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

June 27, 2000

EA-00-094

Mr. J. V. Parrish (Mail Drop 1023)
Chief Executive Officer
Energy Northwest
P.O. Box 968
Richland, Washington 99352-0968

SUBJECT: WNP-2 INSPECTION REPORT NO. 50-397/00-10

Dear Mr. Parrish:

On April 2 through May 20, 2000, the NRC completed a safety inspection at the WNP-2 facility. The enclosed report presents the results of this inspection. The results of this inspection were discussed on May 19, 2000, with Mr. R. Webring and other members of your staff.

The inspectors examined activities conducted under your license as they relate to safety and to compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspectors examined a selection of procedures and representative records, observed activities, and conducted interviews with personnel.

Based on the results of this inspection, the NRC documented four issues in this report. Three issues were evaluated under the significance determination process and were determined to be of very low safety significance (Green). The licensee performed a risk analysis for the remaining issue and it was also determined to be of very low safety significance. These issues have been entered into your corrective action program and are discussed in the summary of findings and in the body of the attached inspection report. Of the four issues, three were determined to involve violations of NRC requirements, but because of their very low safety significance the violations are not cited.

One of the NCVs, however, involves the failure to identify an unreviewed safety question through your 10 CFR 50.59 process. Our investigation determined that an unreviewed safety question exists concerning an unisolable piping run between your control rod drive and reactor core isolation cooling system pump rooms. The condition is contrary to your flooding protection commitments and increases the probability of malfunction of equipment important to safety. You are required to seek NRC approval for this condition unless, through further analysis, you can demonstrate that an unreviewed safety question does not exist or the condition is corrected.

If you contest these noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the

Regional Administrator, Region IV; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Cooper facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/RA/

Linda Joy Smith, Chief
Project Branch E
Division of Reactor Projects

Docket No.: 50-397
License No.: NPF-21

Enclosure:
NRC Inspection Report No.
50-397/00-10

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-397
License No.:	NPF-21
Report No.:	50-397/00-10
Licensee:	Energy Northwest
Facility:	WNP-2
Location:	Richland, Washington
Dates:	April 2 through May 20, 2000
Inspectors:	G. D. Replogle, Senior Resident Inspector, Project Branch E, DRP J. P. Rodriguez, Resident Inspector, Project Branch E, DRP J. M. Keeton, Resident Inspector, Project Branch E, DRP J. F. Melfi, Project Engineer, Project Branch E, DRP M. P. Shannon, Senior Health Physicist, DRS L. T. Ricketson, Senior Health Physicist, DRS A. B. Earnest, Physical Security Inspector, DRS
Approved By:	Linda Joy Smith, Chief, Project Branch E, Division of Reactor Projects

ATTACHMENTS:

Attachment 1: Supplemental Information

Attachment 2: NRC's Revised Reactor Oversight Program

SUMMARY OF FINDINGS

WNP-2
NRC Inspection Report 50-397/00-10

The report covers a 7-week period of resident inspection and announced inspections by health physics and security inspectors. The significance of issues is indicated by their color (green, white, yellow, red) and was determined by the significance determination process in Inspection Manual Chapter 0609.

Cornerstone: Mitigating Systems

- NO COLOR. The inspectors identified that the licensee's 50.59 evaluation, concerning a change to flooding protection features, was inadequate in that it failed to identify an unreviewed safety question. The change to the facility involved accounting for a nonisolable drain line between the reactor core isolation cooling and control rod drive pump rooms. Per the Final Safety Analysis Report the rooms were supposed to be water resistant and not connected. This change was an unreviewed safety question because the connection could result in the malfunction of several additional pieces of equipment important to safety during flooding. The inadequate evaluation was a noncited violation of 10 CFR 50.59 (EA-00-094). The licensee disagreed with the violation.

  10 CFR 50.59 issues are not handled per the significance determination process. However, a risk assessment concerning the licensee's proposed change to the facility was performed and the change was of very low risk significance (Section 1R02.1).

- Green. The inspectors identified two instances (of three checked) where equipment failures were not properly characterized as maintenance preventable functional failures. First, two main steam pressure switches were rendered inoperable for 82 days because of a faulty calibration rig, which was constructed by maintenance personnel. Second, the Division II standby gas treatment system was rendered inoperable when craftsmen miscalibrated the heaters, because of a faulty work package.

  This issue was of very low safety significance because it was administrative in nature and did not, in itself, affect the reliability of the noted safety-related equipment (Section 1R12).

- Green. Reactor core isolation cooling system keepfill pump rebuild procedures were inadequate. The keepfill pump failed because lubrication components were not properly adjusted during refurbishment and a bearing did not receive adequate lubrication. The condition rendered the reactor core isolation cooling system unavailable for 13 hours, when pump replacement was completed. The inadequate work documents constituted a noncited violation of Technical Specification 5.4.1.a.

The inspector determined that this issue was of very low risk significance because of the short reactor core isolation cooling system out-of-service time (Section 1R19).

- Green. During a surveillance, the licensee identified that two condenser vacuum pressure switches were inoperable for 82 days (Licensee Event Report 50-397/2000-01). The condition rendered the main steam isolation valves inoperable for the loss of condenser vacuum event, but the valves would have closed at a higher condenser pressure. The switches were rendered inoperable when they were recalibrated with a faulty calibration rig. The problem was found during the subsequent calibration. Additionally, during the problematic calibration, maintenance and operations personnel noted anomalous initial readings but did not follow plant procedures and document the problem on a problem evaluation request. This effectively circumvented the corrective action program. The inspectors determined that the problem constituted a noncited violation of Technical Specification 3.3.6.1.

