UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-4005

May 9, 2008

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

SUBJECT: COLUMBIA GENERATING STATION - NRC INTEGRATED INSPECTION REPORT 05000397/2008002

Dear Mr. Parrish:

On March 31, 2008, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Columbia Generating Station. The enclosed inspection report documents the inspection results, which were discussed on April 3, 2008, with Mr. Gambhir and other members of your staff.

The inspection 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. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

This report documents two NRC-identified findings and one self-revealing finding of very low safety significance (Green). All of these findings were determined to involve violations of NRC requirements. However, because of the very low safety significance and because these violations were entered into your corrective action program, the NRC is treating these findings as noncited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy. If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011-4005; 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 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) 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 Website at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/ Thomas R. Farnholtz for

Claude E. Johnson, Chief
Project Branch A
Division of Reactor Projects

Docket: 50-397
License: NPF-21

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NRC Inspection Report 05000397/2008002

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SUNSI Review Completed: __TRF__ ADAMS: √ Yes No Initials: __TRF__
√ Publicly Available Non-Publicly Available Sensitive √ Non-Sensitive

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U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-397
License:	NPF-21
Report:	05000397/2008002
Licensee:	Energy Northwest
Facility:	Columbia Generating Station
Location:	Richland, Washington
Dates:	January 1 through March 31, 2008
Inspectors:	Z. Dunham, Senior Resident Inspector, Project Branch A, DRP R. Cohen, Resident Inspector, Project Branch A, DRP
Approved By:	C. E. Johnson, Chief, Project Branch A, Division of Reactor Projects
ATTACHMENT:	SUPPLEMENTAL INFORMATION

SUMMARY OF FINDINGS

IR05000397/2008002; 01/01/2008 - 03/31/2008; Columbia Generating Station; Equipment Alignment, Operability Evaluations, Event Followup.

The report covered a 13-week period of inspection by resident inspectors. Three green noncited violations were identified. The significance of most findings is indicated by their color (Green, White, Yellow, or Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. An NRC identified noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified for Energy Northwest's failure to adequately review a design change to the facility in 1994. The design change installed a bypass line around a residual heat removal pump shutdown cooling suction header isolation valve to bleed pressure from the header. This would be done in the event of leakage past the shutdown cooling suction header reactor coolant system pressure isolation valves. The design change failed to consider the thermal effects of introducing hot reactor coolant system water into the residual heat removal shutdown cooling suction header at a design maximum flowrate of 0.75 gpm. As a result, operation of the bypass line would have resulted in saturation conditions being achieved in the suction header causing flashing across the isolation valves and potentially degrading the valve disk and seating surfaces. This could result in increased reactor coolant system leakage past the isolation valves beyond the capacity of the bypass line. However, in the event of leakage in excess of the ability of the bypass line, Energy Northwest would have received a control room alarm which would have alerted operators to the degraded condition allowing the operators to take prompt action to define the actual leakage and to take actions as needed. Energy Northwest entered the issue into the corrective action program and took immediate action to monitor suction header temperature with the bypass line in service to assure that saturation conditions would not develop.

  This finding was more than minor because it was a design control issue which affected the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, degradation of reactor coolant system pressure isolation valves would occur with the bypass line in service at the maximum allowable design flowrate. This was considered to be a primary system loss of coolant accident initiator contributor (i.e. intersystem loss of coolant accident). The finding was determined to be of very low risk significance (Green) because assuming worst case degradation, the finding would not result in exceeding any Technical Specification limits for reactor coolant system leakage. Additionally, the finding would not have likely affected other mitigation systems resulting in a total loss of their safety function. A crosscutting aspect was not identified due to the performance deficiency occurring in 1994 (Section 1R15).

Cornerstone: Mitigating Systems

- Green. An NRC identified noncited violation of Technical Specification 5.4.1.a was identified for an inadequate emergency support Procedure PPM 5.5.26, “Overriding RHR [Residual Heat Removal] Shutdown Cooling Return Valve Isolations,” Revision 5. The deficient procedure could have resulted in portions of the RHR Trains A and B injection lines inadvertently draining during emergency response to an anticipated transients without scram event. Although Energy Northwest identified the deficiency with Procedure PPM 5.5.26 in June 2006 and had taken action to implement a procedure change, it was not until the inspectors prompted Energy Northwest regarding status of the procedure change and lack of apparent timeliness in issuing a revision to the procedure that Energy Northwest issued the revision. Procedure PPM 5.5.26, Revision 6, was issued on February 6, 2008. As a result of the value added by the inspectors, this finding is considered to be NRC identified.

  The finding was more than minor because it was a procedure quality issue which affected the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, implementation of Procedure PPM 5.5.26 during an anticipated transients without scram condition could have resulted in an inadvertent draining of RHR and subsequent damage to RHR piping and supports during subsequent initiation of injection. The finding was determined to be of very low risk significance (Green) because the finding did not represent an actual loss of safety function, did not represent a loss of system safety function, was not a design or qualification deficiency that resulted in a loss of operability, and was not risk significant due to external initiating events. The deficiency associated with Procedure PPM 5.5.26 would only occur during an anticipated transients without scram which is a non-design bases accident or event. A crosscutting aspect in problem identification and resolution with a corrective action program component [P.1.d] was identified in that the inadequate procedure, although entered into the corrective action program, was not corrected in a timely manner commensurate with safety. This was attributed to a shortage of qualified operations department procedure writers (Section 1R04.2).

Cornerstone: Barrier Integrity

- Green. A self-revealing noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," was identified for the failure to promptly correct a condition adverse to quality to preclude further degradation of a secondary containment sealing surface. As a result of untimely corrective actions to repair a previously identified breach in secondary containment, further degradation of secondary containment occurred due to high winds. Energy Northwest entered the issue into the corrective action program and took action to implement interim corrective actions so that operability of secondary containment was ensured.

  This self-revealing finding was more than minor in accordance with Manual Chapter 0612, Appendix B, because it had an attribute of configuration control and design control that affected the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Using Manual Chapter 0609, Significance Determination Process," Phase 1 work sheet, the inspectors determined that the finding was of very low risk significance (Green) since the finding only represented a degradation of the radiological barrier function provided for the reactor building. Specifically, the finding resulted in significant erosion of the design margin of allowable secondary containment breach size in the reactor building to support standby gas treatment system and secondary containment operability. However, standby gas treatment and secondary containment remained operable during and following the high wind event. A cross-cutting aspect in human performance with a work control component [H.3.a] was identified in that Energy Northwest did not plan and prioritize work activities associated with final repair of the reactor building siding considering the potential for additional high wind events that could further degrade secondary containment. As a result, in February 2008, a high wind event further damaged the reactor building causing additional erosion of the secondary containment design margin for allowable breach size (Section 4OA3.1).

B. Licensee-Identified Violations.

None.

REPORT DETAILS

Summary of Plant Status

Columbia Generating Station operated at 100 percent power for the entire inspection period with the following exceptions. During the course of a routine Reactor Recirculation (RRC) system flow adjustment to lower power on January 24, 2008, a failed master controller caused RRC flow and power to further decrease without operator action. Subsequently, operators took individual control of the RRC flow controllers to stop the flow reduction with a resultant final reactor power of 87 percent. In addition, on March 22, 2008, Reactor Feedwater (RFW) Pump 1B experienced a speed transient due to a malfunctioning governor valve electro-hydraulic operator speed controller associated with Pump RFW-P-1B turbine. Operators stabilized the plant by lowering reactor power with RRC flow to 81 percent power. On March 29, 2008, reactor power was lowered to 60 percent for planned maintenance, including maintenance of Pump RFW-P-1B. Reactor power was being increased to 100 percent power at the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather (71111.01)

.1 Readiness for Impending Adverse Weather Conditions

a. Inspection Scope

The inspectors completed a review of the licensee's readiness for impending adverse weather conditions. The inspectors: (1) reviewed plant procedures, the Updated Safety Analysis Report, and Technical Specifications to ensure that operator actions defined in adverse weather procedures maintained the readiness of essential systems; (2) verified that the licensee implemented plant walkdowns as needed to assure continued operability of safety significant systems; (3) reviewed maintenance records to determine that applicable surveillance requirements were current before the anticipated (severe thunderstorms, tornado warning, high winds) developed; and (4) reviewed plant modifications, procedure revisions, and operator work arounds to determine if recent facility changes challenged plant operation.

