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

June 2, 2003

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

SUBJECT: COLUMBIA GENERATING STATION - NRC TRIENNIAL FIRE PROTECTION INSPECTION REPORT 50-397/03-02

Dear Mr. Parrish:

On March 20, 2003, the NRC completed the onsite portion of the subject inspection at your Columbia Generating Station. Review of additional documentation provided to the NRC subsequent to the onsite inspection was performed from March 31 through April 15, 2003. The enclosed report documents the inspection findings, which were discussed with Mr. Scott Oxenford, and other members of your staff on April 22, 2003.

This triennial fire protection inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

During the inspection, an apparent violation of NRC requirements was identified regarding the vulnerability of safe shutdown equipment to fire induced multiple spurious actuations. These circuit vulnerabilities could, under certain postulated fire scenarios, adversely affect the ability to achieve and maintain safe shutdown of the facility. It is the NRC's understanding that you do not consider these vulnerabilities to be violations of NRC requirements. In order to allow the industry to develop an acceptable approach to resolving this issue, that the NRC can endorse, the NRC will defer any enforcement action relative to these matters while the staff evaluates Nuclear Energy Institute's proposed resolution methodology. In addition, this will provide you time to implement the resolution methodology, once approved, since you took compensatory measures for the identified vulnerabilities.

Based on the results of this inspection, the NRC has identified two findings whose final significance has not yet been determined under the risk significance determination process, but is known to be at least Green (very low safety significance). These findings are being treated as unresolved items, pending determination of their significance. These unresolved items are described in the subject inspection report. In addition, the NRC identified a finding that had very low safety significance (Green). This finding is being treated as a noncited violation, consistent with Section VI.A of the Enforcement Policy. The noncited violation is described in the subject inspection report and was not cited because of very low safety significance and that adequate compensatory measures were implemented. If you contest the violation or significance of the noncited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011; 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/

Charles S. Marschall, Chief
Engineering and Maintenance Branch
Division of Reactor Safety

Docket: 50-397
License: NPF-21

Enclosure:
NRC Inspection Report
50-397/03-02

cc w/enclosure:
Rodney Webring (Mail Drop PE04)
Vice President, Nuclear Generation
Energy Northwest
P.O. Box 968
Richland, Washington 99352-0968

Albert E. Mouncer (Mail Drop PE01)
Vice President, Corporate Services/
General Counsel/CFO
Energy Northwest
P.O. Box 968
Richland, Washington 99352-0968

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

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

Christina L. Perino (Mail Drop PE20)
Manager, Licensing
Energy Northwest
P.O. Box 968
Richland, Washington 99352-0968

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

Dale K. Atkinson (Mail Drop PE08)
Vice President, Technical Services
Energy Northwest
P.O. Box 968
Richland, Washington 99352-0968

Thomas C. Poindexter, Esq.
Winston & Strawn
1400 L Street, N.W.
Washington, D.C. 20005-3502

Bob Nichols
Executive Policy Division
Office of the Governor
P.O. Box 43113
Olympia, Washington 98504-3113

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

Electronic distribution by RIV:
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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION

REGION IV

Docket:	50-397
License:	NPF-21
Report:	50-397/03-02
Licensee:	Energy Northwest
Facility:	Columbia Generating Station
Location:	Richland, Washington
Dates:	March 10 through April 15, 2003
Lead Inspector:	R. P. Mullikin, Senior Reactor Inspector Engineering and Maintenance Branch
Inspectors:	R. L. Nease, Senior Reactor Inspector Engineering and Maintenance Branch G. D. Replogle, Senior Resident Inspector Project Branch E
Approved By:	Charles S. Marschall, Chief Engineering and Maintenance Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR05000397/2003-02; Energy Northwest; 3/10-20/2003 and 3/31/2003 through 4/15/2003; Columbia Generating Station, Triennial Fire Protection Inspection

The inspection was conducted by two regional inspectors and one senior resident inspector. The inspection identified one green finding and two unresolved item which were violations of NRC regulatory requirements. 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.

Cornerstone: Mitigating Systems and Barrier Integrity

Inspector Identified Findings

Cornerstone: Mitigating Systems

- TBD. The inspectors identified a violation of Technical Specification 5.4.1.d (inadequate procedure) because Procedure ABN-CR-EVAC, “Control Room Evacuation and Remote Cooldown,” failed to provide adequate post-fire direction to: (1) assure suppression pool temperatures did not increase above residual heat removal pump temperature limits following depressurization; and (2) assure adequate core cooling with one safety relief valve stuck open.

  This finding is unresolved pending completion of a significance determination. This finding is greater than minor because it impacts the mitigating systems cornerstone and affects the ability of the low pressure coolant injection system to provide adequate core cooling to prevent core damage. This finding was determined to have potential safety significance greater than very low significance because of the lack of credited systems to mitigate the effects of a control room fire. (Section 1R05.3)

Cornerstone: Mitigating Systems and Barrier Integrity

- TBD. The inspectors identified a violation of License Condition 2.C(14) for the failure to take appropriate corrective measures to address a condition adverse to quality affecting the low pressure coolant injection system. During a control room fire, the system has been vulnerable to a water hammer since at least 1997 due to a leaking check valve in Train B of the residual heat removal system. The licensee took over five years to identify the condition and failed to specify appropriate corrective measures to promptly fix the condition.

