ENCLOSURE 2

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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License Nos.:	NPF-87 NPF-89
Report No.:	50-445/97-20 50-446/97-20
Licensee:	TU Electric
Facility:	Comanche Peak Steam Electric Station, Units 1 and 2
Location:	FM-56 Glen Rose, Texas
Dates:	October 12 through November 22, 1997
Inspectors:	Harry A. Freeman, Acting Senior Resident Inspector Rebecca L. Nease, Resident Inspector
Approved By:	Joseph I. Tapia, Chief, Branch A Division of Reactor Projects
Attachment:	Supplemental Information

EXECUTIVE SUMMARY

Comanche Peak Steam Electric Station, Units 1 and 2
NRC Inspection Report 50-445/97-20; 50-446/97-20

Operations

- Operations personnel response to the Units 1 and 2 trips was prompt, appropriate, and characterized by excellent three-way communications and good command and control (Sections O1.2 and O1.3).

- Inadequate procedural guidance, complicated by a senior reactor operator's incorrect understanding of the system's operation, led to an inadvertent actuation of the low-temperature overpressure protection system during the Unit 2 cooldown. This was a violation of 10 CFR Part 50, Appendix B, Criterion V (Section O4.1).

- Weak procedural guidance, poor communications, and a lack of understanding of the operation of the personnel airlock doors resulted in a violation of Technical Specifications related to ensuring containment integrity during core alterations (Section O4.2).

- Operators performed core alterations prior to establishing direct communications with the control room when they installed lighting inside the core. This resulted in a noncited violation of Technical Specifications (Section O4.3).

Maintenance

- Observed surveillances were well controlled and professional with, thorough prejob briefings and excellent communications (Section M1.1).

- Poor control of work activities in the switchyard and personnel error resulted in an automatic trip of the Unit 1 reactor from 100 percent power (Section M1.2).

- The licensee's retrieval of a damaged fuel assembly was performed in a deliberate and professional manner with a thorough prejob briefing. Good radiological, foreign material, and personnel safety practices were observed (Section M1.4).

- The licensee failed to repair all the structural gaps in the Unit 2 emergency core cooling sumps that had been identified in 1994. This was a violation of 10 CFR Part 50, Appendix B, Criterion XVI (Section M1.3).

- A noncited violation was identified when maintenance workers modified bushings in as many as 79 valve actuators without review and documentation as required by procedure (Section M8.1).

Engineering

- The inspectors found that the licensee's evaluation of an issue concerning control room pressurization surveillance testing was weak in that it failed to encompass all potential vulnerabilities (Section E1.1).

- Engineering provided good support to the operation of the facility by providing thorough and conservative evaluations. This support was sometimes provided after the maintenance actions had already been taken (Section E2.3).

Plant Support

- A noncited violation of Technical Specifications was identified when a radiation protection technician failed to maintain constant physical control of a radiation area key (Section R4.1).

Report Details

Summary of Plant Status

Unit 1

Unit 1 began the inspection period at 100 percent power. On October 27, the unit tripped due to a generator output breaker fault which occurred during relay testing.

Unit 1 resumed power operations and, on October 30, the unit reached full power and remained at 100 percent power through the end of the report period.

Unit 2

Unit 2 began the inspection period at 100 percent power. On October 24, during a downpower in preparation for a refueling outage, the reactor was manually tripped from approximately 10 percent power when a control rod malfunction resulted in four dropped control rods.

The licensee commenced the third Unit 2 refueling outage on October 25. During plant cooldown, the low-temperature overpressure protection system was inadvertently actuated. Unit 2 ended the inspection period in Mode 6 with the core being reloaded.

I. Operations

O1 Conduct of Operations

O1.1 General

The conduct of operations observed was characterized by good command, control, and communications. Licensed operator responses to both the Units 1 and 2 trips were professional and well controlled. Reduced inventory control and midloop operations were good. However, several incidents occurred involving errors by experienced personnel.

O1.2 **Unit 1 Reactor Trip**

a. Inspection Scope (93702)

At 10:44 a.m. on October 27, Unit 1 experienced an automatic trip from 100 percent power due to a loss of load and turbine trip. All safety systems functioned as designed. Inspectors reported to the control room and observed the response to the event, including operator monitoring of annunciators, supervisory control, and posttrip actions. Operators exhibited good response to the event as exemplified by manually starting auxiliary feedwater in anticipation of an automatic initiation. Following the trip, Source Range N-31 energized as expected, but immediately failed downscale. This was the only material condition problem experienced as a result of the trip.

b. Observations and Findings

Operator response to the event was prompt and appropriate, operators used clear three-way communications, and the unit supervisor and shift manager demonstrated good command and control in directing the event response. The material condition of the plant was very good and only minor problems were experienced as a result of the trip.

