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

August 7, 2000

Mr. C. L. Terry
TXU Electric
Senior Vice President & Principal Nuclear Officer
ATTN: Regulatory Affairs Department
P.O. Box 1002
Glen Rose, Texas 76043

SUBJECT: NRC INSPECTION REPORT NO. 50-445/00-04; 50-446/00-04 FOR COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2

Dear Mr. Terry:

On July 8, 2000, the NRC completed an inspection at the Comanche Peak Steam Electric Station, Units 1 and 2, facility. The enclosed report presents the results of the inspection. The results of this inspection were discussed with you and other members of your staff.

The inspection was an examination of activities conducted under your license as they relate to safety and to compliance with the Commissions's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations of activities, and interviews with personnel.

One unresolved issue was identified associated with a failure to identify degraded emergency diesel generator jacket water coolers due to excessive fouling. Although the issue has been placed into your corrective action process, it has not yet been evaluated under the NRC risk significance determination process because your review of past operability was not complete at the end of the report period.

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/

Joseph I. Tapia
Project Branch A
Division of Reactor Projects

Docket Nos.: 50-445
50-446
License Nos.: NPF-87
NPF-89

Enclosure:
NRC Inspection Report No.
50-445/00-04; 50-446/00-04

cc w/enclosure:
Roger D. Walker
TXU Electric
Regulatory Affairs Manager
P.O. Box 1002
Glen Rose, Texas 76043

Juanita Ellis
President - CASE
1426 South Polk Street
Dallas, Texas 75224

George L. Edgar, Esq.
Morgan, Lewis & Bockius
1800 M. Street, NW
Washington, D.C. 20036

G. R. Bynog, Program Manager/
Chief Inspector
Texas Department of Licensing & Regulation
Boiler Division
P.O. Box 12157, Capitol Station
Austin, Texas 78711

County Judge
P.O. Box 851
Glen Rose, Texas 76043

Chief, Bureau of Radiation Control
Texas Department of Health
1100 West 49th Street
Austin, Texas 78756-3189

John L. Howard, Director
Environmental and Natural Resources Policy
Office of the Governor
P.O. Box 12428
Austin, Texas 78711-3189

Electronic distribution from ADAMS by RIV:
Regional Administrator **(EWM)**
DRP Director **(KEB)**
DRS Director **(ATH)**
Senior Resident Inspector **(ATG)**
Branch Chief, DRP/A **(JIT)**
Senior Project Engineer, DRP/A **(DNG)**
Branch Chief, DRP/TSS **(LAY)**
RITS Coordinator **(NBH)**

Only inspection reports to the following:
D. Lange **(DJL)**
NRR Event Tracking System **(IPAS)**
CP Site Secretary **(LCA)**
Dale Thatcher **(DFT)**

R:_CPSES\2000\CP2000-04RP-ATG.wpd

RIV:SRI:DRP/A	RI:DRP/A	SRI:DRP/A	DRS	C:DRP/A
ATGody	SCSchwind	NFO'Keefe	GMGood	JITapia
/RA/	***ATGody for***	***JITapia for***	***JBNicholas for***	***/RA/***
08/04/00	08/04/00	08/07/00	08/07/00	08/07/00

OFFICIAL RECORD COPY T=Telephone E=E-mail F=Fax

ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos.:	50-445 50-446
License Nos.:	NPF-87 NPF-89
Report No.:	50-445/00-04 50-446/00-04
Licensee:	TXU Electric
Facility:	Comanche Peak Steam Electric Station, Units 1 and 2
Location:	FM-56 Glen Rose, Texas
Dates:	May 21 through July 8, 2000
Inspectors:	A. Gody, Senior Resident Inspector N. O'Keefe, Senior Resident Inspector, South Texas Project S. Schwind, Resident Inspector J. Dodson, Health Physicist
Approved By:	Joseph I. Tapia, Chief, Project Branch A

ATTACHMENTS:

Attachment 1: Supplemental Information

Attachment 2: NRC's Revised Reactor Oversight Process

SUMMARY OF FINDINGS

Comanche Peak Steam Electric Station
NRC Inspection Report 50-445/00-04; 50-446/00-04 (DRP)

This integrated inspection report covers a 7-week period of resident inspection and announced inspections by regional engineering, emergency preparedness, and radiation specialist inspectors.

