Overpower Rod Stop Alarm,” Revision 0, following Inspection Procedure 71111, Attachment 23, with respect to design-bases documentation, approvals, and tracking. The inspectors reviewed the 10 CFR 50.59 screening, updated procedures, and drawings (WAN 201818).

The inspectors also reviewed Temporary Modification T2-01-10019-2, “Temporary Power to Distribution Panel DPL-434,” Revision 0, following Inspection Procedure 71111, Attachment 23, with respect to design-bases documentation, approvals, and tracking. The inspectors reviewed the 10 CFR 50.59 screening, updated procedures, and drawings (WAN 411017).

b. Findings

No findings of significance were identified.

1EP1 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors evaluated an emergency preparedness drill conducted on July 26, 2001, using Inspection Procedure 71114, Attachment 6. This evaluation included reviewing the scenario and drill objectives, observing licensee performance in the emergency facilities, observing the licensee's critique, and discussing observations and the licensee's findings with emergency preparedness managers. The licensee's Management Critique document was reviewed to determine whether the licensee was effective in identifying and addressing performance problems. Emphasis was placed on confirming proper event classification and timely reporting.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety (OS)

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspectors interviewed radiation workers and radiation protection personnel throughout the radiologically controlled area and conducted independent radiation surveys of selected work areas. No work was performed in high exposure or high radiation areas during this inspection. Therefore, this aspect of the above procedure could not be evaluated. The following items were reviewed and compared with regulatory requirements to assess the licensee's program to maintain occupational exposure as low as is reasonably achievable (ALARA):

- ALARA program procedures
- Radiation Protection Department Self-Assessments
- Processes used to estimate and track exposures
- Plant collective exposure history for the past 3 years, current exposure trends, and 3-year rolling average dose information
- Four radiation work permit packages for refueling outage work activities that resulted in the highest personnel collective exposures during Refueling Outage 2RE08 (Steam

Generator Primary - Eddy Current Testing, Steam Generator Replacement - Feedwater Baseplate Modification, Reactor Head Disassembly/Reassembly, and Steam Generator Primary Nozzle Dams)

- Use of engineering controls including all temporary shielding installations
- Hot spot tracking and reduction program
- Radiological work planning
- A summary of ALARA and radiological worker performance related corrective action reports written since March 1, 2001, and 19 specific Condition Reports (01-3621, 01-3767, 01-3822, 01-3951, 01-4052, 01-4135, 01-4152, 01-4269, 01-4336, 01-4491, 01-4597, 01-4602, 01-4726, 01-5128, 01-5252, 01-5808, 01-7820, 01-13045, and 01-13810)
- Declared pregnant worker dose monitoring controls
- 2000 Annual ALARA Report

b. Findings

No findings of significance were identified.

2OS3 Radiation Monitoring Instrumentation (71121.03)

a. Inspection Scope

The inspector interviewed cognizant licensee personnel and reviewed the following items to determine: (1) the accuracy and operability of radiation monitoring instruments that are used for the protection of occupational workers, and (2) the adequacy of the program to provide self-contained breathing apparatus (SCBA) for entering and working in areas of unknown radiological and/or potential immediately dangerous to life and health (IDLH) areas.

- Calibration, operability, and alarm setpoints, when applicable, of portable radiation detection instrumentation, selected area radiation monitors (N1RA-RE-8052, N2RA-RE-8072, N2RA-RE-8094, and N1RA-RE-8098), temporary area radiation monitors, continuous air monitors, containment high range monitors (A1RA-RT-8050 and C2RA-RT-8051), whole-body counting equipment, electronic alarming dosimeters, teledosimetry, and personnel contamination monitors
- Calibration expiration and source response check currency on radiation detection instruments staged for use
- Calibration source accountability and traceability

- The licensee’s capability for refilling and transporting self-contained breathing apparatus air bottles to and from the control room and operations support center during emergency conditions

- Control room operator and emergency response personnel training and qualifications for use of self-contained breathing apparatus

- The status and surveillance records of self-contained breathing apparatuses staged and ready for use in the plant

- Selected exposure-significant radiological incidents that involved internal exposures, radiation monitoring instrument deficiencies, or self-contained breathing apparatuses since the last inspection in this area

- Condition Reports (2000-17423, 18450, 18486, 18794, and 18817; 2001-1830, 2007, 2009, 2922, 3829, 4028, 4088, 4326, 4459, 4661, 4686, 5107, 6548, 6857, 7705, 7706, 8606, 8901, 9465, and 10174)

