ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket Nos.:	50-445 50-446
License Nos.:	NPF-87 NPF-89
Report No.:	50-445/99-18 50-446/99-18
Licensee:	TXU Electric
Facility:	Comanche Peak Steam Electric Station, Units 1 and 2
Location:	FM-56 Glen Rose, Texas
Dates:	November 14, 1999, through December 25, 1999
Inspectors:	Anthony T. Gody, Senior Resident Inspector Scott C. Schwind, Resident Inspector
Approved By:	Joseph I. Tapia, Chief

ATTACHMENT: Supplemental Information

EXECUTIVE SUMMARY

Comanche Peak Steam Electric Station, Units 1 and 2
NRC Inspection Report No. 50-445/99-18; 50-446/99-18

This inspection included aspects of licensee operations, maintenance, engineering, and plant support. The report covers a 6-week period of resident inspection.

Operations

- The Unit 2 component cooling water system was in good operating condition and in its proper standby condition. A small system leak which had been properly identified and characterized, grew into a notable leak during system operation with Valve 2-HV-4575 in its throttled accident position. Maintenance personnel quickly repaired the leak (Section O2.2).

Maintenance

- Maintenance and surveillance activities were conducted by knowledgeable and professional maintenance personnel. Quality, good skill of the craft, effective communications, and planning were evident in maintenance activities. Maintenance personnel were familiar with and effectively used the licensee's corrective action program to address degraded and/or nonconforming conditions and evaluate generic implications of issues (Section M1.2).

- Failure of an electric chain hoist while lifting the Unit 1 Reactor Coolant Pump 1-03 motor resulted in the 40 ton motor falling 20 to 30 feet before a chain link randomly lodged between the lower chain block and brass guide bar of the hoist and arrested the fall. Licensee personnel in containment were aggressive in verifying all personnel were clear of the area and that there were no injuries. There were no actual safety consequences as a result of this event. However, had the link not randomly lodged in the hoist, the 40 ton motor would have continued to fall, probably landing on the reactor coolant pump flange and damaging the reactor coolant system piping. Although the reactor fuel had been moved to the spent fuel pool, a rapid draindown of the refueling cavity could have exposed personnel in containment to high doses of radiation from the exposed core barrel, which was stored in the refueling cavity. The licensee's root cause investigation was thorough, probing, and expedient in determining the root cause of the event. The licensee's investigation team determined that improper maintenance resulted in misalignment between the spindle shaft and the planetary gear assembly in one of the hoist's drive trains. This led to failure of that drive train and, ultimately, failure of the entire hoist. Despite several opportunities, the licensee failed to recognize symptoms of the gear misalignment and correct it prior to this event (Section M4.1).

Engineering

- The postulated drop of a reactor coolant pump motor onto reactor coolant piping which would result in a reactor coolant leak in Mode 5 with fuel in the reactor would require at least one containment sump for decay heat removal. No procedural controls were developed to maintain at least one sump available while in Mode 5 during heavy load lifts even though abnormal operating procedures directed operators to shift the residual heat removal system suction to the containment sumps following a postulated loss of reactor coolant. The failure to have procedural controls for an available containment sump in Mode 5 during heavy load lifts is a violation of Technical Specification 5.4.1. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as SmartForm 1999-003178-00 (Section E3.1).

- On June 17, 1998, the licensee determined that the hydrogen purge system would not function as described in the Final Safety Analysis Report or the Design Basis Document and appropriately wrote Licensee Event Report 98-005-00. The Final Safety Analysis Report and Design Basis Document state that the system can be operated with containment pressures between 0 and 5 psig. However, at containment pressures above 0 psig, the flow rate through the filter elements would exceed the design limits; therefore, the charcoal adsorber residence stay times required by NRC Regulatory Guide 1.140 and ANSI N509-1976 would not be satisfied. The failure to adequately translate the design requirement for maximum flow rates through the hydrogen purge system into design specifications is a violation of 10 CFR Part 50, Appendix B, Criterion III. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. This condition was entered into the licensee's corrective action program as SmartForm SMF-1999-000487-00 (Section E8.1).

Report Details

Summary of Plant Status

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

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

The inspectors conducted frequent reviews of ongoing plant operations. In general, the conduct of operations was professional and safety-conscious; specific issues and noteworthy observations are detailed in the sections below. Through daily observations of control room activities, the inspectors concluded that both units were operated by knowledgeable operators using good self-verification techniques and communications.

O2 Operational Status of Facilities and Equipment

O2.1 Plant Tours

a. Inspection Scope (71707)

The inspectors conducted tours of accessible portions of the plant:

- Units 1 and 2 safeguards buildings
- Units 1 and 2 control room
- Units 1 and 2 auxiliary building
- Units 1 and 2 fuel handling building
- Units 1 and 2 electrical control building
- Units 1 and 2 turbine buildings

b. Observations and Findings

The inspectors found emergency equipment in proper standby and good material condition. Temporary equipment was stored properly to prevent interaction with safety-related equipment. Floor drains were free of debris and overall cleanliness was good.