None

Report Details

Summary of Plant Status

Both units operated at approximately 100 percent power for the entire report period.

1. REACTOR SAFETY
Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1RO4 Equipment Alignment

.1 Partial System Walkdown

a. Inspection Scope

The inspectors conducted partial inspections of the following risk-significant systems to verify that they were in their proper standby alignment. In addition, the inspectors evaluated the effectiveness of the licensee's problem identification and resolution program in resolving issues which could increase event initiation frequency or impact mitigation system availability.

- Unit 2, Trains A and B emergency diesel generator (EDG) systems
- Unit 2, Trains A and B motor-driven auxiliary feedwater systems
- Train A, spent fuel pool cooling system

The following documents were reviewed by the inspectors during this inspection:

- Operations Procedure SOP-609B, "Diesel Generator System," Revision 8

- Operations Procedure SOP-610B, "Diesel Generator Fuel Oil and Transfer System," Revision 3

- Operations Procedure SOP-809B, "Diesel Generator Rooms Ventilation System," Revision 5

- Operations Procedure SOP-304A, "Auxiliary Feedwater System," Revision 14

- Operations Procedure SOP-506, "Spent Fuel Pool Cooling and Cleanup System," Revision 12

- "Trend Analysis of Shift Operations & Operations Support Departments Smart Forms," April 19, 2000

- Comanche Peak Steam Electric Station System Health Report for the Unit 2 auxiliary feedwater systems

- Comanche Peak Steam Electric Station Final Safety Analysis Report Section 10.4.9, "Auxiliary Feedwater System"

b. Findings

No findings were identified.

.2 Detailed Semiannual System Walkdown

a. Inspection Scope

The inspectors conducted a detailed semiannual inspection of the Unit 2 turbine-driven auxiliary feedwater system to ascertain if the system and its operating procedures were in accordance with the design and licensing bases of the system. Outstanding maintenance work requests and design issues were reviewed to determine if any impacted the system's ability to operate as designed.

The following documents were reviewed by the inspectors during this inspection:

- Operations Procedure SOP-304A, “Auxiliary Feedwater System”, Revision 14
- Comanche Peak Steam Electric Station System Health Report for the Unit 2 auxiliary feedwater systems
- Comanche Peak Steam Electric Station Final Safety Analysis Report Section 10.4.9, “Auxiliary Feedwater System”

b. Findings

No findings were identified.

1R05 Fire Protection

.1 Routine Fire Area Walkdowns

a. Inspection Scope

The inspectors toured the following areas to assess the licensee's control of transient combustible materials, the material condition and lineup of fire detection and suppression systems, the material condition of manual fire equipment and passive fire barriers, and the effectiveness of compensatory measures for degraded equipment:

- Unit 1, Trains A and B switchgear rooms
- Unit 1, Safeguards Building 810 foot corridor
- Unit 1, Train B uninterruptible power supply and distribution room

The following documents were reviewed by the inspectors during this inspection:

- Fire Preplan FPI-107A, “Unit 1 Safeguards Building 810'-6" Elev. Corridor and Sample Room,” Revision 2

- Fire Preplan FPI-108A, “Unit 1 Safeguards Building Electrical Equipment Area Elv. 810'-6",” Revision 2

- Fire Preplan FPI-503, “Unit 1 Battery Rooms Control Bldg. Elev. 792'-0",” Revision 2

- Smart Form SMF-2000-001449-00, fire ratings not transcribed to CAD version of drawing.

- Smart Form SMF-2000-001454-00, fire preplans inconsistent with Fire Protection Report

- Comanche Peak Steam Electric Station Fire Protection Report, Revision 15

b. Findings

No findings were identified.

1R07 Heat Sink Performance

.1 Annual Heat Exchanger Review

a. Inspection Scope

The inspector conducted an annual inspection of the licensee’s testing of the EDG jacket water coolers. The inspection included Units 1 and 2 EDG jacket water cooler cleaning observations, a review of the test methodology, a review of the frequency of testing, and an evaluation of the incorporation of design and licensing bases and test instrument inaccuracies in the test program acceptance criteria.

