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

January 23, 2003

EA-02-107

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

SUBJECT: COLUMBIA GENERATING STATION - NRC SUPPLEMENTAL AND BASELINE INSPECTION REPORT 50-397/02-06

Dear Mr. Parrish:

On December 10, 2002, the NRC completed a supplemental and baseline team inspection at Columbia Generating Station. The enclosed report presents the results of this inspection. On December 10, 2002, we discussed the preliminary results of the onsite inspection with Mr. Rod Webring, Vice President, Nuclear Generation, and other members of your staff.

The inspection consisted of two focus areas. The first focus area was a supplemental inspection to address a previously identified “White” inspection finding. The finding involved multiple safety-related breaker failures and was documented in NRC Inspection Report 50-397/02-05. We performed the supplemental inspection in accordance with NRC Manual Chapter 95001 to address this issue. Causes for this problem included deficiencies in the design control and corrective action areas. The White finding placed your plant in the regulatory response column of the NRC’s Action Matrix - NRC Manual Chapter 0305, “Operating Reactor Assessment Program,” Exhibit 5, dated February 11, 2002. This inspection was conducted to provide assurance that the root and contributing causes of the White inspection finding were understood and to provide assurance that the corrective actions were sufficient to prevent recurrence. The NRC conducted a Regulatory Performance Meeting with Mr. Webring and other members of your staff on December 10, 2002. No additional issues were identified related to the White finding. This report will serve to close this issue.

The second focus area involved your problem identification and resolution activities and was conducted for the reasons outlined in our letter to you dated March 4, 2002. In general, we believed additional attention was warranted based on the identification of several issues related to the cross-cutting area of human performance, including a Yellow finding in the emergency preparedness area. This narrowly focused inspection effort reviewed: 1) your corrective measures related to past human performance problems; 2) your employee concerns program; and 3) your actions to address a small selected group of plant issues. Within these areas the inspection consisted of a selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, one finding was identified, which was determined to be a violation of NRC requirements. However, because of its very low safety significance and because it has been entered into your corrective action program, the NRC is treating this finding as a noncited violation, in accordance with Section V1.A.1 of the NRC's Enforcement Policy. If you deny this noncited violation, you should provide a response with the basis for your denial within 30 days of the date of this inspection report, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region IV; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Columbia Generating Station.

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

Sincerely,

/RA/

Anthony T. Gody, Chief
Operations Branch
Division of Reactor Safety

Docket: 50-397
License: NPF-21

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NRC Inspection Report
50-397/02-06

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-397
License:	NPF-21
Report No.:	50-397/02-06
Licensee:	Energy Northwest
Facility:	Columbia Generating Station
Location:	Richland, Washington
Dates:	December 2-10, 2002
Inspectors:	G. D. Replogle, Senior Resident Inspector, Project Branch E G. F. Larkin, Resident Inspector, Project Branch E J. F. Drake, Operations Engineer, Operations Branch D. L. Stearns, Reactor Inspector, Project Branch E
Approved By:	Anthony T. Gody, Chief Operations Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000397/2002-06, Energy Northwest, 12/2-12/10/2002, Columbia Generating Station, Supplemental Inspection for a White Finding and Baseline Inspection of the Identification and Resolution of Problems; Other.

The inspection was conducted by one senior resident inspector, one resident inspector and two regional reactor inspectors. The significance of most findings is indicated by their color (Green, White, Yellow, 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.

<u>Inspector Identified Findings</u>

Cornerstone: Mitigating Systems

- The U.S. Nuclear Regulatory Commission (NRC) performed this supplemental inspection to assess the licensee's evaluation associated with the failure to meet the requirements of 10 CFR Part 50, Appendix B, Criterion III (Design Control), and Criterion XVI (Corrective Actions). Specifically, the licensee failed to properly verify the design adequacy of replacement Westinghouse DHP-VR 350 breakers to operate in the existing cubicles, without vendor recommended maintenance, and the licensee failed to specify effective corrective measures in response to initial breaker failures, which permitted even more failures to occur and further delayed correcting the condition for several additional months. This finding was previously characterized as having low-to-moderate risk significance ("White") in NRC Inspection Report 50-397/02-05. During this supplemental inspection, performed in accordance with Inspection Procedure 95001, the team verified that the licensee performed a comprehensive evaluation of the White finding. The licensee identified the root cause and several contributing causes and took extensive corrective measures to address the problems.