  This issue was of very low risk significance. The problem would not have increased the risk for core damage and the worst case scenario, leading to condenser failure, would not lead to a large early fission product release (Section 1R22.2).

Report Details

Summary of Plant Status:

At the start of the period, the plant was at 80 percent power due to economic dispatch (load following). Between April 7 and 16, plant power varied between 60 and 80 percent, depending on the power needs of the Pacific Northwest. On April 17, 100 percent power was achieved but was again reduced to 61 percent on April 19. On May 13, power was reduced to 23 percent, to support steam leak repairs in high radiation areas, but was returned to 61 percent the following day. On May 15, power was reduced to 43 percent to support repairs on main steam isolation Valve 28A. Reactor power was increased to 60 percent on May 17 and then to 70 percent on May 18, where it essentially remained for the remainder of the period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R02 Evaluation of Changes, Tests, or Experiments (71111.02)

.1 (Closed) Unresolved Item 50-397/99013-02: as-built reactor building flood protection was not consistent with Final Safety Analysis Report commitments.

The inspector had identified that the licensee was not meeting flooding protection commitments contained in the Final Safety Analysis Report. In part, the Final Safety Analysis Report specified that reactor core isolation cooling and control rod drive pump rooms were individual water-resistant compartments. However, the two compartments were connected via a nonisolable 3-inch-diameter equipment drain line. Additionally, the inspector identified that Calculation 5.51.055, "Flooding Analysis," neglected the equipment drain line and assumed that each room can flood separately but not at the same time. As such, the analysis was inadequate. The Final Safety Analysis Report commitments were made, in part, to demonstrate compliance with 10 CFR Part 50, Appendix A, General Design Criterion 4, which requires flooding protection for equipment important to safety. The Final Safety Analysis Report design provided separation, which is a method acceptable for meeting General Design Criterion 4.

Since the unresolved item was opened, the licensee performed a new flooding analysis and a 10 CFR 50.59 assessment to evaluate the departure from the flood protection commitments. The licensee concluded the issue was not an unreviewed safety question. While only one residual heat removal pump (A or B) would remain following the flood, the licensee specified that operators would use a single train for both reactor injection and containment heat removal. Operators would line up the train with suction from the suppression pool and would maintain the heat exchanger in service. They would flood the reactor and the effluent water would escape through safety relief valves to the suppression pool.

NRC Assessment: The inspector reviewed the licensee's flooding analysis and 10 CFR 50.59 evaluation. The inspector discussed the documents with experts in the NRC's Office of Nuclear Reactor Regulation and determined that the departure from the flooding protection commitments of the Final Safety Analysis Report was an unreviewed safety question, which was contrary to the licensee's determination.

The described condition was an unreviewed safety question because the probability of malfunction of equipment important to safety evaluated in the Final Safety Analysis Report was increased. While the licensee's 10 CFR 50.59 evaluation stated that this was not the case, the inspector observed that the revised flooding analysis resulted in rendering several additional pieces of equipment inoperable. For example, reactor core isolation cooling room flooding consequences, under the licensed design, resulted in rendering reactor core isolation cooling, residual heat removal Train C, and the low pressure core spray system inoperable. However, when the connection between the reactor core isolation cooling and control rod drive pump rooms was considered, the high pressure core spray system was additionally rendered inoperable (equipment important to safety). Similarly, a flood in the control rod drive pump room had only challenged control rod drive and high pressure core spray pump operability. Under the revised flooding analysis, the reactor core isolation cooling system, residual heat removal Train C, and the low pressure core spray system (all systems important to safety) also fail.

The licensee determined that the change to the facility was not risk-significant. The increase in core damage frequency was less than 1E-10. In the risk assessment, the licensee assumed that the cables running through the two rooms would maintain their integrity during flooding. Accordingly, the licensee assumed that the flood only affects the reactor core isolation cooling and control rod drive pumps. More realistic assumptions are permitted in risk assessments than are otherwise allowed in deterministic evaluations.

The failure to submit a license amendment for a change to the facility involving an unreviewed safety question was a violation of 10 CFR 50.59. This regulation permits a change to the facility as described in the Final Safety Analysis Report, without Commission approval, as long as the proposed change does not involve an unreviewed safety question. Based on very low risk significance, this Severity Level IV violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy (EA-00-094). This issue has no color because, in the revised reactor oversight program, 10 CFR 50.59 issues are not processed through the significance determination process. The problem is in the licensee’s corrective action program as Problem Evaluation Request 200-0630 (50-397/00010-01).

Licensee Position: The licensee disagreed with the inspector's determination and stated that the issue was not an unreviewed safety question. The licensee referenced Question E.12 from "Questions and Answers on 10 CFR 50.59 and Nuclear Energy Institute 96-07, Revision 1," Draft, April 4, 2000 (Nuclear Energy Institute composed guidance). The question and answer states:

> Section 4.3.2 of Nuclear Energy Institute 96-07, R1, says that a change that reduces system/equipment redundancy, diversity, separation, or independence requires prior NRC approval. Does this mean reduction from redundancy, diversity, separation, or independence described in the UFSAR [Updated Final Safety Analysis Report]? Or is prior NRC approval required only if the change reduces redundancy, diversity, separation or independence below the level required by the regulations?