  This finding is unresolved pending completion of a significance determination. This finding is greater than minor because it impacts the mitigating systems cornerstone and affects the ability of the low pressure coolant injection system to provide adequate core cooling to prevent core damage. This finding was determined to have potential safety significance greater than very low significance because of the lack of credited systems to mitigate the effects of a control room fire. (Section 1R05.3)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a noncited violation of License Condition 2.C(14) and 10 CFR Part 50, Appendix R, Section III.G.2 for utilizing unapproved manual actions in Procedure ABN-FIRE, “Fire,” in lieu of physical protection to assure than one train of safe shutdown equipment is free of fire damage.

  This finding is greater than minor because it impacts the mitigating systems cornerstone. Inspection Procedure 71111.05, “Fire protection,” issued March 6, 2003, integrated inspector guidance for manual actions associated with licensee commitments to Section III.G.2 of Appendix R. This guidance is that if the manual actions are reasonable and are expected to meet the criteria outlined in Enclosure 2 to Inspection Procedure 71111.05, then the inspectors will identify this issue as a Green finding pending the Commission’s acceptance of the staff initiative to incorporate the use of manual actions into Section III.G.2. The inspectors determined that the manual actions were reasonable and constituted appropriate compensatory measures.
  (Section 1R05.2)

Report Details

1. REACTOR SAFETY

1R05 Fire Protection

The purpose of this inspection was to review the Columbia Generating Station fire protection program for selected risk significant fire areas. Emphasis was placed on verification of the licensee's post-fire safe shutdown capability. The inspection was performed in accordance with the new Nuclear Regulatory Commission (NRC) reactor oversight process using a risk-informed approach for selecting the fire areas and attributes to be inspected. The inspectors used the "Columbia Generating Station Fire Probabilistic Risk Assessment," dated April 24, 2002, to choose several risk-significant areas for detailed inspection and review. The fire areas chosen for review during this inspection were:

- Fire Area R-1, 522' level of the reactor building
- Fire Area RC-2, cable spreading room
- Fire Area RC-3, cable chase
- Fire Area RC-10, control room
- Fire Area RC-14, Division 1 switchgear room

For each of the selected fire areas, the inspectors focused the inspection on the fire protection features and on the systems and equipment necessary for the licensee to achieve and maintain safe shutdown conditions in the event of a fire in those fire areas.

Documents reviewed by the inspectors are listed in the attachment.

.1 Systems Required to Achieve and Maintain Post-Fire Safe Shutdown

a. Inspection Scope

The inspectors reviewed piping and instrumentation diagrams and the list of safe shutdown equipment documented in the licensee's post-fire safe shutdown analysis to verify whether their shutdown methodology had properly identified the components and systems necessary to achieve and maintain safe shutdown conditions for equipment in the fire areas selected for review. The inspectors focused on the following functions that must be ensured to achieve and maintain post-fire safe shutdown conditions.

- Reactivity control capable of achieving and maintaining cold shutdown reactivity conditions

- Reactor coolant makeup capable of maintaining the reactor coolant inventory

- Reactor heat removal capable of achieving and maintaining decay heat removal

- Supporting systems capable of providing all other services necessary to permit extended operation of equipment necessary to achieve and maintain hot shutdown conditions

A sample review was also conducted to ensure that all required electrical components in the selected systems were included in the licensee's safe shutdown analysis. The inspectors identified the systems required for each of the primary safety functions necessary to shut down the reactor. These systems were then evaluated to identify the systems that interfaced with the fire areas inspected and were the most risk significant systems required for reaching both hot and cold shutdown conditions. The following systems were selected for review.

- Automatic depressurization system
- Main steam isolation system
- Residual heat removal system
- Standby service water system

b. Findings

No findings of significance were identified.

.2 Fire Protection of Safe Shutdown Capability and Post-fire Safe Shutdown Circuit Analysis

a. Inspection Scope

The inspectors reviewed licensee documentation to verify that at least one post-fire safe shutdown success path was free of fire damage in the event of a fire in the selected fire areas. Specifically, the inspectors examined the separation of safe shutdown cables, equipment, and components within the same fire areas. Additionally, on a sample basis, the inspectors reviewed the licensee's analysis of electrical protective device (e.g., circuit breaker, fuse, relay) coordination. The inspectors reviewed the licensee's methodology for meeting the requirements of 10 CFR 50.48, and the bases for the NRC's acceptance of this methodology as documented in NRC safety evaluation reports. In addition, the inspectors reviewed license documentation, such as the final safety analysis report, submittals made to the NRC by the licensee in support of the NRC's review of their fire protection program, and deviations from NRC regulations to verify that the licensee met license commitments.

b. Findings

(1) Fire Areas R-1 and RC-10 - Failure to Assure That One Train of Low Pressure Coolant Injection was Free of Fire Damage

Introduction

The inspectors identified an apparent violation involving the failure to follow the requirements of 10 CFR Part 50, Appendix R, Section III.G.2, as committed to in License Condition 2.C(14) of the Columbia Generating Station operating license. Specifically, the apparent violation involved the failure to assure that one train of systems necessary to achieve and maintain hot shutdown conditions from either the control room or emergency control stations is free of fire damage. The licensee credits low pressure coolant injection for accomplishing reactor coolant inventory control and core cooling following a fire. The inspectors found that a fire in either Fire Area R-1 or Fire Area RC-10 could result in the substantial diversion of coolant from the core, during the low pressure coolant injection mode of operation, due to multiple hot short circuits.