O2.2 Unit 2 - Train A Engineered Safety Features Walkdown

a. Inspection Scope (71707)

The inspectors conducted a walkdown of the accessible portions of the Unit 2 component cooling water (CCW) system. In addition, the inspectors conducted general material condition walkdowns of the emergency diesel generators, residual heat removal (RHR) pumps, and spent fuel pool cooling systems.

b. Observations and Findings

Overall, the inspectors found the accessible portions of the Unit 2 CCW system in good operating condition and in its proper standby condition. A minor leak was noted on the flange for Train B containment spray (CS) heat exchanger (HX) CCW outlet Valve 2-HV-4575. While in standby, the leak was only several drips per minute. However, after the system operated with Valve 2-HV-4575 in its throttled accident position, the leak increased to a steady stream of water. The leak was repaired quickly by maintenance personnel and no damage to other standby equipment occurred.

c. Conclusions

Overall, the Unit 2 CCW system was in good operating condition and in its proper standby condition. A small system leak which had been properly identified and characterized grew into a notable leak during system operation with Valve 2-HV-4575 in its throttled accident position. Maintenance personnel quickly repaired the leak.

O2.3 Containment Isolation Valve Walkdown

a. Inspection Scope (71707)

The inspector walked down accessible containment isolation valves in Unit 1. The walkdown was performed in the north and south valve penetrations rooms as well as Room 1-088 which contained nonradiological piping penetrations.

b. Observations and Findings

The Unit 1 north and south valve penetration rooms contained various containment penetrations for radioactive systems. The inspector found that all manual containment isolation valves were in their required positions and were tagged per plant procedure to indicate that they should not be operated without shift manager approval. All required locking devices were installed. The inspector observed that a drip containment was installed underneath Valve 1-HV-4170, reactor coolant Loops 1-01 and 1-04 hot leg sample line orifice isolation valve, in the south valve penetration room; however, there was no indication of valve leakage. The system engineer indicated that there had been some packing leakage from this valve in the past while the valve was opened. There was no safety consequence regarding this leak since it is a remotely operated valve that is normally closed and not used for any postaccident sampling requirements. Housekeeping in valve penetration rooms was generally good; however, some minor debris was noted in a contamination area in the Unit 1 south valve penetration room.

c. Conclusions

A walkdown of manual and remotely operated containment isolation valves in Unit 1 revealed no discrepancies. All valves were in the correct position with locking devices installed as required. There was no evidence of leakage from any of the valves inspected. Housekeeping in the valve penetration area was good.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments

In general, maintenance and surveillance activities were characterized by knowledgeable and professional maintenance personnel. Quality and good skill of the craft was evident in maintenance activities. Effective communications and planning were evident in both maintenance and surveillance activities. An isolated but significant example where maintenance personnel demonstrated a lack of questioning attitude in the past is discussed in Section M4.1 below.

M1.2 Maintenance and Surveillance Observations

a. Inspection Scope (61726, 62707)

The inspectors observed risk significant maintenance and surveillance activities; some are listed below, more detailed observations are discussed in the following sections:

Unit 1, Train A emergency diesel generator surveillance
Unit 2, Train A CCW system flow balance test
Unit 2, Train B CCW system flow balance test
Unit 2, Train B CS pump surveillance
Eight-hour load test on the Unit 1 Battery Charger CP1-EPBCED-05
Unit 2, Train A control room emergency filtration and pressurization system surveillance
Breaker 2EA1-1 replacement and troubleshooting
Emergent Limitorque motor-operated butterfly valve limit switch adjustments
Emergent CCW flange maintenance
Emergent maintenance on main feedwater isolation valve for Steam Generator 1-03
Service water tunnel refurbishment

b. Observations and Findings

Surveillance Activities

The inspectors observed the Unit 2, Train B, CS pump quarterly operability test. This test was performed following maintenance on the CS Pump 2-04 breaker. A thorough brief was conducted by the control room staff. Operators performed the test safely and proficiently. In addition to verifying operability of the breaker, system engineering was present during the test to inspect for leakage from the pump seals on both CS Pumps 2-02 and 2-04 as well as the Train B CS HX flange. Although there was evidence of past leakage in some of these areas, system engineering noted no leakage during the test. Leakage from the HX flange was of particular concern since it would be difficult to remove several of the flange studs for inspection, per code requirements, due to the piping configuration.

The inspectors observed instrumentation and controls technicians performing a portion of an 8-hour load test on Unit 1 Battery Charger CP1-EPBCED-05. The technicians were knowledgeable of the test procedure and were constantly monitoring the performance of the battery charger. Hourly test data was recorded appropriately. The battery charger supplied a minimum of 300 amps at 130 volts throughout the 8-hour duration of the test as required. In the past, the licensee had experienced problems with the 480 volt molded case circuit breakers supplying the battery chargers during this test. These breakers were infrequently operated and tended to trip due to thermal overload during the test. The licensee believed this was due to resistance heating caused by a buildup of an oxide layer on the main contacts of the breaker. To correct this, all of these breakers were replaced and a maintenance activity was created to periodically cycle the breakers to prevent oxide buildup. This corrective action appeared to have been successful.