- Freezing temperatures; January 23, 2008

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignments (71111.04)

.1 Partial Walkdown

a. Inspection Scope

The inspectors: (1) walked down portions of the risk important systems listed below and reviewed plant procedures and documents to verify that critical portions of the selected systems were correctly aligned; and (2) compared deficiencies identified during the walk down to the licensee's corrective action program to ensure problems were being identified and corrected.

- Reactor Core Isolation Cooling following maintenance outage; February 13, 2008

- Emergency Diesel Generator Division 2; February 26, 2008

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

.2 Complete Walkdown

a. Inspection Scope

The inspectors: (1) reviewed plant procedures, drawings, the Updated Safety Analysis Report, Technical Specifications, and vendor manuals to determine the correct alignment; (2) reviewed outstanding design issues, operator workarounds, and corrective action program documents to determine if open issues affected the functionality of the system; and (3) verified that the licensee was identifying and resolving equipment alignment problems.

- Residual Heat Removal (RHR) System Train C; February 1, 2008

The inspectors completed one sample.

b. Findings

Introduction. An NRC identified Green noncited violation (NCV) of Technical Specification 5.4.1.a was identified for a deficient emergency support procedure. As a result, the RHR Trains A and B injection lines were susceptible to inadvertent draining during emergency response to an anticipated transients without scram (ATWS) event. A crosscutting aspect in problem identification and resolution with a corrective action program component [P.1.d] was also identified.

Description. On February 1, 2008, during a corrective action document review of RHR system problems and deficiencies, the inspectors noted that Energy Northwest documented in Condition Report (CR) 2-06-04445, dated June 13, 2006, a deficiency with Procedure PPM 5.5.26, “Overriding RHR Shutdown Cooling Return Valve Isolations,” Revision 5. Procedure PPM 5.5.26 is used to override the shutdown cooling return of Valves RHR-V-53A and RHR-V-53B isolation signals and to override the RHR heat exchanger bypass Valves RHR-V-48A and RHR-V-48B automatic opening signal to allow use of RHR via shutdown cooling return valves as an outside shroud injection source during emergencies. CR 2-06-04445 provided that Procedure PPM 5.5.26 needed steps to secure drywell sprays, wetwell sprays, and wetwell cooling prior to lining up RHR for injection through Valves RHR-V-53A and RHR-V-53B. The concern being that if drywell sprays, wetwell sprays, or wetwell cooling are in service during implementation of PPM 5.5.26, Step 4.1, “ATWS,” that once Valves RHR-V-48A and RHR-V-48B and the RHR heat exchanger outlet isolation Valves RHR-V-3A and RHR-V-3B are closed per the procedure that the downstream piping drains to the drywell or wetwell (depending on the spray lineup in service). The inspectors were concerned that, with this section of piping drained, the RHR piping would be susceptible to water hammer upon initiation of injection via the shutdown cooling return line potentially damaging pipe and supports during emergency response to an ATWS.

The inspectors noted that a procedure revision to address the deficiency had yet to be issued at the time of the inspection and communicated the observation to Energy Northwest. Energy Northwest subsequently issued Procedure PPM 5.5.26, Revision 6, on February 6, 2008, which had been previously prepared and written prior to occurrence of the inspection and had been in the review process. The inspectors reviewed the revised procedure and did not identify any other deficiencies.

The inspectors noted that operator licensing training provided training on specifically prioritizing actions including consideration of securing drywell and wetwell sprays prior to initiating RHR injection via the normal low pressure core spray injection line during emergency operating procedure implementation. However, operator failure to secure wetwell or drywell sprays prior to normal RHR low pressure core injection would not result in draining portions of the RHR system piping. The inspectors concluded that given the unique nature of Procedure PPM 5.5.26 and its use during ATWS conditions which could result in draining of the injection line, that specific directions to secure wetwell spray, drywell spray, and wetwell cooling prior to injection were required to assure successful implementation of the procedure and reactor pressure vessel injection during an ATWS.

The inspectors concluded that although Energy Northwest originally identified Procedure PPM 5.5.26 as deficient, value was added to the issue since the inspectors prompted Energy Northwest to issue the procedure revision. Energy Northwest concluded that the procedure revision was not timely due to an insufficient number of qualified operation’s department procedure writers.

<u>Analysis</u>. The inadequacy of Procedure PPM 5.5.26 is a performance deficiency. Although Energy Northwest identified the procedure deficiency on June 13, 2006, and documented the issue in CR 2-06-04445, a procedure revision was not implemented until February 6, 2008, after prompting by the inspectors. As a result of the value added by the inspectors, this finding is considered NRC identified. The finding was more than minor because it was a procedure quality issue which affected the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that

respond to initiating events to prevent undesirable consequences. Specifically, implementation of Procedure PPM 5.5.26 during an ATWS condition could have resulted in an inadvertent draining of RHR and subsequent damage to RHR piping and supports during subsequent initiation of injection. The finding was determined to be of very low risk significance (Green) because the finding did not represent an actual loss of safety function, did not represent a loss of system safety function, was not a design or qualification deficiency that resulted in a loss of operability, and was not risk significant due to external initiating events. The deficiency associated with Procedure PPM 5.5.26 would only occur during an ATWS which is a nondesign bases accident or event. A crosscutting aspect in problem identification and resolution with a corrective action program component [P.1.d] was identified in that the inadequate procedure, although entered into the corrective action program, was not corrected in a timely manner commensurate with safety.

Enforcement. Technical Specification 5.4.1.a requires in part that the applicable procedures recommended in Regulatory Guide 1.33, "Quality Assurance Program Requirements," Revision 2, be established. Regulatory Guide 1.33, Section 6.r, requires in part that the licensee establish adequate procedures to combat expected transients that may be applicable (ATWS is an expected operational occurrence). Contrary to this requirement, from June 13, 2006, through February 6, 2008, Procedure PPM 5.5.26 was deficient in that, during an ATWS condition, the RHRs A and B injection piping could be susceptible to drain down and an associated water hammer event. Because this finding was of very low safety significance and was entered into the licensee's corrective action program as Action Request/Condition Report (AR/CR) 177711, this violation is being treated as an NRC identified NCV, consistent with Section VI.A.1 of the Enforcement Policy (NCV 05000397/2008002-01; Failure to Take Adequate Corrective Actions to Address Deficient Emergency Procedure). Energy Northwest took immediate corrective action to revise Procedure PPM 5.5.26.

1R05 Fire Protection (71111.05)

.1 Quarterly Inspection

a. Inspection Scope

The inspectors walked down the plant fire areas listed below to assess the material condition of active and passive fire protection features and their operational lineup and readiness. Where applicable, the inspectors: (1) verified that transient combustibles and hot work activities were controlled in accordance with plant procedures; (2) observed the condition of fire detection devices to verify they remained functional; (3) observed fire suppression systems to verify they remained functional; (4) verified that fire extinguishers and hose stations were provided at their designated locations and that they were in a satisfactory condition; (5) verified that passive fire protection features (electrical raceway barriers, fire doors, fire dampers, steel fire proofing, penetration seals, and oil collection systems) were in a satisfactory material condition; (6) verified that adequate compensatory measures were established for degraded or inoperable fire protection features; and (7) reviewed the corrective action program to determine if the licensee identified and corrected fire protection problems.