O1.3 **Unit 2 Downpower and Reactor Trip**

a. Scope (71707, 93702)

On October 24 and 25, the inspectors observed control room operators ramp down reactor power in preparation for a unit shutdown for Refueling Outage 2RFO3 using the following procedures:

IPO-003B, "Power Operations," Revision 4
IPO-004B, "Plant Shutdown from Minimum Load to Hot Standby," Revision 2

b. Observations and Findings

During the downpower, with Unit 2 at 56 percent power, a rod control urgent failure alarm occurred at approximately 11:40 p.m on October 25. The licensee immediately began troubleshooting. At approximately 12:30 a.m., the licensee resumed ramping down power by borating while troubleshooting on the rod control system continued. At 4:04 a.m., with the unit at approximately 10 percent power, three control rods dropped approximately 20 steps and another dropped to the bottom. Operators immediately tripped the reactor and entered the emergency operating procedures. All equipment operated as expected and there were no emergency safety features actuations.

The downpower was well controlled and performed in accordance with procedures. The operators used excellent three-way communications throughout the routine downpower. Operations personnel responded to the rod drop in a deliberate manner without hesitation. The unit supervisor demonstrated excellent command and control, calling out the steps in the emergency operating procedure clearly and methodically. The operators responded well.

O4 Operator Knowledge and Performance

O4.1 Low-Temperature Overpressure Protection Actuation

a. Inspection Scope (71707)

On October 25, the licensee inadvertently actuated the low-temperature overpressure protection system during the Unit 2 cooldown. Both pressurizer power-operated relief valves actuated. After verifying that the pressure was below the actuation setpoint, operators closed the block valves to secure the depressurization. The inspector reviewed the operating procedures, the alarm response procedures, and the technical data manual procedure used during plant cooldown.

b. Observations and Findings

The low-temperature overpressure protection system is an automatic system that protects the reactor vessel and piping systems against pressure transients during low temperature operation. The system continuously monitors reactor coolant temperature and pressure. System temperature is converted into an allowable pressure and then compared to the actual reactor coolant system (RCS) pressure. If pressure approaches the allowable pressure, a control board annunciator will sound. If pressure exceeds allowable pressure, another control board annunciator will sound and one or both pressurizer power-operated relief valves will open if the RCS temperature is ≤350°F.

Plant cooldown was conducted in accordance with Integrated Plant Operating Procedure IPO-005B, "Plant Cooldown from Hot Standby to Cold Shutdown," Revision 2. The inspector noted that, in several locations throughout the procedure, a statement cautioned that RCS pressure and temperature should be maintained within the limits of Technical Data Manual TDM-301B, "RCS Temperature & Pressure Limits," Revision 5. TDM-301B contained a graph showing the pressurizer power-operated relief valve low-temperature overpressure protection setpoints. Neither the TDM nor the integrated plant operating procedure referenced that the system armed when reactor coolant temperature was ≤350°F.

During the cooldown, operators slowed the pressure reduction to maintain pressure between 850 and 900 psig while the safety injection accumulators were being isolated but did not slow the rate of cooldown. The procedure required that the accumulators be isolated prior to reducing pressure below 800 psig. During this time, operators received Alarm 2-ALB-6D, "AT LO TEMP PORV 455A APPROACHING LMT PRESS." This annunciator alarms when pressure is within 20 psig below the reference pressure. Operators referenced the alarm response procedure which required that they refer to TDM-301B in order to determine the RCS pressure and temperature limits. The procedure did not require that pressure be immediately reduced to clear the alarm. Accumulator isolation was completed and operators recommenced the pressure reduction. At approximately 350°F, the low-temperature overpressure protection system actuated with pressure at approximately 735 psig. The power-operated relief valves were shut when pressure dropped below the setpoint. Following the actuation, the operators stated that they believed that the system armed at 320°F.

10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by instructions of a type appropriate to the circumstances. Contrary to this requirement, Procedure IPO-005B, Revision 2, was not appropriate to the circumstances in that the procedure did not require that pressure be maintained below the RCS pressure and temperature limits as specified

in TDM-301B and, as a consequence, both power-operated relief valves opened when pressure was allowed to remain above the limit (50-446/9720-01).

In Special Report 2-SR-97-001 to the NRC Region IV Regional Administrator (licensee letter TXX-97250, dated November 24), the licensee committed to the following: (a) assess if the misconception of the arming temperature is a general knowledge deficiency and conduct appropriate training if deemed necessary, (b) revise Alarm Procedures ALM-0064A and -B to include the arming setpoint, and (c) revise Integrated Plant Operating Procedures IPO-005A and -B to add appropriate provisions to establish temperature and pressure conditions during cooldown which will not unnecessarily challenge low-temperature overpressure protection system actuation.

c. Conclusions

Inadequate guidance in both the integrated plant operating procedure and in the alarm response procedure contributed to an inadvertent actuation of the low-temperature overpressure protection system. Additionally, incorrect knowledge of system operation by a senior reactor operator contributed to the event.

