UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

August 13, 2007

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 05000397/2007003

Dear Mr. Parrish:

On June 29, 2007, 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 July 9, 2007, with Mr. Atkinson 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 three self-revealing findings of very low safety significance (Green). Four of these findings were determined to involve violations of NRC requirements. Additionally, a licensee-identified violation which was determined to be of very low safety significance is listed in this report. However, because of the very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as non-cited violations (NCV(s)) 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/

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

Docket: 50-397
License: NPF-21

Enclosure:
NRC Inspection Report
05000397/2007003

cc w/enclosure:
Sudesh K. Gambhir (Mail Drop PE04)
Vice President, Technical Services
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Dale K. Atkinson (Mail Drop PE08)
Vice President, Nuclear Generation
Energy Northwest
P. O. Box 968
Richland, WA 99352-0968

Cheryl M. Whitcomb (Mail Drop PE03)
Vice President, Organizational
Performance & Staffing/CKO
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Greg Cullen (Mail Drop PE20)
Supervisor, Licensing
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Chairman
Energy Facility Site Evaluation Council
P.O. Box 43172
Olympia, WA 98504-3172

Douglas W. Coleman (Mail Drop PE20)
Manager, Regulatory Programs
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

Chairman
Benton County Board of Commissioners
P.O. Box 190
Prosser, WA 99350-0190

William A. Horin, Esq.
Winston & Strawn
1700 K Street, NW
Washington, DC 20006-3817

Matt Steuerwalt
Executive Policy Division
Office of the Governor
P.O. Box 43113
Olympia, WA 98504-3113

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

Technical Services Branch Chief
FEMA Region X
130 228th Street S.W.
Bothell, WA 98201-9796

Assistant Director
Nuclear Safety and Energy Siting Division
Oregon Department of Energy
625 Marion Street NE
Salem, OR 97301-3742

Special Hazards Program Manager
Washington Emergency Management
Division
127 W. Clark Street
Pasco, WA 99301

Mike Hammond
Department of Homeland Security
FEMA/REP
130 228th Street S.W.
Bothell, WA 98201-9796

Electronic distribution by RIV:
Regional Administrator **(BSM1)**
DRP Director **(ATH)**
DRS Director **(DDC)**
DRS Deputy Director **(RJC1)**
DRS Deputy Director **(WBJ)**
Senior Resident Inspector **(ZKD)**
Branch Chief, DRP/A **(CEJ1)**
Senior Project Engineer, DRP/E **(TRF)**
Team Leader, DRP/TSS **(CJP)**
RITS Coordinator **(MSH3)**

Only inspection reports to the following:
DRS STA **(DAP)**
M. Kunowski, OEDO RIV Coordinator **(MAK3)**
D. Pelton, OEDO RIV Coordinator **(DLP)**
ROPreports
Columbia Site Secretary (CAM5)

SUNSI Review Completed: CEJ ADAMS: ☒ Yes ☐ No Initials: CEJ
√ Publicly Available ☐ Non-Publicly Available ☐ Sensitive ☒ Non-Sensitive

R:_REACTORS_COL\2007\COL2007-03RP-ZKD.wpd

SRI:DRP/A	RI:DRP/A	BC:DRS/EB1	BC:DRS/EB2
ZKDunham	RBCohen	DAPowers	LJSmith
/RA/	E-MJSpivey	/RA/	/RA/
7/30/07	8/1/07	7/23/07	7/23/07

BC:DRS/PSB	BC:DRS/OB	BC:DRP/A	
MPShannon	ATGody	CEJohnson	
/RA/	/RA/	/RA/	
7/23/07	7/23/07	8/13/07	

OFFICIAL RECORD COPY T=Telephone E=E-mail F=Fax

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-397
License:	NPF-21
Report:	05000397/2007003
Licensee:	Energy Northwest
Facility:	Columbia Generating Station
Location:	Richland, Washington
Dates:	March 31, 2007 through June 29, 2007
Inspectors:	Z. Dunham, Senior Resident Inspector, Project Branch A, DRP R. Cohen, Resident Inspector, Project Branch A, DRP R. Lantz, Sr. Emergency Preparedness Inspector, Operations Branch, DRS J. Bashore, Project Engineer, Project Branch B, DRP D. Stearns, Health Physicist, Plant Support Branch, DRS E. Owen, Reactor Inspector, Engineering Branch 1, DRS
Accompanying Personnel:	M. Hayes, Project Engineer, Project Branch A, DRP
Approved By:	C. E. Johnson, Chief, Project Branch A, Division of Reactor Projects
ATTACHMENT:	SUPPLEMENTAL INFORMATION

CONTENTS

SUMMARY OF FINDINGS

IR05000397/2007003; 03/31/2007- 06/29/2007; Columbia Generating Station; Inservice Inspection Activities, Operability Evaluations, Event Followup, Other Activities.

The report covered a 13-week period of inspection by resident and regional inspectors. Five 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. The inspectors identified a Green NCV of 10 CFR 50.55a(g)4 for the failure to meet the requirements of American Society of Mechanical Engineers (ASME) Code Section XI. Specifically, the licensee failed to provide adequate oversight of vendor activities which resulted in an examination on an American Society of Mechanical Engineers Code Class 1 weld being incorrectly accepted. On May 23, 2007, during review of the licensee's radiographic examination of the reactor recirculation line valve replacement welding activities, the inspectors questioned the quality of some of the film results that had been accepted by the licensee. Upon reinspection of the film in question, it was discovered that certain sections of the film were not within the code required density range of 2.0 to 4.0.

  This finding was of more than minor significance because it is associated with the Initiating Events cornerstone attribute of "Equipment Performance" and it affected the associated 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. The inspectors determined that the finding was not appropriate for use with the Significance Determination Process because the finding is not associated with the increase in the likelihood of an initiating event. The acceptable reshoot of the weld determined there were no flaws greater than the acceptance criteria, therefore, there was no increase in the likelihood of an initiating event. While the finding is not suitable for Significance Determination Process evaluation, it has been reviewed by NRC management and is determined to be a finding of very low safety significance (Green). The inspectors also determined that the cause of this finding was related to the work practices component (H.4 c) of the human performance cross-cutting area of NRC Manual Chapter 0305 because the licensee failed to ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety was not supported. (Section 1R08)

Cornerstone: Mitigating Systems

- Green. A self-revealing Green finding was identified for the failure to maintain the design of the station's division 1 and 2 diesel generator output breakers. This resulted in reduced reliability of the breakers to function as designed during surveillance testing. Specifically, the breakers may not reset to a standby configuration to be able to automatically close following opening of the breaker. This would only be applicable during surveillance testing when the breaker was closed while the associated diesel generator was paralleled to an offsite source of power. Energy Northwest implemented immediate corrective actions to declare the breakers inoperable during surveillance testing when the breakers were closed.

  This finding was more than minor because the finding had an attribute of design control which affected the mitigating systems cornerstone objective to ensure the reliability of systems that respond to initiating events to prevent undesirable consequences. Utilizing NRC Manual Chapter 0609 Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, the inspectors determined that the finding was of very low risk significance (Green) because reliability of the breakers was only affected during surveillance testing activities. Additionally, the finding was not associated with a qualification deficiency, did not result in a loss of safety function for a system, did not result in a loss of train for greater than its technical specification allowed outage time, and was not risk significant due to external initiating events. (Section 1R15)

- Green. A self-revealing noncited violation of Technical Specification 5.4.1.a associated with Energy Northwest's failure to establish adequate drawings to support emergent work was identified. As a result, Energy Northwest failed to identify during the work planning process that lifting of a neutral ground on transformer, E-TR-IN/2, to support replacement of the transformer, would result in a loss of neutral ground to a safety-related power panel, E-PP-8AA. This resulted in inoperability of E-PP-8AA.

  The inspectors determined that this self-revealing finding was more than minor because the finding had an attribute of procedure quality which affected the mitigating systems cornerstone objective to ensure the reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to provide adequate engineering drawings showing accurate system configuration impacted the ability to accurately plan and implement work, resulting in inadvertently degrading a power supply for safety-related components. Utilizing NRC Manual Chapter 0609 Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, the inspectors determined that the finding was of very low risk significance (Green) because appropriate compensatory actions were taken for the equipment that was affected by the inoperable power panel. Additionally, the finding was not a qualification issue confirmed not to result in loss of operability, did not represent a loss of safety function for a single train or for the system, and did not screen as potentially risk significant due to external events. (Section 4OA3.1)

- Green. A self-revealing noncited violation Technical Specification 5.4.1.a was identified for a failure to provide adequate procedures for configuration control of pressure switch, RHR-PS-19A, during planned surveillance activities. This contributed to RHR-PS-19A being inadvertently left valved out-of-service following a planned maintenance activity and rendering a pump permissive input to the automatic depressurization system inoperable. Energy Northwest implemented immediate corrective actions to restore RHR-PS-19A to an operable condition and entered the issue into the corrective action program for final evaluation and resolution.

  This self-revealing finding was more than minor because the finding had an attribute of procedure quality which affected the mitigating systems cornerstone objective to ensure the reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to provide adequate procedures for configuration control resulted in the isolation of RHR-PS-19A and a degraded automatic depressurization system function. Utilizing NRC Manual Chapter 0609 Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, the inspectors determined that the finding was of very low risk significance (Green) because although one automatic depressurization system pump permissive pressure switch was inoperable, there was sufficient redundancy in the design of the system to assure that the system remained operable. Additionally, the finding was not a qualification issue confirmed not to result in loss of operability, did not represent a loss of safety function for a single train or for the system, and did not screen as potentially risk significant due to external events. (Section 4OA3.4)

Cornerstone: Emergency Preparedness

- Green. An NRC identified noncited violation of 10 CFR 50.54(q) was identified for Energy Northwest's failure to effectively maintain a standard emergency action level scheme in place when adequate compensatory measures to address out-of-service seismic instruments were not implemented. The seismic monitoring instrumentation provided an input to the station's emergency plan for declaring an unusual event or alert as a result of seismic activity detected on site. The lack of adequate compensatory measures would most likely have delayed accurate classification of an event and therefore adversely affected the ability to promptly implement the site's emergency plan.