- RC-10; Main Control Room; January 10, 2008
- RC-14; Division 1 Switch Gear Room; February 6, 2008
- RC-9; Remote Shutdown Room; February 12, 2008
- R-6; Reactor Core Isolation Cooling Room; February 25, 2008
- R-1; Control Rod Drive Pump Room; February 26, 2008
- RC-4; Division 1 Electrical Switchgear Room; February 27, 2008

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures (71111.06)

.1 Semi-annual Internal Flooding

a. Inspection Scope

The inspectors reviewed the Updated Safety Analysis Report, flooding analysis, and plant procedures to assess susceptibilities involving internal flooding for the plant areas and/or systems listed below. The inspectors verified: (1) that the licensee appropriately identified and entered internal flooding concerns into the corrective action program; (2) verified that operator actions for coping with flooding can reasonably achieve the desired outcomes; and (3) walked down the areas to verify, as applicable, the adequacy of: (a) equipment seals located below the floodline, (b) floor and wall penetration seals, (c) watertight door seals, (d) common drain lines and sumps, (e) sump pumps, level alarms, and control circuits, and (f) temporary or removable flood barriers.

- Floor Drain System; March 5, 2008

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors observed performance tests, reviewed test data from performance tests, or verified the licensee's execution and on-line monitoring of bio-fouling controls for the heat exchangers or heat sinks listed below to verify: (1) test acceptance criteria and results considered differences between testing and design conditions; (2) inspection results were appropriately categorized against acceptable pre-established acceptance criteria; (3) the frequency of testing or inspection was sufficient to detect degradation prior to the loss of the heat removal function; (4) the test results considered instrument uncertainties; and (5) the licensee had established adequate bio-fouling controls.

- Work Order (WO) 01144465; Diesel Cooling Water Heat Exchanger 1A1 Thermal Performance; February 21, 2008

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On January 15, 2008, the inspectors observed testing and training of senior reactor operators and reactor operators to identify deficiencies and discrepancies in the training, to assess operator performance, and to assess the evaluator's critique. The training scenario involved a loss of offsite power (LOOP), an insider operative tripping the main turbine, reactor scram with main steam isolation valve closure, an explosion at SM-4 in the high pressure core spray pump breaker and intruder taking control of plant equipment such that operators were unable to operate equipment required to maintain safety functions.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors reviewed the maintenance activities listed below to: (1) verify the appropriate handling of structure, system, and component (SSC) performance or condition problems; (2) verify the appropriate handling of degraded SSC functional performance; (3) evaluate the role of work control, work practices, and common cause problems; and (4) evaluate the handling of SSC issues reviewed under the requirements of the maintenance rule,10 CFR Part 50 Appendix B, and the Technical Specifications.

- WO 01144439; FDR-V-33 Failed To Open When EDR-V-62 Was Shut; February 5, 2008

- AR/CR 177495; E-BU-C121/441/3X Failed Annual Testing; February 13, 2008

The inspectors completed two samples.

b. Findings

No findings of significance were identified.

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

.1 Risk Assessment and Management of Risk

a. Inspection Scope

The inspectors reviewed the risk assessment activities listed below to verify: (1) performance of risk assessments when required by 10 CFR 50.65 (a)(4) and licensee procedures prior to changes in plant configuration for maintenance activities and plant operations; (2) the accuracy, adequacy, and completeness of the information considered in the risk assessment; (3) that the licensee recognizes, and/or enters as applicable, the appropriate licensee-established risk category according to the risk assessment results and licensee procedures; (4) the licensee implements adequate risk management actions as applicable; and (5) identified and corrected problems related to maintenance risk assessments.

- WO 01128228; Reactor Building Recirculation Air Fan RRA-FN-11 Out of Service for Planned Maintenance; January 24, 2008

- WO 01144760; Permanent Repair Of Reactor Building Siding; February 7, 2008 through March 31, 2008

- WO 01104997; HPCS Diesel Planned 12 Year Overhaul; February 25, through March 1, 2008

- WO 01103988; HPCS Service Water Pump Planned Replacement; February 25, through March 1, 2008

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

.2 Emergent Work Control

a. Inspection Scope

The inspectors reviewed the emergent work control activities listed below to verify: (1) that the licensee performed actions to minimize the probability of initiating events and maintained the functional capability of mitigating systems and barrier integrity systems; (2) that emergent work-related activities such as troubleshooting, work planning/scheduling, establishing plant conditions, aligning equipment, tagging, temporary modifications, and equipment restoration did not place the plant in an unacceptable configuration; and (3) that the licensee identified and assessed risk assessment and emergent work control problems within the corrective action program.

- WO 01140128 and WO 01149262; Startup Transformer and Standby Gas Treatment Out of Service for Planned Maintenance Concurrent with Emergent Secondary Containment Temporary Repairs; February 12, 2008

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors: (1) reviewed plants status documents such as operator shift logs, emergent work documentation, deferred modifications, and standing orders to determine if an operability evaluation was warranted for degraded components; (2) referred to the Updated Safety Analysis Report and design basis documents to review the technical adequacy of licensee operability evaluations; (3) evaluated compensatory measures associated with operability evaluations; (4) determined degraded component impact on any Technical Specifications; (5) used the Significance Determination Process to evaluate the risk significance of degraded or inoperable equipment; and (6) verified that the licensee has identified and implemented appropriate corrective actions associated with degraded components.

- AR/CR 175642; RHR Reactor Pressure Vessel Suction Header Pressure High Alarm; January 3, 2008

- AR/CR 176223; SGT-A Pre-filter High Alarm During Performance of MSP-SGT-B101; January 16, 2008

- AR/CR 177222; NRC Resident Request for RHR-C Documentation; February 6, 2008

- AR/CR 177262; Reactor Building/Turbine Building Siding Damage; February 7, 2008

The inspectors completed four samples.

b. Findings

Shutdown Cooling Suction Header Isolation Valve Leakage

Introduction. An NRC identified NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified for Energy Northwest's failure to conduct an adequate design review of a bypass line installed in 1994 around the RHR Pump 2A shutdown cooling suction header isolation valve. As a result, with the bypass line in service, saturation conditions could be achieved in the shutdown cooling suction header resulting in potential degradation of reactor coolant system pressure isolation valves. No crosscutting aspect was identified due to the performance deficiency occurring in 1994.

Description. On January 3, 2008, main control room staff received annunciator "RHR RPV Suction Shutdown HDR Press High Alarm". This annunciator was initiated by RHR Pressure Switch RHR-PS-18 at 168 psig, indicating that there was increased pressure between Valve RHR-V-8 (20-inch outboard RHR shutdown cooling containment isolation valve), Valve RHR-V-6A (RHR Pump 2A shutdown cooling suction isolation valve), and Valve RHR-V-6B (RHR pump 2B shutdown cooling suction isolation valve). Operators investigated the shutdown cooling suction header locally and noted that a relief valve, Valve RHR-RV-5, appeared to have lifted as designed at 183 psig. This was confirmed by an elevated tail pipe temperature associated with Valve RHR-V-5. Operators referenced alarm response procedure, Procedure PPM 4.601.A4, "601.A4 Annunciator Panel Alarms," Revision 25, Window 1-1, and implemented the applicable procedure steps in response to the alarm condition. In particular, operators bypassed Valve RHR-V-6A via an installed bypass line to successfully lower header pressure to below the annunciator setpoint to clear the alarm. Operators were then able to throttle flow through the bypass line to maintain an approximate pressure of 135 psig in the RHR shutdown cooling suction header.

