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NUCLEAR REGULATORY COMMISSION
REGION IV
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November 17, 2003

J. V. Parrish (Mail Drop 1023)
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SUBJECT: COLUMBIA GENERATING STATION - NRC PLANT DESIGN PILOT, ENCLOSURES 1 AND 3, INSPECTION REPORT 05000397/2003010

Dear Mr. Parrish:

On September 25, 2003, the Nuclear Regulatory Commission (NRC) completed the onsite portion of an inspection at your Columbia Generating Station facility. In-office inspection was continued through October 10, 2003, to review issues associated with the containment atmospheric control system. The enclosed report documents the inspection findings, which were discussed on September 25, 2003, with you and other members of your staff.

This inspection examined activities conducted under your licenses 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.

Based on the results of this inspection, the NRC has identified seven findings of very low safety significance (Green). The findings did not present an immediate safety concern. Several of the findings did call into question the effectiveness of your past Final Safety Analysis Report review and update conducted in response to an NRC request that licensee's inform the NRC of the adequacy and availability of design basis information. Because of the very low significance and because you entered them into your corrective action program, the NRC is treating them as noncited violations, consistent with Section VI.A of the Enforcement Policy. The noncited violations are described in the subject inspection report. If you contest the violations or significance of the noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial(s), 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.

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Sincerely,

/RA/

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

Docket: 50-397
License: NPF-21

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NRC Inspection Report
05000397/2003-010

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**ENCLOSURE**

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

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|---|---|
| Docket: | 50-397 |
| License: | NPF-21 |
| Report : | 05000397/2003010 |
| Licensee: | Energy Northwest |
| Facility: | Columbia Generating Station |
| Location: | North Power Plant Loop<br>Richland, Washington |
| Dates: | September 8 through October 10, 2003 |
| Team Leader: | C. J. Paulk, Senior Reactor Inspector, Engineering and Maintenance Branch |
| Inspectors: | P. A. Goldberg, Reactor Inspector, Engineering and Maintenance Branch<br>J. M. Mateychick, Reactor Inspector, Engineering and Maintenance Branch<br>W. M. McNeill, Reactor Inspector, Engineering and Maintenance Branch<br>W. C. Sifre, Reactor Inspector, Engineering and Maintenance Branch |
| Approved By: | Charles S. Marschall, Chief<br>Engineering and Maintenance Branch<br>Division of Reactor Safety |

SUMMARY OF FINDINGS

IR 05000397/2003010; 09/08/2003 through 10/10/2003; Columbia Generating Station; Plant Design Pilot, Enclosures 1 and 3

The NRC conducted an inspection with five regional inspectors. The inspection identified seven green noncited violations. 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.

NRC Identified Findings

Cornerstone: Mitigation Systems

- Green. A noncited violation of Criterion III of Appendix B to 10 CFR Part 50 was identified for the failure to verify the adequacy of the design of the steam safety/relief lines by performing visual inspections to verify adequate clearances and documenting the results. Calculation ME-02-92-33 required these clearances for main steam safety/relief valve piping so that it could sustain the additional loading caused by a main steam line flooding event.

  The finding was more than minor because it was similar to Example 2.f of Appendix E of Manual Chapter 0612. Specifically, the engineering staff had to perform a reanalysis and an operability evaluation due to this condition. When processed through the Significance Determination Process, the finding screened out as being of very low safety significance because it did not represent an actual loss of the mitigating system, nor had the specific accident condition that could have challenged this system been experienced. (Section 1RDS1b.1.)

Cornerstone: Barrier Integrity

- Green. A noncited violation of Criterion III of Appendix B to 10 CFR Part 50 was identified for failure to prevent Calculation NE-02-85-12, "Secondary Containment Bypass Leakage Limit," Revision 1, from becoming effective prior to receiving license amendment approval. This could have permitted the operation of the plant outside of its license basis.

  This finding was more than minor because it was similar to Example 3.a of Appendix E of Manual Chapter 0612. Specifically, an engineering revision was required to void or deactivate Revision 1 of the calculation. When processed through the Significance Determination Process, the finding screened out as being of very low safety significance because the non-conservative results of the calculation had not been used to operate the plant and there were no inoperable systems or components. (Section 1RDS1b.2.)

- Green. A noncited violation of Technical Specification 5.4.1a. and Criterion V of Appendix B to 10 CFR Part 50 was identified for failure to translate design basis into a procedure. Specifically, there was no procedural direction to place a second train of standby gas treatment into service if the charcoal filters of the first train were to become depleted.