  The finding is of more than minor risk significance because it was related to the cornerstone attribute of response organization performance and affected the Emergency Preparedness cornerstone objective because inability to implement an emergency action level diminishes the licensee's capability to protect the health and safety of the public. Utilizing the "Failure to Comply" flowchart of Manual Chapter 0609, Appendix B, "Emergency Preparedness Significance Determination Process," issued March 6, 2003, the finding was determined to be of very low risk significance (Green) because the finding did not represent a loss of function or degradation of a Risk Significant Planning Standard in that other

seismic recording instruments were available which would permit Energy Northwest to make an accurate classification of the event, although the classification would most likely be substantially delayed beyond 15 minutes from the occurrence of an earthquake. The result was consistent with Section 4.4 of MC 0609, Appendix B, which provided examples where a finding would be of very low risk significance for changes to equipment which creates a condition where an existing emergency action level would not be declared for any alert or notification of unusual event. This finding had crosscutting aspects in the area of problem identification and resolution (corrective action program component) in that Energy Northwest failed to take appropriate corrective actions in response to a previously documented condition report which identified concerns with adequate implementation of the emergency plan with the seismic monitors out of service (P.1(d)). This directly contributed to recurring instances of inadequate compensatory measures being utilized. (Section 4OA5.1)

B. Licensee-Identified Violations.

Violations of very low safety significance, which were identified by the licensee have been reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee's corrective action program. These violations and corrective actions are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status:

The inspection period began with Columbia Generating Station at 100 percent power. On April 7, 2007, the station declared an Alert due to a fire in a transformer located in a safe shutdown building. As a result of subsequent repair activities on the affected transformer, Columbia Generating Station conducted a technical specification required shutdown on April 9, 2007, and entered forced outage FO-07-01. The reactor was subsequently brought critical on April 12, 2007, with the forced outage ending on April 14, 2007. The facility achieved full power operations on April 15, 2007. The facility was essentially operated at full power until May 12, 2007, when the facility was shutdown to begin refueling outage RFO-18. The reactor was subsequently brought critical on June 22, 2007, and RFO-18 exited on June 25, 2007, when the main generator was synchronized to the electrical grid. The plant was subsequently shutdown and entered forced outage F-07-02 following an automatic scram on June 28 from approximately 70 percent power. The scram occurred due to lowering reactor vessel water level when a condensate booster pump was inadvertently tripped while switching oil filters resulting in a low suction pressure trip of both reactor feedwater pumps. The plant was shutdown in Mode 3 at the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness For Seasonal Susceptibilities

a. Inspection Scope

The inspectors completed a review of the licensee's readiness of seasonal susceptibilities involving extreme high temperatures. 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) walked down portions of the systems listed below to ensure that adverse weather protection features were sufficient to support operability, including the ability to perform safe shutdown functions; (3) evaluated operator staffing levels to ensure the licensee could maintain the readiness of essential systems required by plant procedures; and (4) reviewed the corrective action program to determine if the licensee identified and corrected problems related to adverse weather conditions.

- Diesel Generators - Division 1 and 2; June 11, 2007

- Circulating Water Pumphouse; June 18, 2007

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.

- Residual Heat Removal Train A; April 5, 2007
- Standby Service Water Train A; May 29, 2007
- Residual Heat Removal Train B; June 11, 2007
- High Pressure Core Spray; June 22, 2007

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Quarterly Inspection

a. Inspection Scope

The inspectors walked down the plant areas listed below to assess the material condition of active and passive fire protection features and their operational lineup and readiness. 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.

- Fire Area RC - 3/1; Radwaste Vertical Cable Chase at the 467', 501', and 525' elevations; April 6, 2007
- Fire Area RC - 6/2; Division 2 Battery Room; April 6, 2007
- Fire Area RC - 14/1 Division 1 Switchgear Room; April 6, 2007
- Fire Area U/R2; Drywell; May 15, 2007
- Fire Area R-8; LPCS Pump Room; June 4, 2007
- Fire Area R-1/1; General Equipment Area; June 5, 2007

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

.2 Annual Inspection

a. Inspection Scope

The inspectors observed an unannounced fire brigade drill on April 26, 2007, to evaluate the readiness of licensee personnel to prevent and fight fires, including the following aspects: (1) use of protective clothing, (2) use of breathing apparatuses, (3) placement and use of fire hoses, (4) entry into the fire area, (5) use of fire fighting equipment, (6) brigade leader command and control, (7) communications between the fire brigade and control room, (8) searches for fire victims and fire propagation, (9) smoke removal, (10) use of pre-fire plans, and (11) adherence to the drill scenario. The licensee simulated a fire in the Turbine Building East Mezzanine 471 foot level.

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

1R08 Inservice Inspection Activities (71111.08(G))

From May 21-24, 2007, the inspectors performed Inspection Procedure 71111.08, “Inservice Inspection Activities.” Inspection Procedure 71111.08 requires a minimum sample size, for boiling water reactors, of one for Sections 02.01. The inspectors fulfilled the requirements of Inspection Procedure 71111.08.

02.01 Inspection Activities Other Than Steam Generator Tube Inspections, PWR Vessel Upper Head Penetration Inspections, Boric Acid Corrosion Control

a. Inspection Scope

The scope of this inspection is to verify that inservice inspection activities are being performed in accordance with American Society of Mechanical Engineers (ASME) Code and other applicable regulatory requirements. The scope of this inspection is to review activities associated with reactor coolant system pressure boundaries and piping connected to the reactor coolant system, reactor vessel internals, and other risk significant piping system boundaries. The inspectors focused the inspection by selecting a majority of components from the reactor recirculation system and reactor vessel internals.

The inspectors reviewed seven volumetric examinations and six surface examinations. From those 13 examinations, the inspectors observed one ultrasonic (UT) examination and five radiographic (RT) examinations. In addition, the inspectors observed the ongoing visual examination of the reactor vessel internals for the Boiling Water Reactor Vessel and Internals Project (BWRVIP). The inspectors verified that each examiner held qualifications to perform each examination.

Partial List of Records

Report No.	Component	Component ID	Method
RT 2327	Report of Radiographic Inspection	XI-4 to XI-8	RT
RT 2001	Report of Radiographic Inspection	XI-5-1	RT
RT 2002	Report of Radiographic Inspection	XI-7-1	RT
RT 2003	Report of Radiographic Inspection	XI-8-1	RT
RT 2006	Report of Radiographic Inspection	XI-6-1	RT
RT 2012	Report of Radiographic Inspection (reshoot)	XI-7-1	RT
3RRP-001	Report of Dye Penetrant (PT) Inspection	XI-1 pipe bore	PT
3RRP-002	Report of Dye Penetrant Inspection	XI-2 pipe bore	PT
5-07-1-2	Report of Dye Penetrant Inspection	XI-2 weld	PT

5-07-1-3	Report of Dye Penetrant Inspection	XI-3 weld	PT
5-07-1-4	Report of Dye Penetrant Inspection	XI-4 weld	PT
5-07-1-8	Report of Dye Penetrant Inspection	XI-8 weld	PT
R18-RPV-01	ISI/NDE Examination Evaluation Sheet	Reactor Pressure Vessel Head Weld	UT

The inspectors reviewed the site procedures to verify that recordable indications were dispositioned in accordance with ASME Code or an NRC approved alternative. During the performance of the inspection activities, one indication was identified as exceeding ASME allowed sizing, however further review of this weld indicated that the flaw was not the result of any active degradation mechanism.

The inspection procedure requires verification of one to three welds that the welding process and welding examinations were performed in accordance with ASME Code Class 1 or 2 requirements or an NRC approved alternative. The inspectors reviewed the Section XI replacement of a valve on the Reactor Recirculation Line and the associated welds for installation.

b. Findings

Introduction. The inspectors identified a Green NCV of 10 CFR 50.55a(g)4 for the failure to meet the requirements of ASME Code Section XI. Specifically, the licensee failed to provide adequate oversight of vendor activities which resulted in an examination on an ASME Code Class 1 weld being incorrectly accepted.

Description. On May 23, 2007, during review of the licensee's radiographic examination of the RRC-V-51B and RRC-V-52B reactor recirculation line valve replacement welding activities, the inspectors questioned the quality of some of the film results that had been accepted by the licensee. Specifically, the density readings of the film were called into question. It was discovered that the licensee, due to the infrequency of these examinations, had no Level III inspectors on staff that were qualified to evaluate radiographic film. As a result, the licensee's basis for acceptance of radiographic film was the acceptance from the vendor and the acceptance from the Authorized Nuclear Inservice Inspector (ANII). There was no other quality review performed by the licensee.

Upon discovery of the inspectors' questioning of the radiographic film quality, the licensee proceeded to write a condition report to address the issue. A hold was placed on the spool piece to not move it into the plant until the issue could be resolved. The licensee was able to utilize a Level III inspector visiting from another site, as well as a licensee worker who had previous certification as a Level III inspector to reinspect the film for quality. A densitometer available on site was calibrated to work with radiographic film. Upon reinspection of the film in question, it was discovered that certain sections of the film were not within the code required density range of 2.0 to 4.0. All of the welds were re-shot and reinspected by the vendor and the licensee in accordance with a subsequent condition report. These radiographs returned acceptable density readings and showed acceptable weld quality.

Film density is a measure of the degree of film darkening. It is a log of the ratio of the intensity of light incident on the film (I_o) to the intensity of light transmitted through the film (I_t). The licensee-qualified radiographic procedure used the code standard density of 2.0 to 4.0 for acceptable viewing with common film viewers. The density of the film in question read 4.5 to 4.67 in some locations.

Based on the results of this violation, the licensee made the determination to postpone the replacement from the current Cycle 18 refueling outage to the Cycle 19 refueling outage in order to accommodate acceptable radiographic examination.

Analysis. The inspectors determined that the failure to identify a radiograph that was unacceptable by procedural and ASME Code standards was a performance deficiency and a violation of NRC requirements. This finding was of more than minor significance because it is associated with the Initiating Events cornerstone attribute of "Equipment Performance" and it affected the associated 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. Absent NRC intervention, the licensee would have relied on an unacceptable radiograph for an unspecified period of service, which may have placed the reactor coolant pressure boundary welds at increased risk for undetected cracking, leakage, or component failure.

The inspectors determined that the finding was not appropriate for use with the Significance Determination Process (SDP) because the finding is not associated with the increase in the likelihood of an initiating event. The acceptable reshoot of the weld determined there were no flaws greater than the acceptance criteria, therefore, there was no increase in the likelihood of an initiating event. While the finding is not suitable for SDP evaluation, it has been reviewed by NRC management and is determined to be a finding of very low safety significance (Green).

The inspectors also determined that the cause of this finding was related to the work practices component (H.4 c) of the human performance cross-cutting area of NRC Manual Chapter 0305 because the licensee failed to ensure supervisory and management oversight of work activities, including contractors, such that nuclear safety was not supported.

Enforcement. 10 CFR 50.55a(g)4 requires in part, that throughout the service life of a boiling or pressurized water reactor facility, components classified as ASME Code Class 1, 2, or 3 must meet requirements of Section XI. Section XI, Article IWA-2231 of the ASME Code states that for radiographic examinations employing either X-ray equipment or radioactive isotopes, the procedure shall be as specified in Section V, Article 2. Section V, Article 2 of the Code states that the maximum density for single or composite radiographic viewing shall be 4.0. Plant Procedures Manual 8.3.395 accepts vendor procedure QC-RT-1 for use at the licensee facility. Section 2.2.2.2 of this procedure also states that the maximum density for single or composite viewing of radiographs shall be 4.0. Contrary to these requirements, the licensee had accepted radiographs with densities ranging from 4.5 to 4.67 for Class 1 welded components. Because of the very low safety significance of this finding and because the issue was entered into the licensee's corrective action program (CR#: 2-07-04949, 2-07-05014, and 2-07-05357), it is being treated as a non-cited violation (NCV), consistent with Section VI.A of the Enforcement Policy (NCV 05000397/2007003-01).

02.05 Identification and Resolution of Problems

a. Inspection Scope

The inspectors reviewed 10 condition reports which dealt with inservice inspection activities and found that the corrective actions were appropriate. From this review, the inspectors concluded that the licensee had an appropriate threshold for entering issues into the corrective action program and has procedures that direct a root cause evaluation when necessary. The licensee also had an effective program for applying industry operating experience.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

a. Inspection Scope

On April 23, 2007, 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 failure of a reactor feedwater level control instrument, and an earthquake that causes; a trip of one circulating water pump, a failure of circulating water pump house level switch, failure of makeup for circulating water basin and spray ponds, and a circulating water piping rupture. Also the crew was evaluated on their response to lowering circulating water basin level which resulted in a loss of circulating water pumps, and a reactor recirculating system leak in the drywell that resulted in the need to spray both the wetwell and the drywell.