Documents reviewed during this inspection included:

- NRC Inspection Report 50-445(446)/90-38, Inspection of Licensee Implementation of NRC Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment”

- Comanche Peak Steam Electric Station, Units 1 and 2 Technical Specification 3.7.9, “Ultimate Heat Sink” and its’ bases

- Comanche Peak Steam Electric Station Final Safety Analysis Report Section 9.5.5, “Diesel Generator Cooling Water System”

- Technical Support Procedures Manual Procedure TSP-503, “Emergency Diesel Generator Reliability Program, Revision 3

- Station Administrative Manual Procedure STA-734, “Service Water System Fouling Monitoring Program,” Revisions 0, 1, and 2

- SMART Form 2000-1548, Failure to Properly Implement Diesel Generator Jacket Water Cooler Fouling Monitoring Program
- Calculation ME-CA-0011-3075, Revision 1, “Emergency Diesel Generator Jacket Water Cooler Fouling Factor Analysis”
- Calculation ME-(B)-391, Revision 2, “Service Water Flow to Emergency Diesel Generator Jacket Water Cooler”
- Calculation 2-ME-0042, Revision 0, “Minimum Allowable Service Water Flow to the Emergency Diesel Generator”
- Comanche Peak Steam Electric Station Design Basis Document DBD-ME-011, “Diesel Generator Sets,” Revision 11

b. Findings

Background

NRC Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment,” was issued to all holders of operating licenses or construction permits for nuclear power plants on July 18, 1989, because operating experience revealed that compliance with the above requirements was questionable. NRC Generic Letter 89-13, requested licensee’s to: (1) implement a program to significantly reduce the incidence of service water system flow blockage; (2) conduct a test program to verify heat transfer capability of all heat exchangers cooled by the service water system; (3) ensure that corrosion, erosion, coating failures, silting, and biofouling cannot degrade the performance of the safety-related systems supplied by service water; (4) confirm that the service water system will perform its intended function in accordance with the licensing basis for the plant; and (5) confirm that maintenance practices, operating and emergency procedures, and training that involves the service water system are adequate to ensure that safety-related equipment cooled by the service water system will function as intended.

Inspection Results

On May 31, 2000, the inspector requested that the service water system engineer demonstrate how STA-734, “Service Water System Fouling Monitoring Program,” Revision 2, was implemented for the EDG jacket water coolers. The service water system engineer informed the inspector that he would not be available until June 5 because he planned to be out of the area. On June 5, the system engineer wrote SMART Form 2000-1548, which indicated that, although raw data from EDG testing was obtained, it was not: (1) subjected to the semiannual heat exchanger trending specified in STA-734 paragraph 6.2.2.3.A; and (2) that licensee Procedure TSP-503, “Emergency Diesel Generator Reliability Program,” Revision 3, had removed the recommendation to conduct service water flow trending. On June 7, the licensee established a corrective action (RESL-2000-1548-01) to address the deficiency and was assigned to the system engineer.

The inspector noted that trending had not been conducted as specified in STA-734 for all four EDG during the fall of 1998, for the Unit 1 EDGs during the spring of 1999, and for the Unit 2 EDGs in the fall of 1999. In summary, the Unit 1 EDG jacket water coolers had not been trended for approximately 1 1/2 years and the Unit 2 EDG jacket water coolers had not been trended for about one year. During that period, several of the Units 1 and 2 Train B EDG jacket water coolers were frequently fouled beyond the acceptance criteria contained within STA-734, Section 6.2.2.2A, and were thus considered degraded. To illustrate the issue, for example, from November 1998 to June 2000, the Unit 2 Train B EDG jacket water cooler fouling only satisfied the acceptance criteria contained in STA-734, Section 6.2.2.2A, for the short period between March and May 1999, and the degraded condition was not placed into the licensee's corrective action program. The inspector requested the specific dates when the other EDG coolers did not meet the acceptance criteria contained in STA-734, but the information was not provided by the licensee before the end of the report period.

10 CFR Part 50, Appendix B, Criterion XVI, states, in part, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances are promptly identified and corrected." Contrary to the above, because trending of operational data from monthly EDG testing was not performed in accordance with STA-734, the licensee failed to promptly identify that, on a number of occasions during 1999, EDG jacket water cooler fouling had increased to the point where they could not meet the acceptance criteria specified in STA-734, Section 6.2.2.2A, a condition adverse to quality.

