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

July 20, 2001

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

SUBJECT: COLUMBIA GENERATING STATION - INSPECTION REPORT NO. 50-397/01-03

Dear Mr. Parrish:

On June 23, 2001, the NRC completed an inspection at your Columbia Generating Station for the period April 1 through June 23, 2001. The enclosed integrated inspection report presents the results of this inspection. The in-office emergency preparedness inspection results were presented to Mr. T. Messersmith and other members of your staff in a telephone conversation on May 7, 2001. The radioactive material processing and transportation inspection results and the inservice inspection results were discussed on May 10 and 24, respectively, with Mr. R. Webring and other members or your staff. The access control and performance indicator verification inspection results were discussed on May 25 with Mr. G. Smith and other members of your staff. The remaining inspection results were discussed with you and other members of your staff on June 27, 2001.

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

Based on the results of this inspection, the inspectors identified four issues of very low safety significance (Green). These issues involved: (1) inadequate corrective actions to ensure the adequacy of an alternate decay heat removal method; (2) inappropriate testing of drywell unidentified leak-rate instruments; (3) the failure to test secondary containment isolation valves in accordance with your Inservice Testing Program; and (4) the failure to properly post a high-high radiation area. These issues were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they were entered into your corrective action program, the NRC is treating these issues as noncited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the violations or the significance of the 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, U.S. Nuclear Regulatory Commission, Region IV, 6111 Ryan Plaza Drive, Suite 400, Arlington Texas 76011; the Director, Office of

Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Columbia Generating Station.

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/

William B. Jones, Chief
Project Branch E
Division of Reactor Projects

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

Enclosure:
NRC Inspection Report
50-397/01-03

cc w/enclosure:
Chairman
Energy Facility Site Evaluation Council
P.O. Box 43172
Olympia, Washington 98504-3172

Rodney L. Webring (Mail Drop PE08)
Vice President, Operations Support/PIO
Energy Northwest
P.O. Box 968
Richland, Washington 99352-0968

Greg O. Smith (Mail Drop 927M)
Vice President, Generation
Energy Northwest
P.O. Box 968
Richland, Washington 99352-0968

D. W. Coleman (Mail Drop PE20)
Manager, Regulatory Affairs
Energy Northwest
P.O. Box 968
Richland, Washington 99352-0968

Albert E. Mouncer (Mail Drop 1396)
General Counsel
Energy Northwest
P.O. Box 968
Richland, Washington 99352-0968

Paul Inserra (Mail Drop PE20)
Manager, Licensing
Energy Northwest
P.O. Box 968
Richland, Washington 99352-0968

Thomas C. Poindexter, Esq.
Winston & Strawn
1400 L Street, N.W.
Washington, D.C. 20005-3502

Bob Nichols
State Liaison Officer
Executive Policy Division
Office of the Governor
P.O. Box 43113
Olympia, Washington 98504-3113

Lynn Albin
Washington State Department of Health
P.O. Box 7827
Olympia, WA 98504-7827

Electronic distribution from ADAMS by RIV:
Regional Administrator **(EWM)**
DRP Director **(KEB)**
DRS Director **(ATH)**
Senior Resident Inspector **(GDR)**
Branch Chief, DRP/E **(WBJ)**
Senior Project Engineer, DRP/E **(GAP)**
Section Chief, DRP/TSS **(PHH)**
RITS Coordinator **(NBH)**
Scott Morris **(SAM1)**
NRR Event Tracking System **(IPAS)**
Columbia Site Secretary **(LEF1)**
Dale Thatcher **(DFT)**
William Dean **(BYD)**

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-397
License:	NPF-21
Report:	50-397/01-03
Licensee:	Energy Northwest
Facility:	Columbia Generating Station
Location:	Richland, Washington
Dates:	April 1 through June 23, 2001
Inspectors:	G. D. Replogle, Senior Resident Inspector, Project Branch E, DRP J. F. Melfi, Resident Inspector, Project Branch E, DRP B. D. Baca, Health Physicist, Plant Support Branch, DRS D. R. Carter, Health Physicist, Plant Support Branch, DRS C. A. Clark, Reactor Inspector, Engineering and Maintenance Branch, DRS P. J. Elkmann, Emergency Preparedness Inspector, DRS M. P. Shannon, Senior Health Physicist, Plant Support Branch, DRS J. L. Taylor, Reactor Inspector, Engineering and Maintenance Branch, DRS
Approved By:	W. B. Jones, Chief, Project Branch E, Division of Reactor Projects

ATTACHMENT: Supplemental Information

SUMMARY OF FINDINGS

IR 05000397-01-03; on 4/1-6/23/2001; Energy Northwest; Columbia Generating Station. Integrated Inspection Report; Refueling Outage; Surveillances; EAL & Emerg. Plan Chngs.; Access Control.

The report covers a 12-week period of routine resident inspection from April 1 through June 23, 2001. The inspection identified four findings that had very low safety significance, which were all noncited violations. The significance of the findings is indicated by their color (Green, White, Yellow, or Red) using Manual Chapter 0609 “Significance Determination Process." The NRC’s program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

Cornerstone: Mitigating Systems

- Green. A noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI (Corrective Actions), was identified for inadequate corrective measures taken for an issue identified during a previous outage. Plant personnel had failed to identify an appropriate method of alternate decay heat removal with both trains of shutdown cooling inoperable. The licensee's revised method for alternate decay heat removal was inadequate because it did not meet Technical Specifications and Bases commitments, in that operators the could not place the system in service within 1 hour and the system would become inoperable in response to a loss of reactor cavity water level. The violation is more than minor because it had a credible impact on safety in that the alternate decay heat removal path would not have been available if vessel level decreased and it would not be available for up to 5 hours after needed. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The problem is entered into the licensee's corrective action program as Problem Evaluation Request 201-1360.