<u>Description</u>

The licensee's fire safe shutdown path involves the use of the automatic depressurization system as the method for controlling reactor pressure. Pressure would be controlled by opening automatic depressurization system valves and depressurizing into the suppression pool. This depressurization of the reactor would bypass hot shutdown, and a cold shutdown cooling path would be implemented by removing decay heat with the residual heat removal system operated in the alternate decay heat removal mode. This mode of operation utilizes the suppression pool as the suction source and pumps water to the residual heat exchangers and then through the low pressure coolant injection path into the reactor.

Five automatic depressurization system valves are required to achieve depressurization of the reactor pressure vessel. The licensee takes no credit for any high pressure system for injection of coolant into the core, including the control rod drive system, the main feedwater system, the reactor core isolation cooling system, or the high pressure core spray system. None of these systems are included in the licensee's safe shutdown equipment list and, thus, cannot be credited for any fire scenario.

The inspectors found a potential vulnerability of the low pressure coolant injection mode of operation using the residual heat removal system due to the fire-induced spurious actuations of certain valves during a fire in either the reactor building (Fire Area R-1) or the control room (Fire Area RC-10).

- Valves RHR-V-16B and RHR-V-17B are two in-series, normally closed valves that branch off from residual heat removal system Loop B and go to the drywell spray ring header. The branch line is 16" in diameter but reduces to a 12" line at the spray ring header. Residual heat removal system Loop B is an 18" line but reduces to 14" and eventually 12" before injecting into the reactor vessel.

  The licensee's Post-Fire Safe Shutdown Analysis NE 02-85-19 stated that Valves RHR-V-16B and RHR-V-17B are closed during both normal operation and post-fire safe shutdown operation so that a fire related loss of power would not cause the valves to open. For a design basis fire in any redundant fire area or the main control room more than one spurious actuation (hot short) would be required to energize both of these valves to the open position.

- Valves RHR-V-40 and RHR-V-49 are two in-series, normally closed valves that branch off from residual heat removal system Loop B and go to the radwaste building drain collection tank. The branch line is 4" in diameter off of the 18" Loop B line.

> The licensee’s Post-Fire Safe Shutdown Analysis NE 02-85-19 stated that Valves RHR-V-40 and RHR-V-49 are closed during both normal operation and post-fire safe shutdown operation so that a fire related loss of power would not cause the valves to open. For a design basis fire in any redundant fire area or the main control room more than one spurious actuation (hot short) would be required to energize both of these valves to the open position.

The licensee’s Analysis NE 02-85-19 stated that a fire in the control room or the reactor building could cause both series valves to spuriously open. However, the licensee did not analyze for this since they maintain that they are not required to postulate for more than one spurious operation occurring during a fire event.

Design and Safety Analysis NES-7, “Safe Shutdown Analysis,” stated in the post-fire safe shutdown analysis assumptions and criteria (Section 5.8), that "Any and all, one at a time, means that during a fire, all potential spurious actuations shall be considered to occur, but one at a time only, and not in conjunction with or concurrently with any other postulated spurious actuation(s)." Although the licensee's fire protection staff appeared to indicate that they will evaluate every possible spurious actuation and its effects on safe shutdown, the evaluation was based on a single occurrence (individually) and a mitigation strategy for each was developed separately. (NOTE: The licensee does not apply the single occurrence assumption to high/low pressure interface analyses nor to the high impedance fault analysis.)

The NRC staff position, as discussed in the enclosure to Enforcement Guidance Memorandum 98-002, "Disposition of Violations of Appendix R, Sections III.G and III.L Regarding Circuit Failures," Revision 2, regarding the industry expression "any and all, one at a time" was that every possible spurious actuation must be considered and that spurious actuations of multiple components, whose cables are exposed to the same fire, should be assumed to occur concurrently in time (not necessarily simultaneously). These multiple spurious actuations must be considered as part of a safe shutdown analysis. Additionally, the cumulative effects of such multiple spurious operations on safe shutdown capability must be evaluated.

<u>Analysis</u>

This finding had more than minor safety significance because it impacted the mitigating systems cornerstone and affected the ability of the low pressure coolant injection system to provide adequate core cooling to prevent core damage.

<u>Enforcement</u>

License Condition 2.C(14) of the facility operating license states that the licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in Appendix F of the final safety analysis report (FSAR). The licensee committed to the requirements of 10 CFR Part 50, Appendix R, Section III.G.2 in Appendix F, such that one redundant train of equipment required to safely shut down the plant must be free of fire damage.

Contrary to the above requirements, the licensee failed to assure that one train of low pressure coolant injection was free of fire damage. Valves RHR-V-16B, RHR-V-17B, RHR-V-40, and RHR-V-49 were not protected from fire damage by one of the physical methods described in Appendix R, Section III.G.2.

The licensee disagreed that postulating multiple fire-induced circuit failures is required by NRC regulations or its operating license. Thus, this violation is being treated as an apparent violation in accordance with Section 8.1.7.1, “Fire Induced Circuit Failures,” of the NRC Enforcement Manual, which states that the NRC will defer enforcement action for disputed apparent violations provided the licensee implements reasonable compensatory actions for the identified vulnerabilities. The licensee instituted adequate compensatory actions in the form of fire watches in the affected fire areas. The NRC and the industry are currently working to resolve questions raised by the industry about the adequacy of the existing staff guidance concerning fire-induced circuit failures and the consistency of staff interpretations of both the guidance and the underlying regulatory requirements.