O4.2 Configuration Control of Containment Integrity During Core Alterations

a. Scope (71707)

In following the licensee's investigation and corrective actions concerning recent challenges to maintaining containment integrity during core alteration, the inspectors reviewed the following licensee documents:

Operations Notification and Evaluation Form 97-1384
Operations Notification and Evaluation Form 97-1397
Plant Incident Report 97-1378
Procedure SOP-907B, "Containment Personnel Airlocks," Revision 3
Procedure OPT-408B, "Refueling Containment Integrity Verification," Revision 2.

b. Findings and Conclusions

Loss of Containment Integrity: On November 10, the licensee was performing core alterations with both PAL doors open and de-energized. Technical Specification 3.9.4 requires, in part, that, during core alterations, at least one of the two PAL doors be capable of being closed and that each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be closed. A plant equipment operator responsible for ensuring closure of the PAL noted that several hoses associated with the PAL hydraulic system were disconnected to support maintenance activities. The equipment operator questioned if this configuration would affect his ability to close the inner door. At the time, the licensee was relying on the capability to manually close the inner PAL door for meeting Technical Specification 3.9.4. Operations immediately suspended fuel movement.

The PAL system engineer determined that the inner door could not be considered operable because its pressure equalizing valve, considered part of the PAL door, was open. The open equalizing valve provided a direct containment atmosphere to outside atmosphere pathway that would not have isolated when the inner door was manually closed. A miscommunication between operations personnel and the system engineer resulted in the outer door being declared operable and core alterations were resumed. Upon closer inspection, the system engineer found that the pressure equalizing valve for the outer door was also partially open, rendering the outer door inoperable. Again, fuel movement was suspended. Operators re-energized the outer door and closed its associated pressure equalizing valve, ensured that the outer door was capable of being closed, and then resumed core alterations. Conducting core alterations with the PAL doors being incapable of being closed is a violation of Technical Specification 3.9.4 (50-446/9720-02).

Procedures OPT-408B, "Refueling Containment Integrity Verification," and SOP-907B, "Containment Personnel Airlocks," were weak in that they did not provide clear instructions concerning manual operation of the PAL doors and associated pressure equalizing valves. The licensee revised both procedures to provide precautions and instructions for operating the PAL doors manually to establish containment integrity. In addition, several other weaknesses were identified. The plant equipment operator responsible for PAL door closure was not aware that the PAL doors were de-energized and would have to be closed manually. The licensee determined that few operators fully understood the operational status of the airlock and pressure equalizing valves when the PAL doors were required to be manually operated. Core alterations were resumed prior to verification of the operability of the outer door due to a miscommunication between operations and the system engineer.

Near Miss: On November 11, with core alterations in progress, the nuclear steam supply system work window manager recognized a potential containment integrity breach when he discovered that the hand hole covers in SG 2-04 had been removed. Another concurrently scheduled work activity, to remove and replace a feedwater instrument isolation valve, would have completed an open pathway through the hand holes, Steam Generator 2-04, and the feedwater instrument isolation line to the outside atmosphere. Upon discovery, the licensee immediately suspended core off-load and initiated an investigation which verified that the valve had been welded in place prior to the time when the hand hole covers had been removed during core alterations. Technical Specification 3.9.4 had not been violated.

The inspectors reviewed this near miss and found that, during the outage, the licensee relied on sequencing work to ensure containment integrity during core alterations. Specifically, when work on containment penetrations inside containment had been completed, those penetrations were closed, and work packages were issued to allow work on penetrations outside containment. Eddy current work in Steam Generator 2-04 had been completed and the steam generator hand hole covers were installed.

However, since the work package was not closed and the clearance was still in effect, a technician removed the hand hole covers for some followup eddy current investigation. Operations personnel did not authorize the hand hole cover removal and were unaware that the containment boundary had been opened. During this time frame, the work order for removing and replacing the feedwater instrument isolation valve had been authorized. Had both jobs been conducted at the same time, an air-to-air breach of containment would have occurred during core alterations and would have been a violation of Technical Specification 3.9.4. The inspectors concluded that controls for maintaining containment integrity were weak and, in this case, would not have prevented a breach in containment integrity during core alterations.

The licensee implemented the following intermediate corrective actions to prevent recurrence during fuel reload: (1) no work was permitted on containment boundary equipment during core alterations and all open work order packages were retrieved; (2) the NSSS work window manager was assigned the primary responsibility for containment integrity; and (3) workers were not permitted to sign back on a clearance until the work order had been re-reviewed by operations.

c. Conclusions

The inspectors concluded that the licensee's controls for ensuring containment integrity during core alterations were weak. Weak procedures for operating the PAL doors resulted in the failure to meet the Technical Specification requirement of being capable of closing the PAL door during core alterations. Contributing factors to this included poor communications and operator knowledge deficiencies. Finally, relying on the sequencing of work activities in the outage schedule would not have prevented work activities from occurring concurrently and could have resulted in a breach of containment integrity during core alterations.