> A change that reduces redundancy, diversity, separation or independence described in the USAR requires prior NRC approval. Licensees may, however, without prior NRC approval, reduce excess redundancy, diversity, separation, or independence to the level credited in the licensing basis.

In explaining how the above criteria could be applied, the licensee first emphasized that their commitment to Branch Technical Position APCSB3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," does not include a commitment for protective features (for moderate energy line breaks), such as leak resistant rooms. Final Safety Analysis Report Subsection 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Primary Containment," describes how the licensee meets NRC Branch Technical Position APCSB3-1. In addition, Final Safety Analysis Report Section 3.6.1.11.3, "Method of Analysis for Postulated Moderate-Energy Fluid System Ruptures," describes the methods that were used to evaluate safe shutdown capability following pipe breaks. The licensee believes that, as long as safe shutdown was achievable through this analysis, then any change to the flood protection provisions was acceptable.

The licensee specified that since the NRC had reviewed their approach, with respect to the branch technical position and flooding analysis, and found it acceptable, this NRC review and approval established the acceptable level of equipment redundancy and separation necessary for safe shutdown. Therefore, while some additional equipment would fail because of the change to the facility, the additional failures constitute removal of excess redundancy and separation to the level credited in the licensing basis.

Further NRC Evaluation: The inspector determined that the licensee did not meet the criteria established in referenced Nuclear Energy Institute document. This document specifies, in part:

> Licensees may, however, without prior NRC approval, reduce excess redundancy, diversity, separation or independence to the level credited in the licensing basis.

There is no minimum level of acceptable equipment specified in the WNP-2 licensing basis for flood protection. As such, the change to the facility did not constitute the removal of excess redundancy specified in the licensing basis.

Additionally, while the inspector agreed that the licensee's referenced Final Safety Analysis Report sections did not commit the licensee to water resistant rooms for most pipe breaks in the reactor building, this was not the only portion of the Final Safety Analysis Report that contained flood protection measures. The specific commitments for water resistant rooms were located in Final Safety Analysis Report Sections 3.4.1.4.1.2, "Internal Flood Protection Requirements," and 3.4.1.5.2, "Internal Flood Protection Measures." These flood protection measures were generic to the facility and not only protected equipment important to safety from pipe breaks but also protected the facility from other flooding risk as well, such as from internally generated missiles (i.e., from pump impellers).

The inspectors noted that, during plant construction, the licensee did not originally commit to water resistant rooms in the Final Safety Analysis Report. However, in response to NRC Question 031.099, which asked the licensee to specify internal flood protection design provisions, the licensee updated Sections 3.4.1.4.1.2 and 3.4.1.5.2 (Amendment 5) to include the commitments for water resistant compartments. These flood protection features were evaluated against Standard Review Plan 3.4.1, Section III.5, which asked the reviewer:

> . . . to determine if safety related equipment or components are located within individual compartments or cubicles which act as positive barriers against possible means of flooding . . .

Based on this information, the inspectors determined that the need for water resistant rooms was considered by the NRC and was a part of the original WNP-2 licensing basis.

Finally, the statements of consideration for 10 CFR 50.59 state, in part:

> The intent of the 50.59 process is to permit licensees to make changes to the facility, **provided the changes maintain the level of safety documented in the original licensing basis**, such as in the safety analysis report . . . Margins and equipment functionality, reliability and availability also may be impacted by facility changes . . . Therefore, the criteria for requiring NRC approval were directly related to . . . 2) preserving effectiveness (reliability) of the mitigation systems by **not allowing introduction of different equipment malfunctions** and by limiting increases in probability of malfunction, or reduction in the margin of safety (which reflects the capability of the system) . . .

It was routine for the licensee to commit, in the Final Safety Analysis Reports, to provide margin well above the minimum required to meet the regulations. The licensee's proposed reduction in margin (by removal of a flood protection barrier) warrants NRC review because the change to the facility does not maintain the level of safety documented in the original licensing basis.

Considering the above, the inspectors determined that the problem was still an unreviewed safety question and the licensee is required to restore the licensed condition, demonstrate that the change would not increase the probability of malfunction of equipment important to safety, or seek NRC approval for the facility change.

1R04 Equipment Alignments (71111.04S)

a. Inspection Scope

The inspectors performed a complete equipment alignment verification for the Control Rod Drive System in accordance with Inspection Procedure 71111, Attachment 04.

b. Issues and Findings

There were no findings identified during this inspection.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

The inspectors performed the routine quarterly fire protection inspection of the following areas in accordance with Inspection Procedure 71111, Attachment 05:

- Emergency diesel generator rooms
- Control room
- Reactor core isolation cooling pump room
- Residual Heat Removal Pump Rooms A, B, and C
- Control rod drive system pump room

b. Issues and Findings

There were no findings identified during this inspection.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

The inspectors observed a crew training scenario, assessed operator performance, and witnessed the evaluator critique in accordance with Inspection Procedure 71111, Attachment 11. The simulator control boards were observed to be consistent with the control room configuration.

b. Issues and Findings

There were no findings identified during this inspection.