  Given the licensee's acceptable performance in addressing the issues, the White finding associated with this issue was only considered in assessing plant performance for a total of four quarters in accordance with the guidance in Inspection Manual Chapter 0305, "Operating Reactor Assessment Program." The issue was identified in the first quarter of 2002 (pertinent end of the inspection period for the finding was March 26, 2002); therefore, it will no longer be considered in assessing plant performance after the fourth quarter of 2002.

Identification and Resolution of Problems

- The licensee had taken reasonable corrective measures to address multiple human performance concerns that led to a substantive finding in the cross-cutting area of human performance. Corrective actions were implemented in a timely manner. Additionally, based on the interviews conducted during this inspection and review of selected documents, the licensee properly implemented their employee concerns program and workers at the site felt free to input safety issues into the various problem identification and resolution processes (Section 4OA2).

- Green. The team identified that the licensee failed to take adequate corrective measures in the 1996-1997 time frame to address a significant condition adverse to quality involving multiple debris related containment isolation valve failures. The licensee did not follow through on specified corrective measures and the plant suffered two similar failures in June, 2001 and October, 2002. The debris accumulation was a common mode problem and affected both in-series containment isolation valves. In addition, the team identified that the licensee did not address the continuing operability concern until prompted by the NRC and the licensee's first two attempts at addressing operability were inadequate, in part, because they were based on inaccurate information. The licensee subsequently isolated the containment penetration pending further actions.

  A violation of 10 CFR Part 50, Appendix B, Criterion XVI was identified that is being treated as a noncited violation in accordance with Section VI.A.1 of the NRC Enforcement Policy. The team determined that the issue was more than minor in significance because the problem affected the reactor safety, barrier integrity cornerstone objective. The team utilized the NRC's significance determination process Manual Chapter 0609, Appendix A worksheet and determined that the issue was of very low safety significance (Green). The issue screened out as Green because the problem did not result in an actual open pathway in the physical integrity of the reactor containment or an actual reduction of the atmospheric pressure control function of the reactor containment (Section 4OA2).

Report Details

SUPPLEMENTAL INSPECTION TO ADDRESS "WHITE" FINDING (95001)

01 INSPECTION SCOPE

The U.S. Nuclear Regulatory Commission (NRC) performed this supplemental inspection to assess the licensee's evaluation associated with the failure to meet the requirements of 10 CFR 50, Appendix B, Criterion III (Design Control) and Criterion XVI (Corrective Actions). Specifically, the licensee failed to properly verify the design adequacy of replacement Westinghouse DHP-VR 350 breakers to operate in the existing cubicles, without vendor recommended maintenance, and the licensee failed to specify effective corrective measures in response to initial breaker failures - which permitted even more failures to occur and further delayed correcting the condition for several additional months. The NRC documented this issue in NRC Special Inspection Report 50-397/02-05.

02 EVALUATION OF INSPECTION REQUIREMENTS

02.01 Problem Identification

a. Determination of who (i.e., licensee, self-revealing, or NRC) identified the issue and under what conditions

The issue was self-revealing through several equipment malfunctions. Safety-related breaker mechanism operated cell (MOC) switches associated with several Westinghouse DHP-VR 350 breakers failed to reposition in concert with the breakers. The MOC switches provide signals and permissives to various plant components. Malfunctions occurred on June 29, 2001, November 19, 2001, January 17, 2002, and February 13, 2002. For the first two malfunctions, one train of standby service water was rendered inoperable. In the final two instances, operators received erroneous control room alarms with no actual equipment impact. Overall, this problem was generic in nature and potentially impacted all 22 safety-related breakers that were replaced during the last refueling outage.

b. Determination of how long the issue existed, and prior opportunities for identification

The issue existed since the May - July, 2002 refueling outage, when the licensee installed 22 of the new breakers in the plant, and up to February 14, 2002, when the licensee shut down the plant in accordance with plant technical specifications. The plant shut down was necessitated because the licensee could not resolve the cause of an emergency diesel generator breaker malfunction prior to exceeding the 72 hour technical specification allowed outage time.