The RHR shutdown cooling suction header is supplied from RRC Loop A to provide reactor coolant system water to either RHR Pumps 2A or 2B. The pumps then recirculate the water through the respective RHR heat exchanger, where the water is cooled, prior to the water being returned to the reactor coolant system. The RHR shutdown cooling header is primarily designed for low pressure applications during reactor plant shutdown. In addition, the suction header is isolated by two primary containment isolation valves inside the drywell by parallel valves Valve RHR-V-9 (20-inch inboard RHR shutdown cooling containment isolation valve), and Valve RHR-V-209 (0.75-inch check valve).

The bypass line consists of isolation valves, a throttle valve, and a restricting orifice designed to limit bypass flow around Valve RHR-V-6A to less than 0.75 gpm. The bypass line was not part of the original design of the facility and was installed in 1994 per Basic Design Change (BDC) BDC-94-0022-0A to address RHR shutdown cooling suction header isolation valve leakage.

Operators concluded that one of the RHR shutdown cooling header containment isolation valves must be leaking by at a rate of less than 0.75 gpm since the bypass line would limit flow to less than that value per design. As a result, shift management directed that RHR shutdown cooling suction header pressure be monitored locally via an installed pressure gage twice a shift. Energy Northwest documented the condition in AR/CR 175642.

The inspectors reviewed Final Safety Analysis Report design documentation and relevant operating experience and noted that similar RHR shutdown cooling suction header isolation valve leakby had occurred at the Clinton Power Station as documented in NRC IR 05000461/2006002. The inspectors noted that in the example at Clinton Power Station, the licensee was installing a similar bypass line to accommodate RHR shutdown cooling suction header isolation valve leakby and that NRC inspectors at Clinton identified that the design change did not consider the thermal effects of introducing hot reactor coolant system water into the header at the design flow rate of the bypass line.

The inspectors noted that BDC-94-0022-0A also did not evaluate thermal effects from the introduction of hot reactor coolant system water leaking into the RHR shutdown cooling suction header at a design bypassed flowrate of 0.75 gpm. Additionally, the associated 10 CFR 50.59 safety evaluation for BDC-94-0022-0A, Question 4 – "May the proposed activity increase the consequences of a malfunction of equipment important to safety evaluated previously in the Licensing Basis Documents,?" concluded the following:

"The modification will not change the function of shutdown cooling for either RHR Loop 'A' or 'B'. The system design parameters (operating pressures and temperatures) will not change based on the new modification and as such will not increase the consequences of a malfunction of equipment important to safety evaluated previously in the Licensing Basis Documents."

The inspectors noted that although the 10 CFR 50.59 evaluation specifically states that operating pressures and temperatures would not change with the bypass line in service, that the BDC did not evaluate the thermal effects of leakby past the shutdown cooling header suction isolation valves. The inspectors informed Energy Northwest of the similar issue at the Clinton Power Station and the concerns noted with 10 CFR 50.59 associated with BDC-94-0022-0A. Energy Northwest documented the concern in AR/CR 176344. As an immediate action, Energy Northwest commenced monitoring RHR shutdown cooling suction header pipe temperature to assure that temperature conditions did not exceed any design requirements.

Energy Northwest subsequently evaluated the thermal effects on the shutdown cooling suction header as documented in Calculation ME-02-08-03, "RHR Shutdown Cooling Line Temperature Profile Analysis," dated February 7, 2008, and determined that saturation conditions could occur downstream of Valves RHR-V-9, RHR-V-209, or RHR-V-8 resulting in the formation of steam voids in the pipe. Energy Northwest engineering staff concluded that operation of the bypass around Valve RHR-V-6A could result in unacceptable steam voiding and that temperature monitoring of the RHR shutdown cooling suction pipe was required to assure that with the bypass line in service that saturation conditions would not occur. Energy Northwest planned to implement a procedure change to direct temperature monitoring of the shutdown cooling suction header in the event that the bypass line was put in service. Energy Northwest engineering staff also evaluated the affect of steam voiding in the pipe and concluded that steam voiding would not result in any adverse affects on RHR pump operation, the low pressure core injection safety function, or shutdown cooling due to the isometric configuration of the piping and its relation to the elevation of the RHR pump suctions. However, flashing of steam across Valves RHR-V-8, RHR-V-9, or RHR-V-209 could result in damaged valve seating surfaces and affect the valves' reactor coolant system pressure isolation valve safety functions.

Analysis. Energy Northwest's failure to adequately review BDC-94-0022-0A to install a bypass line around Valve RHR-V-6A is a performance deficiency. Specifically, the design change failed to consider the thermal effects of introducing hot reactor coolant system water into the RHR shutdown cooling suction header at a design maximum flowrate of 0.75 gpm. This finding was more than minor because it was a design control issue which affected the initiating events cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, degradation of reactor coolant system pressure isolation valves could occur with the bypass line in service at the maximum allowable design flowrate. This was considered to be a primary system loss of coolant accident (LOCA) initiator contributor (i.e. intersystem LOCA). The finding was determined to be of very low risk significance (Green) because assuming worst case degradation, the finding would not result in exceeding any Technical Specification limits for reactor coolant system leakage. Additionally, the finding would not have likely affected other mitigation systems resulting in a total loss of their safety function. A crosscutting aspect was not identified due to the performance deficiency occurring in 1994.

Enforcement. 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews. Contrary to the above, since 1994, an adequate design review was not performed on BDC-94-0022-0A resulting in the failure to evaluate the thermal effect of reactor coolant system water leakage into the RHR shutdown cooling suction header. This could have resulted in use of the bypass around Valve RHR-V-6A without adequate monitoring of the header temperature resulting in steam voiding and potentially affecting the pressure isolation function of Valves RHR-V-8, RHR-V-9, or RHR-V-209. Immediate action was taken by Energy Northwest to monitor suction header pipe temperature with the bypass in service. Because this finding was of very low safety significance and was entered into the licensee's corrective action program as AR/CR 176344, this violation is being treated as an NCV, consistent with Section VI.A.1 of the Enforcement Policy (NCV 05000397/2008002-02; Failure to Control Design of Residual Heat Removal Shutdown Cooling Suction Header Bypass Line).

Operability of RHR Pump 2C, RHR-P-2C, During Suppression Pool Mixing Mode

Introduction. An unresolved item (URI) was identified pending Energy Northwest's evaluation of Pump RHR-P-2C, in the suppression pool mixing mode of operation.

Description. On February 6, 2008, Energy Northwest took the Division 2 Emergency Core Cooling System Keepfill Pump, RHR-P-3, out of service for planned maintenance. Pump RHR-P-3, a nonsafety related pump, normally operates continuously and discharges to both of the Division 2 RHR Pumps RHR-P-2B and RHR-P-2C to assure that the respective pump injection lines are filled and pressurized. Keeping the injection lines filled and pressurized assures that, when Pumps RHR-P-2B or RHR-P-2C are started, that a water hammer event does not occur potentially damaging safety-related piping and supports. As an alternative to Pump RHR-P-3 keeping the system piping filled, in preparation for taking Pump RHR-P-3 out of service, Energy Northwest started Pump RHR-P-2B in the suppression pool cooling mode of operation and Pump RHR-P-2C in the suppression pool mixing mode of operation to keep their respective injection lines pressurized and filled while Pump RHR-P-3 was out of service. During an accident, the systems automatically re-align to their reactor vessel injection lineup for accident mitigation.

The inspectors noted that while Pumps RHR-P-2B and RHR-P-2C were in the alternate lineups, that Energy Northwest considered both pumps available and operable. The inspectors requested the basis for operability because under design basis accident conditions of a coincident LOOP with a LOCA that a water hammer event may occur. Specifically, upon the LOOP, the main system pumps Pumps RHR-P-2B and RHR-P-2C would stop upon the loss of electrical power. As a result of the pump stopping and the piping configuration with the system return lineups to the suppression pool, the injection piping would immediately drain to the suppression pool depressurizing the system injection piping. Following the LOOP, the emergency diesel generators automatically start and in response to a LOCA the emergency safety-related systems, including Pumps RHR-P-2B and RHR-P-2C, automatically start. Upon pump start, a water hammer event would most likely occur due to the depressurized injection piping. Similar water hammer events associated with RHR systems were also provided in NRC Information Notice 87-10, "Potential for Water Hammer During Restart of Residual Heat Removal Pumps," dated February 11, 1987.