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 below listed maintenance activities 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 practices and common cause problems; and (4) evaluate the handling of SSC issues reviewed under the requirements of the maintenance rule, 10 CFR 50 Appendix B, and the Technical Specifications.

- PER 206-0664; Flexible Conduit Jackets Maybe Susceptible to Splitting; April 17, 2007

- Containment Vacuum Breaker System; June 19, 2007

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 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, and (4) the licensee identified and corrected problems related to maintenance risk assessments.

- WO 01128487; Digital Electro-Hydraulic pre-outage design modification work in cable spreading room; April 6, 2007

- WO 01130876; Fuel Pool Cooling Assist Flush; May 9, 2007

- WO 01116722; Securing RHR System A for Reactor Vessel Internal Inspections During RHR System B Maintenance Outage; June 3, 2007

- WO 01131490; Diesel Generator 2 Monthly Operability Test During the Performance of Reactor Protection System Surveillances; June 28, 2007

The inspectors completed four samples.

b. Findings

No findings of significance were identified.

.2 Emergent Work Control

a. Inspection Scope

The inspectors: (1) verified 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) verified 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) reviewed the corrective action program to determine if the licensee identified and corrected Risk Assessment and Emergent Work Control problems.

- WO 01133480; Power Panel E-PP-8AA inoperable and transformer E-TR-IN/2 replacement; April 9, 2007

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.

- PER 207-0135; Circuit Breaker E-CB-DG1/7 Degraded; March 23, 2007

- PER 207-0168; Transformer E-TR-IN/3 Heat Degradation; April 13, 2007
- CR 2-07-03761; Scaffold Proximity to Plant Equipment; April 27, 2007
- CR 2-07-03970; DO-P-1B Failed its Acceptance Flowrate per OSP-DO/IST-Q702; May 3, 2007
- CR 2-07-04219; NRC has questioned the ability of the containment monitoring particulate instrumentation; May 11, 2007

The inspectors completed five samples.

b. Findings

Introduction. A self-revealing Green finding was identified for the failure to maintain the design of the station's division 1 and 2 diesel generator output breakers. This resulted in reduced reliability of the breakers to function as designed during surveillance testing.

Description. On March 22, 2007, the No. 1 diesel generator (DG-1) output breaker, E-CB-DG1/7 (Westinghouse/Cutler-Hammer Model 50-DHP-VR-350), was opened following a scheduled loaded run of DG-1. Following the breaker opening, control room operators noted that a "close permit" light was not lit indicating that the circuit breaker ready to close logic was incomplete. A subsequent inspection of the breaker by Energy Northwest determined that the "push to open" flapper was not reset. Troubleshooting by Energy Northwest determined that the failure of the "push to open" flapper to reset was a result of a design incompatibility between the supervisory function of the circuit to monitor the trip circuit breaker coil and the drop out voltage of the trip circuit breaker coil. Specifically, the diesel generator output breakers incorporate three breaker open indicating lights in a parallel circuit which is then in series with the trip circuit breaker coil. With the breaker open, a small amount of current is supplied through the indicating lights and the trip coil circuit to verify that breaker open position is achieved. With three parallel indicating lights, sufficient current is applied through the trip circuit breaker coil that the voltage drop across the coil may be greater than the drop out voltage of the coil. As a result, following breaker opening when the trip coil is energized, the coil may not reliably dropout and prevent the "push to open" flapper from resetting, preventing the breaker from being in a condition to automatically close if demanded.

Energy Northwest conducted an extent of condition evaluation for the station's other Cutler-Hammer 4160 VAC breakers and determined that DG-2 output breaker, E-CB-DG2/8, was also susceptible to the same failure mechanism, but that the station's remaining safety-related Cutler-Hammer 4160 VAC breakers were not affected. Specifically, the station's other safety-related 4160 VAC breakers utilized two indicating lights in the breaker supervisory circuit. As a result, the applied current flow to the breaker trip coil is lower assuring that the applied voltage is below the coil's drop out voltage and therefore assuring that the breakers would reset to a condition to allow subsequent closure.

As a result, Energy Northwest declared breakers E-CB-DG1/7 and E-CB-DG2/8 operable but degraded because the failure mechanism would only affect the ability of the breaker to reset to a ready to close condition following breaker opening. Therefore, as long as the breaker "close permit" light is verified energized following breaker opening, the breaker "push to open" flapper and trip coil is verified to be reset to assure subsequent breaker closure to support the breakers safety function. However, Energy Northwest determined that a vulnerability existed during surveillance testing when the DG output breaker was closed. Previously, Energy Northwest considered the DG's operable during surveillance testing because the breaker design was to automatically trip open and reclose during a design basis accident coincident with a loss of offsite power to re-energize the station's safety busses. Given the reduced reliability of the breakers to automatically reset to a condition to automatically close, Energy Northwest determined that until a permanent corrective action can be implemented, that the DG's should be declared inoperable during surveillance testing and the appropriate technical specification entered. Energy Northwest documented the issue and operability evaluation in PER 207-0135.

Energy Northwest concluded in PER 207-0135 that the root cause of the failure was an inadequate purchase specification issued when the station's 4160 VAC safety-related circuit breakers were replaced in 2001. The original specification for ordering the replacement breakers did not specify a shop order that would provide the details of the original switchgear and circuit breaker design. Additionally, although the original specification provided trip coil pickup characteristics, it did not provide trip coil drop out characteristics. Energy Northwest determined that this omission led directly to the failure.

Energy Northwest noted during a review of the breaker performance history that E-CB-DG1/7 had failed previously in an identical manner on December 29, 2005, as documented in PER 206-0002. In the evaluation of that breaker failure, Energy Northwest concluded that the breaker had failed as a result of binding and galling between a spacer on the trip coil and the "push to open" flapper causing the failure of the flapper to reset to allow subsequent closure of the breaker. However, the breaker failure was not repeatable by Energy Northwest or the breaker vendor after the breaker had been removed from its cubicle. Therefore, although residual magnetism was considered as a possible failure mechanism for the 2005 breaker failure, it was ruled out since the failure was not repeatable and there was evidence of galling which could result in the "push to open" flapper failing to reset (See IR 05000397/2006002 for a detailed discussion of the previous breaker failure and associated enforcement). Energy Northwest concluded that the December 29, 2005, breaker failure was a precursor to the recent failure. The inspectors considered the previous failure to be a missed opportunity to identify the failure mechanism associated with the trip coil dropout voltage.

<u>Analysis</u>. The performance deficiency associated with this finding was Energy Northwest's failure to adequately control the design of breakers E-CB-DG1/7 and E-CB-DG2/8 when the breakers were previously replaced. Specifically, circuit characteristics that were unique to these breakers was not provided in the original purchase order specifications to the vendor. This resulted in reduced reliability of the

breakers to reset to a standby condition for automatic closure following breaker opening. This self-revealing finding was more than minor because the finding had an attribute of design control which affected the mitigating systems cornerstone objective to ensure the reliability of systems that respond to initiating events to prevent undesirable consequences. Utilizing NRC Manual Chapter 0609 Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, the inspectors determined that the finding was of very low safety significance (Green) because reliability of the breakers was only affected during surveillance testing activities. Additionally, the finding was not associated with a qualification deficiency, did not result in a loss of safety function for a system, did not result in a loss of train for greater than its technical specification allowed outage time, and was not risk significance due to external initiating events. In addition, this finding did not have a crosscutting aspect, because the finding was not reflective of current licensee performance, in that the failure of the "push to open" flapper to reset was a result of a latent design incompatibility between the supervisory function of the circuit to monitor the trip circuit breaker coil and the drop out voltage of the trip circuit breaker coil.

Enforcement. No violations of NRC requirements were identified. Although the design of the diesel generator output breakers was not maintained, the function of the breakers to reset to a condition to automatically close was a design feature of the breakers and not a required safety function (FIN 05000397/2007003-02; Degraded Diesel Generator Output Breakers). Energy Northwest took immediate compensatory measures to declare the affected breakers inoperable during surveillance testing when the breakers are closed. Additionally, Energy Northwest plans to implement a design change to restore reliability of the affected breakers.

1R17 Permanent Plant Modifications (71111.17)

.1 Annual Review

a. Inspection Scope

The inspectors reviewed Energy Northwest's review and approval of the procedures listed below. These procedures tested the installation of the new replacement Control/Pressure Control system prior to reactor startup and during power ascension. These procedures primarily ensured that the installation was performed as designed and that the system is ready for plant startup. There are also sections in the procedures which captures data on dynamic performance to provide baseline information for the new system. The inspectors verified that the procedures ensured that adequate post maintenance testing is performed on the new system prior to plant startup and during power ascension.

- PPM 18.1.14; Modification Test PDC-4934, Turbine Control System Replacement; Revision 2
- PPM 18.1.15; DEH Modification Power Ascension Test; Revision 0

The inspectors completed one sample. .

b. Findings

No findings of significance were identified.

1R19 Postmaintenance Testing (71111.19)

a. Inspection Scope

The inspectors selected the post-maintenance test activities of risk significant systems or components listed below. For each item, the inspectors: (1) reviewed the applicable licensing basis and/or design basis documents to determine the safety functions; (2) evaluated the safety functions that may have been affected by the maintenance activity; and (3) reviewed 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 acceptance criteria were met, plant impacts were evaluated, test equipment was calibrated, procedures were followed, jumpers were properly controlled, the test data results were complete and accurate, the test equipment was removed, the system was properly re-aligned, and 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 01133480; E-TR-IN/2 Replacement; April 12, 2007
- WO 01123413; Standby Service Water Pump 1A Motor Replacement; May 27, 2007
- WO 01107516; MSRV Operability Test; June 8, 2007
- WO 01099243; EDR-V-19 and EDR-V-20 Maintenance; June 17, 2007
- WO 01109427; MSIV Operability Test; June 18, 2007
- WO 01111490; HPCS-P-1 Operability Test; June 22, 2007

The inspectors completed six samples.

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities (71111.20)

.1 Forced Outage FO-07-01

a. Inspection Scope

The inspectors reviewed the following risk significant outage activities for the sample listed below to verify defense in depth commensurate with the outage risk control plan

and compliance with the technical specifications: (1) the outage risk control plan; (2) reactor coolant system instrumentation; (3) electrical power; (4) decay heat removal; (5) heatup and cooldown activities; and (6) licensee identification and implementation of appropriate corrective actions associated with refueling and outage activities.

- Forced Outage FO-07-01; April 9 to April 14, 2007

The inspectors completed one sample.

b. Findings

No findings of significance were identified.

.2 Refueling Outage RFO-18 (May 12, 2007 to June 25, 2007)

a. Inspection Scope

The inspectors reviewed the following risk significant refueling items or outage activities to verify defense in depth commensurate with the outage risk control plan and compliance with technical specifications: (1) the shutdown safety plan; (2) tagging/clearance activities; (3) reactor coolant system instrumentation; (4) electrical power; (5) decay heat removal; (6) spent fuel pool cooling; (6) inventory control; (7) reactivity control; (8) containment closure; (9) reduced inventory conditions; (10) refueling activities; (11) heatup and cooldown activities; and (12) licensee identification and implementation of appropriate corrective actions associated with refueling and outage activities. Additionally, the inspectors conducted closeout inspections of the wetwell and drywell to verify that foreign material was removed from those areas prior to plant startup.