The inspectors observed portions of the monthly operability test on the Unit 2 Train A control room emergency filtration and pressurization system and reviewed the results of the test. This test required the control room heating, ventilation, and air conditioning system to be placed in emergency recirculation mode with the heaters in the pressurization system energized for a minimum of 10 hours. This was completed successfully.

The inspectors observed a Unit 1 Train A emergency diesel generator operability test and several other ancillary surveillance activities, such as vibration monitoring and oil sampling. In addition, the inspector walked down both trains of emergency diesel generators. The Train B Unit 1 emergency diesel generator was found in its appropriate standby condition. Both emergency diesel generators had little or no oil or air leakage. Although the Train A emergency diesel generator appeared to be vibrating more than usual to the inspector, vibration measurements were within acceptance criteria.

The inspector observed the Unit 2 Train A and Train B CCW flow balancing test in accordance with Procedure PPT-P2-6200, "CCW to RHR/CS HX Outlet Valve Flow Control Test," Revision 1, on December 15 and December 8, 1999, respectively. The Train B pre-evolution brief was clear and allowed questions from participating personnel. Communications between personnel in the field and the control room were good. Several problems resulting in emergent maintenance activities were observed during the testing of Train B, each of which are discussed below: (1) the Train B CS HX CCW outlet valve failed to automatically travel to its accident position while conducting step 8.3.5 of PPT-P2-6200 (SmartForm SMF-1999-003381-00), (2) as found flow data for Train B was unsatisfactorily high but later became suspect when engineers noted flow data drifting, (3) previously identified leakage from a flange for Valve 2-HV-4575 significantly increased to the point where it was a continuous stream and maintenance personnel had to torque the flange bolts to resume the test.

<u>Planned Maintenance</u>

The inspector observed portions of the service water refurbishment project. This included completion of painting and preservation of the tunnel. The area was properly surveyed and posted as a radiologically controlled area. All scaffolding used in the tunnel met the required commodity clearance from all seismically qualified structures and systems.

Emergent Maintenance

The inspector observed troubleshooting activities on the Unit 1 Feedwater Isolation Valve 1-FWIV-03 hydraulic actuator. It was noted that the hydraulic pump for the valve actuator was running constantly and could not reach the required shut off pressure. This, by itself, did not affect valve operability since hydraulic pressure was used only to open the valve. The actuator was equipped with a nitrogen reservoir which was used to shut the valve; therefore, the containment isolation safety function of the valve was unaffected. The licensee believed that the hydraulic pump was air bound and developed a written troubleshooting plan to prime the pump. This was done by bleeding nitrogen pressure from the top of the valve actuator, then pressurizing the hydraulic oil reservoir with air while the pump was running. This appeared to correct the problem. The licensee appropriately entered the 4-hour shutdown action statement per Technical Specification 3.7.3 for an inoperable feedwater isolation valve while performing this evolution. This condition was also entered into the licensee's corrective action program as Smart Form SMF-1999-003468-00.

As discussed above, during CCW system flow balancing, the Train B CS HX CCW outlet valve (Valve 2-HV-4575) failed to automatically travel to its accident position. Procedure PPT-P2-6200, step 8.3.5, directed test personnel to simulate a CS actuation signal to Valve 2-HV-4575. Valve 2-HV-4575 failed to stroke when the CS actuation signal was simulated. Test personnel quickly ascertained that Contact 9 on Rotor 3 in the actuator for Valve 2-HV-4575 was not making contact and immediately implemented Procedure PPT-P0-6005, "Safety Related Quarter Turn Motor Operated Valve Testing," Attachment 12. Attachment 12 allowed test personnel to check that the contact finger touches the rotor for contacts that are not made and to bend the L-bracket of the finger to achieve contact. The inspectors became curious as to how Attachment 12 was developed and discussed its origins with the lead motor-operated valve engineer. The lead motor-operated valve engineer provided the inspector with a copy of Operation, Notification, Evaluation (ONE) Form 98-684, dated April 23, 1998. This ONE Form was initiated following a review of closed work orders on April 23, 1998, when a system engineer identified that a failure of the Unit 1 charging control valve to close following a safety injection signal was a maintenance rule functional failure. The Unit 1 charging control valve failed to close because of failed continuity on a limit switch contact. As part of the corrective actions, the licensee implemented changes to motor-operated valve testing and refurbishment procedures directing limit switch contact inspections, cleaning, and adjustments as needed. The inspector questioned the licensee about how many limit switch contact failures had been documented over the years and found that only the two discussed above had failed and the only failure involving the need to bend the L bracket was Valve 2-HV-4575. The licensee appropriately wrote SmartForm SMF-1999-003381-00 which assigned engineering to review any generic aspects of the issue. The inspector reviewed past surveillances and found the last successful demonstration of Valve 2-HV-4575 operability was October 19, 1999.

Also during the Train B CCW flow balancing, some as-found flow data for Train B was unsatisfactorily high. This data later became suspect when engineers noted that the flow data was drifting. Instrumentation and controls technicians were dispatched to the flow transmitter and vented the transmitter. A previous calibration of the transmitter may have introduced air into the sensing lines. After the venting was complete, the flow transmitter responded consistently and accurately. As-left flow rate data for Train B CCW was satisfactory.

