UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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April 12, 2006

J. V. Parrish (Mail Drop 1023)
Chief Executive Officer
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P.O. Box 968
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SUBJECT: COLUMBIA GENERATING STATION - NRC SUPPLEMENTAL INSPECTION REPORT 05000397/2006009

Dear Mr. Parrish:

On March 3, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Columbia Generating Station. The enclosed inspection report documents the inspection findings which were discussed on March 2, 2005, with Mr. D. Atkinson, Vice President, Nuclear Generation, and other members of your staff.

The inspections examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspector reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, no findings of significance were identified.

In accordance with 10 CFR 2.390 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.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

Docket: 50-397
License: NPF-21

Enclosure:
NRC Inspection Report 05000397/2006009

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-397
License:	NPF-21
Report:	05000397/2006009
Licensee:	Energy Northwest
Facility:	Columbia Generating Station
Location:	Richland, Washington
Dates:	February 27 through March 3, 2006
Inspector:	T. R. Farnholtz, Senior Project Engineer, Branch A, Division of Reactor Projects (DRP)
Approved By:	C. E. Johnson, Chief, Project Branch A, DRP
ATTACHMENT:	Supplemental Information

SUMMARY OF FINDINGS

IR05000397/2006009; 2/27/2006 - 3/3/2006; Columbia Generating Station. Inspection Procedure 95001 Supplemental Inspection.

The report covered a one-week period of inspection by a region-based inspector. No violations were identified. 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.

Cornerstone: Initiating Events

The U.S. Nuclear Regulatory Commission performed this supplemental inspection to assess the licensee's evaluation associated with a performance indicator (Unplanned Scrams per 7000 Critical Hours) that crossed the Green-White threshold in the second quarter 2005. The primary reason for this performance indicator being characterized as White was three unplanned scrams in the third quarter of 2004 and two additional unplanned scrams in the second quarter of 2005. Taken together, these events drove the performance indicator from a value of 0.0 scrams per 7000 critical hours as reported in the second quarter of 2004, to a value of 4.9 as reported four quarters later in the second quarter of 2005. The Green-White threshold value is 3.0 scrams per 7000 critical hours. The White-Yellow threshold value is 6.0 scrams per 7000 critical hours. Two of the unplanned scrams were caused by component failures in the turbine digital electro-hydraulic control system. The other three unplanned scrams were caused by a low reactor pressure vessel water level condition following the loss of an operating reactor feed pump. The licensee determined that the underlying cause of all five unplanned scrams was unresolved issues with plant equipment. During this supplemental inspection, performed in accordance with Inspection Procedure 95001, the inspector determined that the licensee, in general, adequately determined the root causes and significant contributing causes of the unplanned scrams and established appropriate corrective actions to prevent recurrence. In addition, the licensee performed an apparent cause evaluation to determine the common causes for these five events.

Report Details

01 INSPECTION SCOPE

The U.S. Nuclear Regulatory Commission (NRC) performed this supplemental inspection to assess the licensee's evaluation associated with a performance indicator (PI) (Unplanned Scrams per 7000 Critical Hours) that crossed the Green-White threshold in the second quarter 2005. The PI returned to Green in the third quarter 2005.

The primary reason for this PI being characterized as White was three unplanned scrams in the third quarter of 2004 and two additional unplanned scrams in the second quarter of 2005. Taken together, these events drove the performance indicator from a value of 0.0 scrams per 7000 critical hours as reported in the second quarter of 2004, to a value of 4.9 as reported four quarters later in the second quarter of 2005. The Green-White threshold value is 3.0 scrams per 7000 critical hours. The White-Yellow threshold value is 6.0 scrams per 7000 critical hours. Two of the unplanned scrams were caused by component failures in the main turbine digital electro-hydraulic control system. The other three unplanned scrams were caused by a low reactor pressure vessel water level condition following the loss of an operating reactor feed pump.

This supplemental inspection was focused on the five unplanned scrams that occurred between July 30, 2004, and June 23, 2005.

02 EVALUATION OF INSPECTION REQUIREMENTS

02.01 <u>Problem Identification</u>

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

This supplemental inspection focused on five reactor scrams which took place between July 30, 2004, and June 23, 2005. Three of these scrams were automatic and two were initiated manually prior to an automatic signal being generated on low reactor pressure vessel water level. All five scrams were self-revealing.

