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

July 30, 2001

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: COMANCHE PEAK STEAM ELECTRIC STATION - NRC INSPECTION REPORT 50-445/01-03; 50-446/01-03

Dear Mr. Terry:

On July 7, 2001, the NRC completed an inspection at your Comanche Peak Steam Electric Station, Units 1 and 2. The enclosed report documents the inspection findings which were discussed on July 11, 2001, with Mr. J. J. Kelly 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 inspectors identified two issues that were evaluated under the risk significance determination process as having very low safety significance (Green). The NRC 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 NRC 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 Comanche Peak Steam Electric Station, Units 1 and 2, facility.

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/

Jeffrey A. Clark, Chief
Project Branch A
Division of Reactor Projects

Dockets: 50-445
50-446
Licenses: NPF-87
NPF-89

Enclosure:
NRC Inspection Report
50-445/01-03; 50-446/01-03

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

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

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Dockets:	50-445 50-446
Licenses:	NPF-87 NPF-89
Report:	50-445/01-03 50-446/01-03
Licensee:	TXU Electric
Facility:	Comanche Peak Steam Electric Station, Units 1 and 2
Location:	FM-56 Glen Rose, Texas
Dates:	April 1 through July 7, 2001
Inspectors:	S. C. Schwind, Resident Inspector N. F. O'Keefe, Senior Resident Inspector, South Texas Project G. L. Guerra, Resident Inspector, South Texas Project D. R. Carter, Health Physicist, Operations Branch
Approved By:	J. A. Clark, Chief Division of Reactor Projects

SUMMARY OF FINDINGS

Comanche Peak Steam Electric Station, Units 1 and 2
NRC Inspection Report 50-445/01-03; 50-446/01-03

IR 05000445-01-03; 05000446-01-03; on 04/01-07/7/2001; TXU Electric; Comanche Peak Steam Electric Station, Units 1 & 2. Integrated Res & Reg Rpt; Flood Prot Measures, Access Control to Radiologically Significant Areas

The inspection was conducted by resident inspectors and a region-based senior health physicist. The inspection identified two Green findings that were noncited violations. The significance of the findings are indicated by their color (green, white, yellow, or red) using MC 0609, "Significance Determination Process" (SDP). 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. Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation.

Cornerstone: Initiating Events

- Green. Inspectors identified a violation of design control requirements for two flooding mitigation features, credited in the flooding analyses, that were never installed in the Unit 2 auxiliary feedwater pump rooms. The licensee evaluated this unanalyzed condition and determined that worst-case flooding could render two auxiliary feedwater trains inoperable.

  This issue was determined to have very low safety significance because the licensee was able to demonstrate that at least one train of auxiliary feedwater would have remained unaffected during flooding. The issue had credible safety impact because specific cases of internal flooding could have had a greater impact on mitigation equipment than was analyzed. A noncited violation was identified for inadequate design control (10 CFR Part 50, Appendix B, Criterion III). Reference Smart Form 2001-001257 (Section 1R06).

Cornerstone: Occupational Radiation Safety

- Green. The inspector identified two occasions, during the Unit 1 refueling outage, when radiation protection personnel failed to survey an area prior to workers entering the area. The first occasion was for failure to survey steam generator platform Loop Room 2/3. The second occasion was for failure to survey the overhead of the pressurizer relief tank room. 10 CFR 20.1501(a) requires each licensee to make or cause to be made surveys that are reasonable under the circumstances to evaluate radiation levels, concentrations or quantities of radioactive material, and the potential radiological hazards. The failure to perform radiological surveys in the above areas was a violation of 10 CFR 20.1501(a). This violation is being treated as a noncited violation and is in the licensee's corrective action program as Smart Forms 2001-1619 and 2001-805, respectively.

  The safety significance of this violation was determined to be very low by the Occupational Radiation Safety Significance Determination Process because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised. This violation was more than minor because the failure to perform a survey has a credible impact on safety and the potential for unplanned or unintended dose (Section 2OS1).