Energy Northwest provided a basis of operability for Pump RHR-P-2B while in the suppression pool cooling mode of operation. To summarize, the basis for operability of both Pump RHR-P-2B (as well as the Division 1 RHR Pump, RHR-P-2A) was predicated on Energy Northwest's response to an apparent violation (AV) (See AV 05000397/1993029-01 and Inspection Reports 05000397/1993029 and 05000397/1995029 for more details). The AV was identified during an NRC review of Licensee Event Report (LER) 93-01, "Inoperable Suppression Pool Cooling Due to Potential Water Hammer," Revisions 0 and 1. The LER provided that water hammer could fail a train of RHR in suppression pool cooling mode due to a LOOP coincident with a LOCA. Subsequent analysis by Energy Northwest, as provided in an enforcement conference documented in "Enforcement Conference and Management Meeting Report 05000397/1993037," and in LER 93-01, Revision 2, determined that Pumps RHR-P-A and RHR-P-2B were operable while operating in the suppression pool cooling mode of operation even with a LOOP coincident with a LOCA. The analysis determined that an accident sequence of a LOOP coincident with a LOCA, occurring while an RHR loop was in the suppression pool cooling (or suppression pool spray) mode of operation was not in the original design basis for the facility. Energy Northwest also provided that General Electric, the plant designer, supported the position, and that limited use of RHR in these operating modes during a LOOP and LOCA resulting in a water hammer event was not sufficiently credible to be included in the design basis accident analysis. Energy Northwest also provided in LER 93-01 that with adherence to limits on duration of RHR operation in the suppression pool cooling (or suppression pool spray) mode of operation that an RHR loop in that lineup would not be declared inoperable. Energy Northwest established the operational limits of Pumps RHR-P-2A and RHR-P-2B to less than an average of 15 hours per week in the suppression pool cooling (or suppression pool spray) mode of operation. Energy Northwest implemented procedure revisions to track and assure that the operational limit was maintained.

Although the inspectors conceded that Energy Northwest established an operability basis for Pump RHR-P-2B, the inspectors noted that Energy Northwest did not have a similar basis for Pump RHR-P-2C while it was operating in the suppression pool mixing mode. Specifically, Energy Northwest's basis for operability for Pump RHR-P-2B was based partly on the reliance of not exceeding a prescribed number of average hours per week. The inspectors questioned the basis for operability of Pump RHR-P-2C if the time operating in the suppression pool mixing mode of operation wasn't similarly tracked and limited. Energy Northwest documented the issue in AR/CR 177222. The inspectors noted that in addition to Pump RHR-P-2C that the low pressure core spray and the high pressure core spray systems were also subject to the same concerns when their respective keep fill pumps were out of service with the pumps operating in a test return lineup mode to the suppression pool. The licensee was still evaluating the condition at the end of the inspection period as provided in Action Request 179672. Consequently, a URI was opened pending an NRC review of Energy Northwest's final evaluation of the acceptability of considering Pump RHR-P-2C operable while in the suppression pool mixing mode of operation (URI 05000397/2008002-03; Operability of RHR-P-2C During the Suppression Pool Mixing Mode of Operation). Energy Northwest issued Operation's Department Night Order 915 to document the pending evaluation and to direct the control room staff to declare RHR-P-2C, low pressure core spray, and high pressure core spray systems inoperable with the systems lined up to return to the suppression pool pending a completion of the analysis.

<u>Analysis</u>. The inspectors will complete an evaluation of any risk and safety significance of the issue during a future inspection of Energy Northwest's analysis.

<u>Enforcement</u>. The inspectors will complete an evaluation of any regulatory aspects of this issue during a future inspection of Energy Northwest's analysis.

1R18 Plant Modifications (71111.18)

.1 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the temporary plant modifications listed below to verify that: (1) applicable 10 CFR 50.59 evaluations and design reviews are adequate for the modification; (2) installation or removal of the modification does not adversely affect system operability and is consistent with modification documents; (3) applicable plant procedures, drawings, and other documents are updated to reflect installation or removal of the modification; (4) post installation and removal testing is adequate and completed satisfactorily; (5) that associated tagouts are appropriately controlled; (6) the cumulative affect of multiple temporary modifications does not adversely affect mitigating systems or radiological boundaries; and (7) the licensee has identified and implemented appropriate corrective actions associated with temporary plant modifications.

- Temporary Modification Request 08-003; Compensatory Measures Installed on the Reactor Building to Stabilize the Siding that Remained on the Building; February 25, 2008

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the post-maintenance test activities listed below and applicable licensing and/or design-basis documents to: (1) determine the safety functions that may have been affected by the maintenance activity; and (2) assess the adequacy of the test procedure to ensure it adequately tested the safety function that may have been affected. The inspectors either witnessed or reviewed test data to verify that: (1) acceptance criteria were met; (2) plant impacts were evaluated; (3) test equipment was calibrated; (4) procedures were followed; (5) jumpers were properly controlled; (6) test data results were complete and accurate; (7) test equipment was removed; (8) the system was properly re-aligned; and (9) deficiencies during testing were documented. The inspectors also reviewed the corrective action program to determine if the licensee identified and corrected problems related to post-maintenance testing.

- WO 01137918; Standby Liquid Control Relief Valve SLC-RV-29A Replacement; January 14, 2008
- WO 01083556; Diesel Starting Air Shuttle Valve DSA-V-79A1 Replacement; January 28, 2008

- WO 01146473; Diesel Cooling Water Pump DCW-P-1C Replacement; January 29, 2008
- WO 01138892; CRD Transponder Card Replacement PMT; February 3, 2008
- WO 01144439; FDR-V-33 Failed To Open; February 5, 2008
- WO 01103988; HPCS-P-2 Pre-Service Test; February 28, 2008

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report, procedure requirements, and Technical Specifications to ensure that the surveillance activities listed below demonstrated that the associated systems tested were capable of performing their intended safety functions. The inspectors either witnessed or reviewed test data to verify that the following significant surveillance test attributes were adequate: (1) preconditioning; (2) evaluation of testing impact on the plant; (3) acceptance criteria; (4) test equipment; (5) procedures; (6) jumper/lifted lead controls; (7) test data; (8) testing frequency and method demonstrated Technical Specification operability; (9) test equipment removal; (10) restoration of plant systems; (11) fulfillment of American Standard of Mechanical Engineers (ASME) Code requirements; (12) updating of performance indicator data; (13) engineering evaluations, root causes, and bases for returning systems to an operable status that did not meet the test acceptance criteria were correct; (14) reference setting data; and (15) annunciators and alarms setpoints. The inspectors also verified that the licensee identified and implemented appropriate corrective actions associated with the surveillance testing.

- WO 01132848; MSP-SGT-B103, “Standby Gas Treatment Filtration System – Unit A Carbon Adsorber Test,” Revision 7; February 14, 2008
- WO 01134689; OSP-CRD/IST-Q701, “Scram Discharge Volume Vent and Drain Valves Operability,” Revision 3; February 19, 2008
- WO 01111020; ESP-B1DG3-Q101; “Quarterly Battery Testing 125 VDC HPCS-B1-DG3,” Revision 10; March 1, 2008
- WO 01104997; “Division 3 Diesel Generator Engine 2/4/6/12 Year Preventive Maintenance;” Revision 1; March 1, 2008

The inspectors completed four samples including: three routine surveillance tests and one inservice test.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

a. Inspection Scope

The inspectors observed an emergency response organization drill which contributed to the drill/exercise performance and emergency response organization performance indicators. For the drills and simulator-based training evolution listed below, the inspectors: (1) observed the training evolution to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities; (2) compared the identified weaknesses and deficiencies against licensee identified findings to determine whether the licensee is properly identifying failures; and (3) determined whether licensee performance is in accordance with the guidance of the NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 5 document acceptance criteria.