When the instrumentation and controls technicians responded to the drifting flow transmitter described above, they observed a continuous stream of water coming from the flange for Valve 2-HV-4575. Valve 2-HV-4575 had a green work request tag on it indicating that the flange had a minor leak (drops per hour). The instrumentation and controls technicians appropriately cordoned off the area because of the minor flooding, informed the control room, and contacted mechanical maintenance personnel. Operators opened the valve fully from its as-left accident position and the leakage decreased significantly. Maintenance personnel torqued the flange bolts and the leakage was reduced to a small drip. Operators stroked the valve several times to verify that the leakage was acceptable. The inspectors observed that all personnel took the appropriate safety precautions and the mechanics properly retorqued the valve flange.

After transferring the Unit 2 Train A switchgear from the preferred source (Startup Transformer XST1) to the alternate source (Startup Transformer XST2), operators noted that the open indication light in the control room for Breaker 2EA1-1 was not lit as expected. Operators in the field confirmed that the local open light on the breaker was also not on as expected even though the breaker was open. Maintenance personnel found that the auxiliary contact switch had overtraveled, resulting in the loss of open indication and loss of remote functionality. The inspector observed the electricians replace the breaker with an enhanced breaker. Electricians demonstrated clear and concise communications and rigorous electrical safety precautions. The replacement breaker appeared to be in good working condition, properly lubricated, and clean. The failure of the breaker was attributed to jack shaft binding and was documented in SmartForm SMF-1999-003427-00.

c. <u>Conclusions</u>

Maintenance and surveillance activities were conducted by knowledgeable and professional maintenance personnel. Quality, good skill of the craft, effective communications and planning were evident in maintenance activities. Maintenance personnel were familiar with and effectively used the licensee's corrective action program to address degraded and/or nonconforming conditions and evaluate generic implications of issues.

M4 Maintenance Staff Knowledge and Performance

M4.1 Failure of Chain Hoist While Lifting Reactor Coolant Pump (RCP) Motor 1-03

a. Inspection Scope (92902)

The inspector observed the licensee performing a heavy lift evolution to replace the Unit 1 RCP 1-03 motor during the refueling outage and the licensee's emergency response and recovery efforts following failure of the chain hoist used to lift the motor. The inspector also observed the licensee's root cause investigation and reviewed the findings and conclusions from that investigation.

b. Observations and Findings

Heavy Lift Evolution and Emergency Response

On October 6, 1999, the licensee was performing a heavy lift evolution to remove the RCP 1-03 motor from the Unit 1 containment building in order to replace it with a rebuilt and upgraded model. The motor weighed approximately 40 tons. This evolution was performed using a 45 ton electric-driven chain hoist rigged to the polar crane main hook which was rated at 175 tons. Use of the chain hoist was necessary to raise the motor approximately 80 feet from its base to the 905 foot elevation since the polar crane's main hook was too large to be lowered into the narrow compartment above the motor. This hoist was procured by the licensee specifically for this task and was originally a manual chain hoist which had been modified by the addition of two separate electric drive trains and a set of chain blocks by the vendor.

The inspector observed the lift from the 860 foot and 872 foot elevations of containment. Two riggers were stationed on top of the lifting platform attached to the motor, per plant procedures, in order to operate the pendant controls for the hoist and to guide the motor out of the compartment. A number of personnel were in the compartment below the motor during the lift, only some of which were directly involved in the lift. The inspector observed that, aside from standing on the load as it was raised, the riggers used appropriate safety precautions and lifted the motor in a slow, controlled manner. As the top of the motor cleared the 872 foot elevation, a failure in the chain hoist allowed the chain to free-wheel through the chain blocks and the motor dropped 20 to 30 feet in a matter of seconds. As the chain was free-wheeling, one link randomly lodged in the lower chain block which arrested the unplanned accelerated descent. The motor stopped approximately 8 feet above its base.

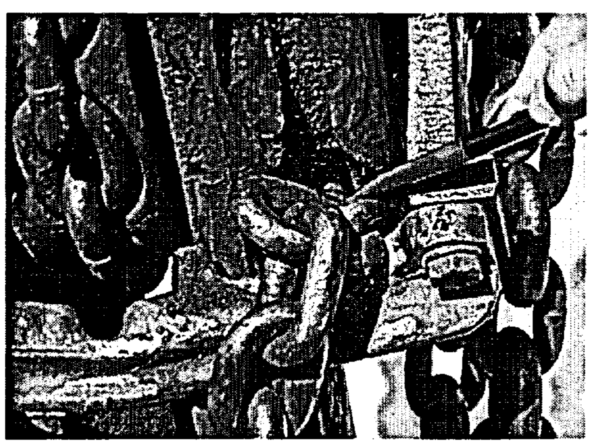
Licensee personnel in containment responded quickly and aggressively to ensure all personnel were evacuated from the area above and below the motor. The riggers were wearing safety harnesses and were able to jump from the lifting rig as it fell and were unharmed. The motor did not come in contact with any structures or equipment as it fell. All nonessential personnel were evacuated from the containment and further access was restricted.