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed equipment failures associated with the: (1) compressed air system; (2) standby gas treatment System B; and (3) main steam pressure Switches MS-PS-56B and -D, to evaluate proper implementation of the licensee's Maintenance Rule Program in accordance with Inspection Procedure 71111, Attachment 12. The following documents were reviewed during this inspection:

- Maintenance Rule Status Report for the Fourth Quarter, 1999

- Maintenance Rule Status Report for the First Quarter, 2000

- System Health Report Summary, Fourth Quarter, 1999
- Problem Evaluation Requests 200-0878 (failures not properly characterized as maintenance preventable); 200-183 (MS-PS-56B/D failure); 200-364 (standby gas train B failure); and 299-0724 (control air system, B train failure).
- Maintenance Rule Program, Revision 3

b. Issues and Findings

The inspectors identified two instances where failures were not properly characterized as maintenance preventable functional failures.

Standby Gas Treatment System: On February 28, 2000, the licensee identified that two heater temperature switches were miscalibrated and the condition rendered standby gas treatment system Train B inoperable. The temperature switches were supposed to be calibrated to approximately 200°F but were mistakenly calibrated to approximately 110 degrees during maintenance on February 22. Additionally, following postmaintenance testing on February 25, an unexpected alarm associated with the first stage heater came in. At the time, operators did not know what caused the alarm but believed that the system remained operable and returned the unit to service. A few days later, the licensee identified that the heater control switches were miscalibrated and declared the unit inoperable. An inadequate work package caused the problem.

> NOTE: An NCV was identified in NRC Inspection Report 50-397/2000-09 for the inadequate work package. The intent of this inspection is solely dedicated to evaluating how the failure was processed in the licensee's Maintenance Rule Program.

The system engineer determined that the failure was not a maintenance preventable functional failure because operators were alerted to a problem during postmaintenance testing (the above noted alarm). However, the inspectors disagreed with the determination because the system was returned to service prior to identification and correction of the problem. Upon further review the licensee agreed with the inspectors conclusion. Problem Evaluation Request 200-0878 was issued to address the oversight. The maintenance preventable function failure did not result in exceeding the Maintenance Rule performance criteria for this system

Main Steam Pressure Switches 56B and D: During a surveillance on January 31, 2000, technicians found condenser vacuum Pressure Switches MS-PS-56B and -56D out of calibration and inoperable. The switches were required to be set at greater than or equal to 7.2 inches of mercury (Hg) vacuum per Technical Specification 3.3.6.1. However, the switches tripped at 2.8 and 2.4 inches Hg vacuum. With the instruments inoperable, the main steam isolation valves would not have closed when required in response to a loss of condenser vacuum event. A loss of condenser vacuum event is described in the Final Safety Analysis Report, Section 15.2.5.

> Note: This issue is covered in more detail in Section 1R22 of this report.

The licensee's investigation found that the two pressure switches were miscalibrated on November 10, 1999, because of a faulty calibration rig and human error. Flow restrictions and leakage in tubing and fittings, between the calibration gage and the pressure switches, caused the problem. Maintenance craftsmen had constructed the calibration rig.

The system engineer determined that the problem was not a maintenance preventable function failure because, based on engineering judgement, the switches would have tripped at a higher value (above the Technical Specification limit) and the main steam isolation valves would have accomplished their safety function prior to condenser failure. However, the inspector asked the engineer to provide the calculation which supported the Technical Specification specified trip value and observed that the calculation demonstrated that the Technical Specification limit was the least conservative value needed to ensure condenser viability during the event.

In addition per the Final Safety Analysis Report, the licensee was committed to Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 1, which endorses NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." NUMARC 93-01, Section 9.4.5, "Maintenance Preventable Functional Failures," specifies:

> Under certain conditions, a SSC [structure system or component] may be considered to be incapable of performing its intended function if it is out of specified adjustment or not within specified tolerances.

Upon further review, the licensee agreed with the inspectors' conclusions and reclassified the failure as maintenance preventable. This problem was also captured in Problem Evaluation Request 200-0878. The maintenance preventable function failure did not result in exceeding the Maintenance Rule performance criteria for this system.

Risk Assessment: The failure to properly characterize maintenance preventable functional failures did not result in a condition that would have impacted core damage frequency or the potential for large early fission product release. Accordingly, this issue is of very low safety significance and is characterized as a Green issue.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation (71111.13)

a. Inspection Scope

The inspectors reviewed the following work prioritization, risk evaluation, and control activities in accordance with Inspection Procedure 71111, Attachment 13:

- Emergent reactor core isolation cooling system maintenance (risk significant component)
- Reactor feedwater pump work (risk significant component)
- Emergent main steam isolation Valve 28A maintenance (event potential)

b. Issues and Findings

There were no findings identified during this inspection.

1R14 Personnel Performance During Nonroutine Plant Evolutions (71111.14)

a. Inspection Scope

On May 15 main steam isolation Valve 28A unexpectedly closed. The inspectors reviewed the Operations response to this event in accordance with Inspection Procedure 71111, Attachment 14. The inspector reviewed the following documents:

- Operations Instructions 9, "Operations Expectations and Standards," dated February 2, 2000

- Problem Evaluation Request 200-811, Operator Failed to Promptly Identify Main Steam Isolation Valve Closure

b. Issues and Findings

The licensee identified that operators did not meet management expectations with respect to board awareness in that the operator took 1 hour and 54 minutes to identify that the valve had closed. Plant computers indicated that the valve closed at 2:54 a.m. but the operator did not note the valve position until 4:48 a.m. Operations Instructions-9, states, in part:

> Operators are alert and attentive to assigned duties; they routinely scan control panels and walk down assigned areas . . .
>
> All on-shift operations crew personnel are responsible for maintaining awareness of plant equipment status and conditions in their assigned area and being alert to the potential for degrading or changing conditions.