This finding is identified as an apparent violation (APV 50-397/0302-01).

(2) Use of Manual Actions

Introduction

The inspectors identified a noncited violation of Section III.G.2 of 10 CFR Part 50, Appendix R, for using manual actions in lieu of physical protection. The inspectors determined that the violation was of very low safety significance.

Description

Abnormal Condition Procedure ABN-FIRE, “Fire,” is the procedure used for response to a fire in the plant other than in the control room. The procedure directed operators on the actions to take for fires in each fire area of the plant. For a fire in the selected Fire Areas R-1, RC-2, and RC-3, Procedure ABN-FIRE directed operators to perform several manual actions outside of the control room. Section II.G.2 requires that circuits that could prevent the operation or cause misoperation of redundant trains of safe shutdown equipment be physically protected from fire damage by one of the three methods specified. Manual actions to respond to misoperations are not listed as an acceptable method for satisfying this requirement. However, the NRC has previously accepted plant-specific manual actions in formal exemption/deviation requests and in safety evaluation reports. Based on inspection results and industry comments, the NRC determined that licensees have, without request for an exemption/deviation from the code, implemented manual actions where the specified requirements of Section II.G.2 cannot be met. The staff concluded that rulemaking would be required to allow licensees committed to Section III.G.2 of Appendix R to substitute manual actions in lieu of Section III.G.2 compliance.

Analysis

This finding had more than minor safety significance because it impacted the mitigating systems cornerstone to safely shut down the plant due to fire damage. Since the manual actions were considered reasonable, the significance determination process was not entered in accordance with the guidance in Inspection Procedure 71111.05.

Enforcement

The inspectors reviewed the manual actions required for a fire in the selected fire areas, performed a plant walkdown of manual actions in selected areas, and observed a simulator scenario for a fire in the cable spreading room, and concluded that the manual actions were reasonable and were expected to meet the criteria in Enclosure 2 of NRC Inspection Procedure 71111.05. However, the inspectors did note some areas where Procedure ABN-FIRE could be improved. These were: (1) assure that timing in the procedure fully agreed with the time line analysis, (2) assure that operators could quickly locate electrical panels outlined in the procedure, and (3) evaluate the adequacy of accessing a ladder to perform manual actions for a fire in Fire Area RC-13 (control room chillers, communications room, and west HVAC chase). The licensee initiated Problem Evaluation Request 203-0956 to review these observations. Thus, in accordance with Inspection Procedure 71111.05 guidance, this issue is identified as a Green finding pending the Commission's acceptance of the proposed NRC staff initiative to incorporate the use of manual actions into Section III.G.2 of Appendix R. The Green finding is an indicator that while compensatory measures in the form of manual actions have been implemented and are acceptable, the licensee has not met the requirements of Section III.G.2 of Appendix R (NCV 50-397/0302-02).

.3 Alternative Safe Shutdown Capability and Implementation

a. Inspection Scope

The inspectors reviewed the systems required to achieve alternative safe shutdown to determine if the licensee had properly identified the components and systems necessary to achieve and maintain safe shutdown conditions from the remote shutdown panel. The inspectors also focused on the adequacy of the systems to perform reactor pressure control, reactor makeup, decay heat removal, process monitoring, and support system functions. The inspectors reviewed Procedure ABN-CR-EVAC, "Control Room Evacuation and Remote Cooldown," which would be used by operators to shut down the reactor in the event of a control room fire with evacuation of the control room. The inspectors also walked through the procedure with licensed and non-licensed operators to determine its adequacy to direct safe shutdown.

b. Findings

(1) Failure to Have Adequate Procedures in Effect for Alternative Shutdown

Introduction

The inspectors identified a noncited violation of Technical Specification 5.4.1.d (inadequate procedure) because Procedure ABN-CR-EVAC failed to provide adequate direction to: (1) assure that containment temperatures did not increase above residual heat removal pump temperature limits following depressurization; and (2) assure adequate core cooling with one safety relief valve stuck open. The inspectors determined that the violation was of very low safety significance.

Description

The inspectors identified two examples of an inadequate procedure due to significant discrepancies between procedural requirements contained in Procedure ABN-CR-EVAC and the licensee's safe shutdown analysis for a control room fire, "GE-NE-L12-00824-01," dated September 1994.

The first example was that Procedure ABN-CR-EVAC provided inadequate direction to ensure that suppression pool temperature did not exceed low pressure coolant injection system design limits during a control room fire event. Exceeding the design temperature limits could challenge low pressure coolant injection system operability. The licensee's safe shutdown analysis assumed that operators would establish "normal shutdown cooling" immediately after manual depressurization. This time-sensitive action helps to ensure that suppression pool water temperature does not increase above the low pressure coolant injection pump temperature limit (204 degrees Fahrenheit). However, the procedure failed to specify time limits for placing shutdown cooling in service. Further, during procedure walkdowns, operators stated that they would not likely place the shutdown cooling in service immediately following depressurization but would wait for a potentially extended period before taking the action.

The second example was that Procedure ABN-CR-EVAC failed to provide adequate instructions to ensure adequate core cooling, assuming a scenario with one safety relief valve stuck open. The licensee's fire protection analysis relied on operator action within 10 minutes to depressurize the reactor, following a reactor scram, to ensure adequate core cooling with the low pressure coolant injection system. In contrast, operators, when walking down the procedure, usually took at least 23 minutes to get to the depressurization step.