  The finding represents a problem identification and resolution crosscutting issue where the licensee’s corrective actions for establishing an alternate decay heat removal path still did not meet the Technical Specification requirements. The finding was determined to be of very low safety significance using the significance determination process Appendix G (Shutdown Operations) for the reactor vessel inventory greater than 23 feet above the reactor vessel flange and the time-to-boil was greater than 2 hours, the finding screened out as Green based on the fact that it did not result in a loss of reactor vessel inventory or a significant loss of thermal margin (Section 1R20).

Cornerstone: Initiating Event

- Green. A noncited violation of 10 CFR Part 50, Appendix B, Criterion XI (Test Control), occurred for inadequate calibration testing of the drywell unidentified leak rate instrument. The Final Safety Analysis Report specifies that the instrument can detect an increase of 1 gpm [gallon per minute] in 1 hour but the lowest point checked during calibration was 1.5 gpm. During the past cycle, this instrument read about 0.8 gpm when actual leakage was about 1.2 gpm. This finding is more than minor and had a potential impact on safety because the drywell unidentified leak rate instrument was not calibrated to detect changes in unidentified leakage consistent with the Final Safety Analysis Report and could impact operator responsiveness to the initial phases of a loss of coolant accident. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The problem is entered into the licensee's corrective action program as Problem Evaluation Request 201-1362.

This finding was determined to be of very low risk significance based on the finding did not contribute to the likelihood of a primary system LOCA and that small changes in reactor coolant system leakage, that may not be detected, were well below the established Technical Specification limits for unidentified leakage. The issue constituted a qualification deficiency and did not result in a loss of system function (Section 1R22).

Cornerstone: Barrier Integrity

- Green. A noncited violation of Technical Specification Surveillance Requirement 3.6.4.2.2 was identified for the failure to test secondary containment isolation Valves FDR-V-219, FDR-V-220, FDR-V-221, FDR-V-222, EDR-V-394, and EDR-V-395 in accordance with the Inservice Testing Program. This issue was more than minor based on the time of discovery, two valves were degraded and would not have passed Code testing. The inspectors identified a lack of design control and design understanding, on the part of plant engineers, associated with six secondary containment isolation valves in the equipment drain and floor drain systems. Inspector identified problems included the failure to perform Technical Specification required testing, an inappropriate change to the Technical Specifications Bases, inaccurate plant drawings, and the failure to meet commitments with respect to describing secondary containment system classifications in the Final Safety Evaluation Report. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The problem is entered into the licensee's corrective action program as Problem Evaluation Request 201-0680.

  The inspectors determined that the issue had very low safety significance because the finding only represents a degradation of the radiological barrier function provided for the secondary containment versus primary containment (Section 1R22).

Cornerstone: Occupational Radiation Safety

- Green. On May 22, 2001, the inspector identified that radiological postings surrounding recirculation Loop A on the 501-foot elevation of the drywell were not in accordance with Technical Specification 5.7.2.(a) requirements. General area radiation levels were as high as 2500 millirem per hour. The failure to post a high-high radiation area is a violation of Technical Specification 5.7.2.(a). The issue was more than minor because the failure to control an area in accordance with Technical Specification requirements has a credible impact on safety and the potential for unplanned or unintended dose. This violation is being treated as a noncited violation consistent with Section VI. A.1 of the NRC Enforcement Policy. This violation is entered into the licensee’s corrective action program as Problem Evaluation Request 201-0886.

The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised (Section 2OS1).

B. Licensee Identified Violations

Violations of very low significance which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee appear reasonable. These violations are listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status:

Operators maintained reactor power at 100 percent at the beginning of the period. On April 9, reactor power started gradually coasting down because of fuel depletion, as expected. On May 18 operators initiated a reactor shutdown from approximately 81 percent reactor power. Operators completed the shutdown on May 19, signifying entry into Refueling Outage 15. The plant remained shut down for the remainder of the period.

1 REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignments (71111.04)

a. Inspection Scope

The inspectors verified partial equipment alignments, for the existing plant conditions, for the following systems while the licensee had the redundant trains out of service.

- Residual heat removal system, Train C
- Residual heat removal system, Train B
- Division II emergency diesel generator with both the Division I and III units inoperable
- High pressure core spray system with the reactor core isolation cooling system out of service

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors performed the routine quarterly fire protection inspection. The inspectors observed the functionality and material condition of the fire protection equipment, detection systems, and passive protection features. The inspectors also verified proper controls for combustible materials and ignition sources. The inspectors reviewed the following areas:

- Drywell
- Division I, II, and III switch gear rooms

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08)

.1 Performance of Nondestructive Examination Activities

The Columbia Generating Station inservice inspection program is committed to the ASME [American Society of Mechanical Engineers] Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” 1989 Edition, with no Addenda for the second 10-year interval. The current Refueling Outage R-15 inservice inspections are scheduled to complete the second period of the second 10-year interval of the program.

a. Inspection Scope

The inspectors reviewed the licensee’s nondestructive examination records for work that was performed for the current outage. This review was performed to verify that nondestructive examination activities were performed in accordance with ASME Boiler and Pressure Vessel Code requirements. Nondestructive examination records reviewed are listed in the Attachment.

The inspectors observed the licensee’s nondestructive examination contractor personnel (General Electric Nuclear Energy) perform the inservice inspection program specified examinations listed below.

System	Component/Weld Identification	Examination Method
Reactor Recirculation System	Loop A valve-elbow weld 24RRC(1)A-19	Ultrasonic Examination
Reactor Recirculation System	Loop A elbow-pipe weld 24RRC(1)A-20	Ultrasonic Examination

During the performance of each examination, the inspectors verified that the correct nondestructive examination procedure was used, procedural requirements or conditions were as specified in the procedure, and test instrumentation or equipment was properly calibrated and within the allowable calibration period. The inspectors reviewed the nondestructive examination certification packages of the contractor personnel and verified that they had been properly certified in accordance with ASME Code requirements. The inspectors also verified that indications revealed by the examinations were compared against the ASME Code-specified acceptance standards and appropriately dispositioned.

b. Findings

No findings of significance were identified.

