ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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Report No.:	50-397/99-09
Licensee:	Energy Northwest
Facility:	Washington Nuclear Project-2
Location:	Richland, Washington
Dates:	July 25 through September 4, 1999
Inspectors:	G. D. Replogle, Senior Resident Inspector J. E. Spets, Resident Inspector P. A. Goldberg, Reactor Inspector, Division of Reactor Safety
Approved By:	Linda Joy Smith, Chief, Project Branch E, Division of Reactor Projects

ATTACHMENT: Supplemental Information

EXECUTIVE SUMMARY

Washington Nuclear Project-2
NRC Inspection Report No. 50-397/99-09

This information covers a 6-week period of resident inspection.

Operations

- The conduct of operations was professional and safety conscious. Operators were generally knowledgeable of important plant parameters and problems (Section O1.1).

- The inspectors identified a violation of Technical Specification 3.5.1, in that operators, with the high pressure core spray system inoperable, did not immediately verify by administrative means that the reactor core isolation cooling system was operable. Operators had isolated the minimum flow line in support of planned maintenance and failed to recognize that the condition rendered the high pressure core spray system inoperable. This Severity Level IV violation is being treated as a noncited violation, consistent with Appendix C of the NRC Enforcement Policy. The violation is in the licensee's corrective action program as Problem Evaluation Request 299-1557 (Section O1.2).

Maintenance

- During Thermolag removal, contractors inadvertently severed electrical cables to 5 components and damaged approximately 70 other cables. While operability of some components was affected, all system safety functions were maintained. Planned corrective measures were acceptable (Section M1.2).

- The inspectors identified a Technical Specification 5.4.1.a violation with two examples. First, maintenance craftsmen did not properly implement a procedure for checking the refueling floor crane hook for yielding. The craftsmen did not obtain the original as-built dimensions to compare with the as-found dimensions, as required. Additionally, they did not question unexpected and illogical data and results, and the licensee had not independently reviewed all the results. Second, engineers failed to complete an engineering evaluation, as required by a plant procedure, for a scaffolding storage rack on the 422-foot reactor building elevation. Annual inspections were not effective at catching the problem earlier. These Severity Level IV violation examples are being treated as a noncited violation. The violation was in the licensee's corrective action program as Problem Evaluation Requests 299-1759, -1760, and -1591 (Sections M1.3 and E2.2).

- Plant material condition was very good, with some problems. New material condition concerns included: (1) the failure of one circulating water pump and (2) a 1-gallon-per-minute leak on a low pressure heater drain valve, which was leaking through the floor and contaminating the condenser bay floor below. Conversely, an inoperable tower makeup pump was returned to service this period (Section M2.1).

Engineering

- Engineering and chemistry provided good support to operations in response to inadvertent Fyrquel hydraulic fluid in-leakage into the primary system. The departments: (1) determined that no more than 2 pints were injected; (2) determined that the fuel would not be affected; and (3) verified that no other plant equipment would be impacted by the incident. The assessment was thorough and comprehensive (Section E2.1).

- The inspectors identified a 10 CFR Part 50, Appendix B, Criterion III (Design Control) violation, with three examples. First, the inspectors found that design constraints associated with the residual heat removal system in the shutdown cooling mode were not properly incorporated into Technical Specification 3.4.9. As such, the system was not always operable consistent with Technical Specification requirements. Second, the inspectors identified that a plant modification had removed the outboard shutdown cooling isolation valve controls from the alternate remote shutdown panel, but Technical Specification 3.3.6.1, note (d), which was rendered obsolete by the modification, was not removed from the Technical Specifications. Third, the licensee identified that Technical Specifications Bases 3.3.1.1.3 incorrectly states that four instrumentation channels of reactor vessel steam dome pressure - high are available. Only two instrument channels are actually available. These Severity Level IV violation examples are being treated as a noncited violation. The violation was in the licensee's corrective action program as Problem Evaluation Requests 299-0574, -0691, and -0548 (Sections E8.1 and E8.2).