Analysis

The inspectors determined that both inadequate procedure examples had more than minor significance because each impacted the mitigating systems cornerstone and affected the cornerstone objective - to ensure the availability, reliability, and capability of the system that responds to the event to prevent undesirable consequences. The finding remains unresolved pending completion of a significance determination.

Enforcement

The failure to provide an appropriate procedure for alternate shutdown is a violation of Technical Specification 5.4.1.d. This requirement specifies, in part, that the licensee establish procedures for fire protection program implementation. License Condition 2.C(14) of the facility operating license states that the licensee shall implement and maintain in effect all provisions of licensee's fire protection program as described in Section 9.5.1 and Appendix F of the final safety analysis report. Section F.4.3 of Appendix F, as updated, states that alternative shutdown systems used in the event of a main control room future must meet the requirements of 10 CFR Part 50, Appendix R, Section III.L. Section III.L.3 states, in part, that procedures shall be in effect to implement alternative and dedicated shutdown capability. However, the licensee failed to ensure that Procedure ABN-CR-EVAC was adequate to implement alternative and dedicated shutdown capability. The licensee entered this finding into the corrective action program as Problem Evaluation Request 203-0956.

This finding and related violation are unresolved pending completion of a significance determination. This finding is identified as an unresolved item (URI 50-397/0302-03).

(1) Inadequate Corrective Action

Introduction

The inspectors identified a noncited violation of License Condition 2.C(14) for failure to take appropriate corrective measures to address a condition adverse to fire protection affecting the low pressure coolant injection system, Train B . During a control room fire, the system has been vulnerable to a water hammer, which could render the train inoperable, since at least 1997. The inspectors determined that the violation was of very low risk significance.

Description

For a control room fire, the licensee credits and protects portions (but not all) of the Division II residual heat removal system and the automatic depressurization system. Credited operator actions, prior to evacuating the control room, include a manual reactor trip and the closure of all main steam isolation valves. The licensee is required to maintain the capability to achieve safe shutdown (cold shutdown) from the remote shutdown panel utilizing only protected systems and components.

The inspectors identified that the licensee failed to take prompt corrective measures to address a long-standing low pressure coolant injection Train B water hammer vulnerability, which could jeopardize system operability. Following control room evacuation, in response to a control room fire, Procedure ABN-CR-EVAC instructed operators to check the status of the Division II residual heat removal system keepfill Pump RHR-P-3 hourly when the primary system Pump RHR-P-2B is not running. However, the licensee did not protect, or credit, the keepfill pump and, if the keepfill pump failed, the system could not maintain system fill for an hour due to a leaky Pump

RHR-P-2B discharge check valve (RHR-V-31B). Consequently, the system could suffer a water hammer if Pump RHR-P-2B started after a loss of fill.

The inspectors reviewed pressure decay test results to check for historical Valve RHR-V-31B leak-tight integrity. The inspectors found that the valve had last demonstrated acceptable performance in 1994 and had leaked excessively since at least 1997 (no data was available between 1994 and 1997). In 1997, the valve could only maintain system fill for about 40 seconds. Between November, 2000 and October, 2002 the licensee conducted seven leakage tests to estimate how long the valve could maintain pressure without losing system fill. The test results varied from test to test, with no particular trend. Calculated loss of fill time ranged from 6 minutes to a few hours. For four of the seven tests the valve could not maintain system fill for greater than the 1 hour procedural specification and in two instances the calculated loss of fill was less than 20 minutes.

The licensee had written Problem Evaluation Request 202-2984 on October 24, 2002, to capture the deficiency (over five years after initial indication), but the licensee took ineffective corrective measures to address the problem. The licensee didn’t plan to repair the valve until the spring, 2005 outage and specified only one compensatory measure, which was a fire watch. However, the corrective action did not include advising operators that a control room fire could result in the loss of the keepfill pump and subsequent potential water hammer. The inspectors considered a fire tour inadequate because it had no impact on preventing a system water hammer when attempting to mitigate a control room fire. In addition, the remote shutdown panel did not have residual heat removal system pressure indication to alert operators to a leaking check valve.

Analysis

The inspectors determined that the issue was greater than minor significance because it impacted the mitigating systems and barrier cornerstones and affected the cornerstone objectives to ensure the availability, reliability, and capability of the system that responds to the event to prevent undesirable consequences. In this instance, the problem affected the ability of Train B low pressure coolant injection to provide adequate core cooling to prevent core damage and to provide adequate decay heat removal from containment to prevent containment failure.

Enforcement

The failure to take prompt corrective measures to address a condition adverse to fire protection (leaking low pressure core spray pump discharge Check Valve RHR-V-31B) is a violation of Columbia Generating Station License Condition 2.C(14), which requires the licensee to implement and maintain in effect all provisions of the approved fire protection program as described in Appendix F of the final safety analysis report. The final safety analysis report, Appendix F, Section C.8 states, in part, that “Plant procedures require that conditions adverse to fire protection, such as . . . deficiencies, . . . defective components . . . are promptly identified, reported and corrected.”

Procedure SWP-FPP-01, “Nuclear Fire Protection Program,” Revision 3, Section 3.5.8, states that “Nonconforming fire protection items shall be identified, reported, dispositioned, and corrected in accordance with SWP-CAP-01.”

Procedure SWP-CAP-01, “Problem Evaluation Requests,” Revision 6, Section 2.1 states that, “ The problem evaluation request process assures the following: . . . conditions adverse to quality [fire protection] are promptly identified and corrected.”