.2 ASME Code Repair and Replacement Activities

a. Inspection Scope

The inspectors reviewed an ASME Section XI Code repair and replacement package for work performed on High Pressure Core Spray Valve HPCS-V-102 to verify repairs and replacements met ASME Code requirements. The work performed on High Pressure Core Spray Valve HPCS-V-102 was performed as part of the “Small Bore Vibration Fatigue Failure Reduction Program.”

b. Findings

No findings of significance were identified

.3 Identification and Resolution of Problems (71152)

a. Inspection Scope

The inspectors reviewed the problem evaluation request records listed below to verify the appropriateness of the corrective actions and that the licensee was identifying inservice inspection problems at an appropriate threshold and entering the problems into the corrective action program:

299-2338   299-2115   200-0560   200-0118   201-0342

b. Findings

No findings of significance were identified

1R12 Maintenance Rule Implementation (71111.12)

a. Inspection Scope

The inspectors reviewed the following documents associated with equipment failures to assess the effectiveness of the Maintenance Rule evaluations:

- Problem Evaluation Request 201-0475, Sparks coming from reactor core isolation cooling pump mechanical seal

- Problem Evaluation Request 201-1055, Loss of 215 KV power source because of lightning strike
- Problem Evaluation Request 201-1041, Division II diesel generator failed to start following maintenance
- Problem Evaluation Request 201-1324, Division III diesel generator failed to start following overhaul
- Problem Evaluation Request 201- 0532, Compressed air system Compressor CAS-C-1C tripped on thermal overloads
- Problem Evaluation Request 201-1259, Containment isolation Valve FDR-V-3 failed to close
- Control Room Logs
- Maintenance Rule Program, Revision 3

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors reviewed the following work prioritization, risk evaluation, and work control activities to evaluate the effectiveness of licensee risk management:

- Compressor CAS-C-1C (risk significant) out of service concurrent with several other pieces of risk significant components
- Shutdown cooling outage (both trains)
- Division I outage
- Division II outage

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following operability evaluations and related documents. The inspectors checked that the licensee properly justified operability and that other components/systems remained available such that no unrecognized increase in risk had occurred:

- Problem Evaluation Request 201-0446, Reactor core isolation cooling Valve RCIC-V-63 back-seated to stop packing leak

- Problem Evaluation Request 201-0732, Scram discharge volume level switch isolated for 3 months (Section 40A7 (1))

- Problem Evaluation Request 201-0984, Linear cracks on control rod blades

- Problem Evaluation Requests 200-2178, 200-0962, 200-2178, and 201-0962, which addressed emergency diesel generator heat exchanger problems

- Nonconventional snubber configuration observed during drywell walkdown

- Problem Evaluation Request 201-1033, Anomalous indications during fuel clad inspections

The inspectors reviewed the following additional documents during this inspection:

- "Duralife Control Blade Indications Safety Evaluation," June 2001,

- Letter from General Electric to Columbia Generating Station, dated June 8, 2001 addressing duralife control blade issues

- General Electric RICSIL 84, "Duralife Control Rod Blade Cracking," dated May 12, 2001

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors evaluated postmaintenance testing for the following activities to determine whether the tests confirmed equipment operability:

- Division III diesel generator testing following the 6-year overhaul
- Reactor core isolation cooling system pump rebuild

- Hydraulic Control Unit 3423 maintenance
- Oscillation power range monitoring system (new system)
- Nobel metal application

Documents reviewed during this inspection included:

- Procedure 2.7.3, "High Pressure Core Spray Diesel Generator," Revision 39

- Procedure OSP-ELEC-M703, “HPCS [High Pressure Core Spray System] Diesel Generator Monthly Operability Test,” Revision 12

- Work Order 01025933, Reactor core isolation cooling outboard seal postmaintenance testing

- Work Order 01026747, Hydraulic Control Unit 3423 directional controller replacement

- Work Orders 010007862 and 010007863, Operational power range monitor testing

- General Electric Evaluation, "Noble Chem Application at Columbia Generating Station," dated May 22, 2001

b. Findings

No findings of significance were identified.

1R20 Refueling Outage (71111.20)

a. Inspection Scope

The inspectors observed the following refueling outage activities and verified that the activities were well controlled and accomplished in accordance with documents appropriate to the circumstances:

- Plant shutdown
- Clearance activities
- Refueling
- Electric power configurations
- Shutdown cooling management and configuration
- Spent fuel cooling operations
- Inventory control
- Containment control
- Drywell closeout
- Overtime controls
- Identification and resolution of problems

The inspectors reviewed the following documents as part of this inspection:

- "R-15 Outage Shutdown Safety Plan," including all changes up to June 23, 2001
- Procedure 3.2.1, "Normal Shutdown to Cold Shutdown," Revision 46
- Procedure 2.8.5, "Fuel Pool Cooling and Cleanup System," Revision 34
- Procedure ABN-FPC, "Fuel Pool Trouble," Revision 1
- Clearance Orders:
    - 00PJD901 - CRD-FCV-2A, Replace stem
    - 00PJF007 - Isolate CRD-FCV-2B
    - 01010000-01 - Refurbish CRD-SPV-186
    - 01012236-01 - Rebuild CRD-SPV-110A
    - 01022046-01 - Rebuild CRD-SPV-110B
    - 01024740-01 - DSA-DY-2 Replacement

b. <u>Findings</u>

A noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI (Corrective Actions), was identified for inadequate corrective measures taken for an issue identified during a previous outage. Plant personnel had failed to identify an appropriate method of alternate decay heat removal with both trains of shutdown cooling inoperable. The finding was determined to affect the mitigating system cornerstone and to be of very low safety significance (Green) using the significance determination process Appendix G (Shutdown Operations).