STA-734, Section 6.2.2.2A, states, in part, that "Test data should be analyzed in accordance with Calculation ME-CA-011-3075 . . . to correlate to a fouling factor for the heat exchanger . . . . This fouling factor will be compared to an acceptable fouling factor which has been calculated for 102°F SW [Service Water] temperature. If the fouling factor is determined to be unacceptable, then the affected DG [emergency diesel generator] system train should be declared INOPERABLE and the appropriate Technical Specification requirements satisfied."

The acceptance criteria specified in STA-734 was appropriately based on the current licensing and design bases information as follows:

(1) The Comanche Peak Steam Electric Station Final Safety Analysis Report (FSAR) Section 9.5.5.1, "Design Bases," states "The emergency diesel generator jacket water cooling system is designed to allow the diesel generator sets to be rapidly loaded and to operate continuously at their maximum ratings. The various components are sized to remove the maximum heat produced by the diesel generator sets using 115°F service water as a cooling medium." FSAR Table 9.5-18, "Jacket Water Cooler Design Parameters," lists a design heat removal rate of 25.5 X 10^6 BTU/hour, a required heat removal rate of 17.0 X 10^6 BTU/hour, and a maximum service water inlet temperature of 115°F. The Comanche Peak Steam Electric Station Supplemental Safety Evaluation Report, Section 9.5.5, states "The staff concluded . . . that the diesel generator cooling water components are adequately sized to remove the maximum heat produced by the

diesel generator sets, using 115°F service water as a cooling medium. On this basis, the staff finds that 115°F service water to the diesel generator cooling water components is acceptable."

(2) Comanche Peak Steam Electric Station Design Basis Document DBD-ME-011, "Diesel Generator Sets," Revision 11, dated March 11, 1999, Section 4.3.1.1.3, states, "The DGJWS [Diesel Generator Jacket Water System] shall be designed to allow the diesel generator sets to be rapidly loaded and to operate continuously at their maximum ratings. The various components are required to be sized to remove the maximum heat produced by the diesel generator sets using 115°F service water as a cooling medium."

The inspector found that the 102°F acceptance criteria appropriately accounted for the service water temperature rise associated with a design basis accident which is documented in Calculations 2-ME-0042, Revision 0, and ME-(B)-391.

The licensee's preliminary review concluded that sufficient margin existed to consider the EDGs operable. To evaluate the significance of the licensee's failure to identify the EDG jacket water cooler fouling in 1999, the inspector planned to review the licensee's past operability and reportability determinations. This documentation was not yet available by the end of the report period. Because the significance of the violation of 10 CFR Part 50, Appendix B, Criterion XVI, could not be determined, it will be tracked as an unresolved item (URI 50-445(446)/200004-01).

On June 22, the inspector discussed the importance of ensuring that the plant design basis is properly translated into the service water system fouling monitoring program. On June 23, the licensee opened resolution RESL-2000-1548-02 which stated, in part, "The purpose of this resolution is to investigate a possible conflict between the design basis as stated in the FSAR and actual plant conditions." The inspector requested a meeting with the design engineer responsible for Calculation ME-CA-011-3075 to discuss how instrument inaccuracy was addressed and what service water flow value should be used to input for the fouling factor calculation. The inspector found that the service water system engineer had been using actual service water flow rates in the fouling factor analysis and questioned whether that was appropriate. These concerns will be addressed when the inspector completes the inspection associated with the unresolved item discussed above.

1R12 Maintenance Rule

.1 Maintenance Rule Functional Failure Review

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.

- Instrument Air Compressor 2-01 reliability

- Main control board annunciator power supply failures

- Chemical and Volume Control System 45 gpm letdown orifice isolation valve air regulator failures

- Diesel Generator 1-01 jacket water leak

The inspectors' review focused on whether the structures, systems, or components (SSC's) that experienced problems were properly characterized with respect to the scope of the program, whether the SSC failure or performance problem was properly characterized, the adequacy of the licensee's significance classification for the SSC, 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 50.65a(1) as applicable. Documents reviewed during the inspection include:

- Station Administrative Manual (STA) - 744, "Maintenance Effectiveness Monitoring Program," Revision 2

- Smart Form SMF-2000-00718-00, Unit 2 instrument air compressor tripped on low oil pressure

- Maintenance Work Order MWO-4-00-129968-00, replace voltage sensing card on Unit 1 Control Room Annunciator Power Supply P39

- Smart Form SMF-2000-001135-00, 2-8149-PR1 bowl gasket blown for second time in one week

- Smart Form SMF-2000-001540-00, 1-01 D/G jacket water leak

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessment and Emergent Work

.1 Unscheduled work activities

a. Inspection Scope

The inspectors evaluated the effectiveness of the licensee's risk assessment for the following emergent at-power work.