Report Details

Summary of Plant Status

Unit 1 began the report period shut down for its eighth refueling outage. On April 22, the unit was synchronized to the grid, completing a 29-day outage. The unit reached 100 percent power on April 28 and continued at approximately 100 percent power for the remainder of the report period.

Unit 2 operated at approximately 100 percent power for the entire report period.

- **REACTOR SAFETY**

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope (71111.01)

The inspector reviewed Station Administrative Procedure STA-634, “Extreme Temperature Equipment Protection,” Revision 3, and Abnormal Operating Procedure ABN-907, “Severe Weather,” Revision 9. The inspector evaluated whether they adequately addressed actions that should be taken to protect safety-related equipment during extreme summer temperatures and severe weather, such as thunderstorms, high winds, and tornados. The inspector also conducted a walkdown of the protected area to assess the threat to risk significant systems from wind-generated missile hazards, due to material stored in the open.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope (71111.04)

On April 18, 2001, the inspectors conducted a partial system walkdown of the Unit 1, Train A battery, using System Operating Procedure (SOP) 605A, “125 VDC Switchgear and Distribution Systems, Batteries and Battery Chargers.” The inspectors also observed the control board for proper indications and assessed the material condition of the battery cells, following replacement of the battery during the Unit 1 refueling outage.

On May 14, 2001, the inspector walked down accessible portions of the Unit 2, Train B containment spray system, using SOP 204B, “Containment Spray System.” The inspector checked the system lineup against the appropriate standby condition and assessed the materiel condition of system components. The inspector observed those mechanical portions of the system from the refueling water storage tank to the containment isolation valves, compared the actual valve positions to the expected positions, and looked for leaking components.

On May 29, 2001, the inspector performed a walkdown of the Unit 1 Train A diesel generator using SOP 609A, "Diesel Generator System," to assess whether it was in a proper standby condition. The walkdown included the switch lineup on the local control panels in the diesel generator room and a valve lineup in the diesel generator room and air compressor room.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope (71111.05)

The inspectors toured fire zones and fire areas to assess the licensee's control of transient combustible materials and the material condition and lineup of fire detection and suppression systems. The inspectors also examined the material condition of manual fire equipment and passive fire barriers and evaluated the effectiveness of compensatory measures for degraded equipment. The following zones or areas were inspected:

- Unit 1 battery rooms, inverter rooms, and hallway (Fire Area EA)
- Unit 1 Train A diesel generator room (Fire Zone 1SG10A)
- Unit 1 Train B diesel generator room (Fire Zone 1SI12A)
- Unit 1 Train A switchgear room (Fire Zone 1SD9)
- Units 1 and 2 control rooms (Fire Zone 1EO65)
- Unit 1 cable spreading room (Fire Zone 1EN64)
- Unit 2 cable spreading room (Fire Zone 2EM63)

In addition, the inspectors reviewed the following response procedures and design basis documents for fires in these areas:

- ABN-803A, "Response to a Fire In the Control Room or Cable Spreading Room," Revision 5

- RXE-92-01A, "Individual Plant Examination," August 1992

- Updated Final Safety Analysis Report

- Design Basis Document DBD-EE-033, Revision 6, "Detailed Control Room Design"

- Fire Preplan Instructions:
  Number 504, Revision 1, Cable Spreading Room
  Number 506, Revision 4, Control Room

Portions of the response procedures were walked down with plant operators and fire protection personnel. The inspectors evaluated the location of repair kits, required by these procedures, and inventoried the contents. The inspectors also checked that operators received recent (November 2000) training on the use of these complicated procedures. The inspectors also assessed whether adequate procedures and communications equipment were available to support operators and the emergency director at the remote shutdown panels.

b Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope (71111.06)

The inspectors conducted an annual inspection of flood protection measures at Comanche Peak. This included a review of flood analysis documentation and calculations to determine areas in the plant susceptible to flooding from internal sources. Based on that review, and a review of the probabilistic risk assessment, a walkdown of the Units 1 and 2 auxiliary feedwater pump rooms was performed to assess the adequacy of flood protection measures. The following documents were reviewed:

- ER-EA-008, “Individual Plant Examination of External Events for Severe Accident Vulnerabilities”
- RXE-92-01A, “Individual Plant Examination”
- Calculation SI-CA-0000-662, Revision 4, “Safeguards Building Unit 1 - Flooding Analysis”
- Calculation CPE-SI-CA-0000-730, Revision 3, “Flooding Shutdown Analysis Calculation”
- Calculation SI-CA-0000-921, Revision 3, “Flooding Analysis - HELB Flows in Unit 1 Safeguards Building Areas”
- PRA Evaluation of the Consequences of Missing Backwater Check Valves In The AFW Pump Room Floor Drains, Revisions 1 and 2
- Smart Form SMF-2001-001257-00
- Safeguards and Auxiliary Building Vents and Drains System Drawings M1/2-0229 and -0236.
- Calculation ME-CA-0206-5187, Revision 0, “Auxiliary Feed Water Pump Room Floor Drains Back Flooding Analysis”

b. Findings

A noncited violation of very low safety significance was identified for inadequate design control when inspectors identified that two flooding mitigation features, credited in the flooding analyses, were never installed in the Unit 2 auxiliary feedwater pump rooms. When the licensee evaluated this unanalyzed condition, it was recognized that worst-case flooding could render two trains inoperable, instead of only one as originally thought.

The inspectors identified that the Unit 2 plant configuration did not match the conditions evaluated in the flooding analyses for the auxiliary feedwater (AFW) system. The system had three pumps located as follows:

Train A motor-driven pump - Room 72
Train B motor-driven pump - Room 73
Turbine-driven pump - Room 74

These rooms were next to each other and shared a floor drain system. The middle room, Room 73, was intended to have an opening to the hallway, approximately 3 feet by 3 feet, below the level of the floor. In Unit 1, this trench was open, but in Unit 2, this opening was sealed. The flooding analyses assumed the trenches were both open, and significant credit was obtained for Train B because flood water drained faster than it could be introduced. Without credit for the trench, Train B would have more impact from flooding.

Additionally, the inspectors noted that Unit 1 floor drains had backwater check valves, but Unit 2 AFW floor drains did not. The flooding analyses assumed no inflow and credited no removal of flood water sources through the drains. Without check valves, the drains should have been evaluated as a flooding source in Unit 2, particularly for postulated flooding in the adjacent AFW pump rooms, since multiple trains could be affected.

The licensee performed detailed flow modeling of the Unit 2 AFW pump rooms to evaluate the combined effect of these two existing conditions. The licensee concluded that worst-case flooding in either motor-driven AFW pump room would have rendered both trains inoperable due to communications through the floor drains between rooms, but that the turbine-driven pump room would neither affect nor be affected by the other rooms. For flooding in the turbine-driven AFW room, only the turbine-driven train would be rendered inoperable. The original calculation results (still valid for Unit 1) concluded that only a single AFW pump would be affected by worst-case flooding.

Significance Determination Process

Entry Conditions

The risk significance of the deficiencies associated with the AFW pump room flooding mitigation features was evaluated in accordance with NRC Inspection Manual Chapter 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for At-Power Situations.”

The inspectors, with assistance from a regional senior reactor analyst, evaluated the significance of this unanalyzed condition. Pipe breaks were postulated for each train in the associated pump discharge line and were assumed to occur while the pump was running and supplying feedwater to steam generators during low power operation. The break was assumed to result in a plant trip from partial loss of feedwater flow. Since the motor-driven pumps were each 50 percent capacity, and the turbine-driven pump was 100 percent capacity, the safety function was maintained in all scenarios.

Significance Determination Process Phase I

The identified condition resulted in degradations in flood protection, which resulted in a degradation in the mitigation systems cornerstone. Since flooding in one motor-driven AFW pump room could render both trains inoperable, the finding was determined to be potentially risk significant using the SDP Phase 1 Screening Worksheet. As a result, a Phase 3 analysis was performed as directed by Question 5 for Mitigating Systems.