There were no licensee self-assessments or audits focusing on radiological incidents that involved personnel internal exposures, radiation monitoring instrument deficiencies, or self-contained breathing apparatuses since the last inspection in this area.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification (71151)

.1 Mitigating Systems

a. Inspection Scope

The inspectors reviewed performance indicators for the period from June 2000 through June 2001, to assess the accuracy and completeness of the indicator reporting. The inspectors used Nuclear Energy Institute (NEI) 99-02, “Regulatory Assessment Performance Indication Guideline,” Revision 1, as guidance for this inspection. The inspectors also reviewed 0PGP05-ZN-0007, “Preparation and Submittal of NRC Performance Indicators,” Revision 1, and “Safety System Performance Indicator Desktop Guidelines,” Revision 0. The following performance indicators were reviewed:

- Safety system functional failures

- Safety system unavailability for the following systems:

  Emergency power
  High head safety injection

Auxiliary feedwater
Residual heat removal

b. Findings

No findings of significance were identified.

.2 Initiating Events

a. Inspection Scope

The inspectors reviewed performance indicator data reported by the licensee in order to assess the accuracy and completeness of the information. The inspectors used Nuclear Energy Institute (NEI) Guidance NEI 99-02, “Performance Indicator Verification,” Revision 1, as guidance for this inspection. Data was reviewed for the following indicators for both units for the second through fourth quarters of 2000, and the first quarter of 2001:

- Unplanned scrams per 7000 critical hours
- Scrams with loss of normal heat removal
- Unplanned power changes per 7000 critical hours

Note: This inspection was performed in April 2001, and inadvertently not included in Inspection Report 498/499/2001-02. It is included here for completeness.

b. Findings

No findings of significance were identified.

4OA3 Event Followup (71153)

.1 Failure to Recognize that Two Evolutions Using the Same System Conflicted

a. Inspection Scope

The inspectors conducted an event followup inspection following an unplanned dilution of the volume control tank during routine operations. The inspectors reviewed Condition Reports 01-14307 and 01-14309. The inspectors discussed the licensee’s analysis of the event, including a draft calculation by reactor engineering of the impact of the dilution without operator action, with the Unit 2 operations manager, the shift supervisor and the unit supervisor. The inspectors discussed procedure usage practices with both Units 1 and 2 operations managers, the operations training manager, and the supervisor of performance assessment, as well as a number of licensed and senior licensed reactor operators to compare the practices to written station program requirements. The following procedures were reviewed:

- 0POP02-CV-0001, “Makeup to the Reactor Coolant System,” Revision 17

- 0POP02-CV-0003, “Mixing of Boric Acid,” Revision 2
- 0POP02-CV-0005, “Chemical and Volume Control System Pre-Start System Alignment,” Revision 16
- 0PGP03-ZO-42, “Reactivity Management Program,” Revision 4
- 0PGP03-ZA-0010, “Performing and Verifying Station Activities,” Revision 25
- Conduct of Operations Manual

b. Findings

Operators failed to recognize that two routine evolutions using the chemical and volume control system conflicted because they did not properly verify that the prerequisites were satisfied. While attempting to add borated water to the reactor coolant system, only pure water was added, challenging operators to avoid an unintended power increase. A Green NCV was identified for failure to follow procedure.

Also, the inspectors concluded that the licensee’s corrective action program did not adequately identify the cause or significance of this event, and instead treated it as minor because operators were able to negate to effect of the error.

System Design and Sequence of Events

The chemical and volume control system was used to make adjustments to the volume and reactivity control of the reactor coolant system. Operators routinely make up for minor system losses and make changes to the chemical concentration of the boric acid in the coolant manually. Adding water would cause the boron content to be diluted, adding positive reactivity to the core.

On September 10, 2001, Unit 2 operators attempted to perform a routine blended makeup (addition of water and boric acid solution from two separate tanks in a specific proportion) to the volume control tank. Earlier in the shift, they had placed one of the boric acid tanks on recirculation while making batch additions of boric acid. Operators failed to recognize that the procedure for batch additions of acid, 0POP02-CV-0003, had not placed the remaining boric acid transfer pump into a condition that would allow it to start automatically in response to a demand (such as a makeup evolution).

After operators started the blended makeup, they recognized that only water was going to the volume control tank and stopped the addition after 42 gallons had been added. This diluted the contents of the tank. Both the licensed operator performing the operation and the licensed operator conducting a peer check had failed to notice that the system alignment did not support the intended operation; one boric acid transfer pump was isolated from the rest of the system while recirculating the boric acid tank, and the other was not aligned for operation. To compensate for the inadvertent dilution, the supervisor directed a boration by adding the amount of acid that was planned, but added it directly to the charging pump suction so that it went into the reactor coolant system before the dilution could have any affect, causing a minimal decrease in reactor power.