  This finding was more than minor because it has the potential for an unfiltered release of radioactive gases. Specifically, the failure to direct a second train of standby gas treatment into service would result in unfiltered air being released to the environment. When processed through the Significance Determination Process, the finding screened out as being of very low safety significance because the plant had not experienced an event or condition that would have required the use of such a procedure. (Section 1RDS1b.3.)

- Green. A noncited violation of Criterion III of Appendix B to 10 CFR Part 50 was identified for failure to establish correct acceptance limits for the inservice testing of standby gas treatment isolation valves. Specifically, the limits established would allow the valves to operate outside of the design requirements for stroke time.

  This finding was more than minor because it was similar to Example 2.a. of Appendix E of Manual Chapter 0612. Specifically, the stroke times for the subject valves typically exceeded the allowable time provided in the Final Safety Analysis Report. When processed through the Significance Determination Process, the finding screened out as being of very low safety significance because the inservice testing engineers determined that the stated design requirement was conservatively low and could be revised such that the valves would remain within acceptable limits. (Section 1RDS1b.4.)

- Green. A noncited violation of Technical Specification 5.4.1b., and Criterion III of Appendix B to 10 CFR Part 50 was identified for failure to develop emergency operating procedures that would accomplish their intended functions. Specifically, the procedures for venting the drywell and wetwell to atmosphere would not vent the gasses through a de-energized standby gas treatment train as required.

  This finding was more than minor because it was similar to Example 4.j. of Appendix E of Manual Chapter 0612. Specifically, the procedures were in a condition that would adversely affect the response to an emergency. When processed through the Significance Determination Process, the finding screened out as being of very low safety significance because it did not represent an actual loss of containment integrity, and the specific accident conditions that could have challenged the barrier have not existed. (Section 1RDS1b.5.)

- Green. A noncited violation of Technical Specification 5.4.1a., and Criterion XI of Appendix B to 10 CFR Part 50, was identified for failure to establish acceptance criteria for satisfying Surveillance Requirement 3.6.3.1.1. Specifically, Procedure OSP-CAC-B701, “CAC-HR-1A Preheater Operability Test," Revision 6, did not identify an acceptance criterion for the minimum air flow through the hydrogen recombiner, as required.

The finding was more than minor because it was similar to Example 3.a. of Appendix E of Manual Chapter 0612. Specifically, procedures required revision to meet regulatory requirements. When processed through the Significance Determination Process, the finding screened out as being of very low safety significance because it did not represent an actual loss of containment integrity, and the specific accident conditions that could have challenged the barrier have not existed. (Section 1RDS1b.6.)

- Green. A noncited violation of Criterion III of Appendix B to 10 CFR Part 50 was identified for failure to accurately translate design requirements, along with supported assumptions, into the determination of the minimum flow for the hydrogen recombiners under all accident conditions.

  The finding was more than minor because it has the potential to permit the accumulation of an explosive gas mixture within the containment. Specifically, the failure to accurately determine the minimum flow required through the recombiners under all conditions could result in releases of radio nuclides after an ignition of an explosive gas mixture in the containment. When processed through the Significance Determination Process, the finding screened out as being of very low safety significance because it did not represent an actual loss of containment integrity, and the accident conditions that could have challenged the containment integrity have not existed. (Section 1RDS1b.7)

# Report Details

## 1. REACTOR SAFETY

Introduction

The NRC has undertaken a pilot inspection program to determine if efficiencies or resource savings can be gained by consolidating selected baseline inspection procedures. This inspection report documents the performance of Attachment 71111.DS, "Plant Design - Pilot," Enclosures 1 and 3. These enclosures are normally performed using Attachment 71111.21, "Safety System Design and Performance Capability," and Attachment 71111.02, "Evaluation of Changes, Tests, or Experiments."

### 1RDS Plant Design (71111.DS)

1RDS1: Safety System Design and Performance

a. Inspection Scope

The NRC conducted an inspection to verify the adequacy of the original design and subsequent modifications to safety systems and to monitor the capability of the selected systems to perform their design basis functions. The team reviewed in detail the containment structures. The primary review prompted parallel review and examination of support systems, such as, containment atmospheric control, standby gas treatment, residual heat removal (containment spray and shutdown cooling modes), and related structures and components.

The team assessed the adequacy of calculations, analyses, engineering processes, and engineering and operating practices that the licensee used for the selected safety system and the necessary support systems during normal, abnormal, and accident conditions. Acceptance criteria used by the NRC inspectors included NRC regulations, the technical specifications, applicable sections of the Updated Safety Analysis Report, design specifications, design bases documents, design requirements documents, procedures, applicable industry codes and standards, and industry initiatives implemented by the licensee's programs.