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

Two of the five scram events reviewed were caused by a failed electronic circuit card in the main turbine digital electro-hydraulic control system. These events were the result of a sudden and unexpected component failure in a system that does not have a redundant or backup system capable of continued control of the main turbine if such a failure occurs. Failures of this type in a system vulnerable to single point failures are not predictable and are random in nature. Therefore, the inspector considered the conditions leading to these events to not have existed prior to the actual event (scram). Also, no reasonable opportunities for prior identification existed for this type of failure.

The reactor scram event of August 15, 2004 was the result of a trip of the in-service reactor feed pump caused by a high water level in the drain tank. The control room operators initiated a manual reactor scram from 18 percent power when the feed pump tripped. The identified cause of the high water level in the drain tank was a displacement of inventory from the condensate storage tanks to the condenser hotwell due to the associated hotwell level controller being adjusted to the high end of the band. Also mentioned in the Problem Evaluation Request (PER 204-1030) was a reactor feed pump suction valve with seat leakage that was contributing to the high water levels in the condensate system. The inspector considered the conditions for this scram to have existed prior to the reactor startup with the plant operators trying to control the condensate system inventory using established water management guidance. A more thorough understanding of the way this system functions under these conditions combined with a better material condition of the leaking feed pump suction valve could have prevented this event. Prior identification was possible if a more questioning attitude had been adopted.

The reactor scram event of August 17, 2004, was a manual scram from 20 percent power initiated following a trip of the only running reactor feed pump. The trip of the feed pump was caused by control room operators improperly filling a feed water heater with condensate following maintenance resulting in a low reactor feed pump suction pressure condition. The inspector concluded that this event was the result of personnel error exacerbated by the lack of a Boiling Water Reactor (BWR) Owners Group recommended time delay on the reactor feed pump low suction pressure trip. Also, the operators demonstrated a lack of understanding of the feed and condensate system. The specific conditions for this scram did not exist prior to the actual event. Prior identification was not a reasonable possibility given these circumstances.

The reactor scram event of June 23, 2005 was an automatic scram from 24 percent power following a loss of feed water injection flow. Reactor feed pump B was suppling feed flow to the reactor pressure vessel (RPV) and the A feed pump was running at lower speed and not developing enough head to inject into the RPV. An electrical technician was assigned to restore a previously defeated low suction pressure trip of the reactor feed Pump B. During this evolution, an error was made which resulted in completing the circuit and causing the feed Pump B to trip on low suction pressure. Feed Pump A did not respond fast enough to avoid the reactor scram. As in the August 17, 2004, scram described above, the inspector concluded that this event was the result of personnel error exacerbated by the lack of a BWR Owners Group recommended time delay on the reactor feed pump low suction pressure trip. The specific conditions for this scram did not exist prior to the actual event. Prior identification was not a reasonable possibility given these circumstances.

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

The two reactor scram events involving the main turbine digital electro-hydraulic control system were documented in NRC Inspection Reports 05000397/2004004 and 0500397/2005003 with no compliance issues identified. The August 15 and 17, 2004, scram events were documented in NRC Inspection Report 05000397/2004004 as self-revealing findings of very low safety significance (Green). The June 23, 2005, scram event was documented in NRC Inspection Report 05000397/2005003 as a self-revealing finding of very low safety significance (Green).

02.02 Root Cause and Extent of Condition Evaluation

a. Evaluation of methods used to identify root cause(s) and contributing cause(s)

To evaluate these five reactor scram events, the licensee utilized the following root cause analysis techniques:

History Review and Analysis
Cause and Effect Analysis
Design Review and Analysis
Equipment Failure Analysis
Difference Analysis
Barrier Analysis
Fault Tree Analysis
Human Error Analysis
Failure Modes and Effects Analysis
Event and Causal Factor Analysis
Change Analysis

The guidance for the use of cause determination techniques is contained in Site Wide Procedure SWP-CAP-02, "Cause Determination," Revision 3. This procedure describes the use of change analysis, barrier analysis, event and causal factor charting, fault tree analysis, and task analysis. In addition, other methods are mentioned including Management Oversight Risk Tree (MORT) analysis. With regard to MORT analysis, SWP-CAP-02 states: "MORT analysis is a time consuming and complex process and requires knowledgeable personnel with extensive training. If it becomes necessary to perform a MORT analysis for a significant event with broad management concerns, the responsible manager should arrange for a trained analyst to lead the investigation."

The inspector discussed the root cause determination process with the corrective action program supervisor. There are five individuals on site that are considered qualified to perform root cause analysis determinations. The qualification requirements for these individuals consist primarily of actively participating in root cause determinations under the instruction of a qualified analyst. No formal or classroom training is provided to learn the techniques involved.