The inspectors identified that the licensee had specified inadequate corrective measures for Problem Evaluation Request 299-0871, dated April 1999. The corrective action document specified, in part, that no adequate method of alternate decay heat removal was specified to meet the requirements of Technical Specification 3.9.8, Action A.1. The inspectors observed that the recommended method stemming from the corrective action document still did not meet the Technical Specification requirements.

The Bases for the noted Technical Specification Action specifies:

> With no . . . shutdown cooling available, an alternate method of decay heat removal **must be established within 1 hour**. In this condition, the volume of water above the reactor vessel flange provides adequate capability to remove decay heat from the reactor core. **However, the overall reliability is reduced because the loss of water level could result in reduced decay heat removal capabilities . . .**

The licensee's "R-15 Outage Shutdown Safety Plan" implemented the Problem Evaluation Request 299-0871 corrective actions. The plan specified entry into Technical

Specification 3.9.8, Action A.1, to support work that rendered both trains of normal shutdown cooling inoperable. The plan also identified the alternate method of decay heat removal as the residual heat removal system, Train B, in the spent fuel pool cooling assist mode. In this lineup, the system takes suction from the spent fuel pool skimmer surge tanks. In addition, the outage plan did not require placing the system in service within one hour but called for placing the system in service prior to exceeding certain spent fuel pool temperature limits. Engineers estimated that it would take between 4 and 6 hours to make the system operational from that point.

The inspectors considered the licensee's plans and corrective actions for Problem Evaluation Request 299-0871 inadequate because:

(1) operators would not establish the system within one hour after entering the Action; and

(2) the system would not remain operational in the event of a loss of water level. The skimmer surge tanks receive water through weirs positioned at the very top of the reactor vessel and spent fuel pool. When level drops (even a few inches) flow into the skimmer surge tanks stops, rendering the alternate decay heat removal system inoperable.

This issue is an example of a 10 CFR Part 50, Appendix B, Criterion XVI (Correction Actions), violation. This regulation, in part, requires the licensee to take effective measures to correct conditions adverse to quality. The finding represents a problem identification and resolution crosscutting issue where the licensee’s corrective actions for establishing an alternate decay heat removal path still did not meet the Technical Specification requirements.

The inspectors determined that the problem constituted an issue of more than minor significance because failure to provide adequate alternate decay heat removal capabilities increases the risk of uncontrolled temperature increases in the reactor cavity. The finding was determined to be of very low safety significance using the significance determination process, Appendix G (Shutdown Operations), for the reactor vessel inventory being greater than 23 feet above the reactor vessel flange and the time-to-boil being greater than 2 hours. The finding screened out as Green based on the fact that it did not result in a loss of reactor vessel inventory or a significant loss of thermal margin. Accordingly, 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 201-1360 (NCV 50-397/01003-01).

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the surveillances listed below to verify that the testing demonstrated system/component capability:

- Procedure OSP-RPV-R801, "Reactor Pressure Vessel Leakage Test," Revision 9
- Procedure TSP/DG2/LOCA-B501, "Standby Diesel Generator DG2 [Division II Emergency Diesel Generator] LOCA [Loss of Coolant Accident] Test," Revision 5
- Procedure TSP/DG3/LOCA-B501, "HPCS [High Pressure Core Spray] Diesel Generator LOCA [Loss of Coolant Accident] Test," Revision 3
- Work Orders 01023001,01019691, 01016150, 1012470, 01007427, 01003759, and 01000039, stroke time testing for secondary containment equipment drain and floor drain valves
- Procedure ISP-FDR/EDR-X301, "Drywell Sump Flow Monitors," Revision 4
- Procedure TSP-CRD-C101, "CRD [Control Rod Drive] Scram Timing with Autoscramtimer System," Revision 4
- Procedure ISP-MS/IST-R101, "MSRV [Main Steam Relief Valve] Setpoint Verification," Revision 3

The inspectors reviewed the following additional documents as part of this inspection:

- Final Safety Analysis Report
- Work Schedules
- Procedure OSP-INST-H101, "Shift and Daily Instrument Checks (Modes 1, 2, 3)," Revision 29
- Reactor Coolant System Leakage Logs
- Control Room Logs
- Procedure OSP-ELEC-M702, "Diesel Generator 2 - Monthly Operability Test," Revision 12
- NRC Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," dated July 2, 1984
- EDM Engine Maintenance Manual, Section 10
- Component Summary Sheets for Valves FDR-V-219, FDR-V-220, FDR-V-221, FDR-V-222, EDR-V-394, and EDR-V-395

b. Findings

.1 **Inadequate Secondary Containment Isolation Valve Testing**

A noncited violation of Technical Specification Surveillance Requirement 3.6.4.2.2 was identified for the failure to test secondary containment isolation valves in accordance with the Inservice Testing Program. The finding was determined to effect the barrier cornerstone and to be of very low (Green) safety significance using the significance determination process.