O4.3 Core Alterations Without Communications

a. Inspection Scope (71707)

On November 8, the licensee inserted an underwater light into the reactor vessel in preparation for fuel off-load prior to establishing direct communications between the control room and the refueling station. The inspector reviewed the cause of the event and the licensee's immediate and planned corrective actions.

b. Observations and Findings

Technical Specifications define core alterations as the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Technical Specification 3.9.5 requires that direct communications be maintained between the control room and personnel at the refueling station during core alterations. Contrary to these requirements, the licensee placed an underwater light inside the reactor vessel prior to establishing direct communications between the control room and the refueling station.

After the light had been placed into the vessel, the fuel handling supervisor was alerted to the fact that this constituted a core alteration. The fuel handling supervisor halted further fuel handling activities. As part of their corrective actions, the licensee issued a lessons learned report on the event, established signs on the refueling bridge to remind personnel of what constituted a core alteration, and conducted refresher training to all fuel handling supervisors and contractor personnel involved with the refueling. The licensee issued a ONE form which will be dispositioned as a plant incident report.

Placing an underwater light into the reactor vessel above the fuel had negligible effect on core reactivity. In the improved standard Technical Specifications, not yet approved for the licensee, core alteration has been redefined to be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The inspector concluded that the event had little actual significance and that the licensee's corrective actions were appropriate. This nonrepetitive, licensee-identified, and corrected violation is being treated as a noncited violation, consistent with Section VII.B.1 of the NRC Enforcement Policy (50-446/9720-03).

c. Conclusions

A noncited violation of Technical Specifications was identified related to performing core alterations prior to establishing direct communications with the control room when operators installed lighting inside the core.

O8 Miscellaneous Operations Issues (92901)

O8.1 (Closed) Violation 50-445(446)/9706-01: failure to establish procedures to ensure that all licensed operators that have their licenses conditioned to wear corrective lenses for the eye, always have available appropriate lenses qualified for a self-contained breathing apparatus. In the enclosed Notice to NRC Inspection Report 50-445(446)/97-06, the NRC concluded that the information regarding the reason for the violation and the corrective actions taken and planned were already adequately addressed on the docket and that a response was not necessary unless the licensee concluded that the descriptions or corrective actions did not accurately reflect their position. The licensee did not respond and this item is closed.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

In general, maintenance and surveillance activities were characterized by knowledgeable individuals and professional performance. Although the planning and preparation prior to the start of the Unit 2 outage was thorough, numerous problems occurred during the outage. This included damaged equipment, work stoppages, and personnel injury. Several of these issues involved mistakes by experienced personnel.

a. Inspection Scope (61726, 62707)

The inspectors observed all or portions of the following work activities:

Turbine-driven Auxiliary Feedwater Pump design modifications
Diesel Generator 2-01 maintenance
OPT-214A, "Diesel Generator Operability Test," Revision 10
OPT-467A, "Train A Safeguards Slave Relay K609 Actuation Test," Revision 3

b. Observations and Findings

The inspectors found the work performed under the above activities to be well controlled and professional. Prework briefings were timely and thorough. Excellent three-way communications were observed throughout the performance of the work. Individual work groups reviewed the work steps and discussed the potential consequences and appropriate responses. Operators performing the work read the steps aloud before performing them.

M1.2 Unit 1 Reactor Trip Review

a. Scope

In followup to the October 27 reactor trip, the inspectors reviewed the licensee's immediate and interim actions for controlling work activities in the switchyard. The inspectors also reviewed the Interface Responsibilities Memorandum, dated July 2, 1997, which described the licensee's process for controlling work in the switchyard.

b. Observations and Findings

On October 27, Unit 1 experienced an automatic trip from 100 percent power due to a loss of load and turbine trip (Section O1.2). Unit 2 was in a refueling outage, and the Unit 1 generator was supplying the grid through Breaker 8010 to the west bus of the 345kV switchyard. The east bus of the 345kV switchyard was isolated to support lightning protection modifications. At the time of the trip, Glen Rose Transmission protection and control technicians were testing Breaker 8000 on the east bus of the 345kV switchyard. Glen Rose Transmission is a separate division within the TU Electric Company. In preparation for testing, technicians had closed Breaker 8010 and opened the two air switches on either side. When Breaker 8000 received the test trip signal to open, an unexpected pole disagreement delayed the opening enough to allow the protective lockout function timer to start and time out. This resulted in a protective lockout relay actuation which opened Breaker 8010, causing a loss of load and subsequent Unit 1 turbine and reactor trips. All work in the switchyard was immediately suspended and was permitted to resume only after licensee management was fully briefed on the scope and potential impact. The licensee determined that the cause of the trip was failure of Glen Rose Transmission technicians to defeat the protective lockout feature on Breaker 8000.