At the time of this event, the reactor fuel had been offloaded to the spent fuel pool, the refueling cavity was flooded, the transfer canal gates were closed, and the core barrel had been removed from the reactor vessel and placed in its stand at one end of the refuel cavity.

Event Recovery

The condition of the chain hoist was unknown following the event. The licensee clearly observed that a chain link was wedged in the lower chain block between the outermost sheave and a brass guide bar across the sheaves (see photo below). It was speculated that this was all that was preventing the motor from descending further. Each of the hoist's drive trains was equipped with an electric and mechanical brake. The brakes had failed to prevent the initial drop, so it was unknown whether any of them were engaged and holding the load.

The inspectors observed the licensee's decision making process during several strategy meetings to plan the recovery of the motor. Despite the sense of urgency, the licensee methodically evaluated several possible strategies, including rigging temporary kevlar slings to the motor as well as shoring the motor from underneath. The licensee ultimately chose to stabilize the chain by shackling opposite passes of chain together as close to the chain blocks as possible so that, in the event the one link of chain became dislodged, the remaining chain would not be able to free wheel through the chain blocks. Once this had been completed, the licensee was able to lower the motor to its base using the polar crane's main hook. This was accomplished without further incident.



Safety Significance

The RCP motor did not free fall during this event, since its decent was slowed by friction generated in the chain hoist. However, none of the brakes on the failed drive train were engaged and the brakes on the intact drive train were destroyed during the event, since they were insufficient to hold the load. The only thing that arrested the fall was the link of chain that lodged in the lower chain block. The postevent examination of the hoist revealed severe damage to the brass retaining bar and its threaded fasteners which indicated that, had the chain lodged in the center of the chain block rather than the edge, the retaining bar may have failed, allowing the motor to continue to fall.

There were no actual safety consequences resulting from this event, since the motor did not come in contact with any structures or equipment. However, the licensee performed an informal evaluation of a worst-case scenario which considered the consequences had the motor fallen onto the pump and seal package resulting in an uncontrolled leak and draining of the refueling cavity. This evaluation assumed that the refueling cavity would have been instantaneously drained to the point where the highly activated belt line region of the core barrel was exposed. This would have flooded the 808 foot elevation of containment and would have rendered the 860 and 905 foot elevations of containment inaccessible due to extremely high radiation levels. In addition, had the motor fallen unarrested, the personnel working below the motor could have been severely injured. The licensee concluded that, if this postulated scenario had occurred, the radiological consequences outside containment would not have exceeded the limits contained in 10 CFR Part 100 and there was no significant risk to the general public. The inspectors agreed there was no significant risk to the public and further concluded that the licensee's scenario was conservative. This conclusion was based on the inspectors review of the actual radiation survey data taken on the core barrel and a determination of the refueling cavity leak rate necessary to expose the belt line region of the barrel. The inspectors concluded that there was only a moderate potential for personnel in containment to exceed their dose limits as a result of this event.

The inspectors reviewed the licensee's administrative controls and procedures used to control this evolution as well as their heavy loads control program. Plant procedures specifically allowed the licensee to perform this evolution in Mode 5 (cold shutdown) and Mode 6 (refueling), as was the case in 1996 when an RCP motor was replaced in Unit 2 at the end of a refueling outage with the unit in Mode 5. These aspects are discussed further in Section E3.1 of this report.

<u>Root Cause</u>

Following this event, the licensee assembled a multidisciplined task team to investigate the root cause of the hoist failure. The team's charter included determination of the root cause of the event as well as a review of plant procedures for moving heavy loads and maintenance practices. The charter did not specifically require a review of the hoist's design basis. Several industry experts as well as an offsite materials laboratory were used during the investigation. The team considered the root cause of the event to be a lack of full engagement between gear teeth on the spindle shaft and the planetary gear assembly in one of the hoist's drive trains. This eventually led to failure of the gears and, ultimately, failure of the entire hoist. The misalignment of the gears had not been identified following a maintenance activity in 1994 during which the hoist had been rebuilt and reassembled incorrectly. The licensee's team also noted the following missed opportunities to identify the condition of the hoist:

- When the hoist was rebuilt in 1994, there was evidence that the spindle shaft did not properly align with the planetary gears. This observation was supported by test results from an offsite materials laboratory which indicated that a set screw hole, which ensures proper alignment, had been redrilled to achieve alignment. In addition, following the event of October 6, 1999, when the hoist was disassembled, this setscrew was not fully engaged as required.

- During a mechanical inspection in 1996, a cover plate could not be re-installed over the planetary gears due to the misalignment. There was also evidence that the gears had been rubbing on the inside of the cover plate. Rather than question and correct the gear alignment problems, the screw holes for the cover plate were elongated so that the plate could be re-installed.

- When the hoist was rigged to the polar crane's main hook in October 1999, the hoist would not initially operate in the down direction. After operating the hoist in the up direction, it would operate in the down direction. No mechanical or electrical inspection of the hoist was performed at this time. This could have been an additional indication of gear misalignment in the hoist.