Contrary to the above, the licensee failed to promptly identify and correct a condition adverse to fire protection. Since 1997, Valve RHR-V-31B has leaked excessively so that, during a control room fire event, the one credited injection source, low pressure coolant injection Train B, was, and still is, at increased risk of water-hammer related damage and failure. The licensee failed to identify the problem for 5 years and, once identified in October 2002, specified inadequate and untimely corrective measures. This finding is in the licensee’s corrective action program as Problem Evaluation Request 203-0997.

This finding and related violation are unresolved pending completion of a significance determination. This finding is identified as an unresolved item (URI 50-397/0302-04).

.4 Emergency Communications

a. Inspection Scope

The inspectors reviewed the adequacy of the communication system to support plant personnel in the performance of alternative safe shutdown functions and fire brigade duties. The inspectors verified that adequate communication equipment was available consistent with the licensing basis. The inspectors performed a review of the electrical power supplies and cable routing for the radio repeater system, sound powered phone system, and plant paging system.

` b. Findings

No findings of significance were identified.

.5 Emergency Lighting

a. Inspection Scope

The inspectors reviewed the emergency lighting system required to support plant personnel in the performance of alternative safe shutdown functions to verify it was adequate to support the performance of manual actions required to achieve and maintain safe shutdown conditions, and for illuminating access and egress routes to the areas where manual actions are required. The locations and positioning of emergency lights were observed during a walkthrough of the control room evacuation procedure. The inspectors also reviewed lighting surveillances to ensure that the licensee properly maintained the lighting.

b. Findings

No findings of significance were identified.

.6 Cold Shutdown Repairs

a. Inspection Scope

The inspectors reviewed equipment operations and capability to determine if any repairs were required in order to achieve cold shutdown. The inspectors noted that the licensee did not require the repair of equipment to reach cold shutdown based on the safe shutdown methodology implemented.

b. Findings

No findings of significance were identified.

.7 Fire Protection Systems, Features, and Equipment

a. For the selected fire areas, the inspectors evaluated the adequacy of fire protection features, such as fire suppression and detection systems, fire area barriers, penetration seals, and fire doors. To do this, the inspectors observed the material condition and configuration of the installed fire detection and suppression systems, fire barriers, and construction details and supporting fire tests for the installed fire barriers. In addition, the inspectors reviewed license documentation, such as NRC safety evaluation reports and deviations from NRC regulations and the National Fire Protection Association code to verify that fire protection features met license commitments.

b. Findings

No findings of significance were identified.

.8 Compensatory Measures

a. Inspection Scope

The inspectors verified, by sampling, that adequate compensatory measures were put in place by the licensee for out-of-service, degraded, or inoperable fire protection and post-fire safe shutdown equipment, systems or features (e.g., detection and suppression systems, or passive fire barrier features).

b. Findings

No findings of significance were identified.

4OA6 Meetings, including Exit

On March 19, 2002, at the conclusion of the inspectors's onsite inspection, the lead inspector debriefed Mr. Scott Oxenford, Plant General Manager, and other licensee management and staff members on the triennial fire protection inspection results.

On April 22, 2003, the lead inspector conducted an telephonic exit meeting with Mr. Scott Oxenford, and other licensee management and staff members, during which the results of the triennial fire protection inspection were characterized.

The licensee was asked whether any materials examined during the inspection should be considered proprietary. A proprietary document was identified and the inspectors agreed to properly dispose of the document immediately.

ATTACHMENT

PARTIAL LIST OF PERSONS CONTACTED

Licensee

P. Ankrum, Licensing Engineer
G. Brastad, Consulting Engineer
J. Civay, Electrical Engineer
D. Coleman, Manager, Performance Assessment & Regulatory Programs
J. Dittmer, Supervisor, Design Engineering
D. Feldman, Manager, Operations
W. Harper, Fire Protection Supervisor, Design Engineering
M. Humphreys, Manager, Engineering
R. Olson, Fire Marshall
S. Oxenford, Plant General Manager
H. Phan, Risk Engineer

ITEMS OPENED, CLOSED AND DISCUSSED

Items Opened, Closed, and Discussed During this Inspection

Opened

50-397/03-02-01	APV	Failure to protect one train of low pressure coolant injection from the effects of a fire. (Section 1R05.2)
50-397/03-02-03	URI	Failure to have adequate procedures in effect for alternative shutdown. (Section 1R05.3)
50-397/03-02-04	URI	Inadequate corrective action (Section 1R05.3)

Opened and Closed

50-397/03-02-02	NCV	Unapproved use of manual actions to protect safe shutdown equipment from the effects of a fire (Section 1R05.2)

LIST OF DOCUMENTS REVIEWED

Procedures

Number	Description	Revision
2.8.7	Fire Protection System	33
ABN-CR-EVAC	Control Room Evacuation and Remote Cooldown	4
ABN-FIRE	Fire	4
ABN-FIRE-CR	Control Room Fire	1
PPM 4.12.1.1	Control Room Evacuation (Historical Procedure)	8
SWP-FPP-01	Nuclear Fire Protection Program	2

Problem Evaluation Requests

296-0718	296-0855	298-0235	201-0323	201-1111	201-1493
201-1623	201-2587	202-0121	202-0600	202-0711	202-0734
202-1162	202-2184	202-2480	202-2537	202-2836	202-2952
202-2984	202-3068	203-0763	203-0772	203-0841	203-0850
203-0857	203-0956	203-0957	203-0962	203-0978	203-0997
203-1003					