The inspectors reviewed the licensee's process for controlling work in the switchyard and found it to be ambiguous and lacking formality. The Interface Responsibilities Memorandum, dated July 2, 1997, provided little guidance concerning review, approval, or coordination of switchyard work. The interface agreement did not require switchyard work to be reviewed by Comanche Peak personnel. Switchyard work was controlled through the work control scheduling process. Typically, only work thought to have a potential for impacting plant operation was shown on the schedule and not all switchyard work was scheduled.

The licensee's investigation revealed several weaknesses associated with control of work in the switchyard. Since few Glen Rose Transmission technicians had site access, the technicians actually performing the work were not necessarily those that attended the prejob briefing. Contrary to the Interface Responsibilities Memorandum, a single point of contact between Operations and Glen Rose Transmission had not been designated. Switchyard breaker testing was performed without procedures, relying on skill of the craft and electrical diagrams. Only that work which was scheduled received Comanche Peak review for impact to the plant. The testing of Breaker 8000 was not scheduled and as a result did not receive Comanche Peak review for impact to the plant.

The licensee has established a task team to develop long-term corrective actions. In the interim, all switchyard work is described on a written plan, reviewed by Comanche Peak management, and briefed by Glen Rose Transmission technicians.

c. <u>Conclusions</u>

The inspectors concluded that the trip was caused by personnel error because Glen Rose Transmission protection and control technicians did not fully understand or review the impact that the breaker testing on the east bus could have on the west bus. In addition, the licensee lacked a formal process for effectively controlling activities in the switchyard.

[illegible] Inadequate Corrective Actions on Containment Sumps

a. Inspection Scope (62707, 92902)

On November 8, the licensee identified 16 gaps in the emergency core cooling system structural support members that were larger than the fine mesh openings. Following completion of the repairs to the Unit 2 sumps, the inspectors visually confirmed that the gaps identified by engineering had been repaired. The inspectors reviewed the work orders originally used to repair the gaps in 1994.

b. Observations and Findings

In 1994, Violation 50-445(446)/9423-03 was issued which stated that the licensee's design control measures did not adequately verify that the Units 1 and 2 containment sump trash racks met design requirements. Structural gaps in each of the emergency core cooling system containment sumps were larger than the fine mesh openings and did not meet design basis requirements. The licensee repaired the Unit 2 gaps under Work Orders 1-94-078046-00 and 1-94-078047-00. On November 8, 1997, the licensee identified nine gaps in the Train A sump and seven gaps in the Train B sump which did not meet the design basis requirements. The gaps were approximately 0.625 inches long and the two widest were between 0.250 inches and 0.375 inches wide. Of these 16 gaps, 8 had been previously identified in 1994, but had not been adequately corrected. When the sump gap issue was originally identified in 1994, the licensee contracted with Westinghouse to perform an engineering evaluation to consider what affect the ingestion of debris into the containment sumps could have on containment spray and emergency core cooling. That evaluation concluded that the gaps did not affect the operability of these safety systems. The inspector verified that the recently identified gaps were bounded by the 1994 evaluation and that the operability of the systems was again not affected.

The inspector reviewed the work orders used in 1994 to repair the Unit 2 sump structures. Neither work order specified the locations of the gaps to be repaired. Instead, the work orders directed the work group to use stainless steel plates to cover openings and reduce the gaps to no more than 0.115 inches. The work orders required that a quality control inspector inspect the structure for holes or gaps, which exceeded the criteria specified in the design change. The design change directed that any hole or gap with a side dimension or diameter greater than 0.115 inches that would allow debris to flow into the sump be repaired. The quality control inspections had been documented as being satisfactory.

FSAR Section 6.2.2.2.7 stated that the fine screen had a 0.115 inch opening to ensure that the spray nozzle orifices and grid assemblies in the reactor core do not clog and that trash racks and screens are provided to preclude clogging of the recirculation lines and any of the system's components. The section stated that the design of the containment spray recirculation sumps satisfied the requirements of NRC Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," dated June 1974. Regulatory Guide 1.82 states that the size of the openings in the fine screen should be based on the minimum restrictions found in systems served by the sump.

The original ONE form which documented the deficiencies in 1994 did not specify the locations of gaps but instead stated that there were approximately 48 gaps in the Unit 2 sumps. The ONE form wording was not clear as to whether there were 48 gaps in each sump or 48 gaps total. Weld data record sheets in the work orders indicated that 71 repairs were made in the Train A sump and that 81 repairs were made in the Train B sump.

10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," requires, in part, that conditions adverse to quality and nonconformances be promptly identified and corrected. Contrary to this requirement, 16 gaps greater than the maximum allowable gap were not promptly identified and corrected. On November 8, 1997, eight previously identified and eight additional structural gaps, that exceeded the Final Safety Analysis Report description of 0.115 inches, were identified in the Unit 2 emergency core cooling sumps. This is a violation (50-446/9720-04).

c. Conclusions

The licensee failed to promptly identify and correct the gaps in the Unit 2 emergency core cooling system containment sumps. The original work orders were poorly written because they did not specify the locations of the gaps to be repaired but instead relied on the work groups to locate and repair the gaps. Several gaps which did not meet the design basis requirements had been identified in 1994.