The licensee had multiple prior opportunities to identify and fix the problems, as noted below:

- The NRC had issued generic communications informing licensee's of the need to perform appropriate breaker maintenance (i.e. NRC Generic Letter 93-38, "Metal-Clad Circuit Breaker Maintenance Issues Identified by NRC Inspections," dated January 15, 1998).

- System engineers were aware of the vendors maintenance recommendations, but did not recognize the importance of the recommendations, despite the clear warnings. Engineers, at the time, felt some pressure to minimize plant maintenance. In addition, engineering reviews during the design process failed to properly consider the different operating and maintenance requirements associated with the newer breaker design.

- Of the four noted malfunctions, the first three malfunctions constituted prior opportunities to identify and correct the issues. Initial problem identification and resolution efforts were narrowly focused on superficial symptoms and lacked rigor.

c. <u>Determination of the plant-specific risk consequences (as applicable) and compliance concerns associated with the issue</u>

An NRC senior risk analyst determined that the combined internal and external events core damage frequency, using the seismic frequency surrogate method, was between a lower bound value of 1.33E-6/year and an upper bound value of 3.16E-6/year. The risk analyst determined that the increase in the large early release frequency, using the seismic surrogate method, was between a lower bound value of 9.08 E-8/year and an upper bound value of 1.78E-7/year. Therefore, the NRC determined that the increased unreliability of the circuit breaker MOC switch was of low-to-moderate safety significance (White).

02.02 <u>Root Cause and Extent of Condition Evaluation</u>

a. <u>Evaluation of methods used to identify root causes and contributing causes - Level of detail of the root-cause evaluation</u>

The licensee performed a comprehensive root-cause evaluation and properly considered the extent of condition. Industry recognized methods utilized during this evaluation included fault tree analysis, change analysis and barrier analysis. The evaluation included in-depth discussion of failed barriers and human performance issues that led to the MOC switch failures. Overall, the licensee identified the following causes and contributing causes:

- The licensee's engineering review considered the new beaker design as a roll-in equivalent replacement for the older design breakers. This conclusion was based on an inadequate review of the replacement breaker qualification test data. Engineers failed to recognize that the qualification tests were conducted with the breakers maintained in a manner consistent with vendor recommendations, which was different that the standard maintenance conducted on breakers and their breaker cubicles at the station.

- Lack of commitment to recommendations from self-assessments. A self assessment had previously recommended that the licensee establish a circuit-breaker program. This was not accomplished. The lack of a program expert, coupled with several competing resource interests, led to less than thorough design and maintenance work products.

- Inadequate engineering turnover. The engineer primarily responsible for the breaker replacement project left the company and provided inadequate turnover to the replacement engineer. The replacement engineer had competing interests and was not completely familiar with the breaker functions and requirements prior to installation. As a consequence, the engineer was not sufficiently familiar with the breaker design and maintenance requirements to properly diagnose initial breaker failures.

b. Consideration of prior occurrences of the problem and knowledge of prior operating experience

The licensee's evaluation concluded that they had missed prior opportunities to catch the problems. As noted previously, engineers had failed to properly evaluate the three earliest malfunctions. Further, the licensee acknowledged that the failure to properly implement past self-assessment recommendations and to ensure that engineers received appropriate turnovers were contributing causes.

c. Consideration of potential common causes and extent of condition of the problem

The licensee appropriately considered generic implications and extent of condition. The licensee determined that the problem scope included all 22 safety-related breakers that were replaced during the last refueling outage. However, only 16 of the safety-related breakers had an active safety function to reposition during an accident.