Calculations

Number	Description	Revision
E/I-02-95-01	Overcurrent Protective Device Settings and Coordination Calculations for 480 Volt Distribution Systems	0
NE-02-85-19	Post Fire Safe Shutdown (PFSS) Analysis	4
NE-02-94-35	System Impacts on Post Fire Safe Shutdown	1

Drawings

Number	Description	Revision
1E048	Nuclear Boiler Systems Valve MS-V-22A (B22-F022A)	14
1E049	Nuclear Boiler Systems Valve MS-V-22B (B22-F022B)	15
1E050	Nuclear Boiler Systems Valve MS-V-22C (B22-F022C)	13

Number	Description	Revision
1E051	Nuclear Boiler Systems Valve MS-V-22D (B22-F022D)	13
E502-2	Main One Line Diagram - Emergency Buses	48
E919	Reactor Building El 522'-0" Location Plan Cable Tray Nodes	10
E920	Reactor Building El 548'-0" Location Plan Cable Tray Nodes	9
E921	Reactor Building El 572'-0" Location Plan Cable Tray Nodes	9
E947	All Buildings Cable Tray Nodes Index, Notes and Reference Drawings	14
E948	Appendix R Post Fire Safe Shutdown (PFSS) Protected Raceways, Reactor Building Elevations 501'-0" and 522'-0"	3
EWD-9E-002	Electrical Wiring Diagram: Residual Heat Removal System Pump RHR-P-2A Breaker RHR-CB-P2A	13
EWD-9E-002A	Electrical Wiring Diagram: Residual Heat Removal System Pump RHR-P-2A Breaker RHR-CB-P2A	1
EWD-9E-003	Electrical Wiring Diagram: Residual Heat Removal System Pump RHR-P-2B (E12-C002B)	19
EWD-9E-004	Electrical Wiring Diagram: Residual Heat Removal System Pump RHR-P-2B Breaker RHR-CB-P2B	16
EWD-9E-004A	Electrical Wiring Diagram Residual Heat Removal System Pump RHR-P-2B Breaker RHR-CB-P2B	0
EWD-9E-004B	Electrical Wiring Diagram: Residual Heat Removal System Pump RHR-P-2B Breaker RHR-CB-P2B	2
EWD-9E-028	Electrical Wiring Diagram: Residual Heat Removal System MOV RHR-V-16B (E12-F016B)	20
EWD-9E-030	Electrical Wiring Diagram: Residual Heat Removal System MOV RHR-V-17B (E12-F017B)	15
EWD-9E-033	Electrical Wiring Diagram: Residual Heat Removal System MOV RHR-V-24A (E12-F024A)	12
EWD-9E-034	Electrical Wiring Diagram: Residual Heat Removal System MOV RHR-V-24B (E12-F024B)	16
EWD-9E-039	Electrical Wiring Diagram: Residual Heat Removal System MOV RHR-V-40 (E12-F040)	11
EWD-9E-078	Electrical Wiring Diagram: Residual Heat Removal System MOV RHR-V-49 (E12-F049)	14

Number	Description	Revision
EWD-9I-005	Electrical Wiring Diagram: Residual Heat Removal System Analog Loop SW-FT-7A, RHR-FT-15A and RHR-ft-37A	8
EWD-9I-006	Electrical Wiring Diagram: Residual Heat Removal System Analog Loop RHR-FT-13, SW-FT-7B, RHR-FT-15B, RHR-FT-15C & RHR-FT-1	7
EWD-25I-022	Electrical Wiring Diagram Primary Containment Atmospheric Monitoring System CMS-LT-2, CMS-PT-4 and CMS-LE-6B	10
EWD-58E-001	Electrical Wiring Diagram: Standby Service Water System SW-P-1A	18
EWD-58E-002B	Electrical Wiring Diagram: Standby Service Water System ESW-P-1A Breaker SW-CB-P1A	3
EWD-58E-003	Electrical Wiring Diagram: Standby Service Water System SW-P-1B	17
EWD-58E-004B	Electrical Wiring Diagram: Standby Service Water System SW-P-1B Breaker SW-CB-P1B	3
EWD-58E-012	Electrical Wiring Diagram: Standby Service Water System MOV SW-V-2A	18
EWD-58E-013	Electrical Wiring Diagram: Standby Service Water System MOV SW-V-2A	13
EWD-58E-014	Electrical Wiring Diagram: Standby Service Water System MOV SW-V-2B	17
EWD-58E-015	Electrical Wiring Diagram: Standby Service Water System MOV SW-V-2B	20
EWD-58E-021	Electrical Wiring Diagram: Standby Service Water System MOV SW-V-12B	23
EWD-58E-022	Electrical Wiring Diagram: Standby Service Water System MOV SW-V-12B	19
EWD-58E-046	Electrical Wiring Diagram: Standby Service Water System MOV RHR-V-68A (E12-F068A)	16
EWD-58E-047	Electrical Wiring Diagram: Standby Service Water System MOV RHR-V-68B (E12-F068B)	22
EWD-58E-050	Electrical Wiring Diagram: Standby Service Water System HX-A Backup Cooling SW-A Inlet SW-V-187A	6

