UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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November 24, 2003

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 PROBLEM IDENTIFICATION AND RESOLUTION INSPECTION REPORT 05000397/2003-009

Dear Mr. Parrish:

On October 9, 2003, the Nuclear Regulatory Commission (NRC) completed a team inspection at Columbia Generating Station. The enclosed report presents the results of this inspection. On October 9, 2003, we discussed the preliminary results of the onsite inspection with Mr. R. Webring, Vice President Nuclear Generation, and other members of your staff.

This inspection was an examination of activities conducted under your license as they relate to the identification and resolution of problems and the compliance with the Commission's rules. Within these areas, the inspection involved examination of selected procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, two findings were identified, which were determined to be violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating the findings as noncited violations, in accordance with Section V1.A.1 of the NRC's Enforcement Policy. If you deny the noncited violations, you should provide a response with the basis for you 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.

Overall, the inspectors concluded that problems were identified, evaluated, and resolved within the context of your problem identification and resolution program. Nonetheless, several problems were identified which indicate a need for improvement in the area of evaluating the cause and extent of problems at your facility. Each of the problems discussed in the report involved ineffective or untimely corrective actions resulting from weak engineering analyses. Corrective actions implemented by your staff to address the cross-cutting issue in the area of human performance appeared to improve performance during the 2003 refueling outage. Nevertheless, findings related to human performance since the refueling outage indicate a need for continued attention.

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
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**ENCLOSURE**

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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| Docket: | 50-397 |
| License: | NPF-21 |
| Licensee: | Energy Northwest |
| Facility: | Columbia Generating Station |
| Location: | Richland Washington |
| Dates: | September 29 - October 9, 2003 |
| Inspectors: | T. McKernon, Senior Operations Engineer, Operations Branch<br>Z. Dunham, Resident Inspector, Project Branch E<br>M. Haire, Operations Engineer, Operations Branch<br>G. Johnston, Senior Operations Engineer, Operations Branch |
| Approved By: | Anthony T. Gody, Chief<br>Operations Branch<br>Division of Reactor Safety |

## SUMMARY OF FINDINGS

IR 05000397/2003-09, Energy Northwest, 09/29-10/09/2003, Columbia Generating Station, biennial baseline inspection of the identification and resolution of problems.

This inspection was conducted by one resident inspector and three regional operations 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.

**Identification and Resolution Problem**

The team found that the licensee, in general, identified problems, entered, and prioritized problems into their corrective action program. Nevertheless, weaknesses were identified in extent of condition reviews and in the development of corrective actions. The team found the corrective actions to address the substantive finding in the cross-cutting area of human performance had resulted in overall improved performance during the 2003 refueling outage. However, the team noted continuing human performance challenges in several areas such as rework, loss of shutdown cooling, and engineering reviews. Furthermore, based upon interviews and review of selected documents, the licensee properly implemented their employee concerns program and workers felt free to input safety issues into the problem identification and resolution program.

### Cornerstone: Barrier Integrity

- Green. The team identified a violation for an inadequate corrective action in accordance 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action” for the licensee’s failure to adequately implement a procedure revision intended to ensure that control room dose limitation requirements in accordance with 10 CFR Part 50, Appendix A, General Design Criterion 19, “Control Room,” were met.

  The failure to implement an effective corrective action was of very low safety significance because the finding only represented a degradation of the radiological barrier function provided to the control room. This issue was entered into the corrective action program as Problem Evaluation Request 203-3643. Therefore, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: Noncited Violation 05000397/2003009-01, Failure to Implement Adequate Corrective Action to Address Increased Control Room In-Leakage (Section 4OA2).

### Cornerstone: Mitigating Systems

- Green. The team identified a violation of 10 CFR Part 50 Appendix B, Criterion XVI, for the failure to promptly correct a condition adverse to quality associated with all safety-related 4160 Vac breakers. The team noted eight instances where truck-operated cell position switches had displayed indication problems, and the licensee had failed to promptly identify and correct a problem associated with seismic qualification. The associated 4160 Vac breakers were used in power circuits for emergency diesel generators, standby service water pumps, and all emergency core cooling system pumps. This issue was more than minor because it affected the reactor safety mitigating systems objective to ensure availability of equipment to respond to initiating events. In addition, this issue was determined to be of very low safety significance because it was a qualification deficiency confirmed not to result in loss-of-function as defined in NRC Generic Letter 91-18.

The violation of 10 CFR Part 50, Appendix B, Criterion XVI, is being treated as a noncited violation in accordance with Section VI.A.1 of the NRC Enforcement Policy because the issue was of very low safety significance and had been entered into the corrective action program in Problem Evaluation Request 203-3693; Noncited Violation 05000397/2003009-02, Failure to Promptly Correct a Condition Adverse to Quality Associated with the 4160 Vac Breaker Truck-Operated Cell Position Switches (Section 4OA2).

## REPORT DETAILS

4. OTHER ACTIVITIES (OA)

**4OA2** Problem Identification and Resolution

a. Effectiveness of Problem Identification

(1) Inspection Scope

The team reviewed items selected across six of the seven cornerstones of safety to determine whether problems were properly identified, characterized, and entered into the corrective action program. The security cornerstone will be evaluated at a later date.

(2) Assessment

Introduction. The team identified a noncited violation (Green) of 10 CFR Part 50, Appendix B, Criterion XVI (Corrective Actions) for the failure to promptly correct a condition adverse to quality associated with all 16 safety-related 4160 Vac breakers. The team noted eight instances where truck-operated cell position switches had displayed indication problems, since installation, and the licensee had failed to properly identify and correct a problem associated with seismic qualification.

Description. During the Spring 2001 refueling outage, the licensee installed new Cutler-Hammer breakers in all safety-related 4160 Vac applications. Sixteen breakers have a safety function to reposition during design basis accidents, including those postulated accidents involving seismic events. The new breakers were utilized in power circuits for emergency diesel generators, standby service water pumps, and all emergency core cooling system pumps. The breakers were installed in the old breaker cubicles with the original auxiliary equipment. Truck-operated cell switches at the rear of each cubicle reposition when the breakers are inserted. The switches provide signals for indication (non-safety function) and breaker close permissives (safety function). If the breaker close permissive contacts fail to remain closed during a seismic event, the breaker may not close on demand (equipment would be rendered inoperable).