The inspectors completed one sample.

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 below listed surveillance activities demonstrated that the SSC's 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 ASME Code requirements;

(12) updating of performance indicator data; (13) engineering evaluations, root causes, and bases for returning tested SSC's not meeting 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 any needed corrective actions associated with the surveillance testing.

- WO 01128917; ISP-MS-Q926; Isolation Condenser Vacuum B & D - CFT/CC; April 4, 2007
- WO 01129093; OSP-LPCS-M102; LPCS Valve Lineup; April 16, 2007
- WO 01010945; TSP-MSIV-B801; Main Steam Isolation Valve Leak Rate Testing; May 17, 2007
- WO 01108823; TSP-RCS-R802; Division 2 High-Low Pressure Interface Valve Leakage Test; May 26, 2007
- WO 01130543; OSP-RHR/IST-Q703; RHR Loop B Operability Test; May 29, 2007
- WO 01108823; TSP-RCS-R802; Division 2 High-Low Pressure Interface Valve Leakage Test; May 26, 2007
- WO 01109661; OSP-RPV-R801; Reactor Pressure Vessel Leakage Test; June 16, 2007

The inspectors completed six samples including 3 routine surveillance tests; 1 inservice test; and 2 containment isolation valve surveillance tests.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report, plant drawings, procedure requirements, and Technical Specifications to ensure that the temporary modification listed below was properly implemented. The inspectors: (1) verified that the modification did not have an affect on system operability/availability; (2) verified that the installation was consistent with the modification documents; (3) ensured that the post-installation test results were satisfactory and that the impact of the temporary modification on permanently installed SSC's were supported by the test; and (4) verified that appropriate safety evaluations were completed. The inspectors verified that licensee identified and implemented any needed corrective actions associated with temporary modifications.

- TMR 07-04; Temporary ground installation for transformer E-TR-IN/2; April 9, 2007

The inspectors completed one sample.

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 inspector performed in-office reviews of Revision 47 to the Columbia Generating Station Emergency Plan, Revisions 19 to Emergency Plan Implementing Procedure 13.1.1A, "Classifying the Emergency - Technical Bases," and Revision 36 to Procedure 13.1.1, "Classifying the Emergency," submitted in March 2007. These revisions (1) added terminology from NRC Regulatory Issue Summary 2006-12, "Endorsement of Nuclear Energy Institute Guidance 'Enhancements to Emergency Preparedness Programs for Hostile Action' " for "hostile action" and "hostile force" to the emergency classification level definitions, and (2) made administrative revisions to Emergency Action Levels 2.2.S.1, 2.2.G.1, 3.1.U.1, 6.1.G.1, 7.1.A.2, and 9.1.U.1 to clarify the intent and simplify the language in the Emergency Action Levels.

The revisions were compared to their previous revisions, to the criteria of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, to the criteria of NEI 99-01, "Methodology for Development of Emergency Action Levels," Revision 2, and to the standards in 10 CFR 50.47(b) to determine if the revisions were adequately conducted following the requirements of 10 CFR 50.54(q). This review was not documented in a safety evaluation report and did not constitute approval of licensee changes, therefore, these revisions are subject to future inspection.

The inspector completed one sample during the inspection.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety [OS]

2OS1 Access Control To Radiologically Significant Areas (71121.01)

a. Inspection Scope

This area was inspected to assess the licensee's performance in implementing physical and administrative controls for airborne radioactivity areas, radiation areas, high radiation areas, and worker adherence to these controls. The inspector used the requirements in 10 CFR Part 20, "Standards for Protection Against Radiation," the technical specifications, and the licensee's procedures required by technical specifications as criteria for determining compliance. During the inspection, the inspector interviewed the radiation protection manager, radiation protection supervisors, and radiation workers. The inspector performed independent radiation dose rate measurements and reviewed the following items:

- Performance indicator events and associated documentation packages reported by the licensee in the Occupational Radiation Safety Cornerstone
- Controls (surveys, posting, and barricades) of three radiation, high radiation, or airborne radioactivity areas
- Radiation work permits, procedures, engineering controls, and air sampler locations
- Conformity of electronic personal dosimeter alarm set points with survey indications and plant policy; workers' knowledge of required actions when their electronic personal dosimeter noticeably malfunctions or alarms
- Physical and programmatic controls for highly activated or contaminated materials (non-fuel) stored within spent fuel and other storage pools
- Self-assessments, audits, licensee event reports, and special reports related to the access control program since the last inspection
- Corrective action documents related to access controls
- Licensee actions in cases of repetitive deficiencies or significant individual deficiencies
- Radiation work permit briefings and worker instructions
- Adequacy of radiological controls, such as required surveys, radiation protection job coverage, and contamination control during job performance
- Dosimetry placement in high radiation work areas with significant dose rate gradients
- Changes in licensee procedural controls of high dose rate - high radiation areas and very high radiation areas

- Controls for special areas that have the potential to become very high radiation areas during certain plant operations

- Posting and locking of entrances to all accessible high dose rate - high radiation areas and very high radiation areas

- Radiation worker and radiation protection technician performance with respect to radiation protection work requirements

The inspector completed 19 of the required 21 samples.

b. Findings

No findings of significance were identified.

2OS2 ALARA Planning and Controls (71121.02)

a. Inspection Scope

The inspector assessed licensee performance with respect to maintaining individual and collective radiation exposures as low as is reasonably achievable (ALARA). The inspector used the requirements in 10 CFR Part 20 and the licensee's procedures required by technical specifications as criteria for determining compliance. The inspector interviewed licensee personnel and reviewed:

- Site-specific ALARA procedures

- ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements

- Interfaces between operations, radiation protection, maintenance, maintenance planning, scheduling and engineering groups

- Integration of ALARA requirements into work procedure and radiation work permit documents

- Shielding requests and dose/benefit analyses

- Assumptions and basis for the current annual collective exposure estimate, the methodology for estimating work activity exposures, the intended dose outcome, and the accuracy of dose rate and man-hour estimates

- Exposure tracking system

- Use of engineering controls to achieve dose reductions and dose reduction benefits afforded by shielding

- Workers' use of the low dose waiting areas

- Source-term control strategy or justifications for not pursuing such exposure reduction initiatives

- Radiation worker and radiation protection technician performance during work activities in radiation areas, airborne radioactivity areas, or high radiation areas

- Corrective action documents related to the ALARA program and follow-up activities, such as initial problem identification, characterization, and tracking

The inspector completed 5 of the required 15 samples and 7 of the optional samples.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

a. Inspection Scope

Cornerstone: Mitigating Systems

- Safety System Functional Failures

The inspectors reviewed data from Energy Northwest's corrective program, maintenance rule data, system health reports, operator logs, and licensee event reports from the first quarter 2006 through first quarter 2007. The inspectors compared the reviewed data against Energy Northwest safety system functional failure performance indicator submitted data to determine if any discrepancies existed. The inspectors utilized NEI 99-02, ""Regulatory Assessment Performance Indicator Guidelines," Revision 4, to verify the accuracy of the data submittal.

The inspectors completed one sample.

Cornerstone: Barrier Integrity

- Reactor Coolant Specific Activity

- Reactor Coolant System Identified Leakage Rate

The inspectors compared the data from surveillance procedures, operator logs, equipment out-of-service logs, and corrective action logs for the last six quarters. The inspectors verified that Energy Northwest calculated performance indicators in accordance with NEI 99-02.

The inspectors completed two samples.

Cornerstone: Occupational Radiation Safety

- Occupational Exposure Control Effectiveness

The inspector reviewed licensee documents from March 1 through May 1, 2007. The review included corrective action documentation that identified occurrences in locked high radiation areas (as defined in the licensee's technical specifications), very high radiation areas (as defined in 10 CFR 20.1003), and unplanned personnel exposures (as defined in Nuclear Engineering Institute 99-02, "Regulatory Assessment Indicator Guideline," Revision 4). Additional records reviewed included ALARA records and whole body counts of selected individual exposures. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. In addition, the inspector toured plant areas to verify that high radiation, locked high radiation, and very high radiation areas were properly controlled. Performance indicator definitions and guidance contained in Nuclear Engineering Institute 99-02, Revision 4, were used to verify the basis in reporting for each data element.

The inspector completed the required sample (1) in this cornerstone.

Cornerstone: Public Radiation Safety

- Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences

The inspector reviewed licensee documents from March 1 through May 1, 2007. Licensee records reviewed included corrective action documentation that identified occurrences for liquid or gaseous effluent releases that exceeded performance indicator thresholds and those reported to the NRC. The inspector interviewed licensee personnel that were accountable for collecting and evaluating the performance indicator data. Performance indicator definitions and guidance contained in Nuclear Engineering Institute 99-02, Revision 4, were used to verify the basis in reporting for each data element.

The inspector completed the required sample (1) in this cornerstone.

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:

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.

.2 Semi-Annual Review to Identify Trends

a. Inspection Scope

As required by Inspection Procedure 71152, Identification and Resolution of Problems, the inspectors performed a review of the licensee's corrective action program and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment and corrective maintenance issues but also considered the results of daily inspector corrective action program item screening discussed in Section 4OA2.1. The review also included issues documented outside the normal corrective action program in system health reports, corrective maintenance work orders, component status reports, and maintenance rule assessments. The inspectors' review nominally considered the six-month period of January through June 2007, although some examples expanded beyond those dates when the scope of the trend warranted. Corrective actions associated with identified trends were reviewed for adequacy.

The inspectors completed one sample.

b. Assessment and Observations

No significant findings or observations were identified. During the review the inspectors noted that Energy Northwest identified the potential adverse trends listed below which were documented in the corrective action program:

- PER 207-0067 documented an adverse trend associated with inadequate manager and supervisor oversight associated with plant operational events.

- PER 207-0107 documented an adverse trend associated with component mis-positioning events.

- PER 207-0167 documented an adverse trend in licensed operator requalification evaluation results. Specifically, data indicates an adverse trend in crew and individual performance on licensed operator requalification dynamic simulator evaluations. Overall pass rate for all licensee operator requalification evaluations is below the station specific internal performance indicator threshold.

- PER 207-0229 documented an adverse trend in the occurrences of foreign material exclusion issues related to reactor vessel internal inspections.

No findings or additional adverse trends were identified.

.3 Annual Sample - PER 203-1991; During the Performance of the EDR Sump De-Sludging Operation a Large Number of Paint Chips were Found on the Bag Filters, and on the Subpile

a. Inspection Scope

On May 15, 2007, the inspectors reviewed PER 203-1991, dated May 26, 2003, which documented that a large number of paint chips were found on bag filters and on the

reactor undervessel subpile floor were identified during the performance of EDR sump de-sludging. Energy Northwest documented that the subpile floor was flaking causing paint chips to occur in the EDR sump. Energy Northwest also documented that preventative maintenance on the subpile floor was insufficient causing the floor coating to peel and flake causing FME issue for valves FDR-V-3 and FDR-V-4. 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 and Observations

No findings of significance were identified.

.4 Problem Identification and Resolution - Radiation Safety Review

a. Inspection Scope

The inspector evaluated the effectiveness of the licensee's problem identification and resolution process with respect to the following inspection areas:

- Access Control to Radiologically Significant Areas (Section 2OS1)
- ALARA Planning and Controls (Section 2OS2)

b. Findings and Observations

No findings of significance were identified.