Number	Description	Revision
EWD-58E-051	Electrical Wiring Diagram: Standby Service Water System HX-B Backup Cooling SW-B Outlet SW-V-187B	7
EWD-58E-052	Electrical Wiring Diagram: Standby Service Water System HX-A Backup Cooling SW-A Outlet SW-V-188A	6
EWD-58E-053	Electrical Wiring Diagram: Standby Service Water System HX-B Backup Cooling SW-B Inlet SW-V-188B	6
EWD-58I-005	Electrical Wiring Diagram: Standby Service Water System Analog Loops SW-FT-8A & SW-PT-32A	17
EWD-58I-008	Electrical Wiring Diagram: Standby Service Water System Analog Loops SW-TE-1BR & SW-PT-32BR	14
EWD-58I-009	Electrical Wiring Diagram: Standby Service Water System Analog Loops SW-FT-8B and SW-PT-32B	15
FM892-1	Fire Barrier and Fire Boundary Plan Ground Floor EL 43'-0" and 441'-0" and Misc Floors	10
FM892-2	Fire Barrier and Fire Boundary Plan Mezzanine Floors EL 467'-0" and 471'-0" and Misc Floors	8
FM892-3	Fire Barrier and Fire Boundary Plan Operating Floors at EL 501'-0" and 507'-0" and 525'-0"	6
FM892-4	Fire Barrier and Fire Boundary Plan Reactor Building Misc Elevations	7
FM892-6	Zones of Limited Combustibles, Reactor Building Plan Miscellaneous Elevations	6
FM892-10	Sprinkler & Hose Station Plans Reactor Building Misc Elevations	4
FM892-12	Access - Egress For PFSS Activities, El 437'-0", 441'-0", 467'-0", 471'-0" and Misc Floors	NI
FM892-13	Access - Egress For PFSS Activities, Operating Floor Plan at El 501'-0" and 525'-0"	0
FM892-14	Access - Egress For PFSS Activities, Reactor Building Misc Plans	NI
M501	Flow Diagram Legend, Symbols and Abbreviations	46
M521-1	Flow Diagram - Residual Heat Removal System Loop “A”	95
M521-2	Flow Diagram - Residual Heat Removal System Loop “B”	98

Number	Description	Revision
M524-1	Flow Diagram - Standby Service Water System - Reactor, Radwaste, D.G. Bldg's and Yard	102
M524-2	Flow Diagram - Standby Service Water System - Reactor, Radwaste, D.G. Bldg's and Yard	95
M524-3	Flow Diagram - Standby Service Water System - Reactor, Radwaste, D.G. Bldg's and Yard	10
M529	Nuclear Boiler -Main Steam System	91
M554	Flow Diagram - H & V, Containment Atmosphere Control System Reactor Building	56
PFSS-1	Appendix R Post Fire Safe Shutdown (PFSS) Division 1 Boundaries One Line Diagram	2
PFSS-2	Appendix R Post Fire Safe Shutdown (PFSS) Division 2 Boundaries One Line Diagram	2
PFSS-3	Appendix R Post Fire Safe Shutdown (PFSS) Remote Shutdown Boundaries One Line Diagram	2
PFSS-4	Appendix R -Post Fire Safe Shutdown (PFSS) RHR & ADS System Alternate Shutdown Cooling Piping and Instrument Diagram	1
PFSS-5	Appendix R -Post Fire Safe Shutdown (PFSS) Nuclear Boiling System - Alternate Shutdown Cooling Instrument Diagram	1
PFSS-6	Appendix R -Post Fire Safe Shutdown (PFSS) Standby Service Water System Piping and Instrument Diagram	1
PFSS-7	Appendix R -Post Fire Safe Shutdown (PFSS) Radwaste Building Control & Switchgear Room HVAC	1
PFSS-9	Appendix R -Post Fire Safe Shutdown (PFSS) Standby Service Water Pumphouse & Diesel Generator BLDG HVAC	1

Miscellaneous Documents

10CFR50.59 Evaluation 95-058 for revising Procedure 4.12.1.1, “Control Room Evacuation and Remote Cooldown,” dated May 16, 1995.

Design and Safety Analysis NES-7, “Safe Shutdown Analysis,” Revision 5

Enforcement Guidance Memorandum 98-002, “Disposition of Violations of Appendix R, Sections III.G and III.L Regarding Circuit Failures,” revision 2

GE-NE-A71-00, “WNP-2 Appendix R Analysis, Letter report,” dated July 7, 1994

GE-NE-L12-00, “WNP-2 Appendix R Analysis,” dated September 1994

Letter GO2-95-013 dated January 25, 1995, from J.V. Parrish, Vice President, Nuclear Operations, WPPSS, to the NRC, Subject: WNP-2 Operating License NPF-21, Revision to Procedures for Control Room Fires.”

Computerized Cable Routing Data

PPM 8.3.293, “RHR Loop B Pressure Decay Test,” performed in 1994

Pressure Decay Data for Various Valve RHR-V-31B Test, performed April 26, 1997, November 22, 2001, May 8, 2002, and October 20, 2002

Quality Department Fire Protection Annual/Triennial Program Audit AU200-007, dated April 27, 2000

Quality’s Integrated Performance Assessment Report, dated March 6, 2003

Quality Services Fire Protection Program Annual/Biennial Audit AU-FPB-01, dated May 10, 2001

Quality Services Fire Protection Program Annual Audit AU-FPA-02, dated June 4, 2002

WNP-2 Pre-Fire Plans

WNP-2 Final safety Analysis Report, Appendix F, “Fire Protection Evaluation,” Amendment 53