- The inspectors identified a negative trend with respect to Technical Specification fidelity. In addition to the three problems identified in this report, four other Technical Specification accuracy and implementation problems were identified as a result of NRC questions during the past 18 months (see NRC Inspection Reports 50-397/98-15 and -99-07). The licensee acknowledged the performance trend and planned to evaluate the condition through the corrective action program (Section E8.3).

Plant Support

- During routine plant tours, the inspectors verified that the emergency preparedness facilities were properly maintained and on-shift staffing was consistent with the emergency plan. No problems were found (Section P2.1).

- During the off-hours call-in drill, a 6-year event, the licensee failed to meet the emergency plan commitment to staff two chemistry technicians within 1 hour of declaring the Alert. Further, the operations support center manager and the team tracker failed to recognize the problem, as they were misled by an erroneous sign-in board that only contained one slot for a chemistry technician signature. Corrective actions were adequate (Section P4.1).

- During routine tours, the inspectors observed protected area illumination levels, maintenance of the isolation zones around protective area barriers, and the status of security power supply equipment. No problems were observed (Section S2.1).

Report Details

Summary of Plant Status

At the beginning of the inspection period the unit was at 99 percent reactor power. Reactor power gradually coasted down to 78 percent by the end of the inspection period because of end-of-cycle fuel depletion.

I. Operations

O1 Conduct of Operations

O1.1 General Comments (71707)

The inspectors conducted frequent reviews of ongoing plant operations. Operators were generally knowledgeable of important plant parameters and problems and were appropriately focused on safety. One problem concerning the failure to complete an emergency core cooling system Technical Specification action statement is discussed in Section O1.2.

O1.2 Failure to Complete High Pressure Core Spray System Action Statement

a. Inspection Scope (71707)

On July 26, 1999, while reviewing operator logs, the inspector observed that the high pressure core spray system minimum flow line was isolated, but the licensee still considered the system operable. The inspectors reviewed the licensee's justification for the operability determination.

b. Observations and Findings

The high pressure core spray system is designed to provide makeup water to the reactor vessel during a small break loss-of-coolant accident. In response to an accident, the pump would start and the injection valve would open. Following the initial injection, the injection valve closes at reactor vessel level 8. The minimum flow valve then automatically opens to ensure that the pump does not overheat and fail, as flow through the pump is required for cooling. As water continues to leave the reactor vessel, the injection valve automatically opens at level 2, and the minimum flow valve closes. During a design basis event, this system response may be repeated many times.

The inspectors determined that the high pressure core spray system was inoperable when the minimum flow line was isolated. Operators had not properly considered all operability requirements for the high pressure core spray system. They believed the system was operable as long as an initial flow path and water source were available, but failed to consider longer-term operability requirements. A minimum flow line and properly operating minimum flow valve are necessary to ensure that the system remains operable throughout the duration of a small break loss-of-coolant accident. The

licensee agreed with the inspectors' assessment and declared the high pressure core spray system inoperable.

The inspectors also observed that operators had missed an earlier opportunity to identify the problem. The high pressure core spray system minimum flow line was isolated on the back shift, but day shift personnel did not adequately question the operability decision during shift turnover.

Technical Specification 3.5.1 requires that all emergency core cooling systems and the reactor core isolation cooling system be operable. With the high pressure core spray system inoperable, Action B.1 requires that the licensee immediately verify, through administrative means, that the reactor core isolation cooling system was operable. Contrary to the above, the high pressure core spray system was rendered inoperable on July 26, at 6:01 a.m., when the minimum flow line was isolated, but the licensee did not immediately verify by administrative means that the reactor core isolation cooling system was operable. The verification was not performed until approximately 11:30 a.m., when the problem was identified by the inspectors. The failure to immediately verify the operability of the reactor core isolation cooling system was a violation of Technical Specification 3.5.1. This Severity Level IV violation is being treated as a noncited violation. The violation was in the licensee's corrective action program as Problem Evaluation Request (PER) 299-1557 (50-397/99009-01).

c. Conclusions

The inspectors identified a violation of Technical Specification 3.5.1, in that operators, with the high pressure core spray system inoperable, did not immediately verify by administrative means that the reactor core isolation cooling system was operable. Operators had isolated the minimum flow line in support of planned maintenance and failed to recognize that the condition rendered the high pressure core spray system inoperable. This Severity Level IV violation is being treated as a noncited violation. The violation was in the licensee's corrective action program as PER 299-1557.