Since Operations Instruction-9 was considered additional operator guidance, above the minimum required by the regulations, this was not a violation of NRC requirements.

The licensee counseled operations staff on the problem and initiated Problem Evaluation Request 200-811 to capture the issue.

The inspector also observed that board awareness issues have continued to be a performance problem at WNP-2. Previous operator board awareness problems were documented in NRC Inspection Report 50-397/99-07. In that report, poor board awareness contributed to two events involving unexpected level decreases in the reactor vessel and the spent fuel pool skimmer surge tank.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed operability determinations associated with the following problems in accordance with Inspection Procedure 71111, Attachment 15:

- Problem Evaluation Request 200-0659 - failure of reactor core isolation cooling system keepfill pump
- Problem Evaluation Request 200-0344 - reactor core isolation cooling system test return valve failed testing

b. Issues and Findings

There were no findings identified during this inspection.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors observed or evaluated the following postmaintenance testing activities and documents to determine whether the tests confirmed equipment operability in accordance with Inspection Procedure 71111, Attachment 19:

- Division II emergency diesel generator Heat Exchanger DCW-HX-1B2 leakage and performance tests
- Division II, emergency diesel generator south bearing flush
- Hydraulic control Units 10-47 and 30-31 testing; Work Orders 00SCN701, 00SCN704, and 00SCN709.
- Division II, 125 Vdc battery testing; Work Order W001008781, Surveillance ESP-B12-Q101, "Quarterly Battery Test," Revision 3
- Reactor core isolation cooling system keepfill pump replacement testing, Work Order 29009713

The inspectors also investigated the cause of the reactor core isolation cooling keepfill pump failure to ensure that replacement pump testing was adequate to ensure pump operability.

b. Issues and Findings

Following the reactor core isolation cooling system keepfill pump replacement, the inboard pump bearing was running at elevated temperatures. The previous pump had failed on April 19 because the same bearing had seized. The pump failure rendered the reactor core isolation cooling system inoperable for approximately 13 hours, until the new pump was installed. The licensee subsequently identified that the pump failure was caused by inadequate lubrication to the pump bearing. The bearings are lubricated by an internal reservoir, but the oil level was maintained at a specific level by the bayonet oiler. The oiler was not properly adjusted during pump rebuild to provide the bearing with adequate lubrication. The pump was rebuilt at WNP-2 and installed in January of this year. Appropriate adjustments were made to the newly installed pump's oiler and the bearing temperatures returned to normal.

The inspectors reviewed pump rebuild Work Order 00LPZ0 (for the failed pump) and determined that the document provided inadequate instructions related to oiler adjustment. Specifically, the work order referenced the vendor manual regarding lubrication requirements. However, the vendor manual was written to address installation and periodic maintenance but not for complete pump rebuild. While the vendor manual stated:

> The correct oil level is maintained 7/32" above the centerline of the tapped opening in frame by the bayonet type oiler.

No provisions were made in the work package to ensure that this adjustment was maintained during pump rebuild.

The failure to provide adequate work instructions regarding the refurbishment of the reactor core isolation cooling system keepfill pump was a violation of Technical Specification 5.4.1.a. This Technical Specification requires procedures for activities covered by Regulatory Guide 1.33. The regulatory guide specifies procedures for maintenance of safety-related equipment. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The problem is in the licensee’s corrective action program as Problem Evaluation Request 200-0659 (50-397/00010-02).

The inspectors determined that the issue was of very low safety significance using the significance determination process (green).

1R22 Surveillance Testing (71111.22)

.1 Routine Observations

a. Inspection Scope

The inspectors observed or reviewed the following surveillance tests in accordance with Inspection Procedure 71111, Attachment 22:

- ISP-MS-Q934, "Division 2 Channel B Isolation Actuation on Reactor Level 2 - CFT/CC[channel functional test/channel check]," Revision 2

- ESP-B1DG3-Q101, "Quarterly Battery Testing 125 VDC HPCS-B1-DG3," Revision 2

- O.P.-EEC-S702, "Diesel Generator 2 Semi-Annual Operability Test," Revision 3
- TOP-C.D.-C101, "C.D. [Control Rod Drive] Scram Timing with Auto Scram timer System," Revision 4

b. Issues and Findings

There were no findings identified during this inspection.

.2 (Closed) Licensee Event Report 50-397/2000-001: inoperable loss of condenser vacuum pressure switches due to inadequate surveillance.

During a surveillance on January 31, 2000, technicians found condenser vacuum Pressure Switches MS-PS-56B and -56D out of calibration and inoperable. The switches were required to be set at greater than or equal to 7.2 inches of mercury (Hg) vacuum per Technical Specification 3.3.6.1. However, the switches tripped at 2.8 and 2.4 inches Hg vacuum. With the instruments inoperable, the main steam isolation valves would not have closed when required in response to a loss of condenser vacuum event. A loss of condenser vacuum event is described in the Final Safety Analysis Report, Section 15.2.5.