02.03 Corrective Actions

a. Appropriateness of corrective actions

The licensee took, or planned to take, extensive short- and long-term corrective actions to address the issues. More specifically, the licensee:

- Trained appropriate engineering and other plant personnel concerning the circumstances that culminated in the identified problems.

- Performed extensive maintenance and testing of each of the 22 safety-related breakers and cubicles prior to plant startup from the forced outage. The tests included appropriate acceptance criteria to account for required closing forces, instrument uncertainty and degradation.

- Planned to complete inspections on the remainder of the MOC switch assemblies (approximately 60 units) utilized with different breaker designs during, or prior to, the next refueling outage, schedule for spring 2003.

- Implemented a monitoring program to ensure that the MOC switch assemblies and "SURE CLOSE" operating mechanisms continued to perform reliably and without noticeable degradation. Engineers established an initial monitoring interval of 2 weeks for a representative sample of breakers. The licensee planned to increase the interval when they gained confidence in breaker capability.

- Modified plant procedures to include appropriate maintenance requirements.

- Modified design procedures to address maintenance design impacts.

- Planned to review vendor recommendations associated with other types of circuit breakers and switchgear to ensure that appropriate maintenance practices are performed.

- Completed a circuit breaker program description and planned to implement a circuit breaker program in the future.

- Reviewed all plant modification scheduled for the next refueling outage to ensure that similar design oversights did not occur.

- Enhanced corrective action procedures to ensure that initial problems with newly installed plant equipment get more thorough evaluations.

- Scheduled periodic effectiveness reviews to further evaluate the corrective measures.

b. <u>Prioritization of corrective actions</u>

The team concluded that the corrective actions were properly prioritized. Actions of an immediate nature were given the highest priority. A completion date and a responsible manager were assigned for each corrective action.

c. <u>Establishment of a schedule for implementing and completing the corrective actions</u>

The licensee's evaluation established a schedule for the completion of the long-term corrective actions by the next refueling outage, scheduled for Spring 2003. The team concluded that this schedule was acceptable.

d. <u>Establishment of quantitative or qualitative measures of success for determining the effectiveness of the corrective actions to prevent recurrence</u>

The licensee scheduled at least two effectiveness reviews to further evaluate the corrective measures, one at 6 months and another at 18 months.

02.04 Conclusions

The licensee appropriately identified the root cause and contributing factors that led to the White finding. The licensee’s identification of the extent of condition was sufficiently broad to capture all of the known affected breakers and to also pursue potential maintenance related problems with breakers of other designs, that were not directly impacted by this particular problem. The team determined that the licensee’s evaluation was thorough and that it provided adequate assurance that all problematic areas were properly addressed.

4. OTHER ACTIVITIES (OA)

4OA2 Problem Identification and Resolution

a. Effectiveness of Corrective Actions Associated with Human Performance Problems

(1) Inspection Scope

The NRC had documented a substantive human performance issue in NRC Inspection Report 50-397/01-04. The issue involved several plant events that were caused by poor human performance. The team reviewed corrective actions associated with that finding, which included: 1) increased contractor training and oversight during outages; 2) increased support to operators during outages; 3) increased resources towards job planning prior to outages; 4) staff coaching sessions; and 5) the use of a human performance simulator. The team also reviewed human performance data and statistics for the past two years, which showed marked improvement for the non-outage period since April 2002.

(2) Issues

Based on a review of the licensee’s records and interviews with plant personnel and managers, the licensee has taken reasonable actions to preclude significant problematic human performance trends. No findings of significance were identified.

b. Effectiveness of Corrective Actions Associated with Selected Plant Issues

(1) Inspection Scope

The team reviewed potentially risk-important plant problems to evaluate the licensee’s ability to properly resolve plant issues and prevent further recurrence for significant conditions adverse to quality. For this effort, the team reviewed 28 directly-related problem evaluation requests and supporting documents, including root-cause analysis, to determine if the licensee's evaluation considered the full extent of the condition, reportability and operability, generic implications, common causes, and previous occurrences. The team also reviewed whether the licensee's evaluation used appropriate industry experience, such as operating event reports and NRC and vendor generic notices. The team's inspection sample included:

- Containment isolation valve FDR-V-4 failure on October 22, 2002
- Standby service water piping thru-wall leakage on July 7, 2002
- Flooding caused by reactor core isolation cooling system relief Valve RCIC-RV-19 (relief valve on lube oil cooler line), on several occasions

(2) Issues

Introduction. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, for the failure to take effective corrective actions to preclude containment isolation valve failures caused by system debris, a known and preventable problem. The original problem surfaced in 1996, but the licensee failed to follow through on planned corrective measures and two additional valve failures were experienced in the past 18 months. In addition, the licensee did not identify a current operability concern until prompted by the NRC and the licensee's first two attempts at addressing operability were inadequate, in part, because they were based on inaccurate information.

Description. **Valve Problems:** In the past 18 months, the licensee experienced two containment isolation valve failures due to system debris - a known and preventable problem. Valves FDR-V-3 and FDR-V-4 are both 3-inch ball valves located in the drywell unidentified leakage rate instrument line.

On June 13, 2001, Valve FDR-V-3 failed to indicate closed during stroke-time testing. The system engineer documented in Problem Evaluation Request 202-1259 that the cause was not conclusively known, but was likely due to debris in the system. In response to the failure, the licensee cleaned and inspected the internals of both valves, but took no action to rid the system of debris. The significance of this particular failure was minimized because the problem occurred during a plant outage when the valve was not required to be operable. However, the failure was an important indicator of the return of sufficient debris in the system to cause valve failures.

On October 7, 2002, with the plant at 100 percent power, Valve FDR-V-4 failed a local leak rate test. The measured leakage was greater than 9000 standard cubic centimeters per minute (sccm), approximately 20 times the acceptance limit of 442 sccm. Again, the licensee took no additional actions to rid the system of debris.

The licensee experienced both valve failures following instances where relatively large amounts of water were flushed from the containment sump, through a system loop seal (a source of debris) and past the valves. Some of the debris remained in the valves and when the valves repositioned, debris would get lodged between the ball and the seat, which caused the failures. The debris consisted of small granules of iron oxide and other small particulate matter.

Upon further inspection, the team identified that the licensee had failed to fully implement planned corrective measures to address similar valve failures experienced in the 1996 time frame. Specifically, in 1996 the licensee experienced three debris related failures. In each case, one of the two valves had either failed to close, failed a local leak-rate test beyond the capacity of the cart to measure (greater than 20,000 sccm), or failed a local leak-rate test beyond the licensee established acceptance criteria. For these problems, the NRC issued a 10 CFR Part 50, Appendix B, Criterion XVI, violation for the failure to take effective corrective actions to address a significant condition adverse to quality, documented in NRC Inspection Report 50-397/96-11. As corrective measures, the licensee planned to: 1) establish a preventative maintenance task to inspect and clean the valves, on a 3-year interval because it took 4 years before failures occurred previously; 2) inspect and clean the drywell sump periodically to prevent debris accumulation; and 3) perform periodic system flushes to prevent the buildup of debris in the loop seal. The licensee had failed to consistently implement the first two action items and had little basis for the 2-year periodicity of the system flushes. Specifically, the sump was cleaned in 1997, but the licensee had no plans to clean the sump again. The licensee did not perform the valve inspect/clean task until the FDR-V-3 valve had already failed in 2001 (4 years after initiating the preventive maintenance task). The licensee performed system flushes on a 2-year periodicity, but had not demonstrated that this interval was appropriate.

Valve Operability: In addition to the above, at the time of the inspection, the licensee had not taken action to clean the system and had not addressed containment isolation valve operability. The team was concerned because a loss-of-coolant accident would send water through the system piping and could transport debris to the valves. When the valves repositioned to perform their safety function, the valves could again fail because of debris intrusion. The team considered this as a common mode problem, potentially affecting both valves.