M1.4 New Fuel Assembly Damage

a. Inspection Scope (62707, 93702)

The inspector responded to the site in response to a fuel handling event in order to verify that the fuel assembly was in a safe condition and did not represent a potential radiological hazard to stored spent fuel. The inspector reviewed the cause of the damage, the licensee's actions to secure the damaged assembly, and the actions taken to prevent future damage.

b. Observations and Findings

While lowering a new fuel assembly into the high density storage racks in Spent Fuel Pool X-02 in high speed, the fuel handler observed a sudden decrease in load cell reading concurrent with unexpected movement of the long-handled tool attached to the assembly. The operator immediately raised the assembly. The operator and fuel handling senior reactor operator observed that the assembly appeared to be damaged. All fuel movement was secured while the licensee conducted an investigation. The licensee suspended placing any more assemblies into the high-density fuel racks until procedural, personnel, and/or equipment weaknesses could be resolved.

The inspector verified that the damaged fuel assembly did not represent a hazard to any spent fuel assembly. The inspector noted that the licensee had stationed a refueling senior reactor operator in the spent fuel pool area to monitor any potential changes to the assembly since it was being held by the long-handled tool and to prevent any unauthorized individual from approaching the refueling bridge. Because the fuel assembly was being held by only 4 of the 25 guide tubes, the licensee secured the assembly to the hoist by using two loops of aircraft cable slung around the lower nozzle. The licensee believed that any lateral movement through the pool could damage the remaining guide tubes, so a temporary design change was implemented to allow the use of the cable prior to removing the assembly from the pool.

The fuel assembly was damaged when an edge of the lower nozzle contacted the edge of the fuel rack during lowering. As the weight of the assembly shifted from the long-handled tool to the fuel rack, the fuel assembly began to lean. A postevent review by the fuel vendor concluded that a 3 degree lean angle could damage the fuel assembly. The vendor also stated that moving the fuel would not be advisable unless seven or eight of the guide tubes remained intact.

The fuel handler was required by procedure to lower the fuel assembly into the high density racks in slow speed until 10 inches had been inserted into the rack. The assembly could then be lowered in high speed. The fuel handler had one hand on the pendent controlling the hoist and the other hand on the long-handled tool to help guide the assembly into the rack. The handler was required to monitor a load cell located on the hoist, guide the assembly into the rack, observe when 10 inches had been inserted, and then shift to fast speed in a smooth motion. Although the fuel handler was a contractor that had a significant amount of experience in fuel movement, he failed to correctly perform this evolution because he did not wait until the assembly was inserted at least 10 inches into the rack before lowering it in high speed.

The licensee's prejob briefing for the retrieval of the damaged fuel assembly was thorough and appropriately focused on personnel, radiological, and equipment safety. Self-checking and verification techniques were stressed. Access to the refueling area was tightly controlled to avoid confusion. Throughout the evolution, the licensee demonstrated proper radiological safety and foreign material exclusion practices. The inspectors concluded that all aspects of this difficult evolution were well controlled and performed in a deliberate and professional manner.

c. <u>Conclusions</u>

The inspector concluded that the damaged fuel assembly was damaged due to operator error. The inspector concluded that the damaged fuel assembly was carefully controlled and moved and that the licensee's plans to suspend further use of the high-density fuel racks until the issue could be resolved was appropriate.

M8 Miscellaneous Maintenance Issues (92902)

M8.1 (Closed) Unresolved Item 50-445(446)/9717-03: modifications made to Fisher-Type 667 air-operated valve actuator bushings without the required review. As previously described in NRC Inspection Report 50-445(446)/97-17, this unresolved item was opened for the inspectors to: (1) evaluate the extent and impact of modifications to the actuator bushings, (2) determine if the appropriate postmaintenance tests were performed on the affected steam generator atmospheric relief valves (ARVs), (3) verify that the licensee corrected the Master Equipment List, and (4) evaluate any generic aspects of the issue.

The licensee completed its investigation and did not identify any additional modifications to actuator bushings, bringing the maximum number of unreviewed modifications to 79, most of which were not safety related. The licensee concluded that the modifications were acceptable because the small holes drilled in the actuator bushings did not affect their functional integrity. The inspectors agreed with this evaluation. The licensee also determined that the steam generator ARVs had been postmaintenance tested after the elastomers were replaced. The inspectors reviewed the Master Equipment List entries for the ARVs and found that they had been corrected. After conducting interviews with maintenance personnel and management and after reviewing the licensee's investigation report, the inspectors did not identify any generic concern related to the issue.