On August 28, 2003, during a corrective action review for trends, the team observed that the licensee had experienced eight instances, since June 2001, where, following breaker cycling, partial breaker indication was lost due to truck-operated cell switch over travel. While only the non-safety-related contacts had actually opened, the inspector identified that the condition could only exist if all contacts (safety and non-safety) were in the over-travel position and very close to opening. Small tolerance differences between the different contacts permitted the non-safety contacts to open while the safety-related contacts remained barely engaged. This was a generic problem affecting all of the newly installed safety-related 4160 Vac breakers. The problem was likely caused by dimensional differences between the old Westinghouse breakers and the newer Cutler Hammer design.

The relatively small movement associated with breaker cycling was sufficient to open some truck-operated cell switch contacts. Therefore, it was indeterminate whether the safety-related contacts would remain in the closed position during a seismic event. A seismic event could subject the breakers to much stronger forces and would increase the potential for switch and contact movement.

The team determined that, due to the over-travel condition, the licensee did not have reasonable assurance that truck-operated cell switches in use at Columbia Generating Station were installed consistent with seismic qualification tests. The licensee's qualification testing documented in Report NSP/RRSM/GCS(00)-347, "Seismic and Environmental Qualification Summary Report of Westinghouse/CH[Cutler Hammer] 50 DHP-VR350-1200A Replacement Circuit Breakers for Columbia Generating Station," Revision 1, contained acceptance criteria that specified that no truck-operated cell switch contacts (safety or non-safety) would remain open during testing. Since the truck-operated cell switches installed in the field had demonstrated decidedly different performance (with contacts remaining open), the team concluded that the installed configuration was different than that actually tested.

The team noted that the licensee had not seismically tested the "in-use" truck-operated cell switch design concurrent with the new Cutler Hammer breaker design. Qualification tests conducted in accordance with Report NSP/RRSM/GCS(00)-347, were performed with a different model of truck-operated cell switch than that commonly used at the facility. The tested truck-operated cell switches had a wider contact surface when compared to the Columbia Generating Station units. Additionally, during original cubicle testing, as documented in "Westinghouse Seismic Qualification Report 47A-00-0147," Revision 3, dated September 1978 (where the same truck-operated cell switches presently used were tested, along with the older style breakers), no abnormal truck-operated cell switch indications were documented. Finally, during breaker installation in 2001, the licensee did not check for truck-operated cell switch alignment, but simply checked for continuity instead. This allowed acceptance of an unqualified configuration.

In addition to the above, the team noted the following previously identified and related problems:

- The licensee received a White finding in NRC Inspection Report 50-397/02-05, dated June 24, 2002, and failed to take effective corrective actions to address design control issues associated with mechanism-operated cell switches. The White finding involved a failure to implement appropriate design controls for the new Cutler Hammer breaker change out. In short, the licensee experienced several operability problems due to the failure to properly evaluate the use of a new breaker design in existing breaker cubicles. The licensee's corrective measures to resolve design problems were not fully effective, as evidence by continuing similar design-related problems associated with the truck-operated cell switches.

- The licensee also failed to correct known performance deficiencies associated with the engineering work group assigned to the breaker project. As a contributing cause to the White finding, the NRC had identified that engineers performed a less than thorough review of initial problems, which led to the existence of the degraded switchgear condition for an extended period of time. The licensee's corrective measures were ineffective to address this higher tier problem, as demonstrated by engineering work that continued to focus on superficial indications and failed to thoroughly investigate the repetitive truck-operated cell switch problems.

- Engineers did not capture all of the initial truck-operated cell switch problems in the licensee's corrective action program. For example, during post-installation testing in June 2001, engineers observed at least three unexpected truck-operated cell switch indication failures. In addition, shortly after startup, one additional failure was observed. However, only one of these early problems was documented on a problem evaluation request. The failure to properly utilize the corrective action program helped to mask the generic significance of the issue. Most of the future failures were sporadic, separated by time, and characterized as "indication only" problems.

- At the time of inspector questioning on August 29, 2003, the licensee had not properly addressed equipment operability associated with the impact of the truck-operated cell switches in a non-seismically qualified configuration.

In response to the team's concerns, the licensee addressed equipment operability and determined that the truck-operated cell switches remained operable, but were degraded. The licensee considered the potential for truck-operated cell switch mis-positioning during an accident to be unlikely. The licensee also took prompt measures to jumper out the safety-related truck-operated cell switches from the affected circuits. The team reviewed the licensee's temporary modification and found it acceptable.

<u>Analysis</u>. The team determined that the issue regarding the truck-operated cell switches was of greater than minor significance because it affected the reactor safety mitigating systems objective to ensure the availability of systems that respond to an initiating event. The team utilized the Significance Determination Process, as described in NRC Manual Chapter 0609, to assess the finding's safety significance. In accordance Appendix A, Phase 1, Mitigating Systems Section, the team determined that the finding was of very low safety significance because it was a qualification deficiency confirmed not to result in loss-of-function in accordance Generic Letter 91-18, "Information to Licensee's Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," Revision 1.

<u>Enforcement</u>. The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI (Corrective Actions), which requires the licensee to take effective corrective measures to correct conditions adverse to quality, such as nonconforming and degraded conditions. Since the 2001 refueling outage, safety-related 4160 Vac breaker truck-operated cell switches were in a non-seismically qualified (over-travel) configuration, a condition adverse to quality. The issue is being treated as a noncited violation, consistent with Section V1.A.1 of the NRC's Enforcement Policy. The licensee entered this issue into their corrective action program as Problem Evaluation Request 203-3693 (NCV 05000397/2003009-02).