The inspectors performed an independent review of the team's findings as well as a review of the design basis for the hoist and agreed with the team's conclusions. In addition, the inspectors found that the design basis document, DBD-ME-006, "Control of Heavy Loads at Nuclear Plants," required that a hoist be considered safety related if a load dropped by that hoist could prevent safe shutdown or result in an offsite radiation release. The RCP motor hoist was not considered safety related despite the fact that its failure could prevent safe shutdown without adequate administrative controls on the containment sumps while in Mode 5. This is discussed further in Section E3.1 of this report.

c. <u>Conclusions</u>

Failure of an electric chain hoist while lifting the Unit 1 RCP 1-03 motor resulted in the 40 ton motor falling 20 to 30 feet before a chain link randomly lodged between the lower chain block and brass guide bar of the hoist and arrested the fall. Licensee personnel in containment were aggressive in verifying all personnel were clear of the area and that there were no injuries. There were no actual safety consequences as a result of this event; however, had the link not randomly lodged in the hoist, the 40 ton motor would have continued to fall, probably landing on the RCP flange and damaging reactor coolant system (RCS) piping. Although the reactor fuel had been moved to the spent fuel pool, a rapid draindown of the refueling cavity could have exposed personnel in containment to high doses of radiation from the exposed core barrel.

The licensee's root cause investigation was thorough, probing, and expedient in determining the root cause of the event. The licensee's investigation team determined that improper maintenance resulted in misalignment between the spindle shaft and the planetary gear assembly in one of the hoist's drive trains. This led to failure of that drive train and, ultimately, failure of the entire hoist. Despite several opportunities, the licensee failed to recognize symptoms of the gear misalignment and correct it prior to this event. The inspectors identified no violation of regulatory requirements regarding the lack of corrective actions for the degraded hoist. However, these missed opportunities to identify problems with the hoist demonstrated a lack of questioning attitude by licensee maintenance personnel and was not consistent with current maintenance performance. The licensee's administrative controls and procedures to ensure that failure of the hoist would not affect safe shutdown safety functions were inadequate since there were no measures in place to assure the availability of at least one containment sump during heavy load lifts in Modes 5 and 6.

M8 Miscellaneous Maintenance Issues (92902, 92700)

M8.1 (Closed) LER 50-445/98-004-00: failure to perform response time testing of the shunt trip feature of a reactor trip breaker. On April 27, 1998, while conducting a postmaintenance review of work documents to replace a failed reactor trip breaker on April 26, 1998, the licensee identified that they had neglected to conduct reactor trip system time response testing on the shunt trip diverse feature as required by Technical Specifications 3.3.2 1c, 1d, and 1e. While in this condition, Unit 1 conducted a reactor startup, transitioning from Mode 3 to Mode 2, contrary to the requirements of Technical Specification 3.0.4. The surveillance test was satisfactorily completed on April 27. This failure constitutes a violation of minor significance and is not subject to formal enforcement action. This violation is in the licensee's corrective action program as ONE Form 98-701.

III. Engineering

E1 Conduct of Engineering

E1.1 General Comments (37551)

In general, timely and appropriate engineering support was evident in day-to-day operations. Through interviews and direct observations, the inspectors found the system engineers aware of system performance and were involved in the identification of adverse trends. Engineering products were typically of high quality and demonstrated a questioning attitude. Two isolated examples where the licensee demonstrated a lack of thoroughness and attention-to-detail while implementing their corrective program are discussed below.

E2 Engineering Support of Facilities and Equipment

E2.1 RCP 1-03 Motor

a. Inspection Scope (37551)

The inspectors conducted an assessment of the licensee's efforts to evaluate the acceptability of use for the RCP 1-03 motor after the unplanned accelerated descent described in Section M4.1 above.

b. Observations and Findings

The inspectors found that the licensee had not placed the need for an evaluation of the acceptability of the RCP 1-03 motor in its corrective action program. This was a concern because a review of the vendor technical manual revealed the following statement, ". . . shipping braces are utilized for securing the shaft in place during shipment to preclude damage to the bearings of the motor. Movement of the motor inside containment and on the dolly should be smooth enough to prevent damage to the motor, as long as reasonable caution is taken." No further clarifying details regarding directional thrust ratings of the RCP motor were provided in the vendor technical manual. In a postoutage letter to the licensee dated October 27, 1999, Westinghouse reported, "Westinghouse EMD [Electric Motor Division] was consulted and determined that it was unlikely any damage occurred to the motor based on a vertical descent and the fact that it made no contact with any plant components during the descent. To further insure the motor was acceptable for operation, a complete one-year inspection was performed, as well as an axial end play check and a 4-hour uncoupled run. No anomalies were observed during these tests that showed the motor unacceptable for operation." The inspector noted that no RCP motor bearing oil samples had been taken following the run after the unplanned accelerated descent.

c. Conclusions

The inspectors found that an evaluation of the acceptability for continued use of the RCP 1-03 motor following the unplanned descent described in Sections M4.1 and E3.1 was not referenced in the licensee's corrective action program. The licensee opened Technical Evaluation 1999-002650-01-00 as part of SmartForm 1999-002650 in response to this finding.