The inspector considered the licensee's root cause determination program to be informal in that the specific techniques to be applied to a particular issue is not specified by procedure. The training provided to the root cause analysts is not sufficient to ensure an adequate and consistent knowledge level for the use of all the available methods. None of the five scram events that were examined used the more formal MORT analysis or similar method to determine the root cause(s). While not required by procedure, the more formal MORT analysis may have provided additional insights to station management such that some of these scrams could have been avoided. The inspector noted that the most recent root cause analysis performed for a reactor scram (PER 205-0428) did exhibit an adequate level of knowledge on the use of the specified methods. Also, recent changes to the corrective action program did address these concerns by establishing additional post analysis reviews.

As a result of the lack of formality, there was an inconsistent quality of the analysis performed across these five events. For example, the root cause analysis performed for the August 17, 2004, scram event (PER 204-1042) failed to establish the correct root cause of the scram. The methods incorporated to determine the root cause for this event were change analysis, barrier analysis, event and causal factor charting, equipment failure analysis, and human error analysis. The root cause was identified as failure of plant operators to demonstrate strict adherence to procedural guidance. In actuality, the problem went much deeper than this and should have included other considerations such as; (1) the cause of the lifting of feedwater system relief valves; (2) possible inadequate or incomplete procedures; (3) inadequate operator training or understanding of the feed and condensate system; (4) or the failure of the station to incorporate the BWR Owners Group Scram Frequency Reduction Committee recommendation to install a time delay on the feed water pump low suction pressure trip; and (5) stagger the feed water pump low suction pressure trip set points or respective time delays. None of these possibilities were considered which resulted in the failure to establish effective corrective actions to address the problems.

The licensee did recognize this inadequate root cause analysis during the root cause determination effort for the June 23, 2005, reactor scram event. The root cause determination for this event (PER 205-0428) was significantly more in-depth and did capture the relevant considerations. However, the inspector noted that this PER did not discuss the reasons for the inadequate root cause determination in PER 204-1042 and that no separate effort was made in the corrective action program to correct this condition at the time, although changes to the corrective action program had been made through management initiative to address these issues. The licensee did generate a Condition Report (2-06-01556) to capture this observation of an inadequate root cause analysis effort in PER 204-1042 during the time of this inspection.

An example of a narrowly focused root cause determination was identified in the case of the August 15, 2004, reactor scram event (PER 204-1030). This effort focused almost exclusively on the condenser hotwell level controller and the way it functions when set at the high end of the band. The method used to establish the root cause of this event was barrier analysis. This PER does consider equipment issues (a leaking reactor feed pump suction valve) and procedural issues (the water management plan) but does not address other possibilities such as human performance, inadequate operations procedures, or inadequate operator training. Also, none of these factors were cited in the root cause and contributing causes for this event. Instead, the cause was centered entirely on the hotwell level controller response. Operations personnel did have some procedural guidance available. Procedure PPM 3.1.1, “Master Startup Checklist” contained an instruction to verify that hotwell level is in the normal operating band with one hotwell level controller in automatic and the standby controller in manual set at 50 percent output. This step had been performed but at a time when different plant conditions existed. When plant conditions changed, operators did not question the deviation from this requirement when the hotwell level controller in manual was set at the high end of the band. In addition, PPM 3.1.1 states to verify that sealed-in and disabled annunciators have been evaluated for reactor plant startup. The sealed-in annunciator for high hotwell level prior to plant startup was not thoroughly evaluated since the hotwell level was high as a result of excessive inventory in the system. Plant operators did not question the off normal condition of the feed and condensate system and the effect this would have on plant startup. The root cause determination did not consider the possibility of inadequate operator understanding of the feed and condensate system as a whole or the contributing factor of a feed water pump suction valve that was leaking past its seat adding inventory to an already excessively full system when establishing the root and contributing causes for this event. During the inspection, the inspector reviewed Work Order 01061163 to verify that the leaking feed water pump suction valve, COND-V-147B, had been repaired. This work was completed during Refueling Outage 17 in 2005.