In general, the team found that the licensee effectively identified problems with the exception of some notable examples. The facility staff generated more problem evaluation reports during this evaluation period than the previous. Nevertheless, there have been a number of instances in which problems existed for a long period of time before problem root causes were identified and other instances where problems were identified but not entered into the corrective action program. Examples included a historically identified issue related to water leakage through floors into electrical equipment spaces, emergency diesel generator bearing degradation, service water low flow, and the truck-operated cell position switches discussed above.

b. Prioritization and Evaluation of Issues

(1) Inspection Scope

The team reviewed approximately 100 problem evaluation requests, and supporting documentation, including root-cause analyses, and analyses associated with justifications for continued operation, to ascertain whether the licensee's evaluation of the problems identified and considered the full extent of conditions, generic implications, common causes, and previous occurrences. In addition, the team reviewed problem evaluation requests to ascertain if the provisions of NRC Generic Letter 91-18, "Resolution of Degraded and Non-Conforming Conditions," and 10 CFR Part 50, Appendix B, were satisfied regarding timeliness of corrective action. Specific items reviewed are listed in the attachment.

(2) Assessment

Overall, the team found that the licensee appropriately prioritized and evaluated issues. Nevertheless, some notable exceptions involving weak engineering review of cause and extent of condition resulted in untimely or ineffective corrective actions. For example, excessive leakage past the Division II residual heat removal pump discharge check valve took approximately 7 years to resolve.

Another example was the weak evaluation of issues related to corrective actions for a reactor scram. The reactor scram occurred on June 26, 2000, when the C-phase current transformer for the differential relay protection system experienced a short-to-ground. This resulted in a main generator trip from a differential current condition, and this caused a subsequent reactor scram from full power. The cause of the short-to-ground was determined to be a wire in the C-phase circuit that had fretted and grounded, due most likely to induced 60Hz vibration on a conduit where the C-phase single lead wire was located, such that fretting occurred from contact by the wire, due to a sharp bend radius where the wire passed through a condulet fitting. The licensee corrected this by ensuring that the bend radius of the wire in the condulet fitting was sufficient to maintain proper insulation. The licensee did not conduct an extent of condition review as a result of this event, presuming a straightforward cause that did not need further review (Problem Evaluation Request 200-1043). Subsequent to this event another reactor scram from a similar cause occurred on June 30, 2003. The licensee noted the similarity of wire insulation fretting caused by contact with condulet fittings in current transformer circuitry and proceeded to do an extensive extent of condition review in Problem Evaluation Request 203-2578. As a result of this review, the licensee determined that replacement of single conductor wiring in the conduit with jacketed wire was warranted. This was done on all potentially affected wiring associated with current transformers in the electrical distribution system. The licensee's actions with regard to the June 26, 2000, reactor scram represented a missed opportunity to utilize extent of condition reviews to preclude recurrence of the reactor scram of June 30, 2003.

Further, the inspectors identified during 2003 that the licensee had failed to take prompt corrective measures to address a condition adverse to quality associated with the Division I emergency diesel generator and had failed to properly evaluate emergency diesel generator operability. Specifically, as early as 1998, plant vibration data indicated that one of the unit's generator bearings was significantly degraded and was continuing to degrade at an accelerated rate. The licensee's operability evaluation was inadequate because it relied on an inappropriate method to determine remaining bearing life. As a result of the NRC concerns, the licensee declared the emergency diesel generator inoperable and, ultimately, shut down the reactor to complete repairs. This issue was reported in NRC inspection report 05000397/2003-04.

Additionally, after errors were identified in the facility's alternate source term submittal, the technical justification to support the use of potassium iodide tablets as a compensatory interim measure was not well documented with bases premised upon the current Final Safety Analysis Report. This issue is further discussed in Section C, below.

The team identified no findings of significance in this area.

c. Effectiveness of Corrective Actions

(1) Inspection Scope

The team reviewed problem evaluation requests, followup assessments for operability, and self-assessments to verify that corrective actions, related to the issues, were identified and implemented in a timely manner commensurate with safety, including corrective actions to address common cause or generic concerns. A listing of specific documents reviewed during the inspection is included in the attachment to this report.

The team also reviewed problem evaluation requests, self-assessments, and licensee generated trending data to evaluate the licensee's performance with respect to the substantive human performance cross-cutting issue regarding procedural compliance described in the annual assessment letter (NRC Report 50-397/2002-01). The NRC noted improvement in the licensee's performance in that area during the next annual assessment letter (NRC Report 50-397/2003-01), but, since the majority of the findings that led to the cross-cutting issue occurred during the R-15 refueling outage, it was determined that the issue should remain open until the effectiveness of corrective actions could be evaluated during the R-16 refueling outage. Therefore, the team focused attention on the effectiveness of corrective actions implemented since R-15 refueling outage to improve human performance in the area of procedural compliance during the R-16 refuel outage. A listing of documents reviewed during the inspection is included in the attachment to this report

(2) Assessment

Introduction. A Green noncited violation was identified for an inadequate corrective action in accordance 10 CFR Part 50, Appendix B, Criterion XVI, “Corrective Action,” for the licensee’s failure to adequately implement a procedure revision intended to ensure that control room dose limitation requirements in accordance with 10 CFR Part 50, Appendix A, General Design Criterion 19, “Control Room,” were met.

Description. In September 2000, the licensee conducted a series of tests utilizing tracer gas decay methodology to determine the total in-leakage into the control room and the associated impact on the control room operators’ dose. These tests were performed in support of the licensee’s planned alternate source term license amendment submittal to the NRC. The test results indicated that the control room in-leakage was approximately 218 cubic feet per minute (cfm). This value exceeded the Final Safety Analysis Report design basis of 10.55 cfm. Additionally, the licensee determined that the design basis thyroid dose limit of 30 rem to the control room operators would have been exceeded during post-accident conditions with the as-found control room in-leakage. The licensee reported this issue to the NRC in Licensee Event Reports 2000-006-00 and 2000-006-01. The licensee also documented the issue in Problem Evaluation Request 200-1570.