- Plant-wide emergency response organization training drill which included a LOOP, an insider operative tripping the main turbine, reactor scram with main steam isolation valve closure, an explosion at SM-4 in the high pressure core spray pump breaker and intruder taking control of plant equipment such that operators were unable to operate equipment required to maintain safety functions; January 31, 2008

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

Cornerstone: Initiating Events

a. Inspection Scope

The inspectors sampled licensee submittals for the performance indicators listed below for the period from first quarter 2007 through the fourth quarter 2007. To verify the accuracy of the data reported during that period, definitions and guidance contained in NEI 99-02, "Regulatory Assessment Indicator Guideline," Revision 5, were used to verify the basis in reporting for each data element. The inspectors reviewed operator logs, LERs, maintenance WO records, and corrective action documents to verify the accuracy of the PIs.

- Unplanned Power Changes per 7,000 Critical Hours; January 10, 2008
- Unplanned Scrams per 7,000 Critical Hours; January 16, 2008
- Unplanned Scrams With Complications; February 12, 2008

The inspectors completed three samples.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As required by Inspection Procedure 71152, "Identification and Resolution of Problems,"and in order to help identify repetitive equipment failures or specific human performance issues for follow-up, the inspectors performed screening of all items entered into the licensee's corrective action program. This was accomplished by reviewing the description of each new corrective action document and periodically attending daily management meetings.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Annual Sample – Pump RFW-P-1B

a. Inspection Scope

On March 25, 2008, the inspectors reviewed Problem Evaluation Request PER 207-0279, dated July, 15, 2007, which documented that reactor feed water Pump 1B governor hunting approximately 200 rpm caused indicated thermal power swings of 97 percent to 103 percent. The inspectors reviewed Energy Northwest's evaluation of the issue considering (1) accurate identification of the problem; (2) evaluation of operability and reportability; (3) consideration of extent of condition and previous occurrences; (4) prioritization of the resolution; (5) assessment of the apparent and contributing causes; and (6) adequacy of corrective actions.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

4OA3 Event Followup (71153)

.1 Reactor Building Siding Damage Due To Wind

a. Inspection Scope

On February 7, 2008, exterior siding panels of the reactor building above the 606 foot elevation were damaged during high winds. For most of the south face and a portion of the west face, the exterior siding panels, sub girts, and resultant exposed insulation was torn free from the inner siding panels and fell to the ground below. Some of these outer siding panels fell onto the roofs of the diesel generator building, the general service building, and the radwaste building. Energy Northwest considered that the exterior siding panels were not part of the secondary containment pressure boundary but instead served to enhance passive secondary containment integrity for the superstructure portion of the reactor building. Consequently, Energy Northwest concluded that secondary containment was operable but degraded. The inspectors observed operator actions and performed walkdowns of the affected areas to evaluate Energy Northwest's response to the wind damage. The inspectors performed an independent assessment of the circumstances of the event and did not identify any significant issues. Energy Northwest took prompt corrective actions to secure the remaining exterior siding panels.

b. Findings

Introduction. A self-revealing NCV of 10 CFR Part 50 Appendix B, Criterion XVI, "Corrective Action," was identified for the failure to promptly correct a condition adverse to quality to preclude further degradation of a secondary containment sealing surface. As a result of untimely corrective actions to repair a previously identified breech in secondary containment, further degradation of secondary containment occurred due to high winds. A crosscutting aspect in human performance with a work control component [H.3.a] was also identified.

Description. On February 7, 2008, the exterior metal siding and insulation for the reactor building superstructure (above elevation 606 feet 10-1/2 inches) was damaged by high winds. For most of the south face and a portion of the west face of the reactor building, the exterior siding panels, sub girts, and resultant exposed insulation was torn free from the inner siding panels and fell to the ground below. Some of these exterior siding panels fell onto the roofs of the diesel generator building, the general service building, and the radwaste building. Subsequently, Energy Northwest performed a visual inspection of the reactor building superstructure interior above the 606 feet 10-1/2 inches elevation and found a degraded seam associated with secondary containment integrity. This degraded seam measured 64 inches in length with a maximum width of 1 inch and was located at the base of the inner siding panels which is part of the secondary containment boundary. The total breach size associated with this degraded seam was approximately 42 square inches. As a result, the remaining margin to maintain secondary containment operable was only 6.4 square inches. Energy Northwest implemented a corrective action to reposition and reattach the loosened inner siding panels to prevent further degradation of secondary containment integrity.

Energy Northwest reviewed the history of any reactor building siding or secondary containment sealing deficiencies and noted that on April 28, 2005, while performing temporary repairs to stabilize loose exterior reactor building siding panels, that a degraded seam was identified in the same location as described above. This condition was documented in CR 2-05-02710 and provided that the degraded seam measured 64 inches in length with a maximum width of 1/16-inch wide for the full length of the seam. Energy Northwest calculated that the total secondary containment breach size associated with this degraded seam was 4 square inches. Energy Northwest performed an extent of condition walkdown of the perimeter of the reactor building 606-foot elevation inner siding to floor seal and discovered multiple locations where the sealant showed signs of aging and also identified that the original installation of this sealant was incomplete. Energy Northwest generated Work Request (WR) 29046348 on April 30, 2005, to repair or reseal any open, depressed, or cracked locations, and also included repair of the degraded seam discussed above. The work associated with WR 29046348, to repair the inner siding to floor seam, was never accomplished. A comparison of the breach size in secondary containment discovered in April 2005 to the breach size discovered in February 2008 showed an increase in area from 4 square inches to 42 square inches.

Energy Northwest's review identified that prior opportunities existed to identify an adverse trend associated with the degrading reactor building siding were as described below and documented in AR/CR 178592:

PER 201-2246; External siding was loose on the north end of the West face of the reactor building near the 606-foot elevation; October 18, 2001.

CR 2-05-01939; Southwest reactor building exterior siding flexing away from the building about 6 inches for a length of about 20 feet; April 1, 2005

CR 2-05-02710; Degraded sealing surface discovered during temporary repair work for loose Southwest reactor building exterior siding; April 28, 2005

CR 2-05-02775; Degraded sealing surface discovered as part of extent-of-condition review of CR 2-05-02710; April 28, 2005

Action Request 9662, Develop a plan to repair reactor building siding to original condition. A repair project was approved by Energy Northwest Plant Health Committee on July 20, 2006. The repair work was scheduled for March 2008; May 14, 2005

CR 2-05-07848; Barrier Impairment Performance Indicator Yellow for three months in a row. This included reactor building south west siding loose in planning status; October 10, 2005

CR 2-06-03377, Radiation waste building to reactor building exterior vertical expansion gap flashing has separated from the reactor building. Concrete anchors were found missing or partially backed out; May 5, 2006.

CR 2-06-04153, Reactor building, exterior metal siding, south side, bottom west edge (about 610-foot elevation), about 8 feet of metal appears detached; June 3, 2006
CR-2-06-06185, Reactor building exterior metal siding is coming loose. There is a gap on the lower end of several panels on the West portion of the South side of the exterior of the reactor building; August 18, 2006

A subsequent root cause evaluation performed by Energy Northwest determined that the primary conditions that allowed the wind damage event to occur were established during initial construction of the reactor building and were attributable to installation deficiencies during construction as well as a change to the original design by the construction vendor that did not meet the specified contractual design specifications.