4OA3 Event Follow-up (71153)

.1 Transformer Fire and Initiation of a Technical Specification Required Plant Shutdown

On April 7, 2007, the licensee experienced a small fire in electrical transformer, E-TR-IN/2. As a result, the licensee declared an Alert per EPIP 13.1.1, "Classifying the Emergency," Revision 36. The licensee commenced repair activities to replace the damaged transformer on April 8. As part of these activities, the neutral ground for the transformer was lifted which caused a loss of ground for safety-related power panel E-PP-8AA rendering it inoperable and placing the plant in a technical specification shutdown condition. Energy Northwest shut down the plant on April 9, 2007.

With the plant shutdown, the licensee implemented a temporary modification to restore the electrical ground for E-PP-8AA. Additional corrective actions included replacement of the damaged transformer, returning E-PP-8AA to an operable status, establishing the failure mechanism that caused the fire, and evaluating the extent of condition for other similar electrical transformers. The inspectors conducted an independent assessment of the event which included a review of EPIP 13.1.1 to determined the applicability for declaring an alert and to ensure that Energy Northwest followed applicable event response procedures. This event was entered into Energy Northwest's corrective action program as PER 207-0160.

b. Findings

Introduction. A Green self-revealing noncited violation of TS 5.4.1.a was identified regarding Energy Northwest's failure to provide adequate drawings. As a result, Energy Northwest failed to identify during the work planning process that lifting of a neutral ground on E-TR-IN/2 would result in a loss of ground to E-PP-8AA.

Description. On April 8, 2007, Work Order 01133480 "Replace Inverter E-TR-IN/2," required that transformer E-TR-IN/2 be disconnected and replaced. To support the replacement, the neutral ground wire at E-TR-IN/2 was lifted. This resulted in a loss of ground reference for power panel E-PP-8AA. The ground reference had been supplied through the E-TR-IN/2 transformer neutral wire to E-PP-8AA. Removing the neutral wire in E-TR-IN/2 and separating the neutral from plant ground in E-PP-8AA resulted in a floating neutral condition. This resulted in blown power supply fuses in the Area Radiation Monitors, the momentary dropout of MS-RLY-K14 (MSIV Inboard Isolation Trip Logic), changes in the intensity of Main Steam Isolation Valve (MSIV) indicating lights, voltage fluctuations on MSIV solenoids and a failed surge protection power strip for the Digital Electrical Hydraulic (DEH) printer. As a result, E-PP-8AA was declared inoperable and TS 3.8.7.A was entered. A plant shutdown was initiated when E-PP-8AA could not be restored to an operable condition within the time restraints specified in the TS action statement.

During a subsequent review of WO 01133480, Energy Northwest determined that the maintenance activity to replace E-TR-IN/2 was inadequate. Specifically, Energy Northwest determined that no drawings explicitly identified the neutral grounding scheme for E-PP-8AA and where the ground was specifically tied to the plant electrical system. As a result, Energy Northwest did not realize that the neutral ground for E-PP-8AA was routed through E-TR-IN/2 when the work was planned.

Analysis. The performance deficiency associated with this finding is Energy Northwest's failure to maintain drawings technically accurate which reflect the as-built condition of the plant. This resulted in the inoperability of E-PP-8AA during emergent maintenance on E-TR-IN/2. The inspectors determined that this self-revealing finding was more than minor because the finding had an attribute of procedure quality which affected the mitigating systems cornerstone objective to ensure the reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to provide adequate engineering drawings showing accurate system configuration impacted the ability to accurately plan and implement work, resulting in inadvertently degrading a power supply for safety-related components. Utilizing NRC Manual Chapter 0609 Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, the inspectors determined that the finding was of very low risk significance (Green) because although the power panel was declared inoperable, it was determined to remain functional to support adequate power for supplied safety loads. Additionally, the finding was not a qualification issue confirmed not to result in loss of operability, did not represent a loss of safety function for a single train or for the system, and did not screen as potentially risk significant due to external events. In addition, this finding did not have a crosscutting aspect, because the finding was not reflective of current licensee performance, in that a latent issue associated with inadequate drawings did not explicitly identify a neutral ground for E-PP-8AA was routed through E-TR-IN/2 when the work was planned.

Enforcement. Technical Specification 5.4.1.a requires, in part, that written procedures shall be established and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," Revision 2, Appendix A, Section 9.a. requires that maintenance that can affect the performance of safety-related equipment should be properly pre-planned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances. Contrary to this requirement, on April 8, 2007, Energy Northwest utilized a plant drawing which was not appropriate to the circumstances to support planned maintenance on E-TR-IN/2 as specified in WO 01133480. This resulted in the subsequent inoperability of a vital equipment power supply, E-PP-8AA. Because this finding was of very low safety significance and was entered into the licensee's corrective action program as PER 207-0163, this violation is being treated as an NCV, consistent with Section VI.A.1 of the Enforcement Policy (NCV 05000397/2007003-03; Failure to Provide Adequate Drawings to Support Emergent Work). Energy Northwest implemented immediate corrective actions to restore E-PP-8AA to an operable condition and to revise the affected drawings.

.2 (Closed) Licensee Event Report (LER) 05000397/2006-001-00; Reactor Trip due to Digital Electro-Hydraulic (DEH) Control System Failure

On October 31, 2006, the reactor tripped from 100 percent reactor power as a result of a reactor protection system actuation due to a failed DEH circuit card. Energy Northwest determined that the circuit card failure occurred as a result of a random failure which could not have been predicted. Previously, Energy Northwest had identified the DEH system as susceptible to single point vulnerabilities as a result of two prior reactor scrams associated with DEH circuit card failures and plans to replace the DEH system with a more robust design during an upcoming refueling outage scheduled to start May 12, 2007. Although three reactor scrams have occurred since 2004 as a result of circuit card failures in the DEH system, Energy Northwest determined that each failure was associated with a different type of DEH card and that the specific failure for each card varied and that there was no discernable trend in individual component failures. No violations of regulatory requirements or findings were identified during the inspectors review of the event. Energy Northwest documented the event in PER 206-0596. This LER is closed.

.3 (Closed) LER 05000397/2006-002-00; Shutdown Cooling Isolation due to Inadequate Procedure Step

On November 3, 2006, shutdown cooling was inadvertently isolated while in Mode 4. The isolation occurred while transferring reactor protection system B to its alternate power supply. During the transfer, electrical disconnect RHR-DISC-V/9 was opened per procedure PPM 2.7.6, "Reactor Protection System ", Revision 23, to prevent a loss of shutdown cooling. Use of RHR-DISC-V/9 during the transfer resulted in an unintended containment isolation signal to RHR-V-9. Energy Northwest concluded that the cause of the event was due to an inadequate procedure step derived from inaccurate technical information provided in procedure SOP-RHR-SDC-BYPASS, "Bypassing RHR Shutdown Cooling Isolation Logic in Mode 4 and 5," Revision 2. See IR 05000397/2007002 for a discussion of a self-revealing NCV associated with this issue. The inspectors completed

a review of the LER and did not identify any other violations of regulatory requirements or findings. Energy Northwest documented the issue in PER 206-0602. This LER is closed.

.4 <u>(Closed) LER 05000397/2007-001-00; Automatic Depressurization System (ADS) Logic Signal Instrument Inadvertently Disabled</u>

a. <u>Inspection Scope</u>

On March 21, 2007, pressure switch, RHR-PS-19A, which provides a permissive to the ADS system for actuation based on adequate pump discharge pressure for residual heat removal pump 2A, RHR-P-2A, was discovered isolated after RHR-P-2A was started for surveillance testing per procedure OSP-LPCS-A702, "Low Pressure Core Spray Keep Fill Integrity Test." This was identified after an alarm, "ADS LPCS/RHR A Pump Running Permissive," did not annunciate as expected. With RHR-PS-19A isolated, Energy Northwest entered TS 3.3.5.1.A for one channel of ECCS instrumentation inoperable. During resolution of the issue, Energy Northwest determined that the probable cause was due to procedure inadequacies. The inspectors reviewed Energy Northwest's assessment of the LER and conducted a review of plant design information and corrective action documents to evaluate the LER.

b. <u>Findings</u>

<u>Introduction</u>. A Green self-revealing NCV of TS 5.4.1.a was identified for a failure to provide adequate procedures for configuration control of RHR-PS-19A during planned maintenance activities. This contributed in RHR-PS-19A being inadvertently left valved out-of-service following a planned maintenance activity.

<u>Description</u>. On March 21, 2007, RHR-PS-19A, which provides a permissive to the ADS system for actuation based on adequate pump discharge pressure, was discovered isolated after RHR-P-2A was started for surveillance testing per Procedure OSP-LPCS-A702, "Low Pressure Core Spray Keep Fill Integrity Test". This was discovered after an expected alarm, "ADS LPCS/RHR A Pump Running Permissive," did not actuate. With RHR-PS-19A isolated, Energy Northwest entered TS 3.3.5.1.A for one channel of emergency core cooling system instrumentation inoperable. TS 3.3.5.1.A was subsequently exited after RHR-PS-19A was unisolated and restored to service.

During resolution of the issue, Energy Northwest could not definitively identify when or how RHR-PS-19A was inadvertently isolated. However, Energy Northwest did identify that on February 17, 2007, that procedure OSP-RHR/IST-Q702, "RHR Loop A Operabilty Test," Revision 24, was conducted, and that on February 22, 2007, ISP-LPCS/RHR-Q901, "RHR A & LPCS Discharge Pressure ADS Trip System A Permissive (By K10A Relay) - CFT/CC," Revision 7, was conducted, both of which manipulated RHR-PS-19A isolation valves. This bounded the time period for which RHR-PS-19A was inoperable from February 17 through March 21, 2007, when the condition was identified. Energy Northwest also concluded that both procedures did not provide adequate direction on how to isolate RHR-PS-19A. Specifically, ISP-LPCS/RHR-Q901 incorrectly specified the wrong component name for an RHR-PS-19A isolation valve and Procedure OSP-RHR/IST-Q702 did not provide any component identification or configuration control

guidance during manipulations at RHR-PS-19A. Energy Northwest identified this as a contributing cause.

Analysis. The performance deficiency associated with this finding is Energy Northwest's failure to establish adequate procedures for the configuration control for RHR-PS-19A during planned maintenance activities per Procedures ISP-LPCS/RHR-Q901 and OSP-RHR/IST-Q702. This contributed to the inoperability of RHR-PS-19A. This self-revealing finding was more than minor because the finding had an attribute of procedure quality which affected the mitigating systems cornerstone objective to ensure the reliability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to provide adequate procedural configuration control resulted in the isolation of RHR-PS-19A and degraded ADS function. Utilizing NRC Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, the inspectors determined that the finding was of very low risk significance (Green) because although one ADS permissive pressure switch was inoperable, there was sufficient redundancy in the design of the ADS system to assure that ADS remained operable. Additionally, the finding was not a qualification issue confirmed not to result in loss of operability, did not represent a loss of safety function for a single train or for the system, and did not screen as potentially risk significant due to external events. In addition, this finding did not have a crosscutting aspect, because the finding was not reflective of current licensee performance, in that a latent issue with a failure to provide adequate procedures for configuration control resulted in the isolation of RHR-PS-19A and a degraded automatic depressurization system function.