O2 Operational Status of Facilities and Equipment

O2.1 Engineered Safety Feature System Walkdowns

a. Inspection Scope (71707)

The inspectors walked down accessible portions of the following safety-related systems:

- High pressure core spray
- Low pressure core spray
- Residual heat removal (RHR), Trains A, B, and C
- Reactor core isolation cooling
- Division I, II, and III emergency diesel generators
- Standby gas treatment system Trains A and B
- Standby liquid control system
- Standby service water system Trains A, B, and C

b. Observations

The systems were found to be properly aligned for the plant conditions and generally in good material condition.

II. Maintenance

M1 Conduct of Maintenance

M1.1 General Comments - Maintenance

a. Inspection Scope (61726, 62707)

The inspectors inspected the following maintenance and surveillance activities:

- Work Order RFK-801, Thermolag removal (event related review)
- Procedure 10.25.11, "Reactor Building Crane MT-CRA-2 Electrical Maintenance Procedure," Revision 9 (record review)
- Procedure 10.4.5, "Reactor (MT-CRA-2) and Turbine Building (MT-CRA-1) Overhead Traveling Crane Inspection, Maintenance and Testing," Revision 7 (record review)
- Procedure MMP-HOI-C101, "Reactor Building Crane Travel Interlock Operability Test," Revision 0 (record review)

b. Observations and Findings

A problem with Thermolag removal is discussed in Section M1.2, and problems associated with a refueling floor crane hook examination are discussed in Section M1.3.

M1.2 Thermolag Replacement

a. Inspection Scope (62707)

During Thermolag removal, the licensee identified that workers had inadvertently damaged numerous safety-related cables. The inspectors observed the licensee's response to the finding.

b. Observations

During Thermolag removal, contractors inadvertently severed electrical cables to 5 components and damaged approximately 70 other cables. Five valves in the reactor plant closed cooling water and the fuel pool cooling systems were rendered inoperable. However, these systems remained operable. Either the valves had no safety functions, or they were rendered inoperable in the safety position, or there were other redundant safety-related components that served to accomplish the safety function. Most other

damage was limited to slight nicks in the cable insulation, and in a few other instances there was limited damage to conductors. In all these additional instances, the licensee determined that component operability was not affected.

As corrective measures, the licensee walked down all potentially-affected cable trays, completed operability evaluations, repaired the severed cables with environmentally-qualified splices, and initiated actions to repair the remainder of the cables with an approved environmentally-qualified configuration. The corrective measures were acceptable.

c. Conclusions

During Thermolag removal, contractors inadvertently severed electrical cables to 5 components and damaged approximately 70 other cables. While operability of some components was affected, all system safety functions were maintained. Planned corrective measures were acceptable.

M1.3 Refueling Floor Crane Inspections and Maintenance

a. Inspection Scope (62707, 61726)

The inspectors reviewed safety-related refueling floor crane (MT-CRA-2) inspection and maintenance records, the Final Safety Analysis Report and other licensee commitments, associated standards, certified vendor information, and plant implementing procedures.

b. Observations and Findings

While most activities were appropriately completed in accordance with procedures and met the recommendations in licensing documents, the inspectors found that crane hook inspections were not appropriately accomplished in accordance with plant procedures. Crane hook measurements were addressed by Procedure 10.4.5, Section 7.3.16, "Hook Inspection (Annual)," and Attachment 9.3, "Inspection and Marking of Hooks." Procedure 10.4.5 required that current hook dimensions, determined during an inspection, and initial hook dimensions, determined prior to initial use, were to be recorded and compared. The procedure specified that if a 15-percent increase in a throat opening dimension or 10-percent wear in a throat section was identified, the hook was to be removed from service. The inspectors found the following specific problems:

- Initial hook dimensions were not available to the craftsmen and were not utilized during the inspections. In one instance, the worker placed his own initials in the space provided for the "initial" dimensions (inspection report dated January 9, 1998). The second checker signed the document without question. In most cases, initial dimensions were recorded but were not the actual initial dimensions for the hook.