Issues

The licensee wrote Condition Reports 01-14307 and 01-14309 following this event. The licensee concluded that operators responded as expected. The licensee coded this as a condition adverse to quality, department level, which was the lower of two classifications for a condition adverse to quality. This coding did not require a cause determination. The event was promptly reviewed by operations management and it was concluded that this was not a reactivity management event as defined in station procedures because there was no undesired impact on core reactivity, and specifically no power increase. The licensee concluded that the two operations should have been procedurally permitted simultaneously, and took action to change the procedures. Senior management ordered a review of these conclusions, but the results of this effort were not completed by the end of this inspection.

The inspectors reviewed the licensee's reactivity management program and concluded that this should have been treated as a reactivity management event as defined in those procedures. The procedure in use, 0POP02-CV-0001, was listed as a reactivity management procedure. The event involved an unplanned dilution which necessitated an unplanned boration to avoid an unplanned power increase from an initial power that was already very slightly above 100 percent. Several key elements of the reactivity management program stated in 0PGP03-ZO-0042 were not met:

- "Goal: Preclude inadvertent positive reactivity additions . . . ."

- "Reactivity Awareness: Station personnel, especially operators, must be aware of how their actions affect reactivity . . . ."

- "Procedure Adherence: Procedural adherence is essential to the safe operation of the reactor."

The inspectors noted that the licensee's Conduct of Operations procedure did not list a blended makeup as a reactivity manipulation. It also required very little supervisory involvement if the intent was to perform reactivity manipulations that were agreed to in a briefing at the beginning of the shift.

The inspectors determined that the operators violated 0POP02-CV-0001 by not satisfying the prerequisite system alignment prior to starting the blended makeup. The procedure prerequisite required that the system be in the prestartup lineup. The inspectors verified that the specified conditions in the prestartup lineup would have allowed successful blended makeup. Failure to follow the prerequisites of Procedure 0POP02-CV-0001 was a violation of Technical Specification 6.8.1 and Regulatory Guide 1.33. This violation is being treated as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy and is in the licensee's corrective action program as Condition Reports 01-14307 and 01-14309 (NCV 499/2001005-02).

Procedure 0PGP03-ZA-0010 states:

> "Operating PROCEDURES are written based on a defined set of plant conditions and equipment availability. PROCEDURE changes are not required to document alternate performance based on conditions different from those assumed if the PROCEDURES can be performed safely. The decision to proceed lies with the assigned supervisor and is based on knowledge of system design and operation and the impact of omitting or resequencing steps... The Shift or Unit Supervisor ensures such an alternate performance does not adversely impact the safety of personnel or equipment, and documents the alternate method in the appropriate PROCEDURE or logbook."

After interviewing a number of licensed operators, operations managers, and the operations training manager the inspectors concluded that operators did not consider this type of system alignment prerequisite, which was common to nearly all system operating and surveillance procedures, to be a literal requirement. The operators considered that having a boric acid tank in recirculation was essentially the same as the prestartup lineup as long as a flowpath could be demonstrated. As a result of this misunderstanding, the special review of 0PGP03-ZA-0010 for not satisfying prerequisites was almost never invoked.

This was not a significant reactivity event, largely because operators recognized the error and took action to override the effects. The licensee calculated that, if left unrecognized, power could have increased 0.3 percent just over 100.3 percent. However, this issue was determined to be more than minor because the violation suggested a programmatic problem in procedure adherence that could have a realistic potential safety or regulatory impact. If left uncorrected, this violation would become a more significant safety and regulatory concern. Understanding and properly adhering to approved procedures is a key element of human performance necessary to support reactor safety. An inadvertent dilution is an intiating event analyzed in the Updated Final Safety Analysis Report, Chapter 15, and this event was bounded by that analysis. Using a Phase 1 Significance Determination Process, this issue screened as having very low safety significance (Green).

<u>Conclusion</u>

The inspectors concluded that the root cause of this event was a failure to recognize that the plant was not in the configuration required by the procedure in use, in part because of a culture that permitted a loose interpretation of what constituted the required system alignment. The licensee's corrective action program focused on the importance of the plant impact after operator actions to mitigate the error, so it did not initially recognize that this event represented a challenge to the reactivity management program, and as a result, under classified the importance and did not perform a cause analysis.