The inspector reviewed the interim and long-term corrective actions as stated in Problem Evaluation Request 200-1570 to determine the timeliness and adequacy of the corrective actions. The licensee’s interim corrective action was to revise Procedure ABN-FAZ, “FAZ,” Revision 1, to direct the control room staff in the event of a ‘Z’ signal in conjunction with a low reactor water level of -161 inches to sample the control room atmosphere for I-131 and to distribute potassium iodide (KI) to all control room personnel if I-131 concentration was greater than 1.4E-7 $\mu$Ci/cc. A ‘Z’ signal would be generated if the reactor building exhaust vent radiation level exceeded 13 millirem per hour (mr/hr) while a low reactor water level of -161 inches corresponded to the top of the active fuel region. The licensee determined that a ‘Z’ signal in conjunction with reactor water level at the top of the active fuel region would be indicative of potential fuel damage and would be an adequate trigger for the commencement of I-131 sampling in the control room. With the distribution of KI to the control room staff, the licensee determined in Follow-up Assessment of Operability 200-1570 that the control room operators’ thyroid dose would be limited to 29.4 rem during design basis accident conditions. By limiting the control room operators’ thyroid dose to 29.4 rem through KI distribution, the licensee determined that the regulatory limit of a 30 rem thyroid dose as required in 10 CFR Part 50, Appendix A, General Design Criterion 19, “Control Room,” would be met. The licensee’s long-term corrective action was to submit to the NRC the planned alternate source-term license amendment and to accept the as-found control room in-leakage of 218 cfm “as is.” The licensee determined that by utilizing alternate source-term methodology with a control room in-leakage of 218 cfm that the control room operators’ doses would meet all regulatory requirements without KI distribution. The licensee submitted the alternate source-term license amendment request to the NRC on December 3, 2001.

On October 2, 2002, the licensee documented in Problem Evaluation Request 202-2772 that an atmospheric dispersion analysis, performed by a contractor, that was utilized in the alternate source term-license amendment request was incorrect. This analysis was also utilized to support the operability evaluation documented in Follow-up Assessment of Operability 200-1570 to justify the acceptability of KI distribution to the control room staff to limit the thyroid dose to acceptable levels. The licensee subsequently withdrew the alternate source-term license amendment request in November 2002 because of the error in this analysis in addition to other issues associated with the submittal. Additionally, the licensee qualitatively assessed the impact of the errors in the atmospheric dispersion analysis to justify the continued acceptability of the interim corrective action to distribute KI to control room staff during a design basis accident. The licensee planned to resubmit the alternate source term license amendment request by April 30, 2004.

The inspector identified three concerns associated with the licensee's interim corrective actions as described below.

- The licensee's reliance on a 'Z' signal and a low reactor water level of -161 inches as an entry condition for sampling the control room atmosphere for I-131 was inappropriate. The inspector determined that during a large break loss-of-coolant accident (LBLOCA), which was the most limiting design basis accident for control room dose, that a 'Z' signal would probably not be generated. The inspector noted that very early in the accident sequence of an LBLOCA that an 'F' signal or an 'A' signal would most likely be generated prior to a 'Z' signal. An 'F' signal would be generated on a high drywell pressure of 1.68 pounds per square inch, while an 'A' signal would be generated on a Low Reactor Vessel Water Level 2 trip of -50 inches. An 'F' signal or an 'A' signal would cause the reactor building exhaust vent to automatically isolate to mitigate any offsite release of radioactivity. However, the radiation monitors that were used to generate a 'Z' signal were also located in the reactor building exhaust vent. With the reactor building exhaust vent path isolated early in an LBLOCA, the radiation monitors would not be able to detect radiation levels indicative of an LBLOCA, which was the accident of concern and, thus, a 'Z' signal would not be generated. The inspector communicated this concern to the licensee who subsequently revised Procedure ABN-FAZ to initiate control room atmospheric sampling for I-131 on an 'A' signal in conjunction with reactor vessel level lowering to the top of active fuel region. The licensee documented this issue in Problem Evaluation Request 203-3643.

- The revision to Procedure ABN-FAZ did not reflect the procedure revision as proposed in Follow-up Assessment of Operability 200-1570, Revision 2. Specifically, Follow-up Assessment of Operability 200-1570 directed revising Procedure ABN-FAZ to use a valid 'Z' signal in conjunction with a valid 'F' signal to commence control room atmospheric sampling for I-131. However, the inspector determined that the actual revision, as documented in

Procedure ABN-FAZ, Revision 1, directed control room atmospheric sampling in the event of a valid 'Z' signal in conjunction with a low reactor water level of -161 inches. The inspector was concerned that the actions required in Follow-up Assessment of Operability 200-1570 to ensure that the dose to control room staff was mitigated were not accurately implemented in the procedure revision. The licensee could not provide a reason for the difference between the required actions as stated in Follow-up Assessment of Operability 200-1570 and the revision to Procedure ABN-FAZ. Additionally, the inspector noted that the interim corrective action to revise Procedure ABN-FAZ was not formally tracked in the licensee corrective action database as a Corrective Action Plan to Problem Evaluation Request 200-1570 in accordance with Procedure Site-Wide Procedures (SWP)-CAP-01, "Problem Evaluation Requests (PERs)," Revision 1. The licensee documented the inspector's concerns in Problem Evaluation Requests 203-3654 and 203-3638.

- The licensee did not adequately evaluate the errors, which were identified with the vendor supplied atmospheric dispersion analysis used to support Follow-up Assessment of Operability 200-1570. After the errors were identified, the inspector noted that the licensee performed a qualitative assessment of the errors, which was not documented to attempt to bound the total control room staff thyroid dose estimate. However, the inspector determined that the qualitative assessment utilized alternate source-term methodology to bound the dose estimate, while Follow-up Assessment of Operability 200-1570 utilized a source-term model consistent with the current licensing design basis. The inspector concluded that utilizing alternate source-term methodology to bound the control room dose estimate was not appropriate since the licensee was not licensed to incorporate alternate source-term as part of the design basis. When challenged by the inspector, the licensee conducted a formal quantitative assessment of the errors associated with the vendor supplied atmospheric dispersion analysis utilizing a source term model consistent with the current licensed design basis. The licensee concluded that with the errors associated with the atmospheric dispersion analysis included in the quantitative assessment, that the control room staff thyroid dose would be below the 30 rem licensed limit assuming KI distribution. The licensee documented the inspector's concern in Problem Evaluation Request 203-3664.