The inspectors identified a Technical Specification Surveillance Requirement 3.6.4.2.2 violation for the failure to test six secondary containment isolation valves in accordance with the Inservice Testing Program since initial plant startup. Several additional issues contributed to the problem and demonstrated a general lack of design control and design understanding by plant personnel. The valves included FDR-V-219, FDR-V-220, FDR-V-221, FDR-V-222, EDR-V-394, and EDR-V-395. At the time of discovery, two valves were degraded and would not have passed Code testing. Other related NRC identified issues included:

- In March 1996, the licensee implemented the Improved Technical Specifications, which specifically required testing the subject valves in accordance with the ASME Code. However, In June I996, Engineers inappropriately changed the Bases for Technical Specification Surveillance Requirement 3.6.4.2.2 to exclude the valves from ASME Inservice Testing requirements. This change required NRC approval because it eliminated testing required by Technical Specifications. Notwithstanding, since the Technical Specifications still stated that "each automatic secondary containment isolation valve" required the testing, the Bases change did not negate the standing requirements. In addition to the Technical Specifications, 10 CFR 50.55a also required testing of these safety-related ASME Code Class 3 valves in accordance with the Inservice Testing Program. Since initial plant startup, engineers did not recognize that the valves are safety-related and ASME Code Class 3, the criteria for inclusion into the Inservice Testing Program.

- The licensee failed to describe secondary containment features in Final Safety Analysis Report Section 3.2.2. Through the Final Safety Analysis Report, the licensee committed to Regulatory Guide 1.70, which specified:

  > This section should identify those fluid systems or portions of fluid systems important to safety and the industry codes and standards applicable to each pressure-retaining component in the system.

  The secondary containment, including the secondary containment isolation valves, constitutes a safety-related pressure retaining fluid system, but the licensee failed to describe the system in Final Safety Analysis Report, Section 3.2.2.

- Plant engineers committed errors in equipment drain and floor drain piping drawings (M-537 and M-539). The drawings erroneously identified the valves as QC II (nonsafety) and Class B31.1 when they are actually QC I (safety-related) and ASME Code Class 3. The licensee used the drawings to determine which valves to include in the Inservice Testing Program.

The inspectors acknowledged that the licensee performed stroke time testing quarterly. However, the inspectors considered the testing inadequate because, contrary to the requirements of the ASME Code, the valves never failed the testing when the stroke times substantially exceeded expectations. For example, as documented in Problem Evaluation Request 201-0484, on March 31, 2001, Valve FDR-V-222, which normally strokes closed in about 25 seconds, stroked closed in 300 seconds. The licensee did not consider the valve inoperable and did not initiate appropriate corrective measures prior to placing the valve back in service.

The inspectors considered the issue's significance more than minor because the failure of the subject valves during an event could permit the transit of fission product material to other plant areas. The inspectors evaluated the significance of this issue using the Significance Determination Process. The inspectors determined that the issue was of very low safety significance (Green) because the finding only represents a degradation of the radiological barrier function provided for the secondary containment (versus primary containment). In addition, the valves were demonstrated capable of closing over an extended period. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The problem is entered into the licensee's corrective action program as Problem Evaluation Request 201-0680 (NCV 50-397/01003-02).

.2 **Inadequate Testing of Drywell Unidentified Leak-Rate Instrument:**

A noncited violation of 10 CFR Part 50, Appendix B, Criterion XI (Test Control), occurred for inadequate calibration testing of the drywell unidentified leak rate instrument. This finding was determined to affect the initiating event cornerstone and to be of very low risk significance (Green).

The inspectors identified that plant procedures specified inadequate calibration testing for the drywell unidentified leak rate instrument. The Final Safety Analysis Report specifies that the instrument can detect an increase of 1 gpm [gallon per minute] in 1 hour. During the past cycle, however, this instrument read about 0.8 gpm when actual leakage was about 1.2 gpm. Per licensee procedures, the calibration included 1.5 as the lowest verified point. The inspectors considered a 1.5 gpm verification point inadequate to ensure that the instrument could detect 1.0 gpm within 1.0 hour. The failure to perform adequate testing to ensure that safety-related equipment can perform satisfactorily in service constitutes a violation of 10 CFR Part 50, Appendix B, Criterion XI (Test Control). The violation was of greater than minor significance because the deficiency could impact operator responsiveness to the initial phases of a loss of coolant accident.

This finding was determined to be of very low risk significance using the Significance Determination Process Phase 1 worksheets. The finding did not contribute to the increased likelihood of either a primary or secondary system LOCA based on the fact that the small changes in reactor coolant system leakage, that may not be detected, were well below the established Technical Specification limits for unidentified leakage. The issue constituted a qualification deficiency and did not result in a loss of system function. This violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy. The problem is entered into the licensee's corrective action program as Problem Evaluation Request 201-1362 (NCV 50-397/01003-03).

.3 **Questionable Diesel Generator Testing Practices**

An unresolved item was identified regarding the testing of the emergency diesel generators. The issue involves a diesel start from a "standby condition" and performance of the biannual emergency core cooling system start test immediately following the monthly test that could result in the diesel being started in a substantially warmer condition than might otherwise be expected in a *normal* standby condition.

The inspectors observed back-to-back diesel generator testing, to meet different Technical Specification Surveillance Requirements, that could compromise the validity of some tests. For example, the licensee scheduled the monthly Division III diesel generator test (slow start and the diesel is gradually brought up to speed) immediately before the biannual emergency core cooling system start test (a much more demanding test, intended to mirror actual accident conditions). The inspector was concerned with the test schedule because Technical Specification Surveillance Requirement 3.8.1.12 (emergency core cooling system start test) specifies that the diesel start from a "standby condition" and performance of this test immediately following the monthly test could result in the diesel being started in a substantially warmer condition than might otherwise be expected in a *normal* standby condition. The inspector noted that several diesel generator Technical Specification Surveillance Requirements require testing the diesels in a standby condition.

The Bases for Technical Specification Surveillance Requirement 3.8.1.12 states:

> For the purpose of this testing, the DGs [diesel generators] must be started from standby conditions, that is, with the engine coolant and oil being continuously circulated and temperature maintained consistent with manufacturer recommendations.