Enforcement. Technical Specification 5.4.1.a requires, in part, that written procedures shall be established and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," Revision 2, Appendix A, Section 8.b.(2)(j) requires that specific surveillance tests should be written for emergency core cooling system tests. Contrary to this requirement, since 1997, procedure ISP-LPCS/RHR-Q901 incorrectly specified the wrong component name for RHR-PS-19A isolation valve and procedure OSP-RHR/IST-Q702 did not provide any component identification or configuration control guidance during manipulations at RHR-PS-19A, subsequently rendering RHR-PS-19A inoperable. Because this finding was of very low safety significance and was entered into the licensee's corrective action program as PER 207-0132, this violation is being treated as an NCV, consistent with Section VI.A.1 of the Enforcement Policy (NCV 05000397/2007003-04; Failure to Provide Adequate Procedural Configuration Control for Pressure Switch, RHR-PS-19A, During Planned Surveillance Activities). The inspectors noted that Energy Northwest reported the inoperable pressure switch in accordance with 10 CFR 50.73(a)(2)(i)(B) for a failure to follow technical specifications. Specifically, with RHR-PS-19A inoperable for greater than 8 days, TS 3.3.5.1.G required that associated supported features be declared inoperable immediately. Contrary to TS 3.3.5.1, associated features supported by RHR-PS-19A were not declared inoperable with RHR-PS-19A inoperable for greater than 8 days. However, this violation is subsumed by the NCV noted above associated with inadequate procedures (NCV 05000397/2007003-04) which contributed to this violation of TS. Energy Northwest implemented immediate corrective actions to restore RHR-PS-19A to an operable condition and planned to revise the affected procedures.

4OA5 Other Activities

.1 (Closed) URI 05000397/2007006-01; Potential Inadequate Compensatory Actions Related to an Out-of-Service Seismic Monitoring Instrument

a. Inspection Scope

The inspectors completed a review of Energy Northwest's ability to effectively implement the Emergency Plan and declare an Unusual Event in a timely manner associated with potential inadequate compensatory actions related to an out-of-service seismic monitoring instrument.

b. Findings

Introduction. The inspectors identified a Green NCV associated with Energy Northwest's failure to establish adequate compensatory measures to ensure the prompt implementation of the Columbia Generating Station Emergency Plan as required by 10 CFR 50.54(q). A problem identification and resolution crosscutting element was identified with the finding.

Description. As discussed in IR 05000397/2007006-01, Section 4OA2.e, an URI was opened pending the NRC's review of Energy Northwest's ability to effectively implement the Emergency Plan during periods of time when seismic monitoring equipment (Seismic Trigger, SEIS-ST-1, and Seismic Switch, SEIS-SS-1) was periodically taken out-of-service to conduct planned maintenance. The inspectors determined that the compensatory measures in place to assure that the Emergency Director would promptly declare the correct emergency classification (Unusual Event or Alert depending on the seismic instrument out-of-service) were inadequate. Specifically, Energy Northwest had revised procedure ABN-Earthquake, "Earthquake," Revision 1, to direct logging on to the US Geological Survey (USGS) website to verify local seismic activity if the seismic monitoring instruments were out-of-service. However, the USGS web site information would be provided in terms of Richter and not local ground acceleration which the station's emergency action levels are based. This was determined to be inadequate to ensure the timely implementation of the emergency plan. Energy Northwest documented these concerns in PER 207-0174 and implemented additional compensatory measures via Night Order 842, dated April 19, 2007. Energy Northwest determined that from February 7, 2006, through April 19, 2007, that the seismic instruments which provided input to the emergency plan had been removed from service periodically without adequate compensatory measures in place. Energy Northwest also concluded that contributing to the issue of inadequate compensatory measures was the station's failure to adequately assess CR 2-05-06739. CR 2-05-06739 documented concerns with procedure ABN-Earthquake and the loss of seismic instrumentation and recommended use of the USGS web site and implementation of a conversion chart from Richter to local ground acceleration. However, although ABN-Earthquake was revised to include reference to the USGS web site, no conversion table was included. Additionally, a later evaluation by Energy Northwest concluded that a conversion from Richter to local ground acceleration was technically not feasible.

Analysis. The failure to provide adequate compensatory actions to support the timely and accurate declaration of a notice of unusual event or alert per the facility emergency plan was a performance deficiency because appropriate emergency classification may not have been made, or would have been significantly delayed. Specifically, risk significant planning standard 10 CFR 50.47(b)(4) requires that a standard scheme of emergency classification and action levels is in use. The finding is of more than minor risk significance because it was related to the cornerstone attribute of response organization performance and affected the Emergency Preparedness cornerstone objective because inability to promptly identify an emergency action level diminishes the licensee's capability to protect the health and safety of the public. Utilizing the "Failure to Comply" flowchart of Manual Chapter 0609, Appendix B, "Emergency Preparedness Significance Determination Process," issued March 6, 2003, the finding was determined to be of very low risk significance (Green) because the finding did not represent a loss of function or degradation of a Risk Significant Planning Standard in that other seismic recording instruments were available which would permit Energy Northwest to make an accurate classification of the event, although the classification would most likely be substantially delayed beyond 15 minutes from the occurrence of an earthquake. The result was consistent with Section 4.4 of MC 0609, Appendix B, which provided examples where a finding would be of very low risk significance for changes to equipment which creates a condition where an existing EAL would not be declared for any alert or notification of unusual event. This finding had crosscutting aspects in the area of problem identification and resolution (corrective action program component) in that Energy Northwest failed to take appropriate corrective actions in response to a previously documented condition report which identified concerns with adequate implementation of the emergency plan with the seismic monitors out-of-service (P.1(d)). This directly contributed to recurring instances of inadequate compensatory measures being utilized.

Enforcement. 10 CFR 50.54(q) requires in part that a licensee follow and maintain in effect emergency plans which meet the standards of 10 CFR 50.47(b). 10 CFR 50.47(b)(4) requires in part that a standard emergency classification and action level scheme be in use. Energy Northwest's Emergency Plan, Revision 47, Table 4-1, Section 9.4, required that an unusual event be declared if the Minimum Seismic Earthquake Exceeded annunciator is received and the control room receives a report from plant personnel who have felt an earthquake or an alert declared if the Operating Basis Earthquake annunciator is received and the control room receives a report from plant personnel who have felt an earthquake on site. Contrary to 10 CFR 50.47(b)(4) and therefore 10 CFR 50.54(q), from the period of February 7, 2006 to April 19, 2007, during numerous maintenance and surveillance activities on the seismic trigger and seismic switch, Energy Northwest failed to effectively maintain a standard emergency action level scheme in place when adequate compensatory measures to address out-of-service seismic instruments were not implemented. This prevented the prompt assessment and classification of an unusual event or alert following an earthquake. Because the failure to maintain in effect a standard emergency action level scheme is of very low safety significance and has been entered into the corrective action program as PER 207-0174, this violation is being treated as a NCV, consistent with Section VI.A of the NRC Enforcement Policy (NCV 05000397/2007003-05, Inadequate Compensatory Actions Related to Out-of-Service Seismic Monitoring Instruments).

.2 Review of Operating Experience Smart Samples (OpESS) FY2007-03, Crane and Heavy Lift Inspection, Supplemental Guidance for IP-7111.20

a. Inspection Scope

Heavy load handling at nuclear power plants may involve risk to stored irradiated fuel and to equipment necessary for a safe shutdown of the reactor. Through the issuance of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July 1980, Generic Letters 80-113, 81-007, and 85-011, and NRC Regulatory Issue Summary 2005-25, the NRC has tried to ensure that the probability of accidents involving dropped heavy loads are kept as low as possible. Because of recent events concerning heavy loads at some nuclear sites, inspectors have completed this supplement inspection to ensure that facilities have implemented and continue to operate in accordance with the guidance listed above.

The inspectors reviewed licensee's procedures and outage plans for crane use inside and outside containment, design basis documents, licensee responses to NUREG-0612, crane maintenance and inspection documents, and interviewed the system engineer.

b. Findings and Observations

No findings of significance were identified. The following inspection items were specifically addressed:

- Determine whether the crane used to lift the reactor vessel head is single-failure-proof:

  The inspectors verified that the crane used to lift the reactor vessel head is single-failure-proof and therefore commitments and review of load drop analysis are not applicable.

- Verify that the licensee has a preventive maintenance program and procedures for inspection and testing in place at the site based on vendor recommendations for their type of crane, and that crane testing and inspection procedures are completed just prior to every outage and just prior to use for reactor disassembly (head lift):

  The inspectors verified that a preventative maintenance program and procedures are in place based on crane testing and inspection procedures that are completed prior to every outage and just prior to use for reactor disassembly.

4OA6 Meetings, Including Exit

Exit Meeting Summary

On April 12, 2007, the inspector presented the emergency plan change inspection results to Mr. M. Reis, Supervisor, Emergency Preparedness. The inspector confirmed that proprietary information was not provided or examined during the inspection.

On May 24, 2007, the inspector presented the occupational radiation safety inspection results to Mr. D. Atkinson and other members of his staff who acknowledged the findings. The inspector(s) confirmed that proprietary information was not provided or examined during the inspection.

On June 27, 2007, the inspectors presented the inspection results to Mr. W. Oxenford and other members of his staff who acknowledged the findings. The inspectors confirmed that proprietary information was returned during the inspection.

On July 9, 2007, the resident inspectors presented the inspection results to Mr. D. Atkinson, 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.

4OA7 Licensee Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy for being dispositioned as an NCV.

- 10 CFR 50, Appendix B, Criterion XVI, “Corrective Actions,” provides in part that measures shall be established to assure that conditions adverse to quality, such as deficiencies are promptly identified and corrected. Contrary to this requirement, on June 1, 2005, Energy Northwest failed to promptly identify a condition adverse to quality. Specifically, as documented in CR 2-05-04561 and CR 2-05-04559, Energy Northwest identified that containment isolation valves, EDR-V-19 and EDR-V-20, failed their as-found local leak rate tests. Corrective actions only consisted of flushing the valves until acceptance criteria was met. Subsequently, in May 2007, both valves again failed their as-found local leak rate tests as documented in PER 207-0213. Energy Northwest disassembled the valves and identified that the valve internals were corroded and that the valve seats were scored. This finding is of very low safety significance because although the administrative limits for local leak rate testing for each of the valves was exceeded, the overall contribution to total leakage through the primary containment was within acceptance criteria.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Energy Northwest

D. Atkinson, Vice President, Nuclear Generation
I. Borland, Manager, Radiation Protection
D. Coleman, Manager, Performance Assessment and Regulatory Programs
L. Cortopassi, Manager, Operations
G. Cullen, Licensing Supervisor, Regulatory Programs
J. Frisco, General Manger, Engineering
D. Gregoire, Supervisor, Licensing
A. Khanpour, General Manager, Engineering
T. Lynch, Plant General Manager
W. Oxenford, Vice President, Technical Services
J. Parrish, Chief Executive Officer
D. Ramey, Engineer, Inservice Inspection
M. Reis, Supervisor, Emergency Preparedness
S. Richter, Engineer, BWR Vessel Inspection Program
F. Schill, Engineer, Licensing
M. Shymanski, Manager, Radiation Protection
R. Torres, Manager, Quality
D. Welch, Engineer, Nondestructive Examination
C. Whitcomb, Vice President, Organizational Performance and Staffing
J. Zimmerschied, Supervisor, Programs

Others:

J. Hair, Authorized Nuclear Inservice Inspector

NRC Personnel

R. Cohen, Resident Inspector
Z. Dunham, Senior Resident Inspector

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Items Opened, Closed, and Discussed During this Inspection

Opened

None

Opened and Closed

50-397/2007003-01	NCV	Failure to Perform Adequate Oversight of Vendor Activities (Section 1R08)

05000397/2007003-02	FIN	Degraded Diesel Generator Output Breakers (Section 1R15)
05000397/2007003-03	NCV	Failure to Provide Adequate Drawings to Support Emergent Work (Section 4OA3.1)
05000397/2007003-04	NCV	Failure to Provide Adequate Procedural Configuration Control for Pressure Switch, RHR-PS-19A, During Planned Surveillance Activities (Section 4OA3.4)
05000397/2007003-05	NCV	Inadequate Compensatory Actions Related to Out-of-Service Seismic Monitoring Instruments (Section 4OA5.1)

Closed

05000397/2007006-01	URI	Potential Inadequate Compensatory Actions Related to an Out-of-Service Seismic Monitoring Instrument (Section 4OA5.1)
05000397/2006001-00	LER	Reactor Trip due to Digital Electro-Hydraulic (DEH) Control System Failure (Section 4OA3.2)
05000397/2006002-00	LER	Shutdown Cooling Isolation due to Inadequate Procedure Step (Section 4OA3.3)
05000397/2007-001-00	LER	Automatic Depressurization System Logic Signal Instrument Inadvertently Disabled (Section 4OA3.4)

Discussed

None.