The licensee's investigation found that the two pressure switches were miscalibrated on November 10, 1999, because of a faulty calibration rig and human error. Flow restrictions and leakage in tubing and fittings, between the calibration gage and the pressure switches, caused the problem. During the surveillance, technicians had found the pressure switches significantly out of the administrative calibration band, but in the conservative direction - 13.6 and 12.8 inches Hg vacuum (false readings). At the time, the technicians brought the surveillance results to the control room supervisor and were instructed to recalibrate the instruments to within the acceptable band. The subsequent calibration adjustments resulted in trip settings that were too low.

The licensee also found that the technicians and the control room supervisor, involved in the November 7, 1999, surveillance, failed to follow plant procedures in that they did not initiate a problem evaluation request. Procedure SWP-CAP-01, "Problem Evaluation Requests," Revision 1, required initiation of problem evaluation requests for abnormal occurrences. The as-found condition of the two pressure switches was abnormal even though the switches were out of calibration in the conservative direction. Both switches were substantially out of calibration, by a similar amount (over 5 inches Hg vacuum) and in the same direction. Historically, these pressure switches demonstrated much better performance. The failure to initiate a problem evaluation request for this condition resulted in circumventing the corrective action program.

As corrective measures, the licensee planned to redesign the calibration rig, enhance reliability, and retrain plant personnel to reinforce requirements for initiating problem evaluation requests. The inspectors considered the licensee's investigation and planned corrective measures acceptable.

The inspectors consulted with regional risk specialists and determined that the risk consequences of the inoperable pressure switches were very low. While the main steam isolation valves would not have closed when required, they would have closed in an actual condenser overpressure condition at 2.4 inches Hg vacuum. All other automatic main steam isolation valve closure capabilities were still in effect. Absent automatic main steam isolation valve closure, operators could have manually closed the valves from the control room. The loss of safety function constituted no additional risk of core damage or large early fission product release.

The failure to maintain main steam system Pressure Switches MS-PS-56B and -D operable was a Technical Specification 3.3.6.1 violation. This Technical Specification requires, in part, operable pressure switches. The allowed outage time for the condition was 12 hours and the switches were inoperable for approximately 82 days. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The problem is in the licensee’s corrective action program as Problem Evaluation Request 200-0183 (50-397/00010-03).

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the following plant temporary modifications in accordance with Inspection Procedure 71111, Attachment 23, with respect to design bases documentation, approvals, and tracking. The inspectors also walked down the modifications to assure that the tags were still in place.

- Temporary Modification Request 99-026, "Eliminate nuisance alarms from the oscillation power range monitors"
- Temporary Modification Request 98-11, "Sand filter for service water spray ponds"

During the inspection the inspectors reviewed the following documents:

- Procedure 1.3.9, "Temporary Modifications," Revision 25
- Procedure SWP-PRO-01, "Description and Use of Procedures and Instructions," Revision 2
- Problem Evaluation Request 200-0629, "Inappropriate use of should statements in temporary modifications procedure"

b. Issues and Findings

There were no findings identified during this inspection.

1OA1 Performance Indicator (PI) Verification (71151)

a. Inspection Scope

The inspectors verified the performance indicator for the reactor core isolation cooling system for period ending April 1, 2000, in accordance with Inspection Procedure 71151.

b. Issues and Findings

There were no findings identified during this inspection.

2. **RADIATION SAFETY**
Cornerstones: Occupational Radiation Safety, Public Radiation Safety

2OS1 Access Controls to Radiologically Significant Areas (71121.01)

a. Inspection Scope

Radiation workers and radiation protection personnel were interviewed concerning their radiation protection work requirements. A number of tours of the radiologically controlled area were conducted. The following items were reviewed:

- Access controls and surveys of three significant high dose work areas in the radiologically controlled area
- Radiation work permits and specified electronic pocket dosimeter setpoints
- Placement of personnel dosimetry
- Radiation postings and barricades used at entrances to high dose rate areas, high radiation areas, and very high radiation areas
- Job coverage by radiation protection personnel
- Health physics prejob briefing for the refueling pool mast and camera work

b. Issues and Findings

There were no findings identified during this inspection.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspector interviewed radiation workers and radiation protection personnel involved in high dose rate and high exposure jobs throughout the radiologically controlled area during routine operations. Independent radiation surveys of selected work areas within the controlled access area were performed. The following items were reviewed:

- ALARA program procedures
- Processes used to estimate and track exposures
- Plant collective exposure history for the past 3 years, current exposure trends, and 3-year rolling average dose information
- Five radiation work permit packages from the outage/online work activities which resulted in the highest personnel collective exposures during the inspection period
- Available data for trends in collective exposures and source term measurements
- Use of engineering controls to achieve dose reductions
- Individual exposures of selected work groups (health physics, operations, and mechanical maintenance)
- Hot spot tracking and reduction program
- Plant related source term data, including source term control strategy
- Radiological work planning
- Selected corrective action documentation involving higher than planned exposures and radiation worker practice deficiencies since the last inspection in this area
- Declared pregnant worker dose monitoring controls

b. Issues and Findings

There were no findings identified during this inspection.