E3 Engineering Procedures and Documentation

E3.1 Control of Heavy Loads

a. Inspection Scope (37551, 92903)

The inspectors reviewed the licensee's program and procedures for controlling heavy loads inside containment and the supporting engineering analyses to ascertain if they were consistent with commitments made in accordance with NUREG-0612, " Control of Heavy Loads at Nuclear Power Plants," submitted to the NRC in 1983. In addition, the licensee's corrective actions following the near miss drop of a 40 ton RCP motor described in Section M4.1 of this report were evaluated against the inspectors findings.

b. Observations and Findings

Following the RCP motor near miss event, the inspectors reviewed the licensee's program for controlling the movement of heavy loads inside containment. The licensee's heavy loads program was outlined in Design Basis Document (DBD) DBD-ME-006, "Control of Heavy Loads at Nuclear Plants." The inspectors noted that the licensee's program closely matched their commitments to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," which was formally transmitted to the NRC in 1983.

Section 4.1.2 of DBD-ME-006, stated, in part, that the crane and associated lifting devices used for handling heavy loads in the containment building shall satisfy the single-failure-proof guidelines of NUREG-0554, "Single-Failure Proof Cranes for Nuclear Power Plants," or the effects of drops of heavy loads shall be analyzed and shown to satisfy the following criteria: (a) postulated doses from a drop are well within the 10 CFR Part 100 limits, (b) K_{eff} remained larger than 0.95 for postulated drops on or near fuel, (c) damage to the reactor vessel or spent fuel pool will not result in the uncovering of fuel, and (d) damage to redundant equipment will not result in a loss of required safe shutdown functions.

Section 5.3 of DBD-ME-006 indicated that the evaluation of heavy loads assumed that such loads are not carried over operating safe shutdown or decay heat removal equipment when the redundant train is not operable, but does note that special precautions are required during shutdown (Operational Modes 5 and 6) inside containment. For example, while in Mode 5, plant Technical Specifications require at least one train of RHR to be operable and operating and, depending on whether the loops are filled, they require either an additional train of RHR be operable or a minimum of two steam generators be filled to at least 10 percent. During operational Mode 5, safe load paths must be followed to prevent the load from traveling over the operable and operating RHR loop.

The inspectors requested a copy of the analysis described in Section 4.1.2 of DBD-ME-006. The licensee indicated that they had performed an analysis but it was either not documented or the documentation was lost. The licensee informed the inspectors that the results of that analysis were contained in their response to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," which was formally transmitted to the NRC in 1983.

Licensee Procedure MDA-304, "Heavy Loads Program," Revision 2, Section 6.5, indicated that the RCP and RCP motors shall be handled only during the cold shutdown or refueling modes and shall follow a safe load path diagram. The inspectors found that Attachment 8.D of MDA-304 appropriately restricted the movement of heavy loads over RCS Loops 1 and 2 when RHR Train A is in operation and RCS Loops 3 and 4 when RHR Train B is operating. These pre-established safe load paths were adequate to prevent an immediate loss of safety function directly from the postulated load drop. However, the inspectors noted that, if an RCP motor drop was postulated in any of the loop compartments while in Mode 5 and RCS leakage resulted, the licensee's abnormal operating procedures (ABNs) relied on operators having the capability to transfer reactor coolant spilled inside containment back to the vessel for core cooling. To successfully transfer coolant from the containment floor back to the reactor vessel, at least one containment sump would need to be available.

ABN-108, "Loss of Coolant - Shutdown," is applicable in operating Modes 4 and 5 and contained provisions to shift RHR suction from the RCS to the containment sumps to maintain decay heat removal following a postulated loss of reactor coolant from the RCS. A review of other applicable procedures for Mode 5 revealed that the licensee had no requirement to maintain a containment sump available while in operational Mode 5. Although the inspector recalled several periods of time during past refueling outages when containment sumps had been covered with plastic to prevent foreign material entry, the licensee could not identify any time when a sump was clearly made inoperable while in operational Mode 5.

Technical Specification 5.4.1 required that written procedures be established, implemented, and maintained as recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, Section 6.a, recommended that procedures be established for combating events such as a loss of coolant. The inspector concluded that procedures were not adequate because no provisions were incorporated into operating procedures regarding containment sump availability while in operational Mode 5 during which successful implementation of ABN-108 relied solely on being able to use the containment sumps for a postulated loss of coolant while shutdown. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as SmartForm 1999-003178-00 and is appropriately required to be addressed prior to entering Mode 5 (NCV 50-445;446/9918-01).

The inspector noted that none of the above findings were discussed in the licensee's review of the RCP motor hoist failure even though documentation stated that the heavy loads program was reviewed.

c. <u>Conclusions</u>

The postulated drop of an RCP motor onto reactor coolant piping which would result in a reactor coolant leak in Mode 5 with fuel in the reactor would require at least one containment sump for decay heat removal. No procedural controls were developed to maintain at least one sump available while in Mode 5 during heavy load lifts even though ABNs directed operators to shift the RHR system suction to the containment sumps following a postulated loss of reactor coolant. The failure to have procedural controls for an available containment sump in Mode 5 during heavy load lifts is a violation of Technical Specification 5.4.1.