In response to the teams’ concerns, the licensee documented an operability evaluation in Problem Evaluation Request 202-3392 and concluded that the valves remained operable. However, the team considered the evaluation inadequate because engineers utilized inaccurate information as a basis for their conclusions. More specifically, the evaluation stated that the valves had never failed a local leak-rate test after a system drain down or flush. The team noted five instances where, after system drain downs, the valves had either failed local leak-rate tests or had failed to reposition closed, the equivalent of a local leak-rate test failure. These failures occurred on January 24, 1996 (FDR-V-4); April 26, 1996 (FDR-V-3); July 12, 1996 (FDR-V-3); June 13, 2001 (FDR-V-3); and October 22, 2002 (FDR-V-4).

In response to the teams’ continued concerns regarding valve operability, the licensee documented a second operability evaluation in Followup Assessment of Operability 202-3392. The team found this operability assessment inadequate for two reasons. First, the assessment relied on inaccurate information. Plant engineers wrote that the sump was not a debris source. This implied that several drain downs performed subsequent to the last failure were sufficient at removing debris from the loop seal. Therefore, the potential for additional debris migrating from the sump to the loop seal was not a problem. However, the team identified that, following the sump cleaning in 1997, the system engineer wrote:

> The FDR drywell sump contained a significant amount of dirt, corrosion product, and paint chips (approximately 1 cubic foot). This sump debris was probably just as much a contributor to the degradation of FDR-V-3 and FDR-V-4 . . . as any debris located within the wetwell loop seal piping.

The second problem with the operability evaluation involved the licensee’s operability focus. Instead of evaluating the operability of each valve as a separate component that must perform its safety function, the licensee focused on the operability of the penetration, where the operability of one valve would satisfy penetration, or containment, operability. The licensee reasoned that it was highly unlikely that both valves would fail simultaneously. The licensee based this assumption on the fact that the valves had never failed simultaneously before. While the team acknowledged the licensee’s assessment could be used as an argument related to risk significance, the team determined that the licensee’s approach was not consistent with the plant technical specifications, which defines operability of individual components (in separate trains) and also specifies operability requirements for each individual valve.

In response to continued concerns, the licensee isolated the containment penetration and planned to obtain additional objective information for their operability basis. The licensee planned to perform several flushes; measure and analyze the debris coming from the system; and perform at least three additional local leak-rate tests of Valves FDR-V-3 and FDR-V-4 following the flushes. The licensee’s plans were found to be sufficient to allow drawing conclusions based on fact.

<u>Assessment</u>. The team evaluated the extent of condition and historical performance of the failed valves. The team noted that the valves had not failed simultaneously in the past, despite being subjected to a common-mode failure mechanism. While the valves had experienced five debris-related failures in the past 7 years, in a majority of instances following sump drain down events, where debris could be transported to the valves and the valves were tested afterwards, the valves had worked as designed. The team found no evidence that the same failure mechanism affected other plant valves.

Based on the above, the team determined that the issue was more than minor in significance because the problem affected the reactor safety, barrier integrity cornerstone objective. The team utilized the NRC’s significance determination process Manual Chapter 0609, Appendix A worksheet and determined that the issue was of very low safety significance (Green). The issue screened out as Green because the problem did not result in an actual open pathway in the physical integrity of the reactor containment or an actual reduction of the atmospheric pressure control function of the reactor containment.

Enforcement. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, which requires the licensee to take effective corrective measures to preclude repetition of significant conditions adverse to quality. In 1996, the licensee experienced three Valves FDR-V-3 and FDR-V-4 failures because of debris in the system, a significant condition adverse to quality. Contrary to the noted requirements, the licensee failed to properly implement effective corrective measures to preclude repetition and on June 13, 2001, and October 7, 2002, Valves FDR-V-3 and FDR-V-4, respectively, suffered additional debris related failures. The licensee entered this issue into their corrective action program as Problem Evaluation Request 202-3471 (NCV 50-397/02006-01).

c. Assessment of Safety-Conscious Work Environment

(1). Inspection Scope

The team interviewed 24 site personnel from various plant organizations, including the Nuclear Safety Issues Program Coordinator, to assess the licensee's safety conscious work environment. These interviews assessed the overall effectiveness of the program and the licensee's staff's willingness to use it. The team also reviewed selected nuclear safety issues program packages to evaluate the characterization of issues, licensee investigations and overall communications.