In response to the inspectors' concerns, the licensee reviewed the manufacturer's recommendations and established a maximum permitted diesel start temperature of 155°F based on the manufacturer's literature, which specified 125 to 155°F as the design temperature range for the keep-warm system. The licensee also ensured the Division III diesel generator was tested under standby conditions. Finally, the licensee performed additional testing to determine the typical diesel generator cooldown rate and reviewed other diesel generator tests for compliance. One Division II diesel generator test was repeated following this review.

The inspectors consulted with a diesel generator Technical Specification expert in the NRC's Office of Nuclear Reactor Regulation to evaluate the adequacy of the licensee's actions. The NRC has identified questions regarding the term "standby condition" and what was meant to specify the normal standby condition expected at most times. In this case, the keep-warm system normally maintains the diesels at approximately 130°F. Testing a diesel at 155°F may not adequately ensure that the diesel can perform its safety function at lower temperatures. The NRC is reviewing the licensee's actions and evaluating the need for further clarification to the licensee's Technical Specifications Bases (compliance backfit analysis). This issue is considered unresolved pending completion of the NRC's review and evaluation on this matter (URI 50-397/01003-04).

1R51 Performance Indicator Verification (71151)

a. Inspection Scope

The inspectors evaluated the following performance indicator data by reviewing operator logs, equipment out-of-service logs, and corrective action program records:

- Reactor coolant system activity
- Reactor coolant system leakage
- Residual heat removal system unavailability

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

a. Inspection Scope

The inspectors performed an in-office review of the following documents against 10 CFR 50.54(q) to determine if the revisions decreased the effectiveness of the emergency plan.

- Revision 27 to the Columbia Generating Station Emergency Plan submitted February 16, 2001

- Revision 7 to Procedure 13.1.1A, “Classifying the Emergency - Technical Bases,” submitted February 22, 2001

- Revision 29 to Procedure 13.1.1, "Classifying the Emergency," submitted February 22, 2001

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstones: Occupational Radiation Safety, Public Radiation Safety

2OS1 Access Control 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 and compared with regulatory requirements:

- Radiation Protection Quality Functional Area Reports AU-RP-02-1 and AU-RP-02-2, Quality Department Forced Outage Report SR200-018, and Self-Assessment Reports SA-00-101 and SA-00-074

- Area postings and other controls for airborne radioactivity areas, radiation areas, high radiation areas, high-high radiation areas, and very high radiation areas

- Radiological surveys involving airborne radioactivity areas and high radiation areas

- Access controls, surveys, and radiation work permits for the following four significant high dose work areas: drywell valve repair - system breach (RWP 30000182-00), reactor disassembly - cavity work (RWP 30000268-01), main steam relief valve maintenance (RWP 30000351-00), and undervessel transverse incore probe tubing removal/installation (RWP 30000362-00)

- ALARA [as low as reasonably achievable] prejob briefing for undervessel transverse incore probe tubing removal/installation

- Dosimetry placement when work involved a significant dose gradient

- Controls involved with the storage of highly radioactive items in the spent fuel pool

- A summary of operational radiation protection corrective action documents (Problem Evaluation Requests) written since May 1, 2000. Twenty-two of these documents were reviewed in detail: PER200-0727, PER200-0753, PER200-0756, PER200-0793, PER200-0815, PER200-0816, PER200-0964, PER200-1014, PER200-1089, PER200-1096, PER200-1334, PER200-1820, PER200-

1956, PER200-1976, PER200-2098, PER201-0027, PER201-0125, PER201-0224, PER201-0527, PER201-0552, PER201-0768, and PER201-0893)

b. Findings

A noncited violation with very low safety significance (Green) was identified for failure to properly post a high-high radiation area. On May 22, 2001, the inspector identified that radiological postings surrounding recirculation Loop A on the 501-foot elevation of the drywell were not in accordance with Technical Specification 5.7.2.(a) requirements. Specifically, an accessible area surrounding recirculation Loop A was posted as a high radiation area, rather than a high-high radiation area. From a review of a radiological survey, the inspector determined that general area radiation levels were as high as 2500 millirem per hour.

The safety significance of this finding was determined to be very low by the Occupational Radiation Safety Significance Determination Process because there was no overexposure or substantial potential for an overexposure, and the ability to assess dose was not compromised. The issue was more than minor because the failure to properly control a high-high radiation area has a credible impact on safety and the potential for unplanned or unintended dose.

Technical Specification 5.7.2.(a) states, in part, that high radiation areas with doses greater than one rem per hour shall be conspicuously posted. Procedure SWP-RPP-01, “Radiation Protection Program,” Revision 4, defines such an area as a high-high radiation area. The failure to post the area in accordance with Technical Specification 5.7.2.(a) is a violation. This violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy. This violation is in the licensee’s corrective action program as PER 201-0886 (NCV 50-397/01003-05).

2PS2 Radioactive Material Processing and Transportation (71122.02)

a. Inspection Scope

The inspector interviewed radiation protection and chemistry personnel involved in radioactive material/waste processing and transportation activities and walked down the liquid and solid radioactive waste processing systems to verify that the current system configuration and operation agreed with the descriptions contained in the Final Safety Analysis Report and the Process Control Program. The following items were reviewed and compared with regulatory requirements:

- Radioactive material/waste processing and shipping procedures

- The status of radioactive waste processing systems that were not operational and/or abandoned in place

- Changes made to the radioactive waste processing systems since the previous inspection in January 2000

- Waste stream sampling procedures and radiochemical sample analysis results for each of the licensee-identified radioactive waste streams for the year 2000

- Scaling factors and calculations used to account for difficult-to-measure radionuclides

- Conduct of the licensee's quality assurance program per 10 CFR Part 20, Appendix G

- Documentation for five nonexcepted package shipments (00-32, 00-40, 00-42, 00-45, and 00-55) that demonstrated shipment packaging, surveying, labeling, marking, placarding, vehicle checks, emergency instructions, waste disposal manifest, shipping papers, and licensee verification of shipment readiness