SDP Phase 3

A Region IV senior reactor analyst (SRA) performed a risk assessment of the degraded flooding mitigation features.

The scenario of interest occurs when the motor-driven AFW pumps are feeding the steam generators during low power operations. During this time, a flooding event in a motor-driven AFW pump room could result in the loss of both motor-driven AFW pumps, resulting in a plant transient. The transient would be either an automatic or manual reactor trip, due to lowering levels in the steam generators. As previously stated, flooding in a motor-driven AFW pump room could render both trains inoperable but would not impact the turbine driven AFW pump. Using this information, the SRA reviewed the sequences and remaining mitigation systems available for a flood induced transient and a transient with a loss of power conversion system initiating events. The Risk Informed Inspection Notebook for Comanche Peak Steam Electric Station was utilized to evaluate the core damage sequences and credit for remaining mitigating systems. The worst case sequence evaluated by the SRA was Sequence 2 on the worksheet for Transient with a Loss of Power Conversion System (TPCS). Sequence 2 involved the following:

TPCS - AFW - Feed and Bleed (FB)

The flooding frequency for the two motor-driven AFW pump rooms was determined to be 1.51E-4/yr. The scenario of interest occurs when the AFW pumps are feeding the steam generators during low power operations. This alignment occurs infrequently, typically only during a plant startup after completion of a refueling outage. The SRA assumed a duration of 5 days (120 hours) per year. Therefore, in one year, the probability of the flooding scenario of interest, and a resultant plant transient, is 1.51E-4/yr x 120 hours/8760 hrs/yr = 2.1E-6.

The following mitigation capability rating was assigned for each of these terms:

AFW - 1 (motor-driven AFW pumps unavailable due to flooding)
FB - 2

No credit was given for the recovery of the failed train. As a result, the total mitigation capability rating for Sequence 2 was determined to be 3.

Using the mitigation capability ratings discussed above, an approximation of the change in core damage frequency (CDF) was calculated as follows:

delta CDF = TPCS x AFW x FB = 2.1E-6 x 1E-1 x 1E-2 = 2.1 E-9/yr (Green)

The analyst also utilized Manual Chapter 0609, Table 4, “Risk Significance Estimation Matrix,” to determine the color associated with this finding. Table 4 was entered with an Initiating Event Likelihood of “G” and a Remaining Mitigation Capability Rating of “3". The color for Cell G3 was determined to be Green. The remaining sequences for TPCS and transient also resulted in Green findings.

The SRA considered the following assumptions used in this analysis to be conservative:

- The analysis assumed that a flooding scenario would result in a loss of the power conversion system. As a result, the most limiting core damage sequence evaluated involved a TPCS. However, the flooding scenario of interest does not directly cause a loss of the power conversion system.

- The evaluation did not give any credit for operators identifying the flooding condition and taking action to stop the flooding prior to rendering both motor-driven AFW pumps inoperable.

The licensee performed an evaluation to assess the risk impact of flooding affecting two AFW pumps and concluded that the change in core damage frequency was 4.75E-9/year.

In response to this issue, the licensee removed the building materials used to block the trench opening in Unit 2. The licensee initiated Smart Form 2001-001257 to evaluate whether it was necessary to install the backflow check valves or change the license basis to reflect the as-built condition.

The licensee stated that the conditions described above had existed since original construction. The plant design flood mitigation features built into Unit 1 (drain system check valves and open trench in Room 73) were not carried forward to Unit 2, but the change was not recognized or evaluated for impact in the flooding analyses. This was considered to be a violation of 10 CFR Part 50, Appendix B, Criterion III, “Design Control.” This violation is being treated as a noncited violation consistent with of the NRC Enforcement Policy and is in the licensee’s corrective action program as Smart Form 2001-001257 (NCV 50-446/0103-01).