.2 <u>(Closed) Licensee Event Report 50-498/2000-005</u>: Reactor operator assumed control room watch with an inactive license on July 17, 2000. The operator had a valid NRC reactor operator license, but did not maintain the license active in accordance with the requirements of 10 CFR 55.53. Specifically, he had not stood watch for the requisite number of hours during the previous quarter. He was called to fill in for an unexpected absence. Although the individual believed that his license had become inactive and let this be known to the control room staff, his qualifications were listed as current on the licensee's "Active License List" in the shift supervisors office. This was subsequently determined to be out of date and the violation was reported to the NRC. The individual stood watch for less than 2 hours before being relieved.

This issue was addressed in the licensee's corrective action program under CR 00-11749. The inspectors reviewed the licensee's extensive corrective actions to prevent recurrence, and found that they adequately addressed the problem.

This violation was of very low safety significance with no color assigned because the individual standing watch had a valid NRC licensee and had been supervised by two Senior Reactor Operators. The license had become inactive at the end of the previous quarter, which had recently ended. This issue was categorized as a licensee-identified noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 498/0105-04) (see Section 4OA7.2).

4OA5 Other

Licensee Strike Contingency Plan (92709)

a. Inspection Scope

In July 2001, as the licensee's contract with the International Brotherhood of Electrical Workers was about to expire, the inspectors used Inspection Procedure 92709 to review the licensee's contingency plan for continued safe operation of the plants in the event that bargaining unit personnel went on strike. The focus of the inspection was to determine if sufficient qualified personnel would be available for safe and secure operation within regulatory requirements. The licensee's plans to ensure current qualifications and training for operators, maintenance personnel, and fire brigade members were reviewed and discussed with appropriate members of licensee management.

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

The inspector presented the radiation monitoring instrumentation inspection results to Mr. W. Dowdy, Manager, Generation Support, and other members of licensee management at the conclusion of the inspection on June 28, 2001. The licensee acknowledged the findings presented.

The inspectors presented the results of the licensed operator requalification inspection to Mr. G. Parkey, Operations Manager, and other members of the licensee's management staff at an exit interview on August 9, 2001. The licensee acknowledged the findings presented.

The inspector presented the ALARA inspection results to Mr. J. Sheppard, Vice President, Engineering and Technical Services, and other members of licensee management at the conclusion of the inspection on September 13, 2001. The licensee acknowledged the findings presented.

The inspectors presented the results of the resident inspection to Mr. T. Cloninger, Vice President, Nuclear Generation, and other members of the licensee's management staff at an exit interview on September 25, 2001. The licensee acknowledged the findings presented.

In each case, the inspectors also asked the licensee whether any materials examined during the inspections should be considered proprietary. No proprietary information was identified.

4OA7 Licensee-Identified Violations - The following findings of very low significance were identified by the licensee and were violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as noncited violations.

	NCV Tracking Number	Requirement Licensee Failed to Meet
.1	NCV 50-498/0105-03	Failure to have a procedure to energize an emergency AC bus from its normal source of power, a procedure required by Technical Specification 6.8.1 and Regulatory Guide 1.33 (Reference Condition Report 01-14699).
.2	NCV 50-498/0105-04	Reactor operator assumes control room watch with an inactive license contrary to 10 CFR 55.53 (Reference Condition Report 00-11749).

ATTACHMENT

Supplemental Information

PARTIAL LIST OF PERSONS CONTACTED

Licensee

A. Aldridge, Supervisor, Fire Protection
M. Berg, Manager Operating Experience Group
W. Bullard, Supervisor, Health Physics
J. Calvert, Operations Training Manager
T. Cloninger, Vice President, Nuclear Generations
W. Dowdy, Manager, Generation Support
R. Gangluff, Manager, Chemistry
E. Halpin, Manager, Operations
S. Head, Manager, Licensing
T. Hurley, Training Supervisor, Operations
J. Johnson, Supervisor, Engineering Oversight
A. Kent, Manager, Testing/Programs
D. Leazar, Manager, Nuclear Fuel Analysis
R. Lovell, Manager, Nuclear Training
M. McBurnett, Director, Quality and Licensing
G. Parkey, Plant General Manager
J. Phelps, Division Manager, Operations
T. Powell, Manager, Health Physics
D. Scoggins, Supervisor, Metrology and Radiological Laboratories
P. Serra, Manager, Plant Protection
J. Sheppard, Vice President, Engineering and Plant Support
M. Sicard, Manager, I&C Maintenance
M. Smith, Supervisor, Quality
M. Tomek, Supervisor, Unit 2 Health Physics
J. Winters, Maintenance Rule Coordinator