- Measured dimensions varied unexpectedly over time without explanation. For example, the current main hook throat opening dimensions were less than previous main hook throat opening dimensions (inspection reports dated September 22, 1998, and February 1, 1991). Additionally, the initial dimension for the auxiliary hook saddle section (inspection report dated September 22, 1998) were half an inch larger than the initial dimensions provided on auxiliary hook Drawing 31A-00, 15, Revision 3, dated November 15, 1974.

- In some instances, the required site rigging coordinator signatures were missing (e.g., Attachment 9.3, dated September 22, 1998; Attachment 9.2, dated January 9, 1998; and Attachment 10.4, dated February 12, 1991). Procedure 10.4.5, Section 7.3.22.b states: "Forward the completed periodic inspection report to the SRC [site rigging coordinator] for review, approval or initiation of any corrective actions."

As corrective measures, the licensee: (1) sensitized craftsmen and the site rigging coordinators to the findings and (2) identified original scribed markings on the crane hook and compared those dimensions with measurements on original design drawings. No excessive yielding was identified. The corrective measures were appropriate to the circumstances.

The failure to properly implement Procedure 10.4.5 was an example of a Technical Specification 5.4.1.a violation, which requires, in part, that procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, be established and implemented. The Regulatory Guide recommends procedures for maintenance and surveillance activities. This Severity Level IV violation is being treated as an example of a noncited violation, consistent with Appendix C of the NRC Enforcement Policy. The problems are in the licensee's corrective action program as PERs 99-1759 and -1760 (50-397/99009-02).

c. Conclusions

The inspectors identified a Technical Specification 5.4.1.a violation, in that maintenance craftsmen did not properly implement a procedure for checking the refueling floor crane hook for yielding. The craftsmen did not obtain the original as-built dimensions to compare with the as-found dimensions, as required. Additionally, they did not question unexpected and illogical data and results, and the licensee had not independently reviewed all the results. This Severity Level IV violation is being treated as an example of a noncited violation. The violation was in the licensee's corrective action program as PERs 299-1759 and -1760.

M2 Maintenance and Material Condition of Facilities and Equipment

M2.1 Review of Material Condition During Plant Tours

a. Inspection Scope (62707)

During this inspection period, the inspectors conducted interviews and routine plant tours to evaluate plant material condition.

b. Observations and Findings

Overall plant material condition was very good, with some exceptions. The following new material condition problems were observed.

- The motor for circulating water Pump B failed when one of the phases shorted to ground. The motor was sent to a vendor for refurbishment, and the licensee planned to return the pump to service in early September. The plant is normally equipped with three 50-percent capacity pumps. If another pump failed, a power reduction would be necessary.

- A low pressure heater drain valve, located on the 471-foot turbine building elevation, was leaking approximately 1 gallon per minute. The water had leaked through floor cracks and rained onto the 441-foot elevation, contaminating a section of the floor. The licensee planned to repair the valve during the next refueling outage.

The following material condition improvements were completed this inspection period:

- Tower Makeup Pump A was repaired and returned to service. The pump had failed in June 1999.

c. Conclusions

Plant material condition was very good, with some problems. New material conditions concerns included: (1) the failure of one circulating water pump and (2) a 1-gallon-per-minute leak on a low pressure heater drain valve, which had leaked through the floor and contaminated the condenser bay floor below. Conversely, an inoperable tower makeup pump was returned to service this period.