1R11 Licensed Operator Requalification

a. Inspection Scope (71111.11)

The inspectors observed operator performance during a training scenario in the control room simulator, which included a partial loss of offsite power and a partial loss of dc power. The inspectors concentrated on the conduct of operations, procedure usage, command and control, and the simulator instructors’ conduct of the training scenario.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope (71111.12)

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.

- Failure of Relay K1 in the Unit 2 Train A diesel generator on October 12, 2000
- Degraded conditions on the Unit 1 Rod Drive Motor Generator 1-02
- Failure of the Unit 1 Train A diesel generator breaker during testing

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope (71111.13)

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

- Emergent work to isolate and back-seat pressurizer spray Valve 2-PCV-455C due to packing leakage

- Replacement of Relay K1 in the Unit 1 Train A diesel generator

- Modifications to the 138 kV switchyard to add power meters on the buses

- Modifications to the component cooling water supply and return lines for Instrument Air Compressors 2-01 and 2-02

- Six-year overhaul of Diesel Fire Pump X-06

- Replacement of the Unit 2 Train A emergency diesel generator breaker for failure to close on demand on May 9, 2001

The inspectors assessed whether emergent work was identified on risk-significant structures, systems, or components and that the licensee took appropriate steps to plan and control the resulting activities. These included the acceptability of any necessary compensatory actions and contingency plans, when applicable.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope (71111.15)

The inspectors selected operability evaluations, conducted by the licensee during the report period, involving risk-significant systems or components. The inspectors evaluated the technical adequacy of the licensee’s operability determination, assessed the compensatory measures taken, and assessed whether 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:

- Quick Technical Evaluation 2001-0826-01 regarding possible generic concerns of a water hammer event in the Unit 1 Train B centrifugal charging pump alternate recirculation flow path

- Evaluation 2001-1011-01 regarding the acceptability of data from the Unit 1 emergency core cooling system flow balance test

- Quick Technical Evaluation 2001-1016-01 regarding the operability of centrifugal charging pump throttle valves due to loose stem nuts

- Smart Form 2001-1032 regarding failure of the Unit 1 Train A diesel generator output breaker (1EG1) during testing

- Smart Form 2001-0994 regarding operation of the Unit 1 Train B containment spray pump with the pump suctions isolated

- Smart Form 2001-1368 regarding the lifting of the Unit 1 Train A containment spray pump discharge relief

- Smart Form 2001-1386 regarding operability of the Unit 2 Train A residual heat removal system due to a failed limit switch in the containment sump suction valve motor actuator (2-8811A)

- Smart Form 2001-001245-00 regarding excessive oil level in Residual Heat Removal Pump 2-02 motor bearing

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope (71111.16)

The inspectors reviewed the licensee’s list of identified operator workarounds. They also reviewed previously identified degraded conditions on equipment not considered operator workarounds. The inspectors assessed the cumulative effects of the conditions on the ability of operators to respond to plant transients.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope (71111.17)

The inspector reviewed a modification to the Unit 1 reactor vessel head conoseal clamps, which replaced them with an articulated clamp design. The inspector focused on the acceptability of the clamp material, quality control during the fabrication process, and installation and inspection of the new clamps.

The inspector reviewed and observed a plant modification to a pipe trench in the Unit 2 safeguards building. The inspector evaluated the modification for restoration of the structure to its original design basis by removing a seal from the trench which would allow postulated flood water to exit the Train B motor-driven AFW pump room.