(2) Issues

The team concluded that the licensee effectively operated their nuclear safety issues program, that issues were properly investigated and dispositioned, that employees properly utilized the licensee's corrective action processes to identify issues, and employees were comfortable using the licensee's nuclear safety issues program. Employees also expressed a willingness to bring issues to the NRC.

4OA6 Meetings

01 Regulatory Performance Meeting

On December 10, 2002, the NRC conducted a Regulatory Performance Meeting with Mr. R. Webring and other members of the licensee's staff concerning the White inspection finding, discussed in this report. The NRC and the licensee discussed the performance deficiencies that led to the White finding and the corrective actions that the licensee took to preclude further repetition.

02 Exit Meeting

On December 10, 2002, the team leader conducted an exit meeting with Mr. R. Webring, Vice President, Nuclear Generation, and other members of the licensee's staff to discuss the results of the inspection. The plant management acknowledged the inspection findings. Plant management discussed the extent of the proprietary information shared during this inspection. The team included none of the proprietary material in this report.

ATTACHMENT

PARTIAL LIST OF PERSONS CONTACTED

Licensee

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D. Atkinson, Vice President, Technical Services
D. Coleman, Manager, Performance Assessment and Regulatory Programs
D. Feldman, Manager, Operations
P. Inserra, Plant Engineering Manager
W. Oxenford, Plant General Manager
C. Perino, Manager, Licensing
J. Peters, Manager, Radiation Services
R. Webring, Vice President, Nuclear Generation
G. Weimer, Corrective Action Program Manger

NRC

W. Jones, Chief, Division of Reactor Projects, Branch E

ITEMS OPENED AND CLOSED

Opened and Closed

50-397/02006-01	NCV	Inadequate corrective measures to address debris related valve failures (Section 4A02)

Closed

50-397/02005-01	VIO	White finding for multiple safety-related breaker malfunctions

DOCUMENTS REVIEWED

Procedures

Procedure SWP-CAP-O1, “Problem Evaluation Requests,” Revision 6

Procedure SWP-CAP-02, “Root Cause Analysis,” Revision 0

Procedures SWP-PRO-01, “Description and use of Procedures and Instructions,” Revision 3

Procedure OSP-INST-H101, “Shift and Daily Instrument Checks (MODES 1, 2, 3),” Revision 35

Procedure ECP-DG1/INSP-B101, “Diesel Generator DG1 Electrical Inspection,” Revision 3

Procedure *8.3.63, “Procedure for Monitoring Pipe Wall Thinning,” 05/18/01

Procedure 2.11.7B, “Drywell Floor Drain Backflush,” Revision 2

Engineering Procedure DES-2-1, “Plant Design Changes,” Revision 5

Engineering Standard PDS-9, Design Preparation and Verification,” Revision 4

<u>Problem Evaluation Requests</u>

296-0045, Valve FDR-V-4 failed to close, 1/19/96

296-0301, Valve FDR-V-3 failed local leak rate test, 5/1/96

296-0554, Valves FDR-V-3 and FDR-V-4 failed stroke time tests, 7/6/96

296-0562, Valve FDR-V-3 failed local leak rate test, 7/12/96

296-0602, Valve FDR-V-4 failed stroke time test, 8/1/96

200-1197, Reactor core isolation cooling system relief Valve RCIC-RV-19T lifting, causing room flooding, 7/16/02

201-0214, Division I diesel generator vibrations showing increasing trend, 2/07/01

201-0476, Reactor core isolation cooling system pressure control valve failed to control pressure and relief Valve RCIC-RV-19T lifted and caused partial room flooding, 3/30/01

201-0957, Standby service water flange and elbow displays cavitation pits, 05/27/01

201-1102, Unacceptable pitting and loss of minimum wall thickness in standby service water piping elbow, 6/04/01