The inspector concluded that the widely varying quality of the root cause evaluations reviewed was due primarily to a lack of rigor and formality in the methods used. With the exceptions of the observations noted above, the inspector considered the licensees root cause evaluations to be adequate to determine the root causes and significant contributing causes of the unplanned scrams. The licensee did perform an apparent cause evaluation (PER 205-0473) to establish common causes and common contributing causes for these five scram events. This evaluation adequately captures the technical and human performance issues for these events but did not identify issues related to the quality of the root cause evaluations. However, these issues were addressed through changes to the corrective action program

b. Level of detail of the root cause evaluation

Procedure SWP-CAP-02 provides guidance on the level of detail to be documented and the elements that should be included in the problem evaluation request. In general, the PER is to provide sufficient information to adequately determine the outcome of the analysis. In some of the cases reviewed, the PER evaluation adequately met this requirement by describing the results of the analysis and referred to attachments or other locations for the details of the root cause analysis. However, one example of lack of detail in this regard was identified. The root cause determination associated with the reactor scram event of August 17, 2004, (PER 204-1042) lists the root cause determination methods used but provides little detail as to the specifics of each effort. In the case of the change analysis, barrier analysis, or event and causal factor analysis, no specifics are included at all. For the equipment failure analysis and human error analysis, some information is provided along with a discussion of the problems identified during the evaluation. The inspector considered this effort to be lacking in detail which resulted in an inadequate cause determination.

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

The licensee’s evaluations included a review of internal (Columbia Generating Station) and external operating experience. In general, these sections included relevant operating experience along with a discussion of how this applies or does not apply to the specific situation. However, in one case these sections were just listings of internal and external operating experience documents with no discussion or context. This was the evaluation performed for the August 17, 2004, reactor scram event (PER 204-1042). The inspector considered this to be of no value in establishing if any operating experience could have provided some insight into this event.

d. Consideration of potential common cause(s) and extent of condition of the problem

The licensee performed an apparent cause evaluation (PER 205-0473) after the performance indicator crossed the Green-White threshold to determine if there were any common cause issues associated with these five reactor scram events. As apparent common causes, this evaluation did identify the single point vulnerabilities which resulted in the two scram events caused by the main turbine digital electro-hydraulic control system card failures. Also, the evaluation stated that the station did not take aggressive actions to implement recommendations from the BWR Owners Group Scram Frequency Reduction Committee or to eliminate single point vulnerabilities until recently (June 2005). In addition, the evaluation identified that problems with the feed and condensate system valves resulted in conditions that increased the potential for human error and, combined with inappropriate human actions, ultimately led to the reactor feed pump trip. As a common contributing cause, the evaluation stated that the station's apparent cause and root cause analysis and corrective actions from previous similar and related events were ineffective. The cause analysis and corrective actions failed to effectively address fundamental driving causes of events in a timely manner. The inspector agreed with these assessments but noted that the evaluation did not specify any corrective actions to address these common causes and common contributing causes. The evaluation makes a statement that recent improvements in the station's corrective action program to develop, approve, and implement effectiveness reviews for root cause analysis corrective actions to prevent recurrence address this cause. Other changes to the corrective action program included improvements in oversight activities such as the establishment of Department Corrective Action Review Boards (D-CARBs) and enhancements to the Corrective Action Review Board (CARB) charter. The inspector concluded, that while these changes had been made, effectiveness reviews and post analysis reviews did not address the specific causes identified in the apparent cause evaluation such as human performance, plant component material condition, or the use of operating experience such as the BWR Owners Group recommendations. However, the inspector noted that the most recently performed root cause analysis done for the June 23, 2005, reactor scram event (PER 205-0428) was of significantly higher quality, contained more detail, and provided more fundamental insights as to the cause of the event than previous evaluations. The causes and corrective actions specified in this evaluation were also applicable to the previous scram events involving the feed and condensate system. The inspector concluded that these actions combined with the corrective action program changes describes above adequately addressed the weaknesses identified during this inspection.

The licensee's procedure for performing cause determinations (SWP-CAP-02) provides specific information on the definition and purpose of extent of condition and extent of cause considerations. The inspector noted that the definitions and requirements included in this procedure, which included the specific questions to be addressed by the evaluation for extent of condition and extent of cause, were addressed with varying degrees of detail and specificity. In some cases, the scope of the extent of condition discussion was confined to a specific issue. For example, the analysis done for the August 15, 2004, reactor scram event (PER 204-1030) limited the extent of condition discussion to identifying other controllers where performance has been unreliable. This was in reference to the hotwell level controller that was identified as having been adjusted to the high end of the band resulting in a reduced response capability. The extent of condition discussion did not address any other potential conditions that may have existed which may have been a problem such as inadequate operations procedures or inadequate training.