The inspector reviewed and observed a modification to the Unit 1 Train B emergency diesel generator which replaced Relay K1 with a newer model. The inspector assessed whether the new relay would provide the same function as the existing model and that the relay had been seismically qualified.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing

a. Inspection Scope (71111.19)

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

- Weld repairs to the Unit 1 Train B chemical and volume control system following a water hammer event during testing

- Unit 1 Train A battery replacement (1ED1 and 1ED3)

- Replacement of Relay K1 in the Unit 1 Train A emergency diesel generator

- Repairs to Boric Acid Transfer Pump 1-01

- Replacement of the air regulator supplying air to the Steam Generator 2-01 Steam Supply Valve (2-HV-2452-2) to the Unit 2 turbine-driven AFW pump

- Time response testing following replacement of Unit 1 Pressurizer Pressure Transmitter 1-PT-457

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope (71111.20)

The inspectors observed activities during the seventh Unit 1 refueling outage. The inspectors evaluated whether plant configuration was controlled in consideration of facility risk, mitigation strategies were developed and properly implemented, and Technical Specification requirements were implemented to maintain the appropriate defense-in-depth. Specific outage activities reviewed and/or observed by the inspectors include:

- Reactor Coolant Pump 1-03 motor replacement
- Core reload
- Reactor coolant system midloop activities
- Reactor coolant system vacuum fill
- Startup preparation

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope (71111.22)

The inspectors evaluated the adequacy of periodic testing of selected important nuclear plant equipment. The inspectors reviewed aspects such as preconditioning; the impact of testing during plant operations; and the adequacy of acceptance criteria, including test frequency and test equipment accuracy, range, and calibration. The inspectors also reviewed 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 test
- Unit 1 Train B safety injection with loss of offsite power test
- Unit 1 Train A containment spray pump test
- Unit 1 Train B diesel generator test
- Unit 2 turbine-driven AFW pump test
- Unit 2 Train A safety injection pump test

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope (71111.23)

The inspector reviewed a temporary plant modification which isolated and back-seated Pressurizer Spray Valve 2-PCV-455C, due to a packing leak. The inspector assessed whether the modification was within the design basis for the reactor coolant system and whether isolation of this valve would present any additional challenges to the operators when implementing the emergency operating procedures.

b. Findings

No findings of significance were identified.

2 RADIATION SAFETY
Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas

a. Inspection Scope (71121.01)

The inspector interviewed radiation workers and radiation protection personnel involved in high dose rate and high exposure jobs during routine operations. The inspector also conducted plant walkdowns within the radiologically controlled area and conducted independent radiation surveys of selected work areas. The following items were reviewed and compared with regulatory requirements:

- Area posting and other controls for airborne radioactivity areas, radiation areas, high radiation areas, locked high radiation areas, and very high radiation areas
- Radiation work permits and radiological surveys involving airborne radioactivity areas and high radiation areas
- Access controls, surveys, and radiation work permits for the following three significant high dose work areas: steam generator eddy current testing (Radiation Work Permit 1-1400, Task 1), steam generator nozzle dam installation (Radiation Work Permit 1-1400, Task 3), and Reactor Coolant Pump 2 seal replacement (Radiation Work Permit 1-1401)
- Dosimetry placement when work involved a significant dose gradient
- Controls involved when handling highly radioactive items
- A summary of corrective action documents written since October 2000 that involved high radiation area and work practice incidents (nine corrective action documents (Smart Forms) were reviewed in detail: 2000-002704, -002830, -002845, -003174, -003354, and -003065 and 2001-000674, -000729, and -000679)
- Radiation Protection 2000 Annual Report and Nuclear Overview Department Evaluations: EVAL-2000-041, EVAL-2000-043

b. Findings

A noncited violation with very low safety significance (Green) was identified for two occurrences of a failure to survey. The first occurrence was a failure to survey the steam generator platform in Loop Room 2/3 prior to personnel entering the area. During the review of circumstances surrounding the removal of the Steam Generator 2 manway diaphragm on March 29, 2001, the inspector determined that decontamination personnel accessed the above platform before an air sample was evaluated (surveyed). At approximately 1:15 a.m., radiation protection personnel performed Steam Generator 2 bowl radiological surveys and sent the air sample to the count room for analyses. The results of this air sample were not known until 3:09 a.m. At approximately 2 a.m., decontamination personnel entered the above loop room to perform their work activities. The results of the platform air sample indicated airborne concentrations of 6.4 derived air concentrations. Licensee procedures require posting airborne radioactivity areas at 0.25 derived air concentrations. The workers were in the area for approximately 30 minutes. The failure to evaluate (survey) an area prior to workers entering it was the first occurrence of a violation of 10 CFR 20.1501(a).