201-1259, Valve FDR-V-3 failed to close, 6/14/01

201-2467, Evidence of pipe wall thinning in nine localized spots of 18" standby service water return piping, 11/06/01

201-2692, Division I diesel generator bearing shows degrading trend, 9/24/02

202-0456, During Division II diesel generator surveillance MOC switch failed, 2/13/02

202-0468, Forced shutdown due to inability to identify cause of breaker malfunction within the Technical Specification allowed outage time, 2/14/02

202-0640, Reactor core isolation cooling system relief Valve RCIC-RV-19T lifting, causing room flooding, 2/28/02

202-0718, Reactor core isolation cooling system relief Valve RCIC-RV-19T lifting, causing room flooding, 3/7/02

202-0927, Corrective measures for initial breaker MOC switch failure were insufficient to prevent three subsequent malfunctions, 3/26/02

202-1979, Additional thinned wall areas found downstream of standby service water system Valve SW-V-12A, 7/7/02

202-1977, Pinhole leak on weld downstream of standby service water system Valve SW-V-12B, 07/07/02

202-1979, Thinned walled areas found downstream of Valve SW-V-12A, 07/07/02

202-1991, Cavitation pitting on body of Valve SW-V-12B, 07/09/02

202-2293, Reactor core isolation cooling system relief Valve RCIC-RV-19T lifting, causing room flooding, 8/7/02

202-2954, Valve FDR-V-4 failed local leak rate test, 10/22/02

202-3084, Reactor core isolation cooling system relief Valve RCIC-RV-19T lifting, causing room flooding, 11/4/02

202-3097, Inadequate response to industry information concerning circuit breaker reliability, 11/6/02

202-3256, Review of NDE UT thickness data for Valve SW-V-12A down stream piping indicates the original projected wall loss rate has changed, 11/20/02

202-3392, Valves FDR-V-3 and FDR-V-4 operability concerns, 12/4/02

Maintenance Work Orders

01052820, Drain and flush of Valves FDR-V-3 and FDR-V-4, 12/02

01010302, Inspect Valves FDR-V-3 and FDR-V-4 internals and perform IST and local leak rate testing, 6/14/02

SU70, Inspect Valves FDR-V-3 and FDR-V-4 internals and perform IST and local leak rate testing, 6/19/97

01050729, Valve FDR-V-4 failed stroke time test, troubleshoot, repair and test, 10/27/02

Design Change Package

DCP 86-0324-OG, SW Valves Modification

Miscellaneous

Drawing Number M977, DG-GEN-DG 12 & 2 External Lube Oil Reservoir Diesel Generator Building EL 441'0", Revision 2

Plant Modification Request 02-86-0324-1, “Balance of Design Requirements for Standby Service Water Keep Full Systems,” 4/15/87

NRC Generic Letter 89-08, “Erosion/Corrosion Induced Pipe Wall Thinning,” dated 7/20/89

NRC Generic Letter 87-01, “Thinning of Pipe Walls In Nuclear Power Plants,” 7/9/87

“Pipe Wall Thinning Monitoring Program Plan,” Revision 3

Washington Public Power Supply System, General Information Handbook, Administrative Procedures Manual, GIH 3.4.10, “Nuclear Safety Issues Program,” Revisions 0 and 1

Various Nuclear Safety Issues Program logs and packages for 1999, 2000, 2001, and 2002

Energy Northwest response to Violation 50-397/02005-01, 8/21/2002

“Breaker MOC Force Performance Monitoring & Trending Status,” 11/15/02

Division I and II bearing vibration data, from 1994 to 2002.

SMS 97000787, Recommendation to de-sludge the under-vessel sump periodically, 6/9/97

“Interim Plant Operations Committee Meeting Minutes/Activities 02-49.01,” 12/11/2002

Calculation EQ-02-97-0, Reactor core isolation cooling system room flooding, Revision 0

Calculation 5.20.08, Reactor core isolation cooling system flow restrictors, Revision 3

Final Safety Analysis Report

Technical Specifications