Analysis. The inspector determined that the licensee's failure to ensure that Procedure ABN-FAZ was adequately revised to ensure that KI would be distributed to the control room staff during a design basis accident was a performance deficiency. Specifically, the failure to distribute KI with the as-found control room in-leakage of 218 cfm early in a design basis accident would result in the thyroid dose to the control room staff exceeding the 10 CFR Part 50, Appendix A, General Design Criteria 19 limit of 30 rem. Because the finding affected the control room envelope, which is part of the reactor safety barrier integrity cornerstone, this finding was greater than minor. The inspector utilized Manual Chapter 0609, "Significance Determination Process," Appendix A, to determine the risk significance of the finding and determined that the issue was of very low risk significance (Green) because the finding only represented a degradation of the radiological barrier function provided for the control room.

Enforcement. On September 14, 2000, the licensee revised Procedure ABN-FAZ to implement an interim corrective action to ensure that the General Design Criteria 19 thyroid dose limit of 30 rem would be met following the identification of increased control room in-leakage of 218 cfm. On October 2, 2003, the inspector determined that the revision to Procedure ABN-FAZ was inadequate, in that, the interim corrective action relied on a 'Z' signal in conjunction with a low reactor water level of -161 inches. During a design basis loss-of-coolant accident a 'Z' signal would not be generated. On October 3, 2003, the licensee revised Procedure ABN-FAZ to adequately direct the control room staff to sample the control room atmosphere for I-131. Regulation 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," required that for significant conditions adverse to quality that measures shall assure that corrective action is taken to preclude repetition. Contrary to the above, the licensee failed to implement an adequate interim corrective action to ensure that General Design Criteria 19 thyroid dose limits were met. The inspector determined this to be a violation of 10 CFR Part 50, Appendix B, Criterion XVI. Because the failure to implement an effective corrective action was of very low safety significance and had been entered into the corrective action program in Problem Evaluation Request 203-3643, this violation is being treated as a noncited violation, consistent with Section VI.A of the NRC Enforcement Policy: Noncited Violation 05000397/2003009-01, Failure to Implement Adequate Corrective Action to Address Increased Control Room In-Leakage.

Human Performance

The team concluded that the licensee's corrective actions resulted in some improvement in human performance issues related to procedural compliance. Overall, through a focused effort of training and increased supervisory oversight, the licensee's rate of human performance errors with consequence was cut in half between R-15 and R-16. Additionally, the number of NRC findings related to procedural compliance was also cut in half between R-15 and R-16. The team found that corrective actions to address the substantive finding in the cross-cutting area of human performance had resulted in overall improvement during the 2003 refueling outage. Nevertheless, human performance continued to be a challenge in several areas such as maintenance rework, loss of shutdown cooling, and engineering reviews.

Finally, the team concluded that the effectiveness of corrective actions was acceptable but in need of improvement with regard to fully understanding the extent of problems and developing comprehensive and effective corrective actions.

d. Assessment of Safety-Conscious Work Environment

(1) Inspection Scope

The team interviewed six supervisors, three systems engineers, three maintenance craft personnel, and five program managers, including the employee concerns program coordinator. These interviews assessed whether conditions existed that would challenge a safety conscious work environment.

(2) Assessment

The team concluded, based on information collected from interviews, that these employees were willing to identify and enter issues into the corrective action program. The team found no findings of significance in this area.

4OA6 Meetings

Exit Meeting

On October 9, 2003, the team leader conducted an exit meeting with Mr. R. Webring, Vice President, Nuclear Generation, and other members of the licensee's staff. The plant management acknowledged the inspection findings. The licensee did not identify any material reviewed by the team to be proprietary.

# ATTACHMENT

## PARTIAL LIST OF PERSONS CONTACTED

Licensee

T. Altman, Performance Management
J. Bekhazi, Maintenance Manager
I. Borland, Radiation Protection Manager
D. Coleman, Manager, Performance Assessment and Regulatory Programs
Y. Derrer, Operations Experience Coordinator
J. Engbarth, Assistant, Vice President Technical Services
K. Engbarth, Quality Support Supervisor
A. Fahnestock, Training Supervisor
D. Feldman, Acting Plant General Manager/Acting Vice President Nuclear Generation
M. Ferry, Quality Support Supervisor
R. Feuerbacher, Reator/ Fuels Engineering Manager
R. Fuller, Reactor Maintenance Manager
J. Gillespie, Administrator Assistant
M. Humphreys, Engineering Manager
P. Inserra, Plant Engineering Manager
S. Jerrow, Operations Manager
C. King, Manager, Chemistry
D. Mand, Work Control Manager
C. McDonald, Acting Training Manager
T. Mitts, Root Cause Analyst
A. Mounger, Acting Chief Executive Officer
L. Poznanski, Operations Support Specialist
G. Prior, Root Cause Analyst
L. Pritchard, Problem Evaluation Request Coordinator
S. Rickter, Root Cause Analyst
S. Scammon, Resource Protection Manager
C. Sly, Licensing Engineer
S. Taylor, Plant Tracking Log Administrator
R. Webring, Vice President, Nuclear Generation

NRC

G. Repolgle, Senior Resident Inspector
B. Jones, Region IV Branch Chief, Division of Reactor Protection, Branch E

## ITEMS OPENED AND CLOSED

Opened and Closed

| | | |
|---|---|---|
| 50-397/03009-01 | NCV | Failure to implement adequate corrective actions to address increased Control Room In-Leakage |
| 50-397/03009-02 | NCV | Failure to promptly correct a condition adverse to quality associated with the 4160 VAC breaker truck-operated cell position switches |

DOCUMENTS REVIEWED

Self Assessments

SA-02-084 “Self-Assessment of the process for Classification of Action Requests as Equivalent Changes” 10/8/02

SA-2003-0012 “PDC Document Change Only Process” 5/8/03

SA-2003-0031 “Self-Assessment Design and Engineering Labor Estimate Accuracy” 4/30/03