III. Engineering

E2 Engineering Support of Facilities and Equipment

E2.1 Hydraulic Fluid Leak

a. Inspection Scope (37551)

Equipment operators identified a digital electrohydraulic controls system hydraulic fluid leak in the turbine building. Some of the fluid migrated down a valve stem and into the reactor coolant system, and some became captured in the lagging. The inspectors observed the engineering and chemistry response to the event.

b. Observations and Findings

On August 8, 1999, equipment operators identified a hydraulic fluid (Fyrquel) leak originating from a turbine governor valve body plug. Much of the fluid was spraying as a mist into the turbine building, but some was liquefying on the floor, was adsorbed in lagging, and was pooling in locations on the governor valve itself. The licensee isolated the leak, which included closing the governor valve, and performed repairs. The licensee determined that the auto-ignition temperature for the Fyrquel was well above the maximum expected lagging and piping temperatures. Therefore, no fire hazard was present.

Shortly after the event, control room operators reported higher-than-normal main steam line radiation levels (slightly higher than normal) and reactor coolant conductivity levels (about five times normal). The elevated parameters peaked within a few hours and decayed to normal levels in a few days. Chemistry personnel sampled the reactor coolant and verified the presence of Fyrquel in the coolant. The following actions were accomplished:

- Chemistry personnel determined, through a detailed analysis, that the amount of Fyrquel in the reactor coolant was limited to 2 pints. The fluid chemically broke down in the reactor and was removed by the reactor water cleanup system and through gasification.

- Engineers found that the Fyrquel had seeped past the governor valve stem into the seal steam system and had migrated into the reactor vessel. The filter demineralizers were not effective at removing this type of contaminant from the condensate water.

- The licensee checked industry reports for similar events and potential consequences. While one other nuclear plant had inadvertently injected Fyrquel into the reactor and that event was related to subsequent fuel failures, there was little similarity in the circumstances and resultant conditions. At the other utility, between 50 and 100 gallons of the fluid were injected. Additionally, there were numerous other contributors to the fuel failures.

- Engineers consulted with the fuel vendors to determine the potential impact on the fuel. The fuel vendors indicated that the small amount of Fyrquel injected should result in no adverse fuel effects.

- Engineers systematically identified all plant components that may have come into contact with the Fyrquel. Considering the component materials and the amount and concentration of Fyrquel exposure in each case, the engineers determined that no plant components were adversely affected.

The inspectors considered the licensee's investigation to be thorough and comprehensive.

c. Conclusions

Engineering and chemistry provided good support to operations in response to inadvertent Fyrquel hydraulic fluid in-leakage into the primary system. The departments: (1) determined that no more than 2 pints were injected; (2) determined that the fuel would not be affected; and (3) verified that no other plant equipment would be impacted by the incident. The assessment was thorough and comprehensive.

E2.2 Scaffolding Evaluations

a. Inspection Scope (37551)

The inspectors performed engineering evaluations for scaffolding storage racks located in safety-related areas.

b. Observations and Findings

The inspectors identified that no engineering evaluation was completed for a scaffold storage rack on the 422-foot elevation of the reactor building. Procedure 10.2.53, "Seismic Requirements for Scaffolding, Ladders, Man-Lifts, Tool Gang Boxes, Hoists, and Metal Storage Cabinets," Revision 17, required an engineering evaluation for the structure. The storage rack mass was approximately 3000 pounds, and it was secured to a safety-related seismic support. The engineering evaluations were necessary to ensure that these structures did not adversely impact the plant or analysis, as described in the Final Safety Analysis Report.

The rack was installed approximately 6 years ago, which was prior to the procedural requirement for an engineering evaluation implemented approximately 3 years ago. However, Procedure 10.2.53, Section 7.1.9 required an adequacy verification at least every 12 months to ensure that the storage rack was in compliance with the procedural requirements. As such, the annual inspections were inadequate, in that they should have identified the problem earlier.

The failure to properly implement Procedure 10.2.53 was a second Technical Specification 5.4.1.a violation. This Severity Level IV violation example is being treated

as a noncited violation, consistent with Appendix C of the NRC Enforcement Policy. The problem is in the licensee's corrective action program as PER 299-1591 (50-397/99009-02).

c. Conclusions

The inspectors identified a second example of a Technical Specification 5.4.1.a violation, in that an engineering evaluation was not completed, as required by a plant procedure, for a scaffolding storage rack on the 422-foot reactor building elevation. Annual inspections were not effective at catching the problem earlier. This Severity Level IV violation example is being treated as a noncited violation, consistent with Appendix C of the NRC Enforcement Policy. The problem is in the licensee's corrective action program as PER 299-1591.