On April 3, 2001, during a tour of the Unit 1 reactor building, the inspector identified a ladder leading to the top of a concrete wall in the overhead of the pressurizer relief tank room on Elevation 835 foot. No radiological information was attached to the ladder. When questioned, radiation protection personnel could not provide a survey of the radiological conditions in the overhead of the pressurizer relief tank room. The licensee performed a survey of the area and determined that radiation levels were less than 1.0 millirem per hour and contamination levels were less than 1000 disintegrations per 100 centimeters squared. The failure to perform a survey prior to a worker entering an area was the second occurrence of the above violation.

The safety significance of the first occurrence of this violation was determined to be very low by the Occupational Radiation Safety SDP because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised. The safety significance of the second occurrence of this violation did not affect the cornerstone since there were no actual or potential consequences which could have been significantly greater as a result of a single minor, reasonable alteration of the circumstances. Both occurrences of this violation were determined to be more than minor because the failure to perform a survey has a credible impact on safety and the potential for unplanned or unintended dose.

10 CFR 20.1501(a) requires licensees to make or cause to be made surveys that may be necessary to comply with regulations and are reasonable under the circumstances to evaluate the magnitude and extent of radiation levels and concentrations or quantities of radioactive material. 10 CFR 20.1003 defines survey as “an evaluation of the radiological conditions and potential hazards incident to the use or presence of radioactive material. When appropriate such an evaluation includes a physical survey of the location of radioactive material and measurement or calculation of levels of concentrations or quantities or radioactive material present.” The failure to perform a survey of the above areas is a violation of 10 CFR 20.1501(a). These occurrences are being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee’s corrective action program as SMART Forms 2001-1619 and 2001-805 (Noncited Violation 50-445;-446/0103-02).

4 OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Initiating Event Cornerstone

a. Inspection Scope

The inspector reviewed performance indicator data submitted by the licensee regarding the initiating event cornerstone to determine its accuracy and completeness. The data reviewed included the number of unplanned scrams, the number of scrams with loss of normal heat removal, and the number of unplanned power changes for both Units 1 and 2 for the first quarter of 2001.

b. Findings

No findings of significance were identified.

.2 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspector reviewed corrective action program records for Technical Specification required locked high radiation areas, very high radiation areas, and unplanned exposure occurrences since October 2000 to confirm that these occurrences were properly recorded as performance indicators. Radiologically controlled area entries with exposures greater than 100 millirems were reviewed, and selected examples were examined to determine whether they were within the dose projections of the governing radiation work permits. Whole-body counts or dose estimates were reviewed if the radiation worker received a committed effective dose equivalent of more than 100 millirems.

b. Findings

No findings of significance were identified

.3 Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences

a. Inspection Scope

The inspector reviewed radiological effluent release program corrective action records, licensee event reports, and annual effluent release reports documented since October 2000 to determine if any events exceeded the performance indicator thresholds.

b. Findings

No findings of significance were identified.

4OA3 Event Followup (71153)

.1 (Closed) Licensee Event Report (LER) 50-445/01-002-00, "Primary Plant Ventilation System Negative Pressure Boundary Test Has Been Performed Non-Conservatively with the Ventilation System Supply Units' Pneumatic Dampers in the Closed Position."

The primary plant ventilation system negative pressure boundary test procedure allowed the pneumatic intake dampers to be closed during the test. These dampers are assumed to fail open on a loss of instrument air during a design basis accident; therefore, the test did not satisfy Technical Specification Surveillance Requirement 3.7.12.4. After this issue was identified, the procedure was modified and the test was satisfactorily performed with the intake dampers open. This condition had no actual or credible impact on safety and constituted a violation of minor significance and is not subject to formal enforcement action. The licensee documented this condition in Smart Form SMF-2001-0447.