Technical Specification 6.8.1 requires, in part, that the licensee establish, implement, and maintain procedures covering the activities referenced in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Appendix A, Section 9, requires procedures for performing maintenance. Licensee Procedure STA-206, "Review of Vendor Documents and Vendor Technical Manuals," Revision 19, stated that vendor documents or correspondence that will be used for design, testing, or other input for CPSES activities shall receive review and approval on a vendor document review traveler or be incorporated into an applicable vendor technical manual prior to final acceptance and approval of the activity. Inspectors concluded that the licensee violated Technical Specification 6.8.1 in that maintenance personnel failed to follow procedure when they modified as many as 79 Fisher-Type 667 valve actuator bushings (some of which were associated with safety-related valves) without proper documentation or review. This nonrepetitive, licensee-identified, and corrected violation is being treated as a noncited violation consistent with Section VII.B.1 of the NRC Enforcement Policy (50-445(446)/9720-05).

III. Engineering

E1 Conduct of Engineering

E1.1 Control Room Pressurization Unit Surveillance Plant Incident Evaluation

a. Inspection Scope (37551)

The inspector reviewed Plant Incident Evaluation 97-934 which addressed control room pressurization unit surveillance test acceptance criteria inconsistencies. The evaluation was initiated following questions raised by the inspector as documented in NRC Inspection Reports 50-445(446)/97-17 and 50-445(446)/97-18. The inspector reviewed the evaluation to assess the thoroughness and quality of the self-assessment.

b. Observations and Findings

The inspector found that the evaluation did not include any type of formal charter documenting the purpose or scope of the evaluation. Consequently, the inspector was unable to assess whether the evaluation fully met the licensee's expectation that all of the issues that should have been addressed be resolved. Nevertheless, the evaluation thoroughly reviewed the issue of leaving the as-left values of pressurization flow above the design basis value.

The inspector reviewed the data provided in the evaluation and noted that there appeared to be a correlation between the measured flow rates and the dates that the test was performed. Tests conducted during the summer months were generally measured at higher flow than tests conducted during the winter. Additionally, the inspector noted that the licensee was committed to ANSI/ASME N510-1980, "Testing of Nuclear Air-Cleaning Systems," which stated that the number of readings taken to determine flow through the duct should not be less than 16. The inspector noted that the procedure only required 12 readings. The licensee issued another ONE form to address both the adequacy of the evaluation and why only 12 readings were required.

c. Conclusions

The inspector concluded that, while the evaluation was conducted by qualified personnel who thoroughly reviewed the specific issue of exceeding the design basis, it did not encompass all potential weaknesses in testing the pressurization units. The licensee initiated additional ONE forms to address the additional questions raised by the inspector.

E2 Engineering Support of Facilities and Equipment

E2.1 Dual Train Component Cooling Water System Outage

a. Inspection Scope (37551)

The inspector reviewed 10 CFR 50.59 Evaluation SE 97-81 which concerned a dual train component cooling water system outage during the Unit 2 refueling outage following the reactor core offload.

b. Observations and Findings

Prior to the Unit 2 refueling outage, the inspector questioned the nuclear steam supply system work window manager regarding the safety evaluation prepared for conducting a dual train outage. The work window manager was not aware of any evaluation. Within the next few days, the licensee informed the inspector that D. C. Cook had made a 10 CFR 50.72(b) notification for being outside design basis for conducting a dual train component cooling water system outage. The licensee informed the inspector that they had pulled their planned dual train outage out of the schedule until they had a chance to complete their safety evaluation.

The inspector reviewed the licensee's safety evaluation. The licensee concluded that the planned dual train component cooling water system outage with the reactor core fully offloaded did not represent an unreviewed safety question. The inspector found that the licensee's evaluation was thorough and technically correct.

E8 Miscellaneous Engineering Issues (92902)

E8.1 (Closed) Inspection Followup Item 50-445(446)/9718-02: seismic qualification of integrated leak rate test rig. This item was left open to review the licensee's evaluation of the connection. In ONE Form 97-1103, the licensee concluded that, during a design basis earthquake, the rig could potentially have an undesirable interaction with one source range nuclear instrument cable. Although the licensee concluded that a damaged source range nuclear instrument cable would not have prevented the safe shutdown of the reactor, the licensee determined that the connection should be modified to assure conservatism for future operations. The inspector concluded that the licensee's evaluation was thorough and that the proposed modification was conservative.

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

The inspectors observed good radiological practices being implemented by all plant personnel. Workers were familiar with their radiological work permit requirements. On one occasion, the inspectors observed a crane operator swinging potentially contaminated reactor vessel studs outside of the controlled area boundary to avoid hazards to personnel inside the area. After the inspector pointed out that the studs had to be surveyed prior to passing outside of the boundary, the crane operator ensured that the studs remained inside the boundary during the movement.