SA-2003-0045 “Project Milestone Performance” 9/11/03

Integrated Performance Assessment Report 1/1-6/30/03

Problem Evaluation Request s:

PER 200-1570; (SPER) The Control Room Envelope Unfiltered Inleakage Exceeded the Maximum Allowed in Design Basis Document; September 13, 2000

PER 203-1532; (QA) Inaccurate Emergency Plan Time Estimates for the Plant and Nearby Facility Evacuation; May 9, 2003

PER 203-1531; (QA) Errors and Inconsistencies were Identified in the Emergency Plan Evacuation Time Estimate Study; May 9, 2003

PER 202-1452; Changes to Emergency Evacuation Process Could Impact Estimated Time for Exclusion Area Evacuation; May 10, 2002

PER 202–2372; Recent Access Road Construction Activity is Thought to Affect Site Evacuation Time Estimates; August 15, 2002

PER 203-3664; NRC IR 03-09 FAO 200-1570 May Have Been Affected by Errors in Scientech Calculation 19030-M-04, Revision 1; October 6, 2003

PER 203-3638; NRC IR 03-09; No Corrective Action Created for Compensatory Measures Associated with FAO 200-1570 (CREFS); October 2, 2003

PER 203-3654; NRC IR 03-09; The Required Actions for FAO 200-1570 Were Improperly Incorporated into Procedure ABN-FAZ; October 3, 2003

PER 203-3643; NRC IR 03-09; Compensatory Measure Prescribed for FAO 200-1570 is Inadequate for All Postulated Accident Conditions; October 2, 2003

PER 202-2772; Atmospheric Dispersion Results Provided by a Contractor are Incorrect; October 2, 2002

PER 202-2827; A Calculation Performed by a Contractor with Erroneous Results was Used for the AST Submittal to the NRC; October 9, 2002

PER 203-1866; Significant Water Hammer Noise During Restart of RHR-P-2A in Shutdown Cooling; May 21, 2003

PER 202-2984; Pressure Decay Trend Data for LPCS-V-3, & RHR-V-31C Indicate that They Leak Through Substantially; October 24, 2002

PER 203-2384; RHR-SYS-B Only Meets Technical Specification Requirement for Pressure Decay, Not Appendix R; June 14, 2003

PER 203-1224; (SPER) Requirements of Appendix R Safe Shutdown Fire Analysis not Clearly Defined in Procedure and Other Plant Documents; April 3, 2003

PER 203-1045; WO 01052884 Was Completed But Did Not Correct the PER 202-3450 Deficient SW-FI-61 Tubing Support Condition as Intended; March 31, 2003

PER 202-3450; ½" Tubing Clamp Unistrut Support for the SW-FI-61 Instrument Drain Valve Has Been Damaged; December 9, 2002

PER 297-0349; Spring Nuts Found Installed Incorrectly on Battery Rack for E-B1-1; 5/1/ 1997

PER 203-1049; Multiple Channel (Spring) Nut Installation Deficiencies Found for Supports Off of P5000 Unistrut Members; March 31, 2003

PER 297-0361; Battery Rack Unistrut Bolts Were Incorrectly Installed; April 30, 1997

PER 203-2560; Additional Deficient Unistrut Channel (Spring) Nut Installations Identified During Completion of PERA 203-1329-01; June 27, 2003

PER 203-1329; Potential Generic Installation Deficiencies for Unistrut Channel (Spring) Nut Installations on P5000 Unistrut Members; April 23, 2003

PER 202-3234; E-B1-2 Cell #18 Did Not Meet A & B Limits for Voltage as Discovered During ESP-B12-Q101; November 18, 2002

PER 203-0146; E-B1-2 Cell #18 Did Not Meet Category A & B Limits for Voltage During ESP-BAT-W101; January 15, 2003

PER 203-3084; During Performance of ESP-B11-Q101, Two Battery Cells Failed to Meet Category B Limits for Cell Voltage; August 18, 2003

PER 203-2411; Unexpected NSSSS Outboard Isolation and Interruption of Shutdown Cooling During the Performance of TSP-CONT/ISOL-B501

PER 200-0878; Two Maintenance Rule Functional Failure Determinations Were Found By the NRC Resident to be Non-Conservative; May 25, 2000

PER 200-1582; Mis-Classification of Maintenance Rule Program Functional Failures and Maintenance Preventable Functional Failures; September 14, 2000

PER 200-1583; M-Rule Performance Criteria Assumptions Not Supported by Subsequent Data; September 14, 2000

PER 200-0626; During Performance of ESP-B11-Q101 Four Cells Did Not Meet Category B Limits for Cell Voltage; April 12, 2000

PER 202-2428; The Pilot Cell of E-B1-1 Did Not Meet the Category A or B Limits. It Did Meet Category C Limits; August 21, 2002

PER 202-2761; Cell 83 of Battery Bank E-B2-1 Was Found to Have a Voltage Level Below the A and B Limits on the Quarterly Surveillance. The Cell Read 2.11 V; October 2, 2002

PER 203-3111; While Performing ESP-BAT-W101 PER 01058671 It Was Discovered That Pilot Cell #15 Was Below the Category C Limit of 2.07 VDC; August 20, 2003

PER 203-3125; ESP-B21-Q101 Was Inaccurate with Respect to Technical Specifications 3.8.6 Table 3.8.6-1; August 22, 2003

PER 297-1003; The Switch from the JCO Format to the FAO Format has Orphaned a JCO that is Being Actively Reviewed by the NRC; December 11, 1997

PER 203-1861; Loss of Shutdown Cooling - During Performance of WO 01059072-01 Contact B4 on Relay MS-RLY-K72 was Determined Vice MS-RLY-K72A; May 21, 2003

PER 203-3127; Failure to Comply With TSAS 3.6.1.3.C Completion Time; August 22, 2003