2PS2 Radioactive Material Processing and Transportation (71122.02)

a. Inspection Scope

The inspector reviewed independent laboratory analyses of samples taken from four boxes of waste sent to the Washington low-level radioactive waste disposal site in early December 1999. The licensee subsequently retrieved the waste when it became aware of the possibility that the boxes contained mixed waste.

b. Issues and Findings

There were no findings identified during this inspection.

3. **SAFEGUARDS**
Cornerstone: Physical Protection

3PP1 Access Authorization (IP 71130.01)

a. Inspection Scope

The inspector completed the following inspection elements:

- Reviewed licensee event reports and safeguards event logs to identify problems in the access authorization program

- Reviewed procedures, audits, and self-assessments of the following programs/areas: behavior observation, access authorization, fitness-for-duty, supervisor, escort, and requalification training

- Interviewed five supervisors/managers and five individuals who had escorted visitors into the protected and/or vital areas to determine their knowledge and understanding of their responsibilities in the behavior observation program

- Reviewed condition reports, licensee event reports, safeguards event logs, audits, and self-assessments for the licensee's access control program, and selected security events reports.

- Interviewed key security department and plant support personnel to determine their knowledge and use of the corrective action reports and resolution of problems regarding repair of security equipment.

b. Issues and Findings

There were no findings identified during this inspection.

3PP2 Access Control (IP 71130.02)

a. Inspection Scope

The inspector completed the following inspection elements:

- Reviewed licensee event reports and safeguards event logs to identify problems with access control equipment

- Reviewed procedures and audits for testing and maintenance of access control equipment and for granting and revoking unescorted access to protected and vital areas

- Interviewed security personnel concerning the proper operation of the explosive and metal detectors, X-ray devices, and key card readers

- Observed licensee testing of access control equipment and the ability of security personnel to control personnel, packages and vehicles entering the protected area

- Reviewed procedures to verify that a program was in place for controlling and accounting for hard keys to vital areas

- Reviewed the licensee's process for granting access to vital equipment and vital areas to authorized personnel having an identified need for that access

- Reviewed condition reports, licensee event reports, safeguards event logs, audits, and self-assessments for the licensee's access control program, and selected security events reports.

- Interviewed key security department and plant support personnel to determine their knowledge and use of the corrective action reports and resolution of problems regarding repair of security equipment.

b. Issues and Findings

There were no findings identified during this inspection.

4. OTHER ACTIVITIES

4OA2 Performance Indicator Verification (IP 71151)

.1 Radiation Safety Performance Indicator Verification

a. Inspection Scope

The inspector reviewed corrective action program records for high-high radiation areas, very high radiation areas, and unplanned exposure occurrences for the past 12 months to confirm that these occurrences were properly recorded as performance indicators. Radiologically controlled area exit transactions with exposures greater than 100 millirem for the past 12 months were reviewed, and selected examples were investigated to determine whether they were within the dose projections of the governing radiation work permits. Additionally, radiological effluent release program corrective action records, licensee event reports, and annual effluent release reports documented during the past 4 quarters were reviewed to determine if any events exceeded the PI thresholds.

b. Issues and Findings

There were no findings identified during this inspection.

.2 Safeguards PI Verification

a. Inspection Scope

The inspector completed the following inspection elements:

- Reviewed the licensee's program for collection and submittal of performance indicator data. Specifically, a random sampling of security event logs, maintenance logs, and corrective action reports were reviewed for the following program areas:

  (1) Fitness-for-duty/personnel reliability program performance

  (2) Personnel screening program performance

  (3) Protected area security equipment performance index.

- Reviewed the licensee's security tracking, trending, and analysis of perimeter security equipment problems.

b. Issues and Findings

There were no findings identified during this inspection.

4OA5 Other

.1 (Closed) Licensee Event Report (LER) 50-397/1999S01 Failure To Take Compensatory Measure Within Required Time Frame. This LER was a minor issue and was closed.

.2 Inspectors reviewed the following inspection followup items (IFI) and determined that no further action is required. These items are closed.

- IFI 50-397/9720-07: Effectiveness of Final Safety Analysis Report Upgrade Program

- IFI 50-397/9809-06: Division I emergency diesel generator failures and performance criteria

4OA6 Meetings

.1 Exit Meeting Summary

The physical security inspector presented the safeguards inspection results to Mr. D. Atkinson and other members of licensee management at an exit meeting on April 13, 2000.

The health physicist inspectors presented the radiation safety inspection results to Mr. R. Webring and other members of licensee management on May 11, 2000.

The senior resident inspector presented the remainder of the inspection results to Mr. R. Webring and other members of licensee management at an exit meeting on May 19, 2000. The licensee did not agree with a noncited violation involving an unreviewed safety question (Section 1R02.1).