E8 Miscellaneous Engineering Issues (92903, 92700)

E8.1 <u>(Closed) LERs 50-445:446/98-005-00 and 50-445;446/98-005-01</u>: functional requirements of the hydrogen purge system not in accordance with design. On June 17, 1998, the licensee determined that the hydrogen purge system would not function as described in the Final Safety Analysis Report or the DBD. This system is common to both Units 1 and 2 and is a backup means for controlling combustible gases in containment following a loss of coolant accident. The system consists of two trains, each designed to allow either containment building to be vented to the outside atmosphere through high efficiency particulate air filters and charcoal adsorption beds. The Final Safety Analysis Report and DBD state that the system can be operated with containment pressures between 0 and 5 psig and is capable of exhausting a designed airflow of 700 cubic feet per minute (cfm). However, at containment pressures above 0 psig, the flow rate through the filter elements would exceed 700 cfm; therefore, the charcoal adsorber residence stay times required by Regulatory Guide 1.140 and American Nuclear Standards Institute (ANSI) N509-1976 would not be satisfied.

Appendix B, Criterion III of 10 CFR Part 50 requires, in part, that measures be established to assure that applicable regulatory requirements and the design basis for systems required to mitigate the consequences of a postulated accident are translated into specifications, drawings, procedures, and instructions. Contrary to this requirement, the design requirement for a maximum flow of 700 cfm through the hydrogen purge system was not adequately translated into design specifications for the system. This Severity Level IV violation is being treated as a noncited violation, consistent with Section VII.B.1.a of the NRC Enforcement Policy. This condition was entered into the licensee's corrective action program as Smart Form SMF-1999-000487-00 (NCV 50-445;446/9918-02)

IV. Plant Support

R1 Radiological Protection and Chemistry Controls

R1.1 General Comments (71750)

Radiation workers were found to be knowledgeable of their radiation work permit requirements and demonstrated appropriate radiation worker practices in the radiologically controlled area. Drip containments and radioactive liquid hoses were properly labeled and tracked.

S1 Conduct of Security and Safeguards Activities

S1.1 General Comments (71750)

During routine tours of the facility, the inspectors observed the conduct of security and safeguards activities. The inspectors observed that changing plant security barriers and lighting needs were well controlled during routine plant operation and security officers were alert and attentive to their duties.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the results of the routine resident inspection to members of the licensee's management team on December 30, 1999. The licensee acknowledged the findings presented. The licensee indicated that the unplanned descent of the RCP motor was an unfortunate occurrence that they planned to correct. Whether they plan to classify the hoist as safety-related or procure a nonsafety-related single-failure-proof hoist was still not decided. The licensee also indicated that they planned to continue their engineering review of the containment sump availability questions and determine the best course of action from a personnel and plant safety perspective. No proprietary information was identified.

ATTACHMENT

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

J. J. Kelley, Vice President, Nuclear Engineering and Support
D. J. Reimer, Technical Support Manager
J. M. Ayres, Plant Support Overview Manager
R. D. Walker, Regulatory Affairs Manager
R. D. Carver, Independent Safety Evaluation Group Manager
J. A. Taylor, Design Bases Engineering Manager
M. Lucas, Maintenance Manager
D. Moore, Operations Manager
S. Ellis, Shift Operations Manager

INSPECTION PROCEDURES USED

IP 37551	Onsite Engineering
IP 61726	Surveillance Observations
IP 62707	Maintenance Observations
IP 71707	Plant Operations
IP 71750	Plant Support Activities
IP 92700	Onsite Followup of Written Reports of Nonroutine Events at Power Reactor Facilities
IP 92902	Followup - Maintenance
IP 92903	Followup - Engineering

ITEMS OPENED AND CLOSED

Opened

50-445;446/9918-01	NCV	Inadequate abnormal operating procedures for loss of coolant while shutdown (Section E3.1)
50-445;446/9918-02	NCV	Failure to translate design requirements into instructions for the hydrogen purge system (Section E8.1)

Closed

50-445;446/98-004-00	LER	Failure to perform response time testing of the shunt trip feature of a reactor trip breaker (Section M8.1)
50-445;446/98-005-00 50-445;446/98-005-01	LER	Functional requirements of the hydrogen purge system not in accordance with design (Section E8.1)
50-445;446/9918-01	NCV	Inadequate abnormal operating procedures for loss of coolant while shutdown (Section E3.1)
50-445;446/9918-02	NCV	Failure to translate design requirements into instructions for the hydrogen purge system (Section E8.1)

LIST OF ACRONYMS USED

ABN	Abnormal Operating Procedures
ANSI	American Nuclear Standards Institute
CCW	component cooling water
CS	containment spray
DBD	Design Basis Document
HX	heat exchanger
LER	licensee event report
ONE	Operation, Notification, Evaluation
RCP	reactor coolant pump
RCS	reactor coolant system
RHR	residual heat removal