The minimum sample size for this procedure is one risk-significant system for mitigating an accident or maintaining barrier integrity. The team completed the required sample size by reviewing the containment structure and supporting systems.

b. Findings

b.1. Alternate Shutdown Cooling Mode of Residual Heat Removal

Introduction

The team identified a finding of very low safety significance involving a noncited violation of Criterion III of Appendix B to 10 CFR Part 50 for the failure to verify the adequacy of the design of the as-built configuration of the safety/relief valve down-comer piping.

Description

Columbia Generating Station engineers performed Calculation ME-02-92-33, "Main Steam Line Flooding," Revision 0, to demonstrate that the main steam safety/relief valve piping and pipe supports were structurally adequate to sustain the additional loading caused by the main steam line flooding. The main steam line would be flooded when the safety/relief valves are used in conjunction with the suppression pool as the alternate shutdown cooling path, when the shutdown cooling mode of the residual heat removal system was not available. The engineers determined that the safe shutdown earthquake horizontal displacement, at some nodes of the discharge line for Main Steam Safety/Relief Valve MRSV-3B, were as much as 8 inches. The team noted that the engineer concluded in the calculation that a visual inspection was to be performed during Refueling Outage R-7 to verify that the large pipe displacement would not cause an impact between the safety/relief valve pipe and any other piping or equipment in the vicinity.

When requested, the engineers were not able to retrieve adequate documentation regarding the total scope of the visual inspection to determine if the clearances were sufficient to withstand the displacements expected during a seismic event. The inspectors were informed by a design engineer that the actual clearances were less than 3 inches.

Engineers performed an initial operability assessment when they initiated Problem Evaluation Request 203-3559. The engineers concluded that the potential existed for seismic movements to result in the piping impacting other safety-related components. Based on engineering judgement and the use of more recent computer codes, the engineers concluded that, under seismic loading conditions, the impact condition would not result in failure or rupture of the large bore main steam safety/relief valve discharge lines. The engineers determined that the main steam safety/relief valves were operable but nonconforming. The team reviewed the engineers' assessment and had no findings of significance.

Analysis

The team found the mitigating systems cornerstone was affected because of the potential of for the loss of the alternate shutdown cooling method under these conditions. The issue affected the attributes of design control and human performance.

The team considered this finding more than minor since the finding fit with Example 2.f of Appendix E of Manual Chapter 0612, "Power Reactor Inspection Reports," June 20, 2003, in that, the engineering staff had to perform a reanalyses and an operability evaluation due to this condition.

The team found that this issue resulted from a performance deficiency of very low safety significance. The team assessed this finding as Green because it did not represent an actual loss of the mitigating system. Also, the specific accident condition that could have challenged this system has not existed. Columbia Generating Station personnel implemented corrective actions to ensure continued operability.

A contributing cause of this violation was considered to have cross-cutting implications in the problem identification and resolution area. That is, this finding was the direct result of the engineering staff's failure to identify the limitations on the movement of the steam lines as a condition adverse to quality when the concern was first identified in 1992, and to assure that the required inspections were performed. The team attributed this to the overall performance during that time frame.

Enforcement

Criterion III of Appendix B to 10 CFR Part 50, "Design Control," states, in part, that "[m]easures shall be established to assure that applicable regulatory requirements and the design basis, as defined in §§50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. . . . The design control measures shall provide for verifying or checking the adequacy of design, such as, by the performance of design reviews, [or] by the use of alternate or simplified calculational methods . . ."

Contrary to the above, the measures established to perform such verification were inadequate, in that, engineering, maintenance, and operations personnel did not verify the adequacy of the design of the steam safety/relief lines by performing visual inspections and documenting the results. An engineer initiated Problem Evaluation Request 203-3559 and entered this finding into the corrective action program.

Because of the very low safety significance of the finding, and because the finding has been entered into the corrective action program, the team treated this as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy.
(NCV 05000397/2003010-001)

b.2. Secondary Containment Bypass Leakage Limits

Introduction

The team identified a finding of very low safety significance involving a noncited violation of Criterion III of Appendix B to 10 CFR Part 50 for the loss of control of design basis information on the standby gas treatment system when a revision of a calculation for determining the maximum allowable containment bypass leak rate ( a design document) was issued (made effective) with a result that exceeded the licensed limit.