- Unit 2 rod control system shutdown Bank C step demand counter replacement
- Tightening actuator adapter plate bolts on Valve 2-HV-2491B

When the need for emergent work was identified on risk-significant SSCs, the inspectors verified that the licensee took appropriate steps to plan and control the resulting activities, including the acceptability of any necessary compensatory actions and contingency plans when applicable. Documents reviewed during the inspection include:

- Smart Form SMF-2000-001672-00, actuator adapter plate bolts loose on 2-HV-2491B-MO
- Work Order 4-00-131104-00, full length rod control system panel 2-LFL-01
- Work Control Instruction WCI-203, “Weekly Surveillances/Work Scheduling,” Revision 12

b. Findings

No findings were identified.

1R14 Nonroutine Plant Evolutions and Events

a. Inspection Scope

On May 17, 2000, Unit 1 experienced a minor transient on the main turbine due to an intermittent fault on the electrohydraulic control system. The inspector observed the licensee diagnose the fault and return to 100 percent power.

On June 1, 2000, Unit 1 experienced a small reactor coolant system pressure transient following pressurizer pressure and level control channels being shifted in preparation for an operational test of pressurizer level instrument Channel 1-L-0459. The inspector reviewed operational data on the transient and observed the licensee troubleshoot the problem.

On June 15, 2000, Unit 1 experienced a small heater drain system transient as a result of failing to identify the impact of increasing the scope of planned maintenance to replace a broken valve positioner. The inspector observed operations respond to the transient, diagnose the maintenance error, and restore the plant to normal operation and reviewed the licensee's corrective actions.

b. Findings

No findings were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors selected operability evaluations conducted by the licensee during the report period involving risk-significant systems or components to review. The inspectors evaluated the technical adequacy of the licensee’s operability determination, verified that appropriate compensatory measures were implemented, and verified that the licensee considered all other pre-existing 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.

- Potential inadequate venting of the Unit 1 safety injection system
- Potential degraded condition on valves ½-HV-4696 due to loose fasteners on adapter between the actuator and yoke

The following documents were reviewed by the inspectors during this inspection:

- Station Procedure STA-421, "Initiation and Processing of Smart Forms," Revision 8
- Comanche Peak Steam Electric Station Updated Final Safety Analysis Report
- Comanche Peak Steam Electric Station Technical Specifications
- Evaluation QTE-2000-001555-01-00, Unit 1 safety injection high point vents not located at system high point
- Smart Form SMF-2000-001555-00, Unit 1 safety injection system may not be properly vented
- Evaluation CPSES-9800580, evaluation of the applicability of Information Notice 97-40
- Evaluation EVAL-2000-001672-01-00, potentially degraded condition for 1/2-HV-4696

b. <u>Findings</u>

No findings were identified.

1R19 <u>Postmaintenance Testing</u>

a. <u>Inspection Scope</u>

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

- Unit 2 intermediate range nuclear instrument compensating voltage power supply replacement
- Unit 2, Train A Containment Spray Pump 2-03 bearing replacement

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, Design Basis Documents, and selected calculations were also reviewed to determine the adequacy of the acceptance criteria listed in the test procedures. The inspectors reviewed the following documents during this inspection:

- Work Order WO-3-99-334421, measure ac ripple and dc output setting on Power Supply 2-NQ-NQ201A
- Evaluation QTE-2000-001559-01-00, as found ac ripple and dc output settings on Power Supply 2-NQ-NQ201A out of tolerance
- Work Oder 4-98-118947-00, increase IB and OB bearing diametrical clearance per MSM-CO-7308
- Operations Procedure OPT-205B, “Containment Spray System,” Revision 7
- Smart Form SMF-2000-001725-00, CP2-CTAPCS-03 shaft alignment found to be out of manufacturer’s tolerance

b. Findings

No findings were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors evaluated the adequacy of periodic testing of the following important nuclear plant equipment, including aspects such as preconditioning; the impact 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; test failure evaluations; jumper control (if applicable); and the effectiveness of the licensee’s problem identification and correction program. The following surveillance test activities were observed by the inspectors:

- Unit 1, Train B safety injection pump operability test
- Unit 1, feedwater isolation partial stroke test
- Unit 2, Train A solid state protection system actuation logic test
- Unit 2, turbine-driven auxiliary feedwater system surveillance test

The inspectors reviewed the following documents during the inspection:

- Operations Procedure OPT-511A, “FW Section XI Isolation Valves,” Revision 9

- Operations Procedure OPT-447B, “Mode 1, 3 and 4 Train A SSPS Actuation Logic Test,” Revision 3

- Operations Procedure OPT 204A, “SI System,” Revision 8

- Comanche Peak Technical Specifications

- Comanche Peak Updated Final Safety Analysis Report

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Transportation (71122.02)

a. Inspection Scope

The inspector interviewed licensee personnel, walked down liquid and solid radioactive waste processing systems, and reviewed the following items, to determine if the licensee is meeting the objective of this cornerstone which is to ensure adequate protection of public health and safety from exposure to radioactive material released into the public domain from routine operations.

- Radioactive material processing and shipping procedures

- The status of radioactive waste process equipment that was not operational and/or abandoned in place

- Changes made to the radioactive waste processing systems since the last inspection in February 1999

- Waste stream mixing and/or sampling procedures, methodology for waste concentration averaging, and waste classification procedures

- Radio-chemical sample analysis results for each of the radioactive waste streams

- The use of scaling factors and calculations used to account for difficult to measure radionuclides

- Changes in waste stream composition due to changing operational parameters and analysis updates

- Shipment packaging, surveying, labeling, marking, placarding, vehicle checks, emergency instructions, disposal manifest, shipping papers provided to the driver, and licensee verification of shipment readiness
- Transport cask Certificates of Compliance and cask loading and closure procedures
- Transferee's licenses and state/DOT permits
- Conduct of radioactive waste processing and radioactive material shipment preparation activities
- Training program for the conduct of radioactive waste processing and radioactive material shipment preparation activities
- Eight nonexcepted package shipment records
- Licensee Event Reports, Special Reports, audits, and self assessments related to the radioactive material and transportation programs performed since the last inspection in February 1999
- Smart forms written against the radioactive material and shipping programs since the previous inspection in February 1999

b. Findings

No findings were identified.

4. OTHER ACTIVITIES
Cornerstone: Mitigating Systems

4OA1 Performance Indicator Verification (Mitigating Systems)

a. Inspection Scope

The inspectors conducted a review of the licensee's Units 1 and 2 mitigating systems performance indicator data for the first quarter of 2000 to determine its accuracy and completeness.

b. Findings

No findings were identified.

40A6 Management Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. M. Blevins and Mr. C. Lance Terry and to other members of licensee management at an exit meetings on June 29 and July 18, 2000. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Licensee

Bhatty, O., Senior Engineer
Calder, R. D., Executive Assistant
Evans, T., Engineering Supervisor
Feist, C., Design Basis Engineer
Flores, R., System Engineering Manager
Hope, T. A , Regulatory Compliance Manager
Kelly, J. J., Vice President, Nuclear Engineering and Support
Krishnan, G., System Engineer
Merka, G., Regulatory Compliance
Riemer, D. J., Technical Support Manager
Terry, C. L., Senior Vice President and Principal Nuclear Officer
Walker, R. D., Regulatory Affairs Manager

NRC

Allen, D., Project Engineer
Freeman, H. A., Enforcement Specialist
Goldberg, P. A., Sr. Reactor Inspector
Graves, D. A., Senior Project Engineer
Jaffe, D., Senior Project Manager
O'Keefe, N. F., Senior Resident Inspector
Tapia, J. I., Chief, Project Branch A
Vasquez, M., Enforcement Specialist

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-445(446)/200004-01	URI	Failure to identify degraded EDG jacket water coolers due to excessive fouling

LIST OF DOCUMENTS REVIEWED

Listing of radioactive waste and material shipments from February 1999 through June 1, 2000.