E8 Miscellaneous Engineering Issues (92700, 92903, 37551)

E8.1 (Closed) Unresolved Item 50-397/99004-01: design basis of the RHR system shutdown cooling mode.

While reviewing plant procedures and control panel labels, the inspector identified that design limitations precluded the licensee from meeting Technical Specification requirements for operability. Specifically, portions of the RHR system were analyzed to 295°F (48 psig saturation pressure), but Technical Specification 3.4.9 required that the system be operable in the shutdown cooling mode at 358°F (135 psig). Accordingly, the system was not designed to be operable in all conditions required by the Technical Specifications.

While the licensee was previously aware of the inconsistency, no action was taken to change the Technical Specification. Instead, normal operating procedures and control panel labels were changed to place restrictions on system operation. Operator training was also provided to avoid operation in an unanalyzed condition. The guidance instructed operators to refrain from operation in the shutdown cooling mode at pressures greater than 48 psig.

The inspectors also found that the administrative measures were not consistently adequate to ensure system operation within design limits under all conditions. For example, Procedure 4.12.11, "Control Room Evacuation and Remote Cooldown," Revision 32, permitted entry into shutdown cooling at less than 135 psig, and Procedure 2.4.2, "Residual Heat Removal System," Revision 35, permitted entry into shutdown cooling at greater than 48 psig postaccident. Additionally, labels on Control Room Panel H13-P601 permitted entering shutdown cooling at greater than 48 psig postaccident.

Corrective measures to address the inspectors' concerns included: (1) submitting a Technical Specification amendment request to change the operability requirements to be consistent with the design; (2) initiating changes to procedures and other instructions to ensure the design limits are consistently followed; and (3) initiating actions to ensure that no other licensing-based conflicts exist. The licensee also determined that substantial analysis would be required to allow raising the shutdown cooling initiation

temperature. The planned and implemented corrective measures were adequate to address the specific findings.

The failure to properly translate design limitations into Technical Specifications, plant procedures, and other instructions was an example of a 10 CFR Part 50, Appendix B, Criterion III (Design Control) violation. This requirement states, in part, that measures shall be established to assure that the design bases are correctly translated into specifications, procedures, and instructions. This Severity Level IV violation example is being treated as a noncited violation, consistent with Appendix C of the NRC Enforcement Policy. The violation is in the licensee's corrective action program as PERs 299-0574 and -0691 (50-397/99009-03).

E8.2 (Closed) Unresolved Item 50-397/99004-04: Technical Specification Bases inaccuracies associated with RHR Valves 8 and 9.

The inspectors had identified that Technical Specification Table 3.3.6.1-1, note (d), was no longer applicable. Note (d) states "Only the inboard trip system required in Modes 1, 2 and 3, as applicable, when the outboard valve is transferred to the alternate remote shutdown panel and the outboard valve is closed." However, a plant modification negated the need to transfer the outboard valve (RHR-V-8) to the alternate remote shutdown panel, making the note moot. Engineering had failed to identify the need to remove the obsolete instruction from the Technical Specifications.

In addition, the licensee had identified that Technical Specification Bases Section B 3.3.6.1, "Primary Containment Isolation Instrumentation," was in error. The bases incorrectly stated that there were four reactor high pressure isolation instrumentation channels, when only two have existed since original construction. The inspectors verified that this information was consistent with historical versions of the Final Safety Analysis Report. The licensee had inappropriately submitted boilerplate information from NUREG-1434, "BWR Standard Technical Specifications," in the Improved Technical Specifications submittal, and failed to modify the bases section with plant-specific information. Since the facility had not changed, the inspectors determined that 10 CFR 50.59 was not applicable.