In addition, parts of some extent of condition discussions did not appear to be on point. For example, the extent of condition discussion contained in the evaluation for the June 23, 2005 reactor scram event (PER 205-0428) has an extensive writeup concerning the station's varying level of quality of procedures. The evaluation stated that this and other events demonstrated an over-reliance on procedures instead of a strong self-verification of conditions, required actions, results, and actual risk and consequences in order to take appropriate actions to identify and mitigate risk. The inspector understood the licensee's point of encouraging a strong understanding of the task being performed and the importance of a questioning attitude but was concerned that this statement sent an inappropriate message to plant personnel to not trust the procedures being utilized. If station procedures are not adequate to achieve the desired result, a specific corrective action would be appropriate to address this concern. No specific procedures were specified in this evaluation.

02.03 Corrective Actions

a. Appropriateness of corrective action(s)

In general, the specified corrective actions were appropriate to the root and contributing causes that were identified in the root cause determination evaluations. However, as stated above, in some cases the root cause determination was not adequate to capture all aspects of the problems leading to the event. As a result, the specified corrective actions may not have addressed all possible causes and contributing causes. The licensee did identify in their apparent cause evaluation performed in response to the White performance indicator that corrective actions from previous similar and related events were ineffective. The inspector agreed with this assessment and concluded that at least two of the reactor scram events associated with the trip of the reactor feed pump to have been preventable had effective corrective actions been established following the August 15, 2004, reactor scram event and if the recommendations of the BWR Owners Group had been adopted in a timely manner. The inspector noted that the corrective actions identified in the most recent event of June 23, 2005, (PER 205-0428) did address aspects of this type of reactor scram event involving the feed and condensate system that would be effective in preventing recurrence.

Three of the five PERs reviewed did not contain corrective actions specifically tied to an identified root or contributing cause. This increased the probability that the specified corrective actions would not be successful in preventing recurrence because they lacked focus and may not have addressed the identified causes. The inspector attributed this administrative issue to the inconsistent quality of the evaluations performed.

b. Prioritization of corrective actions

The inspector did not identify any specific methods utilized to prioritize the specified corrective actions based on risk significance or regulatory compliance. However, no examples of inappropriate prioritization were noted. The inspector considered the prioritization of the established corrective actions to be consistent with risk consequences.

c. Establishment of schedule for implementing and completing the corrective actions

The licensee established adequate schedules for completion of the specified corrective actions. As appropriate, some corrective actions were tied to scheduled refueling outages while others were more short term such as procedure revisions or training updates. The inspector did not identify any specific concerns with the scheduling or completion of established corrective actions.

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

The licensee established effectiveness reviews for each of the evaluations reviewed. In some cases, the effectiveness reviews were relatively long term due to the scheduled implementation of the corrective actions. For example, the effectiveness review for the evaluation associated with the reactor scram of June 15, 2005, (PER 205-0424) required a self-assessment to be performed in six months to ensure progress on replacement of the main turbine digital electro-hydraulic control system to address the single point vulnerability issues associated with this system. The inspector interviewed station personnel involved in this effort and determined that progress was being made for implementation during the next scheduled refueling outage. The evaluation specified the method, attributes, success criteria, and timing of this action in specific terms. The inspector identified no concerns in this area.

03 **MANAGEMENT MEETINGS**

Exit Meeting Summary

On March 2, 2006, the inspector (T. Farnholtz) presented the inspection results to Mr. D. Atkinson, Chief Executive Officer, and members of his staff who acknowledged the findings. The inspector confirmed that proprietary information was provided or examined during the inspection and returned at the conclusion of the inspection.

ATTACHMENTS

Persons Contacted

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S. Wood, Supervisor, System Engineering

Documents Reviewed

Problem Evaluation Request 204-0972
Problem Evaluation Request 204-1030
Problem Evaluation Request 204-1042
Problem Evaluation Request 205-0424
Problem Evaluation Request 205-0428
Problem Evaluation Request 205-0473
Problem Evaluation Request 204-0843
Problem Evaluation Request 205-0443
Problem Evaluation Request 204-0969

Site-Wide Procedure SWP-CAP-02, “Corrective Action Program, Cause Determination,” Revision 3

Condition Report 2-04-03232
Condition Report 2-06-01556

Columbia Generating Station Single Point Vulnerability Identification and Reduction Project Plan

BWR Owners Group Scram Frequency Reduction Committee Report dated November 18-21, 2002

Work Order 01061163

Acronyms

BWR	boiling water reactor
CAP	corrective action program
CARB	corrective action review board
D-CARB	department corrective action review board
DEH	digital-electro hydraulic
MORT	management oversight risk tree
NRC	Nuclear Regulatory Commission
PER	problem evaluation request
PI	performance indicator
RPV	reactor pressure vessel
SWP	site wide procedure