PARTIAL LIST OF DOCUMENTS REVIEWED

Section 1R01: Adverse Weather Protection

Procedures

SOP-HVAC/DG-LU; Diesel Generator and Cable Cooling HVAC System Breaker Lineup; Revision 0

SOP-HVAC/DG-START; Diesel Generator and Cable Cooling HVAC System Start; Revision 0

SOP-WARM WEATHER-OPS; Warm Weather Operations; Revision 1

SOP-HOT WEATHER-OPS; Hot Weather Operations; Revision 2

SOP-HVAC/CW-LU; Circulating Water Pumphouse and Electrical Buildings HVAC System Breaker Lineup; Revision 0

SOP-HVAC/CW-START; Circulating Water Pumphouse and Electrical Buildings HVAC System Start; Revision 0

Drawings and Diagrams

M551; Flow Diagram HVAC Circ. & M U Water, S.W. & Diesel Generator Bldg.; Revision 56

E503; Auxiliary One Line Diagram; Sheet 9

Section 1R04: Equipment Alignment

Procedures

SOP-RHR-LU; RHR System Valve and Breaker Lineup; Revision 0

SOP-HPCS-LU; HPCS Valve and Breaker Lineup; Revision 0

SOP-HPCS-STBY; Placing HPCS in Standby Status; Revision 2

Drawings and Diagrams

M521-1; Flow Diagram Residual Heat Removal System Loop "A"; Revision 100

M520; Flow Diagram HPCS and LPCS Systems; Revision 94

Corrective Action Documents

2-07-02550	2-07-01862	2-06-09252	2-06-09254
2-06-08965	2-06-08803	2-07-05727	2-07-05712
2-07-05706	2-07-03233	2-07-03067	2-07-03713
2-07-03919	2-07-04387	2-07-04738	2-07-05120
2-07-05235	2-07-05287	2-07-05318	2-07-05338
2-07-05452	2-07-05562	2-07-05656	2-07-05917
2-07-06358	2-07-06339	2-07-06312	2-07-06250
2-07-05873	2-07-05835	2-07-05562	2-07-05452
2-07-05324	2-07-05318	2-07-05180	2-07-06356

Miscellaneous

System Health Report Columbia Generating Station RHR, October to December 2006

Component Classification Evaluation Record C92-0866; HPCS-V-10; Revision 0

Section 1R05: Fire Protection

Procedures

PPM 1.3.10A; Control Of Ignition Source; Revision 13

PPM 1.3.10C; Control Of Transient Combustibles; Revision 11

ISPM-2; Compressed Gases; Revision 1

CGS Pre-Fire Plan

Final Safety Analysis Report; Appendix F

National Fire Protection Association NFPA-10, 1984 Revision

Miscellaneous

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

Columbia Generating Station Pre-Fire Plans; Revision 3

1R08: Inservice Inspection Activities

Procedures

NUMBER	TITLE	REVISION
8.3.395	Plant Procedures Manual: Radiography Procedures	2
ISI-PDI-210-MD	Manual Ultrasonic Procedure for Examination of Nozzle Inner Corner Radius Areas in Accordance with ASME Section XI, Including Appendix VIII	4
PDI-UT-6	Generic Procedure for the Ultrasonic Examination of Reactor Pressure Vessel Welds	F
PDI-UT-7	Generic Procedure for the Manual Ultrasonic Through-Wall and Length Sizing of Ultrasonic Indications in Reactor Pressure Vessel Welds	F
QCI 3-1	Liquid Penetrant Examination Instructions	11
QCI 3-3	Liquid Penetrant Examination - Columbia Generating Station - ISI	7
QCI 3-4	Liquid Penetrant Examination - Section VIII	5
QCI 4-1	Magnetic Particle Examination	9
QCI 4-3	Magnetic Particle Examination - Columbia Generating Station - ISI	8
QCI 4-4	Magnetic Particle Examination Section VIII	7

QCI 6-1	Ultrasonic Examination - Thickness Measurements	6
QCI 7-1	Visual Examination	9
QCI 7-3	Visual Examination - Component Supports	6
QCI 7-4	Visual Examination of Containment	2

Miscellaneous

NUMBER	TITLE	REVISION
	Columbia Generating Station Response to 2007-051, Dissimilar Metal Weld Inspection History	0
2-2120	ASME Section XI Plan for RRC-V-51B and RRC-V-52B Books 1 and 2	0
IR-2007-255	Combined Vessel Head Nozzle Inner-Corner Region and Nozzle-to-Shell Weld Examinations	01/2007

Inspection Reports

NUMBER	TITLE	REVISION
RT 2327	Report of Radiographic Inspection	0
RT 2001	Report of Radiographic Inspection	0
RT 2002	Report of Radiographic Inspection	0
RT 2003	Report of Radiographic Inspection	0
RT 2006	Report of Radiographic Inspection	0
RT 2012	Report of Radiographic Inspection (reshoot)	0
3RRP-001	Report of Dye Penetrant Inspection	0
3RRP-002	Report of Dye Penetrant Inspection	0
5-07-1-2	Report of Dye Penetrant Inspection	0
5-07-1-3	Report of Dye Penetrant Inspection	0
5-07-1-4	Report of Dye Penetrant Inspection	0
5-07-1-8	Report of Dye Penetrant Inspection	0
R18-RPV-01	ISI/NDE Examination Evaluation Sheet	0

Problem Evaluation Requests

207-0096
207-0138
207-0207

Condition Reports

2-05-05605
2-05-05911
2-06-01111
2-06-03829
2-06-04699
2-06-05314
2-06-05951
2-07-04949
2-07-05014
2-07-05357

Section 1R11: Licensed Operator Requalification

Procedures

PPM LR000069 Columbia Generating Station Simulator Training; Revision 7

Section 1R13: Maintenance Risk Assessments and Emergent Work Control

Procedures

PPM 1.3.57; Barrier Impairment; Revision 21

PPM 1.3.14; Risk Assessment and Management for Maintenance/Surveillance Activities; Revision 15

PPM 1.3.68; Work Management Process; Revision 11

PPM 3.1.1; Master Startup Checklist; Revision 35

PPM OSP-ELEC-M702; Diesel Generator 2 - Monthly Operability Test; Revision 29

PPM ISP-MS-Q935; DIV 2 Channel D Isolation On Reactor Level 2 - CFT/CC; Revision 4

PPM ISP-MS-Q939; Group 1 Isolation - Main Steam Line Pressure Low - Channel B and D - CFT/CC; Revision 6

ISP-MSQ919; Division II RHR-B & C (LPCI MODE) Acuation and RCIC Isolation on High Drywell Pressure Channels B and D CFT/CC; Revision 5

Work Orders and Work Requests

WO 01128487	WO 01133637	WO 01116722	WO 01130876
WO 01133480	WO 01131490		

Corrective Action Documents

CR 2-07-03121	CR 2-07-03564	CR 2-07-03186	CR 2-07-03288
PER 207-0163			

Miscellaneous

Barrier Impairment Permit 07-0095

Fire Protection Engineering Evaluation 1.1, Item 40; Re-analysis of NRC Information Notice 88-60; Revision 0

Section 1R15: Operability Evaluations

Procedures

PPM 10.2.53; Seismic Requirements for Scaffolding, Ladders, Man-Lifts, Tool Gang Boxes, Hoists, Metal Storage Cabinets, and Temporary Shielding Racks; Revision 25

Drawings and Diagrams

EWD-47E-007; Electrical Wiring Diagram Standby AC Power System Diesel Generator 2 Breaker E-CB-DG2/8; Revision 17

EWD-46E-049; Electrical wiring diagram AC Electrical Distribution System 4.16 KV and 6.9 KV Switchgear Circuit Breaker Details; Revision 3

Work Orders and Work Requests

WO 01133801

Corrective Action Documents

PER 207-0135	PER 206-0360	CR 2-07-04219	PER 207-0201
PER 207-0168	CR 2-07-03296	CR 2-07-03761	CR 2-07-03970

Miscellaneous

AD-07-0394; Temporarily install two 1 hp fans, one in the vicinity of E-TR-IN/2 and one in the vicinity of E-TR-IN/3, to direct air flow to these transformers; Revision 0

5059SCREEN-07-0072; Temporarily install two 1 hp fans, one in the vicinity of E-TR-IN/2 and one in the vicinity of E-TR-IN/3, to direct air flow to these transformers; Revision 0

Section 1R17: Permanent Plant Modifications

Procedures

PPM 18.1.14; Modification Test PDC-4934, Turbine Control System Replacement; Revision 2

PPM 18.1.15; DEH Modification Power Ascension Test; Revision 0

PPM SOP-DEH-OPS; DEH Pressure Control; Revision 4

PPM SOP-DEH-START, Electro-Hydraulic Fluid System Startup; Revision 2

PPM SOP-MT-GV/OPTIMIZATION, Main Turbine Generator GV Optimization; Revision 1

PPM SOP-MT-TRIP/TEST, Main Turbine Generator Trip Functional Tests; Revision 2

PPM OSP-MS-M701, Bypass Valves Test; Revision 6

PPM OSP-MT-B401, Main Turbine Overspeed Channel CFT; Revision 3

PPM OSP-MS-Q701, Turbine Valve Surveillance; Revision 10

PPM OSP-RPS-Q401, Turb TV Closure CFT; Revision 1

PPM OSP-MT-new, QuadVoter Surveillance Test; Revision 1

PPM ISP-MS-B701, Turbine Bypass System Functional Test; Revision 2

PPM 2.7.1A, 6900 Volt and 4160 Volt AC Electrical Power Distribution System; Revision 16

Drawings and Diagrams

M959; Electro-Hydraullic Fluid System Flow Diagram; Revision 14

E520; Turbine Generator Control; Revision 32

Work Orders and Work Requests

WO 01133217

Corrective Action Documents

CR 2-07-04005	CR 2-07-04222	CR 2-07-04272	CR 2-07-04631
CR 2-07-04735	CR 2-07-05168	CR 2-07-05279	CR 2-07-05807
CR 2-07-05905	CR 2-07-06173	CR 2-07-06326	CR 2-07-06490