The failure to revise Technical Specification Table 3.3.6.1-1, note (d), to be consistent with the current design and operation of the plant, and the failure to ensure that the instructions contained in Technical Specification Bases Section B 3.3.6.1 was consistent with the plant design, were the second and third examples of a 10 CFR Part 50, Appendix B, Criterion III, violation. These Severity Level IV violation examples are being treated as a noncited violation, consistent with Appendix C of the NRC Enforcement Policy. This violation is in the licensee's corrective action program as PERs 299-0548 and -0691(50-397/99009-03).

E8.3 Negative Trend Associated with Technical Specification Fidelity Issues

In addition to the Technical Specification problems discussed in Sections E8.1 and E8.2, above, the NRC has identified four other Technical Specifications-related problems in the past 18 months, as follows:

(1) Technical Specification 4.3.1.2.b (fuel storage requirements) was not properly changed to be consistent with a revised safety analysis that specified a more restrictive fuel storage configuration (NRC Inspection Report 50-397/99-07).

(2) The Mode 4 and 5 surveillance requirements for condensate storage tank level did not ensure the committed reserve volume of 135,000 gallons (NRC Inspection Report 50-397/98-15).

(3) The postaccident environmental effects were not adequately considered when deriving the allowed value for reactor vessel water Level 1 (NRC Inspection Report 50-397/98-15).

(4) A Technical Specification-required battery surveillance test was not incorporated into procedures. Consequently, the service test for Division 2, Battery E-B1-2, was missed (NRC Inspection Report 50-397/98-15).

Considering the number of Technical Specifications-related problems identified, and the relatively small number of Technical Specifications activities inspected, the inspectors were concerned with the overall fidelity of the licensee's Technical Specifications submittal and implementation. In response to the inspectors' concerns, the licensee initiated PER 299-1828 to review the problem further. This is considered an inspection followup item pending further NRC review of the licensee's evaluation (50-397/99009-04).

E8.4 (Closed) Inspection Followup Item 50-397/96201-06: Verify the Appendix R alternate shutdown activities for RHR valves are accomplished from the remote shutdown panel.

The licensee changed the design of the RHR motor-operated valve limit switches for the remote shutdown mode by rewiring the connections to bypass the control room. This change eliminated potential short locations that would affect the control circuit in case of fire in the main control room when the controls were transferred to the remote shutdown panel. The same modification was not performed on the alternate shutdown motor-operated valve control circuits. The licensee's representative stated that the requirements of 10 CFR Part 50, Appendix R, for alternate shutdown capability for the RHR system would be met with the current design. During this inspection period, the inspectors reviewed the design, the Final Safety Analysis Report, Appendix F, and related plant procedures. The inspectors agreed with the licensee's conclusions and determined that the Appendix R requirements were met.

E8.5 (Closed) Licensee Event Reports 50-397/1998-012-00 and -01: 24-month Surveillance Requirement 3.8.4.7 was not completed for the Division 2, 125 VDC Battery E-B1-2.

This problem was previously addressed in NRC Inspection Report 50-397/98-15, Section E2.6.b, and the NRC issued Violation 50-397/98015-06 for the failure to perform the surveillance test per the Technical Specification requirements. No new issues were revealed by this licensee event report.

IV. Plant Support

P2 Status of Emergency Preparedness Facilities, Equipment and Resources

P2.1 General Comments (71750)

During routine plant tours, the inspectors verified that the emergency preparedness facilities were properly maintained and that the licensee maintained at least the minimum staffing required by the emergency plan. No problems were found.

P4 Staff Knowledge and Performance in Emergency Preparedness

P4.1 Emergency Preparedness Drill

a. Inspection Scope (71750)

The inspectors observed the 6-year off-hours emergency preparedness drill.

b. Observations and Findings

On July 29, 1999, the licensee conducted the 6-year off-hours call-in drill. The Alert was declared at approximately 7:55 p.m. Most required positions were filled consistent with emergency plan commitments, but the licensee failed to meet the commitment to staff two chemistry technicians within 1 hour of declaring the Alert. Further, the operations support center manager and the team tracker failed to recognize the problem, as they were misled by an erroneous sign-in board that only contained one slot for a chemistry technician signature. The manager appropriately investigated the problem after being prompted by an exercise controller.