Description

Engineering personnel issued, on January 18, 2000, Calculation NE-02-85-12, "Secondary Containment Bypass Leakage Limit," Revision 1. The results of the revision changed the bypass leakage limit from 0.74 scfh to 15.26 scfh. Using calculated values for source term, engineers apply the secondary containment bypass leakage to insure that accident conditions will not result in radioactive doses to the public or to operators exceeding regulatory limits. Technical Specification 3.6.1.3.10 allows a limit of 0.74 scfh and the leak rate testing also uses that limit. Engineers revised the calculation for bypass leakage based on a proposed license amendment incorporating a reduced source term. Engineering personnel did not plan to change the technical specifications to reflect the revised 15.26 scfh until approval, by the NRC, of the amendment to the technical specifications using the revised accident source term. Lack of design control, however, resulted in making the revised bypass leak-rate calculation effective before receiving NRC approval for the revised source term.

Analysis

The team found this affected the Barrier Integrity cornerstone. The issue affected design control and human performance attributes of this cornerstone. The team considered this finding more than minor because it was similar to Example 3.a. of Appendix E of Manual Chapter 0612. Specifically, an engineering revision was required to void or deactivate Revision 1 of the calculation. An effect of issuing Calculation NE-02-85-12, Revision 1, was that the plant could be operated outside of its license. The team verified that the total leakage had been maintained within the limit specified in the technical specifications.

The team found this issue resulted from a performance deficiency of very low safety significance because the leakage had been maintained below 0.74 schf, the higher limit had not been used as an acceptance value, and there had been no event to challenge the integrity of the containment. A contributing cause of this violation was considered to have cross-cutting implications in the problem identification and resolution area. That is, this finding was the direct result of the engineering staff’s failure to identify that the plant could be operated outside of its licensed limits. The team attributed this to the engineering staff’s not implementing the design change process correctly.

Enforcement

Criterion III of Appendix B to 10 CFR Part 50, "Design Control," states, in part, that "[m]easures shall be established to assure that applicable regulatory requirements and the design basis, as defined in §§50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions."

Contrary to the above, those measures established to assure the correct translation of design information were inadequate. Specifically, the measures did not prevent the issuance of a calculation that could allow operation of the plant outside of its licensed limits. Engineering personnel issued Problem Evaluation Request 203-3370 and placed this finding in the corrective action program.

Because of the very low safety significance of the finding, and because the finding has been entered into the corrective action program, the team treated this as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000397/2003010-002)

b.3. Operation of the Standby Gas Treatment System

Introduction

The team identified a finding of very low safety significance involving a noncited violation of Criterion III of Appendix B to 10 CFR Part 50 for the loss of control of design basis information on the standby gas treatment system when engineering personnel failed to incorporate design requirements into a procedure.

Description

The team found that the Final Safety Analysis Report, Section 6.5.1.2, states that, given high radiation in the standby gas treatment system, an operator will start the second train. The team found that there was no procedure that addressed this activity.

The standby gas treatment system consists of two trains used to remove fission products from containment under post-accident conditions. If the charcoal filters of one train were to become depleted, it would be necessary to manually switch to the other train of the system to remove radio-nuclides from the air.

Analysis

This finding was more than minor because it has the potential to affect the licensee’s response team an unfiltered release of radioactive gases. Specifically, the failure to direct a second train of standby gas treatment into service would result in unfiltered air being released to the environment when the charcoal in the running train became depleted. The team considered this more than minor because a procedure had to be changed or developed to address this requirement. The plant has not experienced an event or condition that would have required the use of such a procedure.

The team considered this to affect the Barrier Integrity cornerstone. The issue affected design control and human performance attributes of this cornerstone. The engineering staff failed to translate the design basis into a procedure.

The team found this issue resulted from a performance deficiency of very low safety significance because there had been no event to challenge the functionality of the standby gas treatment system. A contributing cause of this violation was considered to have cross-cutting implications in the problem identification and resolution area. That is, this finding was the direct result of the failure of engineering personnel to identify that design information was not correctly translated into required procedures. The team attributed this to the engineering personnel not knowing the design requirements.

Enforcement

Technical Specification 5.4.1a. requires written procedures shall be established, implemented, and maintained covering activities including applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978.

Criterion V of Appendix B to 10 CFR Part 50, "Instructions, Procedures, and Drawings," states, in part, that "[a]ctivities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished."

Contrary to the above, the measures established to assure that design information was correctly translated into procedures were inadequate, in that, the requirement to start a second train of standby gas treatment was not incorporated into a procedure. The engineering staff issued Problem Evaluation Request 203-33416 and entered this finding into the corrective action program.