.2 (Closed) Licensee Event Report 50-445;446/01-003-00, "Technical Specification Limiting Condition For Operation Applicability Not Consistent With Testing Allowed in Turbine Overspeed Protection System Test."

The licensee identified that the turbine overspeed protection system test procedure allowed both channels of the P-4 interlock (turbine trip on reactor trip) to be de-energized in Modes 2 and 3. However, Technical Specification 3.3.2 requires this interlock to be operable in Modes 1, 2, and 3. Furthermore, they identified two instances where this test was performed in Mode 3 which violated this Technical Specification requirement. In both instances, there was no actual or credible impact on safety since a low steam line pressure signal would have generated a safety injection actuation signal under the given plant condition, which would have accomplished the same function of isolating steam to the main turbine. Therefore, this violation of Technical Specifications is of minor significance and is not subject to formal enforcement action. The licensee documented this condition in Smart Form SMF-2001-0512.

.3 (Closed) Licensee Event Report 50-445/01-004-00, "Steam Generator Tube Plugging due to Stress Corrosion Cracking"

Technical Specification 5.6.10 requires a report be submitted to the NRC when steam generator tube inspections indicate defects on more than one percent of the total number of tubes inspected. During the eighth Unit 1 refueling outage, tube inspections indicated defects on a total of 209 tubes which met this reporting requirement. The inspection results coincided with the predicted 211 tubes which would require plugging during this outage and no new failure mechanisms were identified. No violations of NRC requirements were identified based on a review of this report.

4OA5 Other

4OA6 Meetings, Including Exit

Exit Meeting Summary

The inspectors presented the inspection results to Mr. J. J. Kelly and other members of licensee management at exit meetings on April 6 and July 11, 2001. The licensee acknowledged the findings presented. No proprietary information was identified.

On June 15, 2001, the inspector presented additional inspection results to Mr. D. Wilder, Manager, Radiation and Industrial Safety, and others members of the licensee staff via teleconference. The licensee acknowledged the findings presented.

ATTACHMENT

KEY POINTS OF CONTACT

PARTIAL LIST OF PERSONS CONTACTED

Licensee

M. Blevins, Deputy to Senior Vice President and Principal Nuclear Officer
J. Curtis, Manager, Radiation Protection
D. Davis, Maintenance Manager
S. Ellis, Operations Manager
R. Flores, Deputy to Vice President - Nuclear Engineering and Support
T. Hope, Regulatory Compliance Manager
J. Kelly, Vice President - Nuclear Engineering and Support
M. Lucas, Manager, Nuclear Overview
D. O'Connor, Supervisor, Radiation Protection
D. Moore, Plant Manager
M. Sunseri, Manager, System Engineering
C. Terry, Senior Vice President and Principal Nuclear Officer
R. Walker, Manager, Regulatory Affairs
D. Wilder, Manager, Radiation and Industrial Safety
C. Wilkerson, Senior Licensing Engineer

NRC

S. Schwind, Resident Inspector, Project Branch A

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-446/0103-01	NCV	Inadequate Design Control Results in Increased Impact from Internal Flooding in AFW Rooms (Section 1R06)
50-445;-446/0103-02	NCV	Failure to survey (Section 2OS1)

Closed

50-445/01-002-00	LER	Primary Plant Ventilation System Negative Pressure Boundary Test Has Been Performed Non-Conservatively with the Ventilation System Supply Units' Pneumatic Dampers in the Closed Position (Section 4OA3)
50-445;446/01-003-00	LER	Technical Specification Limiting Condition For Operation Applicability Not Consistent With Testing Allowed in Turbine Overspeed Protection System Test (Section 4OA3)
50-445/01-004-00	LER	Steam Generator Tube Plugging due to Stress Corrosion Cracking (Section 4OA3).
50-446/0103-01	NCV	Inadequate Design Control Results in Increased Impact from Internal Flooding in AFW Rooms (Section 1R06)
50-445;-446/0103-02	NCV	Failure to survey (Section 2OS1)

Discussed

None.