There are eight chemistry technicians normally employed at WNP-2. One is required to be on-shift at all times, and a second technician was supposed to travel to the site in response to the Alert. Any of the seven remaining technicians could have responded. Approximately 30 minutes into the drill, the first chemistry technician attempted to call in to the computerized phone system. He had inadvertently made a mistake when punching the numbers on his touch-tone phone and spent an additional 25 minutes attempting to call in again (the line was busy). The technician failed to follow training instruction, which specified that he was only to spend 5 minutes attempting to call in, then he was to travel to the site. None of the remaining chemistry technicians responded to the page or the computerized auto-dialer system.

As a short-term corrective measure, the licensee established a dedicated on-call chemistry technician. This technician is required to continuously wear his pager and remain fit for duty. Additionally, training concepts were reinforced with plant personnel and actions were initiated to correct the sign-in board to provide a signature block for the second chemistry technician. The short-term corrective measures were acceptable. Longer-term corrective measures will be addressed through implementation of the corrective action program.

c. Conclusions

During the off-hours call-in drill, a 6-year event, the licensee failed to meet the emergency plan commitment to staff two chemistry technicians within 1 hour of declaring the Alert. Further, the operations support center manager and the team tracker failed to recognize the problem, as they were misled by an erroneous sign-in board that only contained one slot for a chemistry technician signature. Corrective actions were adequate.

S2 Status of Security Facilities and Equipment

S2.1 General Comments (71750)

During routine tours, the inspectors observed protected area illumination levels, maintenance of the isolation zones around protective area barriers, and the status of security power supply equipment. No problems were observed.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of licensee management on September 8, 1999. The licensee acknowledged the findings presented. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

P. J. Inserra, Licensing Manager
J. A. McDonald, Production Manager
W. S. Oxenford, Operations Manager
S. A. Boynton, Quality Assurance Manager
J. V. Parrish, Chief Executive Officer
D. C. Perry, Radiation Operations Supervisor
D. J. Poirier, Maintenance Manager
G. O. Smith, Vice President - Generation/Nuclear Plant General Manager
R. L. Webring, Vice President - Operations Support

INSPECTION PROCEDURES USED

IP 37551:	Onsite Engineering
IP 61726:	Surveillance Observations
IP 62707:	Maintenance Observations
IP 71707:	Plant Operations
IP 71750:	Plant Support
IP 92700:	On-Site Event Report Followup
IP 92903:	Engineering Followup

ITEMS OPENED AND CLOSED

Opened

50-397/99009-04	IFI	Negative trend associated with Technical Specification fidelity issues (Section E8.3).

Opened and Closed

50-397/99009-01	NCV	Failure to complete Technical Specification action statement for inoperable high pressure core spray system (Section O1.2).
50-397/99009-02	NCV	Two procedure violation examples for failing to accomplish refueling floor crane hook inspections and failing to complete an engineering evaluation for scaffolding (Sections M1.3 and E2.2).
50-397/99009-03	NCV	Three examples of Technical Specification accuracy problems (Sections E8.1 and E8.2).

Closed

50-397/99004-01	URI	Design basis of the RHR system, shutdown cooling mode (Section E8.1).
50-397/99004-04	URI	Technical Specification Bases inaccuracies associated with RHR Valves 8 and 9 (Section E8.2).
50-397/96201-06	IFI	Verify Appendix R alternate shutdown activities for RHR valves are accomplished from the remote shutdown panel (Section E8.4).
50-397/1998-012-00 50-397/1998-012-01	LER	24-month Surveillance Requirement 3.8.4.7 was not completed for the Division 2, 125 vdc Battery E-B1-2 (Section E8.5).

LIST OF ACRONYMS USED

CFR	Code of Federal Regulations
IFI	inspection followup item
LER	licensee event report
NCV	noncited violation
NRC	U.S. Nuclear Regulatory Commission
PER	problem evaluation request
psig	pounds per square inch, gage
RHR	residual heat removal
URI	unresolved item