The licensee acknowledged the findings presented during each meeting. Additionally, following each meeting the inspectors asked the licensee whether or not any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

J. Parrish, Chief Executive Officer
D. Atkinson, Acting Vice President, Operation Support
A. Barber, Supervisor, Quality Services
D. Bennett, Manager, Chemistry
I. Borland, Radiation Protection Manager
S. Boynton, Quality Assurance Manager
J. Givin, Supervisor, Security Force
J. Gloyn, Supervisor, Access Authorization, Fitness For Duty, and Security Training
P. Ensure, Licensing Manager
R. James, Operations Supervisor, Radiation Protection
D. Martin, Security Manager
C. McDonald, Supervisor, Health Physics/Chemistry/General Employee Training
C. Nordhaus, Planning Supervisor, Radiation Protection
S. Oxenford, Plant Manager
J. Peters, Manager, Radiological Services
J. Pierce, Support Supervisor, Radiation Protection
D. Poirier, Maintenance Manager
R. Sherman, Engineer, Licensing
R. Webring, Vice President - Acting Generation/Nuclear Plant General Manager

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed During this Inspection

50-397/00010-01	NCV	Reactor building flood protection not consistent with Final Safety Analysis Report (1R02.1)
50-397/00010-02	NCV	Inadequate work package causes reactor core isolation keeping keepfill pump failure (1R19)
50-397/00010-03	NCV	Inoperable loss of condenser pressure switches due to inadequate surveillance (1R22.2)

Previous Items Closed

50-397/99013-02	URI	Reactor building flood protection not consistent with Final Safety Analysis Report (1R02.1)
50-397/97020-07	IFI	Effectiveness of the Final Safety Analysis Report upgrade program (40A5.2)
50-397/98009-06	IFI	Division I emergency diesel generator failures and performance criteria (40A5.2)
50-397/2000-001	LER	Inoperable loss of condenser pressure switches due to inadequate surveillance (1R22.2)
50-397/1999S01	LER	Failure to take compensatory measure within required time frame (40A5.1)

LIST OF ACRONYMS USED

ALARA	as low as reasonably achievable
CFR	Code of Federal Regulations
IFI	inspection followup item
LER	licensee event report
NCV	noncited violation
NRC	Nuclear Regulatory Commission
PI	performance indicator
URI	unresolved item

PARTIAL LIST OF DOCUMENTS REVIEWED

Procedures

- SWP-RPP-01, “Radiation Protection Program,” Revision 3
- GEN-RPP-01, “ALARA Program Description,” Revision 2
- GEN-RPP-02, “Radiation Work Permit,” Revision 2
- GEN-RPP-04, “Entry into, Conduct in, and Exit from Radiologically Controlled Areas,” Revision 4
- GEN-RPP-13, “ALARA Committee,” Revision 3
- HPP 11.2.2.5, “ALARA Job Planning and Reviews,” Revision 10
- HPP 11.2.2.11, “Exposure Evaluations for Maintaining TEDE ALARA,” Revision 3
- HPP 11.2.7.1, “Area Posting,” Revision 13
- HPP 11.2.7.3, “High, High High, and Very High Radiation Area Controls,” Revision 17
- HPP 11.2.13.1, “Area Radiation and Contamination Surveys,” Revision 9
- HPP 11.2.13.8, “Airborne Radioactivity Surveys,” Revision 6
- HPI 12.70, “RWP and ALARA Planning Processes,” Revision 7

Radiation Work Permits

- 30000095 00 Refurbish Hydraulic Control Unit
- 30000100 00 Motor Operated Valve Testing on RCC-MO-131 in Reactor building 548' Fuel Pool Cooling Heat Exchanger Room
- 29000003 02 Equipment Operation/Investigation
- 29000189 02 Main Steam Relief Valve
- 29000288 00 Plant Equipment Operation/Investigation

Reviews and Assessments

- Quality Department Surveillance Report, “R-14 Outage,” (9/17-10/26/99)
- Quality Department Surveillance Report, “SR200-006,” (3/16/00)
- Quality Department Surveillance Report, “SR200-009,” (4/19/00)
- Quality Department Audit Report, “Process Control Program,” Audit AU200-001
- Annual Radiation Protection Review of WNP-2 During 1999 Performed for Energy Northwest
- Energy Northwest Refuel Outage 14 Report
- Quality Assurance Audit Report No. 299-049, “Fitness for Duty and Personnel Access Data System
- Quality Assurance Audit Report No. 299-042, “Security Programs Surveillance Frequency Review”
- Quality Assurance Audit Report No. 298-055, “WNP-2 Security Program

Safeguards Documents

- Safeguards Event Logs from April 1, 1999 through March 30, 2000
- Fitness-for-Duty 6-Month Reports For 1999
- WNP-2 Physical Security Plan, Revision 41

ATTACHMENT 2

NRC'S REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety	**Radiation Safety**	**Safeguards**
•Initiating Events	•Occupational	•Physical Protection
•Mitigating Systems	•Public	
•Barrier Integrity		
•Emergency Preparedness		

To monitor these seven cornerstones of safety, the NRC used two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the significance determination process and assigned colors of GREEN, WHITE, YELLOW, OR RED. GREEN findings are indicative of issues that, while they may not be desirable, represent little effect on safety. WHITE findings indicate issues with some increased importance to safety, which may require additional NRC inspections. YELLOW findings are more serious issues with an even higher potential to effect safety and would require the NRC to take additional actions. RED findings represent an unacceptable loss of safety margin and would result in the NRC taking significant actions that could include ordering the plant shut down.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing incremental degradation in safety: GREEN, WHITE, YELLOW, or RED. The color for an indicator corresponds to levels of performance that may result in increased NRC oversight (WHITE); performance that results in definitive, required action by the NRC (YELLOW); and performance that is unacceptable but still provides adequate protection to public health and safety (RED). GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an action matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action as described in the matrix. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.