Miscellaneous

50.59 Evaluation; 5059-06-0003; Revision 1

FSAR Section 3.5.1.3; Turbine Missiles

FSAR Section 7.7.1.5; Turbine Pressure Regulator and Control System

FSAR Section 10.4.4; Turbine Bypass System

FSAR Section 15.1.3; Pressure Regulator Failure - Open

FSAR Section 15.1.4; Inadvertent Safety/Relief Valve Opening

FSAR Section 15.2.1; Pressure Regulator Failure - Closed

FSAR Section 15.2.2; Generator Load Rejection

FSAR Section 15.2.3; Turbine Trip

FSAR Section 15.2.5; Loss of Condenser Vacuum

FSAR Section 15.2.7; Loss of Feedwater Flow

FSAR Section 15.3.1; Recirculation Pump Trip

FSAR Section 15.5.1; Inadvertent High Pressure Core Spray Startup

Technical Specification 3.3.1.1-1; Reactor Protection System Instrumentation

Technical Specification 3.3.2.2.1; Feedwater and Main Turbine High Water Level Trip Instrumentation

Technical Specification 3.3.4.1.1; End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

Technical Specification 3.7.6; Main Turbine Bypass System

Section 1R19: Post Maintenance Testing

Procedures

OSP-SW/IST-Q701; Standby Service Water Loop A Operability; Revision 16

MSP-MS/IST-R101; MSRV Accessories Operability - IN-SITU; Revision 3

OSP-MS/IST-Q701; MSIV Closure Test - Shutdown; Revision 10

OSP-HPCS/IST-Q701; HPCS System Operability Test; Revision 27

TSP-EDR/X23-C801; LLRT of EDR-V-19 and EDR-V-20; Revision 0

Drawings and Diagrams

Reference Number; Title; Revision or Date of Document

Work Orders and Work Requests

WO 01123413	WO 0113480	WO 01082070
WO 01111490	WO 01109427	WO 01107516

Corrective Action Documents

CR 2-07-06174	CR 2-07-04734	CR 2-07-04702	CR 2-07-05725
CR 2-07-05184	CR 2-07-04943	CR 2-07-04764	

Miscellaneous

Reference Number; Title; Revision or Date of Document

Section 1R20: Refueling and Other Outage Activities

Procedures

PPM 2.14.1; Refueling Bridge Operation; Revision 40

PPM 6.3.2; Fuel Shuffling and/or Offloading and Reloading; Revision 19

SOP-RHR-SDC-BYPASS; Bypassing RHR Shutdown Cooling Isolatoin Logic in Mode 4 and 5; Revision 4

PPM 10.3.21; Reactor Pressure Vessel Disassembly; Revision 22

PPM 3.2.1; Normal Plant Shutdown; Revision 56

PPM 8.3.422; Noblechem Process Application; Revision 6

PPM 6.3.5; Full Core Verification; Revision 10

PPM 6.3.28; Nuclear Component Transfer List Preparation; Revision 14

PPM 6.3.2; Fuel Shuffling and/or Offloading and Reloading; Revision 19

SOP-CAVITY-DRAIN; Reactor Cavity and Dryer Separator Pit Draining; Revision 2

PPM 3.4.4; Natural Circulation; Revision 3

PPM 3.4.1; Minimizing the Potential of Draining the Reactor Vessel or Cavity; Revision 10

ICP-RHR-Q901; RHR SDC Mode High Flow Isolation - CFT/CC; Revision 4

ABN-FPC-ASSIST; Fuel Pool Cooling Assist; Revision 4

PPM 18.1.13; Verification and Validation of RHR/FPC Assist; Revision 0

PPM OSP-RCS-C102; RPV Non-Critical Cooldown Surveillance; Revision 6

PPM 10.4.12; Crane, Hoist, Lifting Device and Rigging Program Control; Revision 23

PPM 10.4.3; Sling Inspection, Maintenance and Testing; Revision 8

PPM 10.3.21; Reactor Pressure Vessel Disassembly; Revision 22

PPM 10.4.5; Reactor (MT-CRA-2) and Turbine Building (MT-CRA-1) Overhead Traveling Crane Inspection, Maintenance and Testing; Revision 20

PPM 10.4.11; Design, Fabrication, Testing and Control of Below the Hook Lifting Devices; Revision 11

<u>Work Orders and Work Requests</u>

WO 01125274	WO 01123565	WO 01124233	WO 01118553

<u>Corrective Action Documents</u>

CR 2-07-04281	CR 2-07-04306	CR 2-07-04256	CR 2-07-04946
CR 2-07-04669	CR 2-07-04803	CR 2-07-04688	CR 2-07-04997
CR 2-07-05005	CR 2-07-05164	CR 2-07-04912	CR 2-07-04439
CR 2-07-00246	CR 2-06-09122	CR 2-06-09119	CR 2-06-08455
CR 2-06-06483	CR 2-06-05477	CR 2-05-09598	CR 2-05-04633
CR 2-05-03647	CR 2-05-02868	CR 2-05-02399	

<u>Miscellaneous</u>

Shutdown Safety Plan - Forced Outage FO-07-01; Revision 0

5059-07-0003; Loose Parts Evaluation For Lost Bush in Reactor Pressure Vessel; Revision 0

R-18 Outage Shutdown Safety Plan; Revisions 0 through 2

LDCS-FSAR-07-002; Clarify FSAR Description of RHR/FPC Assist Mode and Discuss Head Spray Line Flow in RHR/FPC Assist Mode

50559-07-002; RHR/FPC Assist Alternate Cooling Paths; Revision 0

Energy Northwest Foreign Material Loss of Integrity Notification Recovery Plan; Dated May 29, 2007

ASME B30.2-1996; Overhead and Gantry Cranes; Revision 1996

Energy Northwest Letter GO2-01-127; Columbia Generating Station Operating License NPF-21 Request for Amendment Spent Fuel Storage and Handling (Additional Information); Dated September 13, 2001

Section 1R22: Surveillance Testing

Procedures

TSP-MSIV-B801; Main Steam Isolation Valve Leak Rate Testing; Revision 4

PPM 10.24.241; Flow Makeup and Pressure Decay Leak Rate Testing; Revision 0

ISP-MS-Q926; Isolation Condenser Vacuum B & D - CFT/CC; Revision 9

OSP-RHR/IST-Q702; RHR LOOP A Operability Test; Revision 25

TSP-RCS-R802; Division 2 High - Low Pressure Interface Valve Leak Test; Revision 2

OSP-LPCS-M102; LPCS Valve Lineup; Revision 0

OSP-RPV-R801; Reactor Pressure Vessel Leakage Test; Revision 16

Drawings and Diagrams

M529; Flow Diagram Nuclear Boiler - Main Steam System Reactor Building; Revision 95

Work Orders and Work Requests

WO 01128917	WO 01129093	WO 01130543	WO 01108823
WO 01108823			

Corrective Action Documents

CR 2-07-04469	CR 2-07-04467	CR 2-07-05061
PER 207-0163		

Miscellaneous

Primary Containment Leakage Rate Testing Program; December 2004

Work Orders and Work Requests

WO 01133480

Miscellaneous

TMR 07-004; To install a jumper to maintain continuity for the neutral for E-PP-8AA until E-TR-IN/2 can be replaced

Section 4OA1: Performance Indicator Verification

Corrective Action Documents

2-06-02711	2-06-04342	2-06-05000	2-06-05817
2-06-08905			

Miscellaneous

NEI 99-02; Regulatory Assessment Indicator Guideline; Revision 4

LER 2006-001-00

LER 2006-002-00

OSP-INST-H101; Shift And Daily Instrument Checks (Modes 1, 2,3), Revision 58

CSP-I131-W101; Reactor Coolant Isotopic Analysis For I-131 Dose Equivalent, Revision 3

PPM 2.11.3; Equipment Drain System, Sections 5.6 and 5.7, Revision 24

PPM 2.11.5; Floor Drain System, Section 5.11, Revision 32

PPM 12.5.33; Reactor Coolant Sampling, Revision 8

CI 10.17; Iodine, Revision 6

CSP-I131-W101; Reactor Coolant Isotopic Analysis for I-131 Dose Equivalent; Revision 3

NEI 99-02; Regulatory Assessment Performance Indicator Guideline, Revision 4

ISP-FDR/EDR-M401; Drywell Sump Flow Monitors - CFT; Revision 5

ISP-FDR/EDR-X301; Drywell Sump Flow Monitors - CC; Revision 7

LCO Log Number 10917; Technical Specification Inoperable Equipment/LCO/RFO Status Sheet

Section 4OA2: Identification and Resolution of Problems

Drawings and Diagrams

M537; Flow Diagram Equipment Drain System Reactor Building; Revision 70

Work Orders and Work Requests

WO 01070201	WO 01130433

Corrective Action Documents

PER 203-1991	PER 207-0251	PER 207-0249	PER 207-0245
PER 207-0240	PER 207-0238	PER 207-0229	PER 207-0227
PER 207-0216	PER 207-0213	PER 207-0210	PER 207-0240
PER 207-0207	PER 207-0201	PER 207-0200	PER 207-0174
PER 207-0169	PER 207-0167	PER 207-0164	CR 2-07-04561
CR 2-05-04561	CR 2-07-06437	CR 2-05-04559	CR 2-07-06462
CR 2-07-06466	CR 2-07-06472	CR 2-07-06469	CR 2-07-06467
CR 2-07-01918	CR 2-07-03301	CR 2-07-06082	CR 2-07-06071
CR 2-07-06057	CR 2-07-06051	CR 2-07-06050	CR 2-07-06042
CR 2-07-06048	CR 2-07-06016	CR 2-07-06019	CR 2-07-06014
CR 2-07-06010	CR 2-07-06002	CR 2-07-05992	CR 2-07-04382
CR 2-07-03689	CR 2-07-03690	CR 2-07-03685	CR 2-07-03683
CR 2-07-03681	CR 2-07-03672	CR 2-07-03617	CR 2-07-03612
CR 2-07-03619	CR 2-07-03611	CR 2-07-03608	CR 2-07-03535
CR 2-07-03411	CR 2-07-03432	PER 207-0168	PER 207-0163
PER 207-0160	CR 2-07-03454	CR 2-07-03393	CR 2-07-03422
CR 2-07-03391	CR 2-07-03384	CR 2-07-03375	CR 2-07-03417
CR 2-07-03412	CR 2-07-03397	CR 2-07-03082	CR 2-07-03054
CR 2-07-06139	PER 207-0042	CR 2-07-01830	CR 2-07-00908

PER 207-0062	CR 2-07-01560	CR 2-07-01834	CR 2-07-01587
CR 2-07-01584	CR 2-07-01560	CR 2-07-01368	CR 2-07-01076
CR 2-07-00908	CR 2-07-00794	CR 2-07-00655	CR 2-07-00280
CR 2-07-00218	CR 2-07-00095	CR 2-07-01658	CR 2-07-01711
CR 2-07-01709	CR 2-07-01702	CR 2-07-01196	CR 2-07-01204
CR 2-07-08866	CR 2-07-09063	CR 2-07-01175	CR 2-07-01162
CR 2-07-01174	CR 2-07-01167	CR 2-07-00813	CR 2-07-00799
CR 2-07-00908			

Miscellaneous

PTL H201027

Section 4OA3: Event Follow-up

Procedures

EPIP 13.1.1; Classifying the Emergency; Revision 36

Corrective Action Documents

CR 2-07-03246	PER 207-0160	CR 2-07-03133

Miscellaneous

Control Room Operator Logs from April 7, 2007