NRC

J. Collaccino, Mechanical Engineering Branch, NRR
J. Huang, Mechanical Engineering Branch, NRR
J. Pulsipher, Plant Systems Branch, NRR
T. Pruitt, Senior Reactor Analyst, Region IV

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-498/0105-01	NCV	Procedure that was inappropriate to the circumstances because it did not identify the

		impact of removing the load sequencer from service, which was a violation of Technical Specification 6.8.1 and Regulatory Guide 1.33 (Section 1R13)
50-499/0105-02	NCV	Failure to verify that the system was aligned as required by prerequisites prior to attempting a blended makeup to the volume control tank (Section 4OA3)
50-498/0105-03	NCV	Failure to have a procedure to energize an emergency AC bus from its normal source of power as required by Technical Specification 6.8.1 and Regulatory Guide 1.33 (Section 4OA7)
50-498/0105-04	NCV	Reactor operator assumes control room watch with an inactive license contrary to 10 CFR 55.53 (Section 4OA7)

<u>Closed</u>

50-498/0105-01	NCV	Procedure that was inappropriate to the circumstances because it did not identify the impact of removing the load sequencer from service, which was a violation of Technical Specification 6.8.1 and Regulatory Guide 1.33.4 (Section 1R13)
50-499/0105-02	NCV	Failure to verify that the system was aligned as required by prerequisites prior to attempting a blended makeup to the volume control tank (Section 4OA3)
50-498/0105-03	NCV	Failure to have a procedure to energize an emergency AC bus from its normal source of power as required by Technical Specification 6.8.1 and Regulatory Guide 1.33 (Section 4OA7)
50-498/0105-04	NCV	Reactor operator assumes control room watch with an inactive license contrary to 10 CFR 55.53 (Section 4OA7)
50-498/2000-005	LER	Reactor operator assumes control room watch with inactive license (Section 4AO3)

Discussed

None

LISTS OF ACRONYMS USED

AFW	auxiliary feedwater
ALARA	as low as reasonably achievable
CFR	Code of Federal Regulations
CR	condition report
ESF	engineered safety feature
IDLH	immediately dangerous to life and health
LER	licensee event report
LLRT	local leak rate testing
NCV	noncited violation
NEI	Nuclear Energy Institute
SCBA	self-contained breathing apparatus
SDG	standby diesel generator
SSC	structure, system, or component
WAN	work authorization number

DOCUMENTS REVIEWED

The following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of the inspection and to support any findings:

Operator Requalification Program Inspection

Procedures:

Licensed Operator Requalification Plan (2000-2001)

Conduct of Operations, Revision 13

0PGP03-ZT-0132, Licensed Operator Requalification, Revision 4

LOR-GL-0001, Licensed Operator Requalification Training Program Guidelines, Revision 6

LOR-GL-0002, Licensed Operator Requalification Annual and Biennial Evaluation Guidelines, Revision 6

LOR-GL-0003, Licensed Operator Requalification Examination Bank Guidelines, Revision 1

LOR-GL-0004, Licensed Operator Requalification Two Year Training Plan Guidelines, Revision 2

LOR-GL-0005, Guideline for Physical Notification & Responsibilities, Revision 4

LOR-GL-0006, Licensed Operator Requalification Conduct of Simulator Training Guidelines, Revision 8

Licensed Operator Requalification End of Cycle Reports:

LOR001	LOR003	LOR0011
LOR001a	LOR004	LOR0012
LOR002	LOR005	

LOR Remediation Plans

July 19, 2000	April 11, 2001
July 26, 2000	April 26, 2001
October 16, 2000	May 1, 2001
February 21, 2001	May 9, 2001

Curriculum Review Committee Meeting Minutes

January 25, 2000	July 25, 2000	December 5, 2000
February 22, 2000	August 22, 2000	February 6, 2001
May 23, 2000	September 19, 2000	May 1, 2001
June 6, 2000	October 31, 2000	

Condition Reports:

01-2639
00-0058

Simulator Scenarios:

EXAM 014-5, Revision 0
EXAM014-10, Revision 0

Job Performance Measures:

JPM079.01, Revision 5	JPM014.01, Revision 8
JPM010.01, Revision 5	JPM012.02, Revision 8
JPM018.01, Revision 6	JPM015.02, Revision 8
JPM012.01, Revision 8	JPM016.02, Revision 4
JPM048.01, Revision 6	JPM019.02, Revision 8