Shipping Documentation Packages 1999-15, 1999-16, 1999-44, 1999-66, 1999-72, 1999-78, 2000-03, 2000-08.

Smart forms 1999-000424, 1999-001746, 1999-001779, 1999-003385, 1999-003421, 2000-000214, 2000-000637, 2000-000682, 2000-00838, 2000-001235, 2000-001553.

10 CFR Part 61 Analysis data packages for 1998 and 1999.

Radiation Protection Self Assessment Report, August 25, 1999
Radiation Protection Self Assessment Report, November 30, 1999
Nuclear Overview Department Evaluation Report, EVAL-1999-029, July 16 to August 3, 1999.
Nuclear Overview Department Evaluation Report, EVAL-2000-008, February 25 to March 8, 2000.

Station Administration Manual Procedures:
STA-709, “Radioactive Waste Management Program,” Revision 7
STA-713, “Process Control Program,” Revision 0

Radwaste Systems Manual Procedures:
RWS-109A, “Condensate Polishing System,” Revision 8
RWS-301, “Radwaste Solidification,” Revision 8

Radiation Protection Procedures:
RPI-202, “Receipt of Radioactive Material,” Revision 7
RPI-204, “Dry Active Waste Processing,” Revision 9
RPI-205, “Wet Waste Processing,” Revision 3
RPI-206, “Liquid Process Filter Control,” Revision 10
RPI-210, “Radioactive Waste Cask Handling,” Revision 3
RPI-215, “Waste Stream Sampling,” Revision 2
RPI-216, “Sample Validation and Data Base Maintenance,” Revision 4
RPI-230, “Radioactive Material Shipments,” Revision 1
RPI-231, “Exclusive Use Vehicle Inspection,” Revision 2
RPI-232, “Characterizing Radioactive Material for Shipment,” Revision 3
RPI-233, “Verification of License to Receive Radioactive Material,” Revision 1
RPI-234, “Packaging Radioactive Material for Shipment,” Revision 5
RPI-235, “Marking and Labeling Radioactive Material Packages for Shipment,” Revision 3
RPI-237, “Placarding Radioactive Material Shipments,” Revision 3
RPI-238, “Radioactive Material Shipment Surveys,” Revision 5
RPI-239, “Radioactive Material Shipment Documentation,” Revision 4
RPI-240, “Shipment of Radioactive Waste to Waste Processors,” Revision 3

Training Material:
IS21.DOT.HM1, “DOT Hazardous Materials Shipping,” Revision 03-03-00
RP21.RMC.RW2, “DOT/NRC Shipping Regulations,” Revision 05-03-99
RP26.RMC.SU1, “Survey Radioactive Shipments,” Revision 01-04-99
RP26.RMC.RE1, “Operate Lift Truck Equipment,” Revision 01-04-99
RP26.RMC.SU2, “Perform a Box Loading Survey,” Revision 02-11-99
RP26.RMC.WS2, “Sample Radioactive Waste Streams,” Revision 04-13-94
RP26.RMC.WW2, “Process High Integrity Containers,” Revision 03-29-99
RP26.RMC.DW1, “Process Dry Active Waste,” Revision 01-04-99
RP26.RMC.FL2, “Handling of Process Filters,” Revision 03-29-99
RP26.RMC.PG1, “Package Radioactive Material for Shipment Using Strong Tight Containers,” Revision 02-23-99
RP26.RMC.PG2, “Package Radioactive Material for Shipment Using Specification Packages,” Revision 02-23-99
RP26.RMC.LS2, “Load, Shore, and Brace Radioactive Shipments,” Revision 04-13-94

RP26.RMC.RC2, “Handling Radioactive Waste Casks,” Revision 07-14-99
RP26.RMC.ML2, “Mark, Label, and Placard Radioactive Shipments,” Revision 04-13-94
RP26.RMC.EU2, “Perform Exclusive Use Vehicle Inspection,” Revision 07-23-99

ATTACHMENT 2

NRC'S REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) 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 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	·Occupational	·Physical Protection
·Mitigating Systems	·Public	
·Barrier Integrity		
·Emergency Preparedness		

To monitor these seven cornerstones of safety, the NRC used 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, or 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. 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.