Analysis. The performance deficiency associated with this finding was Energy Northwest's failure to take prompt corrective actions for a condition adverse to quality. Specifically, a failure to perform a permanent repair of a breach in secondary containment, which occurred in April 2005, in a timely manner led to a further degradation of secondary containment barrier integrity in February 2008. This self-revealing finding was more than minor in accordance with Manual Chapter 0612, Appendix B, because it had an attribute of configuration control and design control that affected the barrier integrity cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Using Manual Chapter 0609, Significance Determination Process," Phase 1 work sheet, the inspectors determined that the finding was of very low risk significance (Green) since the finding only represented a degradation of the radiological barrier function provided for the reactor building. Specifically, the finding resulted in significant erosion of the design margin of allowable secondary containment breach size in the reactor building to support standby gas treatment system and secondary containment operability. However, standby gas treatment and secondary containment remained operable during and following the high wind event. A crosscutting aspect in human performance with a work control component [H.3.a] was identified in that Energy Northwest did not plan and prioritize work activities associated with final repair of the reactor building siding considering environmental conditions that could impact plant structures. Specifically, Energy Northwest did not factor the potential for additional high winds which could have further degraded secondary containment due to reactor building siding damage. As a result, in February 2008, a high wind event further damaged the reactor building causing additional erosion of the secondary containment design margin for allowable breach size.

Enforcement. 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires in part, that conditions adverse to quality, such as deficiencies and nonconformances are promptly identified and corrected. Contrary to this requirement, Energy Northwest did not take appropriate corrective actions to preclude further degradation of secondary containment as documented in AR/CR 177262. Specifically, Energy Northwest failed to identify an adverse trend associated with degrading reactor building siding from April 005 to February 2008 as documented in AR/CR 178592. As a result, further degradation of the available design margin of secondary containment occurred in February 2008. Because this finding is of very low safety significance and was entered into the corrective action program as AR/CR 177262, this violation is being treated as an NCV, consistent with Section VI.A of the Enforcement Policy (NCV 05000397/2008002-04; Failure to Prevent Further Degradation of Secondary Containment). Energy Northwest took immediate corrective actions to regain the loss margin associated with the breach in secondary containment and to mitigate further degradation of secondary containment integrity and the reactor building siding.

.2 Pump RFW-P-1B Speed Excursion and Unplanned Reactor Power Transient

a. Inspection Scope

On March 23, 2008, the inspectors evaluated Energy Northwest's response to a reactor level and reactor power transient during normal plant operations due to an unanticipated Pump RFW-P-1B speed transient. As a consequence, both reactor feed pump low suction pressure alarms were received followed by a reactor pressure vessel high alarm at 40.5 inches. The increased feed water flow to the reactor vessel caused a slight increase in reactor power which reached a peak of 101.73 percent from a value of 100 percent. Operators took immediate action to lower reactor power with RRC flow to stabilize the plant. After the RFW pump low suction pressure alarms cleared and the feedwater system and reactor vessel water level responded normally, the control room supervisor discontinued the reduction in RRC flow with reactor power stabilizing at 81 percent. Energy Northwest concluded that an electro-hydraulic operator associated with the turbine governor for Pump RFW-P-1B had potentially malfunctioned initiating the transient. The inspectors performed an independent assessment of the circumstances of the event by reviewing plant instrumentation and recorders, interviewing plant personnel, and reviewing plant procedures and logs. Additionally the inspectors verified that fuel thermal limits were not exceeded during the event.

b. Findings

No findings of significance were identified.

.3 Control Rod Misposition

a. Inspection Scope

On March 29, 2008, the inspectors evaluated Energy Northwest's response to a control rod misposition during a control rod sequence exchange. Reactor power was at approximately 58.5 percent. Control Rod 34-43 was at position 10 and was to be withdrawn to Position 40. Instead, the operator inadvertently inserted the control rod to Position 4. This resulted in a 0.5 percent decrease in reactor power. Energy Northwest subsequently concluded that no thermal limits were exceeded. The inspectors performed an independent assessment of the circumstances of the event and did not identify any significant issues. Additionally the inspectors verified that fuel thermal limits were not exceeded during the event.

b. Findings

No findings of significance were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On April 3, 2008, the resident inspectors presented the inspection results to Mr. Gambhir, Vice President – Technical Services, and other members of his staff, who acknowledged the findings. The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

On May 9, 2008, the resident inspectors conducted a telephonic exit to discuss inspection results with Mr. Humphreys, Licensing Supervisor, who acknowledged the findings.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee:

D. Brown, Manager, Operations
G. Cullen, Manager, Regulatory Programs
J. Frisco, Acting Plant General Manager
S. Gambhir, Vice President, Technical Services
M. Humphreys, Supervisor, Licensing
W. LaFramboise, General Engineering Manager
T. Lynch, Acting Vice President, Nuclear Generation
J. Parrish, Chief Executive Officer
F. Schill, Technical Specialists V
M. Shymanski, Radiation Services Manager

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000397/2008002-03	URI	Operability of RHR-P-2C During the Suppression Pool Mixing Mode of Operation (Section 1R15)

Opened and Closed

05000397/2008002-01	NCV	Failure to Take Adequate Corrective Actions to Address Deficient Emergency Procedure (Section 1R04.2)
05000397/2008002-02	NCV	Failure to Control Design of Residual Heat Removal Shutdown Cooling Suction Header Bypass Line (Section 1R15)
05000397/2008002-04	NCV	Failure to Preclude a Recurrence and Further Degradation of Secondary Containment (Section 4OA3.1)

Closed

None.

Discussed

05000397/93-29-01	AV	Failure to Take Effective Corrective Action for Residual Heat Removal (RHR) System Water Hammer (Section R15)

LIST OF DOCUMENTS REVIEWED

Section 1R04: Equipment Alignment

Procedures

SOP-DG2-STBY; Emergency Diesel Generator (DIV 2) Standby Lineup, Revision 9

SOP-SW-STBY; Placing Standby Service Water in Standby Status; Revision 1

Drawings and Diagrams

M524-1; Flow Diagram Standby Service Water System, reactor, Radwaste, D.G. Buildings and Yard; Revision 112

M512-1; Flow Diagram Diesel Oil and Miscellaneous Systems Diesel Generator Building; Revision 40

M512-4; Flow Diagram Diesel Oil and Miscellaneous Systems Diesel Generator Building; Revision 8

WOs and WRs

WO 01143579

Corrective Action Documents

AR/CR 176945 AR/CR 177072 AR/CR 178360

Miscellaneous

FSAR; Section 8.3

Section 1R05: Fire Protection

Procedures

Columbia Generating Station Pre-Fire Plans; Revision 3

Columbia Generating Station Final Safety Analysis Report; Appendix F; Amendment 57

National Fire Protection Association NFPA-10; 1984 Revision

Section 1R06: Flood Protection

Procedures

PPM 2.11.5; Floor Drain System; Revision 32

ME 02-02-02; Calculation for Reactor Building Flooding Analysis; Revision 2

Drawing and Diagram

M539; Flow Diagram Floor Drain System Reactor Building; Revision 82

WOs and WRs

WO 01071941

Corrective Action Documents

PER 203-0443

Miscellaneous

PTL A 199418

Section 1R07: Heat Sink Performance

Procedures

PPM 8.4.54; Thermal Performance Monitoring Of DCW-HX-1A1 and DCW-HX-1A2; Revision 8

WO

WO 01144465

Corrective Action Documents

AR/CR 178433

Miscellaneous

Calculation ME-02-92-241; Calculation For DCW Heat Exchanger 1A1 and 1A2 Performance Monitoring Data Evaluation; Revision 01