R4 Staff Knowledge and Performance

R4.1 Locked High Radiation Area Key Control

a. Inspection Scope (71750)

On November 8, the licensee discovered that positive control of a locked high radiation area key had not been maintained. After contacting the individual who signed for the key, the licensee located the key in a bag at the steam generator platform. The inspector reviewed the circumstances and corrective actions surrounding the control of the locked high radiation area key.

b. Observations and Findings

Technical Specification 6.12.2, "High Radiation Area," requires, in part, that areas accessible to individuals with radiation levels greater than 1000 mrem/h at 30 cm shall be provided with locked doors to prevent unauthorized entry and that the keys shall be maintained under the administrative control of the shift manager on duty and/or radiation protection supervision.

Radiation Protection Instruction RPI-110, "Radiation Protection Shift Activities," Revision 6, provided administrative instructions on the control of keys to locked high radiation areas. Section 6.5.1.4 required that radiation area keys only be issued to the security shift lieutenant, the shift manager, or a radiation protection qualified individual. Section 6.5.1.3 required that the individual, to whom the key is issued, maintain constant physical possession of the key.

Contrary to the requirements of RPI-110, a radiation protection technician to whom the key was issued failed to maintain constant physical possession of the key when the key was given to a contractor to unlock a cover on a steam generator manway. When the oncoming crew conducted a key inventory, they discovered that the key was missing. When contacted, the technician to whom the key was issued informed the lead technician that he had directed that the key be left in a bag on the steam generator platform, where it was subsequently found.

The inspector found that the area where the key was located was controlled by radiation protection and that the key was not used by an individual to gain unauthorized access to a locked high-radiation area. As a corrective measure, the radiation protection manager reiterated the requirements of the procedure and his expectations that the procedures be followed. Training was conducted with all radiation protection personnel.

c. Conclusions

Failure of the radiation protection technician to maintain constant physical control of the radiation area key was a violation of Technical Specification 6.12.2. This nonrepetitive, licensee-identified, and corrected violation is being treated as a noncited violation consistent with Section VII.B.1 of the NRC Enforcement Policy (50-445(446)/9720-06).

S1 Conduct of Security and Safeguards Activities

Throughout the inspection period, the inspectors observed alert security officers appropriately manning their assigned posts. On one morning, the inspectors noted that fog had significantly reduced visibility. The inspectors verified that security had appropriately responded as required by the security plan.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the results of the inspection to members of licensee management at the conclusion of the inspection on November 25. The licensee stated that they had not yet completed their investigation into the emergency core cooling system sump gap issue. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

M. R. Blevins, Plant Manager
J. R. Curtis, Radiation Protection Manager
D. L. Davis, Nuclear Overview Manager
J. J. Kelley, Vice President, Nuclear Engineering and Support
M. L. Lucas, Maintenance Manager
D. R. Moore, Operations Manager
C. L. Terry, Group Vice President, Nuclear Production

INSPECTION PROCEDURES USED

37551 Onsite Engineering

61726 Surveillance Observations

62707 Maintenance Observations

71707 Plant Operations

71750 Plant Support Activities

92901 Followup - Plant Operations

92902 Followup - Maintenance

92903 Followup - Engineering

93702 Prompt Onsite Response To Events At Operating Power Reactors

ITEMS OPENED AND CLOSED

Opened

50-446/9720-01	VIO	Inadequate procedure resulted in both power-operated relief valves opening when pressure was allowed to remain above the low-temperature overpressure protection system limit (Section O4.1).
50-446/9720-02	VIO	Conducting core alterations with the personnel airlock doors incapable of being closed (Section O4.2).

50-446/9720-03	NCV	Failure to establish communications prior to placing an underwater light in the reactor vessel (Section O4.3).
50-446/9720-04	VIO	Failure to promptly identify and correct 16 nonconformances in the ECCS sumps including eight previously identified structural gaps (Section M1.3).
50-445(446)/9720-05	NCV	Failure to follow procedure when modifying Fisher-Type 667 valve actuator bushings without proper documentation or review (Section M8.1).
50-446/9720-06	NCV	Failure of the radiation protection technician to maintain constant physical control of the radiation area key was a violation (Section R4.1).

Closed

50-446/9720-03	NCV	Failure to establish communications prior to placing an underwater light in the reactor vessel (Section O4.3).
50-445(446)/9720-05	NCV	Failure to follow procedure when modifying Fisher-Type 667 valve actuator bushings without proper documentation or review (Section M8.1).
50-446/9720-06	NCV	Failure of the radiation protection technician to maintain constant physical control of the radiation area key was a violation (Section R4.1).
50-445(446)/9706-01	VIO	Failure to establish procedures to ensure that all licensed operators that have their licenses conditioned to wear corrective lenses for the eye, always have available appropriate lenses qualified for a self-contained breathing apparatus (Section O8.1).
50-445(446)/9717-03	URI	Review scope of accumulator bushing modification and generic aspects of maintenance modifications (Section M8.1).
50-445(446)/9718-02	IFI	Review of seismic qualification of integrated leak rate test rig (Section E2.1).