- Applicable transport cask Certificates of Compliance and cask loading and closure procedures

- Transferee licenses and State Department of Transportation permits

- Training program for personnel responsible for the conduct of radioactive waste processing and radioactive material/waste shipment preparation activities

- Quality Assurance Audit AU200-001, "Process Control Program"

- 2000 Radiological Effluent Release Report, Section 4, "Solid Radwaste"

- Nine problem evaluation requests related to the radioactive material/waste processing and shipping program (200-0111, 200-0262, 200-0271, 200-0285, 200-0297, 200-0754, 200-0946, 200-1758, and 200-1826)

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspector reviewed corrective action program records for high radiation areas, high-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 four quarters were reviewed. Selected examples were investigated to determine whether they were within the dose projections of the governing radiation work permits.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspector reviewed radiological effluent release program corrective action records, licensee event reports, and annual effluent release reports documented during the past four quarters to determine if any doses resulting from effluent releases exceeded the performance indicator thresholds.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems

The inspectors documented a corrective action problem in Section 1R20 of this report.

4OA3 Event Followup (71153)

.1 (Closed) Licensee Event Report 50-397/2001-002: Noncompliance with Technical Specification 3.3.1.1 due to a scram discharge volume level instrument isolation valve misconfiguration. This issue is addressed in Section 4OA7(1), Licensee Identified Violations.

4OA5 Other

.1 (Closed) URI 50-397/00-13-01: Performance Indicator Verification - evaluation of accurate notification

NRC inspectors had been unable to verify the licensee evaluation of the accuracy of offsite notifications that were included in the Drill and Exercise Performance performance indicator. The licensee had not proceduralized the definition of notification accuracy. Licensee evaluators had applied multiple evaluation standards in evaluating the accuracy of notifications.

Frequently Asked Question 242, posted January 10, 2001, clarified and provided additional guidance for the evaluation of offsite notification accuracy as applied to performance indicator measurement. During in-office inspection, the inspectors verified that the licensee revised Attachment 7.6, “Drill/Exercise Performance,” to

Procedure 1.10.10, "Regulatory Assessment Performance Indicator Reporting," to incorporate the guidance provided in Frequently Asked Question 242.

4OA6 Management Meetings

Exit Meeting Summary

An emergency preparedness inspector presented the inspection results to Mr. T. Messersmith, Corporate Emergency Preparedness, Safety and Health Officer, and other members of licensee management in a telephone conversation on May 7, 2001. A health physics inspector presented the radioactive material processing and transportation inspection results to Mr. R. Webring, Vice President, Operations Support, and other members of licensee management on May 10, 2001. A senior reactor inspector presented the inservice inspection results to Mr. R. Webring, Vice President, Operations, Mr. G. Smith, Vice President-Generation, and other members of licensee management on May 24, 2001. A senior health physics inspector presented the access control and performance indicator verification inspection results to Mr. G. Smith, Vice President-Generation, and other members of licensee management on May 25, 2001. The senior resident inspector presented the remaining inspection results to Mr. J. Parrish, Chief Executive Officer, and other members of licensee management on June 27, 2001. The licensee acknowledged the inspection results during each meeting. Following the meetings, the inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. Some proprietary information was identified by the licensee but no mention of any proprietary details were made in this report.

4OA7 Licensee Identified Violations The following findings of very low significance were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG 1600, for being dispositioned as noncited violations (NCVs).

	NCV Tracking Number	Requirement Licensee Failed to Meet
(1)	NCV 50-397/01003-06	Technical Specification 3.3.1 requires, in part, that all scram discharge volume hi level instrument switches be operable. Contrary to the Technical Specification, one instrument was inoperable from February 5 to May 2, 2002, because of a human performance error. A technician failed to properly reposition the instrument isolation valve during a previous surveillance. The licensee placed this issue into the corrective action program as Problem Evaluation Request 201-0732 and reported the event to the NRC in Licensee Event Report 2001-002. The inspectors determined that the issue had very low safety significance (Green) because one inoperable

instrument did not result in a loss of the safety function to scram the plant on high scram discharge volume.

(2) NCV 50-397/01003-07

10 CFR 20.1501(a) states, in part, that each licensee shall perform surveys that are reasonable to evaluate radiation levels and potential radiological hazards. On June 29, 2000, the licensee identified that a localized area of residual heat removal Room B had not been surveyed and posted as a high radiation area following plant shutdown 3 days earlier. General radiation levels were as high as 120 millirem per hour. The failure to perform a radiological survey is a 10 CFR 20.1501(a) violation. This event is described in the licensee's corrective action program, reference Problem Evaluation Request 201-1089. This is being treated as a noncited violation.

The safety significance of this finding was determined to be very low (Green) by the Occupational Radiation Safety Significance Determination Process because there was no overexposure or substantial potential for an overexposure, and ability to assess dose was not compromised.

ATTACHMENT

Supplemental Information

PARTIAL LIST OF PERSONS CONTACTED

Licensee

J. Parrish, Chief Executive Officer
M. Allen, Energy Northwest Executive Board Member
D. Atkinson, Manager, Engineering
D. Bunch, Energy Northwest Executive Board Member
T. Coates, Energy Northwest Executive Board Member
J. Cockburn, Energy Northwest Executive Board Member
D. Coleman, Manager, Performance Assessment and Regulatory Programs
D. Feldman, Manager, Operations
V. Harris, Acting Maintenance Manager
T. Messersmith, Corporate Emergency Preparedness, Safety and Health Officer
R. Sherman, Licensing Engineer
W. Oxenford, Plant General Manager
J. Peters, Manager, Radiological Services
R. Sherman, Acting Manager, Licensing
G. Smith, Vice President - Generation
R. Webring, Vice President - Operation Support
S. Wood, Manager, Chemistry