Calculation ME-02-92-244; Calculation For Minimum Heat Transfer Rate Required For DCW Heat Exchangers A and B; Revision 0

Section 1R12: Maintenance Effectiveness

Drawings and Diagrams

M539; Flow Diagram Floor Drain System Reactor Building; Revision 82

M537; Flow Diagram Equipment Drain System Reactor Building; Revision 71

WOs and WRs

WO 01144439	WR 29063980

Corrective Action Documents

AR/CR 177495	CR 2-07-09285	CR 2-07-00351	CR 2-04-02754	CR 2-07-09141
AR/CR 177146	AR/CR 177495			

Miscellaneous

FSAR Section 9.5.3.2.4

Generic Letter 86-10; Implementation Of Fire Protection Requirements; Dated April 24, 1986

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures

OSP-CONT-M102; Secondary Containment Integrity Verification; Revision 4

Drawings and Diagrams

CVI-Drawing 210-29.6; Reactor Building North Elevation; Revision 4

WOs and WRs

WO 01128228	WO 01104997	WO 01103988	WO 01149362	WO 01141175

Corrective Action Documents

AR/CR 177303	AR/CR 177846	AR/CR 177313	AR/CR 177276	AR/CR 177262
PER 207-0470				

Miscellaneous

Calculation CE-02-088-11; Reactor Building Interior Siding Panel Analysis; February 8, 2008

Drawing CVI 210-29.6; Reactor Containment; Revision 9

0.59SCREEN-08-0028; Modifications To Damaged Reactor Building Siding; February 09, 2008

50.59SCREEN-08-0030; Modifications To Damaged Reactor Building Siding; Dated February 11, 2008

FSAR; Sections 3.3, 3.8 and 6.2

Barrier Impairment Permit 08-0014; Secure Interior Reactor Building Panels, February 07, 2008

Root Cause Evaluation; AR/CR 177262; Reactor Building Wind Damage; February 2008

Section 1R15: Operability Evaluations

Corrective Action Documents

AR/CR 175642 AR/CR 1776223 AR/CR 177222 AR/CR 177262

Miscellaneous

Calculation CE-02-088-11; Reactor Building Interior Siding Panel Analysis; February 8, 2008

Night Order 902

Section 1R18: Plant Modifications

WOs and WRs

WO 01149262 WR 29066269

Miscellaneous

Temporary Modification Request (TMR) 08-003; Compensatory Measures Installed On The Reactor Building To Stabilize The Siding That Remained On The Building; February 25, 2008

Section 1R19: Post Maintenance Testing

WOs and WRs

WO 01103988 WO 01137918 WO 01083556 WO 01146473 WO 01144439
WO 1104997AA WO 01138892

Section 1R22: Surveillance Testing

WOs and WRs

WO 01132848 WO 01134689 WO 01111020 WO 01138892

Section 1EP6: Drill Evaluation

Miscellaneous

2008 Team D Drill Report; January 15, 2008

Section 4OA1: Performance Indicator Verification

Corrective Action Document

AR/CR 179094

Miscellaneous

Operator Logs

NEI 99-02; Regulatory Assessment Performance Indicator Guideline; Revision 2

Section 4OA2: Identification and Resolution of Problems

Corrective Action Documents

AR/CR 179242	AR/CR 179256	AR/CR 179263	AR/CR 179163	AR/CR 179173
AR/CR 179179	AR/CR 179187	AR/CR 179190	AR/CR 179115	AR/CR 179117
AR/CR 179070	AR/CR 178737	AR/CR 179025	AR/CR 179039	AR/CR 179035
AR/CR 179025	AR/CR 178966	AR/CR 178976	AR/CR 178979	AR/CR 178972
AR/CR 178850	AR/CR 178853	AR/CR 178863	AR/CR 178885	AR/CR 178898
AR/CR 178899	AR/CR 178754	AR/CR 178516	AR/CR 178539	AR/CR 178559
AR/CR 178563	AR/CR 178669	AR/CR 178573	AR/CR 178575	AR/CR 178576
AR/CR 178582	AR/CR 178606	AR/CR 178484	AR/CR 178450	AR/CR 178440
AR/CR 178434	AR/CR 178355	AR/CR 178370	AR/CR 178381	AR/CR 178395
AR/CR 178360	AR/CR 178365	AR/CR 178345	AR/CR 178346	AR/CR 178052
AR/CR 178089	AR/CR 177799	AR/CR 177805	AR/CR 178170	AR/CR 177495
AR/CR 177984	AR/CR 177998	AR/CR 178008	AR/CR 178018	AR/CR 177481
AR/CR 177482	AR/CR 176614	AR/CR 176513	AR/CR 176271	AR/CR 176203
AR/CR 176195	AR/CR 176188	AR/CR 176174	AR/CR 176170	AR/CR 176025
AR/CR 176028	AR/CR 175913	AR/CR 175852	AR/CR 175106	AR/CR 177881
AR/CR 177882	AR/CR 177892	AR/CR 177894	AR/CR 177896	AR/CR 177901
AR/CR 177905	AR/CR 177906	AR/CR 177908	AR/CR 177909	PER 207-0279
PER 298-2140	CR 2-04-02101	PER 201-2246		

Section 4OA3: Event Followup

Procedures

Controlled Release Fastener Installation Procedure For Reactor Building; Revision A; Dated October 15, 1979

Procedure PPM 9.3.10; Control Rod Sequence Exchange; Revision 14

Procedure SWP-RXE-01; Reactivity Management Program; Revision 1

Drawings and Diagrams

CVI-Drawing 210-29.6; Reactor Building North Elevation; Revision 4

Drawing S814; Structural Reactor Building Elevations; Revision 5

WOs and WRs

WO 01149262	WO 01034077	WR 29020364	WO 01147686	WO01138358

Corrective Action Documents

CR 2-05-01939	CR 2-05-02710	CR 2-05-02705	CR 2-05-02775	CR 2-05-00277
CR 2-05-02101	CR 2-06-03377	CR 2-04-02101	AR/CR 177309	AR/CR 177313
AR/CR 177303	AR/CR 177846	AR/CR 177308	AR/CR 177276	AR/CR 177262
CR 2-05-01939	AR/CR 178973	AR/CR 178592	AR/CR 178367	AR/CR 177313
PER 207-0470	PER 201-2246	PER 298-2140	PER 201-2246	

Miscellaneous

PTL 225378

PTL 226777

PTL 222367

PTL 245034

Action Request 1224459

Miscellaneous

Calculation CE-02-088-11; Reactor Building Interior Siding Panel Analysis; February 8, 2008

Drawing CVI 210-29.6; Reactor Containment; Revision 9

50.59SCREEN-08-0028; Modifications To Damaged Reactor Building Siding; February 09, 2008

50.59SCREEN-08-0030; Modifications To Damaged Reactor Building Siding; Dated February 11, 2008

FSAR; Sections 3.3, 3.8 and 6.2

Barrier Impairment Permit 08-0014; Secure Interior Reactor Building Panels, February 07, 2008

Root Cause Evaluation; AR177262; Reactor Building Wind Damage; February 2008

NCR2 88-050; Controlled Release Fasteners – Engineering Requirements; Dated February 19, 1988

LIST OF ACRONYMS USED

AR/CR	Action Request/Condition Report
ATWS	Anticipated Transients without Scram
AV	Apparent Violation
BDC	Basic Design Change
CFR	Code of Federal Regulations
CR	Condition Report
IMC	Inspection Manual Chapter
IR	Inspection Report
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
NCV	Noncited Violation
NRC	Nuclear Regulatory Commission
RFW	Reactor Feedwater
RHR	Residual Heat Removal
RRC	Reactor Recirculation
SSC	Structure, System, and Component
URI	Unresolved Item
WO	Work Order
WR	Work Request
USAR	Updated Safety Analysis Report