PER 203-2510 QA Adverse Trend – Human Performance RE Crane Ops R-16

PER 203-1095 Adverse Trend – Rad Protection Precursers to Declining Performance

PER 203-3533 Adverse Trend – High Rad Area Performance

PER 203-2649 Adverse Trend – Damage to Temporary Cables

PER 203-3537 Maintenance Industrial Safety PI Yellow for 3rd Month

PER 203-3529 RMC Laborers Exposure PI Red

PER 203-3528 RP Exposure PI Red

PER 203-3196 Adverse Trend – Personnel Error Rate for Operations Continues in Red

PER 201-1171 Inadvertent RHR Isolation

PER 203-2411 Unexpected Loss of Shutdown Cooling

PER 203-1861 Loss of Shutdown Cooling

PER 202-3471 Poor Root Cause Analysis for FDR-V-3, 4, and 15 blockage

PER 203-1789 Lack of Justification Documents for MSIV LLR Testing Methodology

PER 203-1130 Perceived Incongruency Between T.S. LCO 3.0.5 and 3.6.1.3

PER 203-1073 Revising FAO-202-3471 for Frequency of Cleaning FDR-V-3/4

PER 203-1014 Inconsistent Implementation of Effective Operability Determinations

PER 203-0201 NDE UT Data for Piping Btwn FDR-V-3 and 15 Indicates Wall Thinning

PER 203-0173 FDR-V-570 Excessive Leakage Concern

PER 203-0028 Metal Tag Left on Operator Spring of FDR-V-3

PER 202-3529 FDR-V-3/4 Equipment History Research

PER 202-3476 FDR-FT-38 Inoperable Due to Debris

PER 202-2932 Invalid Leak Rate Determination at FDR-FT-38

PER 201-1259 FDR-V-3 Failed Stroke Time Testing

PER 203-1447 New TSC Copier Not Evaluated for DG1 Loading Impact

PER 203-1464 Unauthorized Access Into Vital Area

PER 203-1516 Rad. Postings Found to be Inadequate by Management Tour

PER 203-1535 Unauthorized Access Into Vital Area

PER 203-1583 Unescorted Access Granted Prior to Completion of Background Check

PER 203-1640 Unauthorized Access Into Vital Area

PER 203-1659 Potentially Radioactive Tools Found in DG-2 Room

PER 203-1671 Safeguard Cabinet Lock Unsecured and Unattended

PER 203-1720 Outage Worker Granted RCA Access Without Radworker Training

PER 203-1758 Weld Record Issued and Welds Performed Without QC Review

PER 203-1807 RRC-V-20 Not Bench Tested Prior to Installation

PER 203-1846 PER 203-1671 Initiated Late

PER 203-1851 Unauthorized Access Into Vital Area

PER 203-1865 Four Individuals Contaminated During Maintenance

PER 203-1894 Four Safeguards Drawings Found Without Proper Designation

PER 203-1925 Foreign Material Control Lost During RFW-DT-1A Overhaul

PER 203-2003 RCIC-V-76 Minimum Stem Diameter Less Than Analysis Assumed

PER 203-2005 Unapproved Fasteners Were Installed on RCIC-MO-69

PER 203-2011 Incomplete Welds Inspected and Approved by QC Inspector

PER 203-2032 Unauthorized Access Into Vital Area

PER 203-2039 Unescorted Access Granted Without FFD Tracking

PER 203-2208 Plant Access Termination for Temporary Employee Occurred Late

PER 203-2299 Area Near RWCU Header Had Dost Rate >1000 mrem/hr at 30 cm

PER 203-2347 Security Weapon Left Unattended/Unsecure

PER 203-2453 Unauthorized Vehicle in Protected Area

PER 203-2551 Unauthorized Access Into Vital Area

PER 203-2579 As-Found Set Pressure Exceeded Nominal Value by >10%

PER 203-2645 RCIC Made Inoperable due to Inadvertent Closure of RCIC-V-63

PER 203-2680 Unescorted Vehicle

PER 203-2693 Inaccuracies/Inconsistencies Btwn RCIC License and Design Documents

PER 203-2732 Unauthorized Access Into Vital Area

PER 203-2885 Security Failed to Provide Adequate Security Measures

PER 203-2966 Multiple Termination Problems Found During CR Panel Clean/Inspect

PER 203-2995 Unauthorized Access Into Vital Area

PER 203-3209 Some Important OE Documents Are Not Being Tracked as OERPER's

PERs for Maintenance Rule Systems in A(1) Status (for period ending 7/31/03): 203-0644, 202-2774, 203-1423, 202-2280, 201-0695, 201-0744, 203-0353, 201-2869, 203-1424, 202-2260

Licensee Event Reports:

LER 2000-006-01; Plant Outside Design Basis for Control Room Emergency Filtration System Unfiltered In-Leakage Based Upon Tracer Gas Testing; December 6, 2000

LER 2003-001-00; Residual Heat Removal (RHR) B Train Potentially Inoperable During a Design Basis Event Due to Apparent Inability of System to Adequately Maintain Pressure as Assumed in Appendix R Analysis; June 2, 2003

LER 2003-005-00; Shutdown Cooling Isolation Caused by Procedure Deficiency; August 14, 2003

LER 05000397-2000-007  Manual Reactor Scram due to Loss of Condenser Vacuum

LER 05000397-2003-003  Loss of Shutdown Cooling

Follow-up Assessment of Operabilitys:

FAO 200-1570; The Control Room Emergency Filtration System (CREFS) is Operable but Non-Conforming, Based on Post-LOCA Administration of KI; September 13, 2000

Procedures:

ABN-FAZ; FAZ; Revision 5

PPM 13.2.1; Guidance for Administering Potassium Iodide (KI); Revision 15

PPM 8.3.417; Control Room Envelope Unfiltered Inleakage Test; Revision 0

OSP-WMA-B701; Control Room Ventilation System A Pressurization Flow Test; Revision 4