ITEMS OPENED AND CLOSED

Items Opened, Closed, and Discussed During this Inspection

Opened

50-397/01003-04	URI	(Section 1R22)	Questionable diesel generator testing

Opened and Closed During this Inspection

50-397/01003-01	NCV	(Section 1R20)	Inadequate corrective actions to address alternate decay heat removal method
50-397/01003-02	NCV	(Section 1R22)	Inadequate secondary containment isolation valve testing
50-397/01003-03	NCV	(Section 1R22)	Inadequate drywell leak-rate instrument testing
50-397/01003-05	NCV	(Section 2OS1)	Failure to post a High-High Radiation Area
50-397/01003-06	NCV	(Section 4OA7)	Scram discharge volume level switch inoperable for three months
50-397/01003-07	NCV	(Section 4OA7)	Failure to perform a radiation survey

Previous Items Closed

50-397/00-13-01	URI	(Section 4OA5)	Performance Indicator Verification - evaluation of accurate notification
50-397/2001-002	LER	(Section 4OA7)	Scram discharge volume level switch inoperable for 3 months

Previous Items Discussed

None

PARTIAL LIST OF DOCUMENTS REVIEWED

The following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of the inspection and to support any findings:

PROCEDURES

NUMBER	DESCRIPTION	REVISION
PDI-UT-1	PDI-UT-1 Columbia Generating Station Site Specific Addenda; GE-Performance Demonstration Initiative Generic Procedure for the Ultrasonic Examination of Ferritic Piping Welds	0/B
PDI-UT-2	PDI-UT-2 Columbia Generating Station Site Specific Addenda; GE-Performance Demonstration Initiative Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds	0/B
PDI-UT-3	PDI-UT-3 Columbia Generating Station Site Specific Addenda; GE-Performance Demonstration Initiative Generic Procedure for the Ultrasonic Through Wall Sizing in Pipe Welds	0/B
PDI-UT-5	PDI-UT-1 Columbia Generating Station Site Specific Addenda, GE-Performance Demonstration Initiative Generic Procedure for the Straight Beam Ultrasonic Examination of Bolts and Studs	0/E

PROBLEM EVALUATION REQUESTS

200-0118	200-0560	201-0342	299-2115	299-2338

LIQUID PENETRANT EXAMINATION DATA SHEET, REPORT NUMBERS

5-01-10-1
5-01-10-2

VISUAL EXAMINATION DATA SHEETS; REPORT NUMBERS

2HPV-004	2SWV-014	2SWV-018	2MSV-129	2MSV-133
2RIV-007	2SWV-015	2MSV-127	2MSV-131	2MSV-134
2RIV-008	2SWV-016	2MSV-128	2MSV-132	2MSV-135
2SWV-013	2SWV-017			

VISUAL EXAMINATION DATA SHEET/COMPONENT SUPPORTS; REPORT NUMBERS

2HV-159	2HV-169	2HV-179	2HV-189	2HV-198
2HV-160	2HV-171	2HV-180	2HV-190	2HV-199
2HV-161	2HV-172	2HV-181	2HV-191	2HV-201
2HV-162	2HV-173	2HV-182	2HV-192	2HV-202
2HV-163	2HV-174	2HV-183	2HV-193	2HV-204
2HV-164	2HV-175	2HV-184	2HV-194	2HV-205
2HV-165	2HV-176	2HV-185	2HV-195	2HV-206
2HV-166	2HV-177	2HV-186	2HV-196	2HV-207
2HV-167	2HV-178	2HV-188	2HV-197	2HV-208
2HV-168				

MAGNETIC PARTICLE EXAMINATION DATA SHEET; REPORT NUMBERS

2RIM-09	2LPM-008	2RHM-040	2RHM-046	2RHM-052
2RIM-10	2LPM-009	2RHM-041	2RHM-047	2RHM-053
2RIM-11	2HPM-010	2RHM-042	2RHM-048	2RHM-054
2RIM-12	2RHM-036	2RHM-043	2RHM-049	2RHM-055
2LPM-005	2RHM-037	2RHM-044	2RHM-050	2RHM-056
2LPM-006	2RHM-038	2RHM-045	2RHM-051	2RHM-057
2LPM-007	2RHM-039			

EXAMINATION SUMMARY SHEET, FOR ULTRASONIC: DATA REPORT NUMBERS

R15-005	R15-024	R15-032	R15-039	R15-088
R15-006	R15-025	R15-033	R15-040	R15-089
R15-021	R15-026	R15-034	R15-046	R15-090
R15-022	R15-027	R15-035	R15-048	R15-120
R15-023	R15-031	R15-038	R15-087	

WNP-2 INSERVICE INSPECTION (ISI) EVALUATION SHEET: EVALUATION SHEET NUMBERS

1-050
2-021
2-022
2-023
2-024
2-025

MISCELLANEOUS DOCUMENTS

NUMBER/DATE	DESCRIPTION	REVISION
May 21, 2001	Inservice Inspection Program Plan (ISI) Distribution Package 2001-307	N/A
N/A	Inservice Outage Plan for Outage R15, Spring 2001	2
May 15, 2001	Schedule for R15 Inservice Inspections (ISI) and ISI Related Activities	N/A

WNP-2 INSERVICE INSPECTION (ISI) EVALUATION SHEET: EVALUATION SHEET NUMBERS

1-050
2-021
2-022
2-023
2-024
2-025

MISCELLANEOUS DOCUMENTS

NUMBER/DATE	DESCRIPTION	REVISION
AU200-011/ May 16, 2000	Engineering Audit AU200-011, Section 4.0	N/A
May 23, 2001	ASME Section XI Repair and Replacement Package for Welds FW-XI-60 and FW-XI-65 (Valve HPCS- V-102)	N/A