OSP-RHR-A702; RHR Loop B Keep Fill Integrity Test; Revision 1

ABN-FIRE; Fire; Revision 6

ABN-CR-EVAC; Control Room Evacuation and Remote Cooldown; Revision 6

PPM 10.2.10; Fastner Torque and Tensioning; Revision 19

TSP-CONT / ISOL-B501; Containment Isolation - LSFT; Revision 4

ESP-B11-Q101; Quarterly Battery Testing 125 VDC E-B1-1; Revision 5

PPM 1.3.66; Operability Determination

CAPI 1.3 Station Event Free Clock Program

CAPI 1.4 Station Human Performance Event Rate

CAPI 1.6 Human Performance Review Board

Other:

Calculation 19030-M-04; Control Room, EAB, and LPZ Doses Following a LOCA; Revision 1

PTL R156729; (Annually) Review and Update Population Study and Evacuation Time Estimate in Conjunction with Annual EPlan Review; March 3, 1999

PTL A188372; Perform and Evaluation (Time-Motion) of Existing Site Population Density to Determine if Existing Evacuation Time Estimates are Appropriate; May 8, 2002

PTL R156728; (Annually) Review and Update the EPlan. Coordinate Review with Offsite Agencies, Including DOE, PER 0654 P.3, 10 CFR 50 App E; March 3, 1999

POC Meeting Minutes / Activities 00-43; October 25, 2000

SE-00-0054; PER 200-1570 Disposition As-Is for Control Room Ventilation System; October 12, 2000

Night Order 180; September 13, 2000

C-31637; NCS Corporation Control Room Envelope Inleakage Testing at Columbia Nuclear Station 2000 - Final Report; October 16, 2000

PTL A194426; Develop Tests to Ensure the ECCS Systems Will Remain Pressurized Long Enough for the Keep Fill Pumps, On Loss of Normal Power to be Re-energized by the Diesel Generators; November 26, 2002

Calculation ME-02-03-08; Pressure Calculation for Vapor Bubble Formation in RHR-A and RHR-B Trains; May 30, 2003

Night Order 528; October 2, 2003

PTL H170190; During the Upcoming M-Rule Self Assessment, Perform a Review of the FF/MPFF Determinations to Find Out if There Are Other Non-Conservative Determinations; June 26, 2000

G02-01-116; Columbia Generating Station, Operating License NPF-21 Resubmittal Plan - Request for Amendment to Secondary Containment and Standby Gas Treatment System Technical Specifications; August 16, 2001

G02-02-136; Columbia Generating Station, Docket No. 30-397 License Amendment Request - Alternative Source Term Response to Request for Additional Information; August 29, 2002

CGS R-16 Daily Outage Fliers

Energy Northwest NEWS – August 21, 2003 (Article Re: Drywell Cleaning)

Periodic Functional Area Report – February 28, 2001

STOP WORK ORDER for PER 203-2770-11

PERRG Meeting Agenda and Attachments 9/30/03

PERRG Meeting Agenda and Attachments 10/1/03

GIH 2.9.2 Performance Indicators - Draft

Bi-Monthly Trend Report for Human Performance – May and June 2003

CGS Quest for Excellence Program – Draft Document

OI-9 #1065

OI-9 #3938

WOT #010554801

WOT #010554701

R-16 PER Trending by Selected Categories (OE)

R-15 Outage Assessment

R-16 Outage Assessment Draft

Human Performance Analysis Memos by Gary Weimer

Station Personnel Error Rate Graph – July 1999 to August 2003

CAP Monthly Indicator Report for August 2003 and September 2003

External Operating Experience Documents and Actions Tracking Matrix

Additional PER's Reviewed:

| | | | | | |
|---|---|---|---|---|---|
| 203-1480 | 203-1481 | 203-1489 | 203-1542 | 203-1545 | 203-1546 |
| 203-1641 | 203-1645 | 203-1647 | 203-1664 | 203-1674 | 203-1683 |
| 203-1686 | 203-1694 | 203-1700 | 203-1703 | 203-1705 | 203-1718 |
| 203-1741 | 203-1742 | 203-1749 | 203-1761 | 203-1786 | 203-1800 |
| 203-1804 | 203-1831 | 203-1855 | 203-1861 | 203-1871 | 203-1874 |
| 203-1877 | 203-1880 | 203-1893 | 203-1912 | 203-1940 | 203-1945 |
| 203-1946 | 203-1974 | 203-1985 | 203-2033 | 203-2047 | 203-2053 |
| 203-2061 | 203-2076 | 203-2080 | 203-2081 | 203-2082 | 203-2110 |
| 203-2122 | 203-2157 | 203-2160 | 203-2180 | 203-2207 | 203-2212 |
| 203-2215 | 203-2233 | 203-2239 | 203-2253 | 203-2258 | 203-2269 |
| 203-2324 | 203-2325 | 203-2339 | 203-2345 | 203-2359 | 203-2361 |

| | | | | | |
|---|---|---|---|---|---|
| 203-2432 | 203-2440 | 203-2470 | 203-2475 | 203-2506 | 203-2510 |
| 203-2513 | 203-2556 | 203-2602 | 203-2613 | 203-2688 | 203-2794 |
| 203-2800 | 203-2819 | 203-2832 | 203-2855 | 203-2871 | 203-2876 |
| 203-2882 | 203-2913 | 203-2976 | 203-2977 | 203-2979 | 203-2994 |
| 203-2997 | 203-3038 | 203-3050 | 203-3067 | 203-3068 | 203-3083 |
| 203-3100 | 203-3120 | 203-3125 | 203-3147 | 203-3171 | 203-3173 |
| 203-3175 | 203-3204 | 203-3205 | 203-3257 | 203-3273 | 203-3288 |
| 203-3289 | 203-3295 | 203-3315 | 203-3319 | 203-3345 | 203-3367 |
| 203-3370 | 203-3373 | 203-3377 | 203-3379 | 203-3386 | 203-3415 |
| 203-3416 | 203-3419 | 203-3423 | 203-3435 | 203-3443 | 203-3461 |
| 203-3467 | 203-3475 | 203-3476 | 203-3480 | 203-3495 | 203-3505 |
| 203-3509 | 203-3515 | 203-3516 | 203-3564 | 203-3570 | |

Work Orders:

01013296
01013299
01038152
01051159
01051160
01057842
01058462
01058522
01058749
01058845
01059072
01059101
01059102
01059231