Because of the very low safety significance of the finding, and because the finding has been entered into the corrective action program, the team treated this as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000397/2003010-003).

b.4. Inservice Testing of Standby Gas Treatment Valves

Introduction

The team identified a finding of very low safety significance involving a noncited violation of Criterion III of Appendix B to 10 CFR Part 50 for the failure to establish correct stroke-time acceptance limits for inservice testing of standby gas treatment system valves.

Description

The inservice testing engineers did not establish correct acceptance limits for 10 valves in the standby gas treatment system (Valves SGT-V-1A, SGT 4A-1, SGT 4A-2, SGT 4B-1, SGT 4B-2, SGT 5A-1, SGT 5A-2, SGT 5B-1 and SGT 5B-2). Final Safety Analysis Report, Sections 6.5.1.2 and 6.5.1.4, require all valve stroke times to be less than 4 seconds for the standby gas treatment system. The action limit for the stroke times, as stated in the inservice testing program, and its procedures, was 6 to 8 seconds. Valve stroke-time testing since late 2000 to the present indicated that 8 valves failed to meet the 4-second requirement 85 to 93 percent of the time, 1 valve failed only 21 percent of the time, and 1 valve failed 100 percent of the time.

The inservice testing engineers recognized this discrepancy. However, they did not change the procedure because they anticipated an amendment to the Final Safety Analysis Report. The basis for their decision was a justification for continued operation associated with Problem Evaluation Request 297-1003 and the anticipated approval for the use of a new accident source term methodology.

Analysis

The team considered this finding to be similar to Example 2.a of Appendix E of Manual Chapter 0612, and, therefore, more than minor. Specifically, the stroke times for the subject valves typically exceeded the allowable time provided in the Final Safety Analysis Report. The valves met the inservice testing program requirements, however, test engineers failed to require corrective action before returning the valves to service, despite the failure to meet the Final Safety Analysis Report stroke-time criteria.

The team considered this to affect the Barrier Integrity cornerstone. The issue affected design control and human performance attributes of this cornerstone. The inservice testing engineers failed to translate the design requirements into correct acceptance limits in the inservice testing program and its implementing procedure. While the majority of the valves had stroke times in excess of the design basis, the inservice testing engineers evaluated the requirement and determined that, with a stroke time of 10 seconds, the plant would remain within its design requirements for maintaining barrier integrity.

The team found this issue resulted from a performance deficiency of very low safety significance. A contributing cause of this violation was considered to have cross-cutting implications in the problem identification and resolution area. That is, this finding was the direct result of the failure of engineers to identify that design requirements were not correctly translated into required procedures. The team attributed this to the engineers not knowing or understanding the design requirements.

Enforcement

Criterion III of Appendix B to 10 CFR Part 50, "Design Control," states, in part, that "[m]easures shall be established to assure that applicable regulatory requirements and the design basis, as defined in §§50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions."

Contrary to the above, the measures established to assure that design information was correctly translated into procedures were inadequate, in that, an acceptance criterion for valve stroke times was incorporated into the inservice testing program that allowed the operation of the valves outside of the design requirements, as stated in the Final Safety Analysis Report. The inservice testing engineers issued Problem Evaluation Request 203-3373 and entered this finding into the corrective action program.

Because of the very low safety significance of the finding, and because the finding has been entered into the corrective action program, the team treated this as a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy.
(NCV 05000397/2003010-004)

b.5. Emergency Operating Procedures (EOPs)

Introduction

The team identified a finding of very low safety significance involving a noncited violation of Technical Specification 5.4, “Procedures,” and Criterion III of Appendix B to 10 CFR Part 50. Specifically, the team identified examples of emergency operating procedures, which would not accomplish the stated purpose of the procedures for venting combustible gases from primary containment through a de-energized train of the standby gas treatment system.

Description

The branch for Primary Containment Gas on the EOP Flow Chart 5.2.1 directs the use of the containment atmospheric control system hydrogen/oxygen recombiners to limit the buildup of the gases to less than combustible concentrations. Should combustible concentrations of hydrogen and oxygen be reached, EOP 5.2.1 directs that the recombiners be secured and an emergency venting of the primary containment be performed in accordance with EOP 5.5.20, “Emergency Wetwell Venting With High Hydrogen and Oxygen Concentrations,” Revision 5; or EOP 5.5.21, “Emergency Drywell Venting With High Hydrogen and Oxygen Concentrations,” Revision 5.

Both EOP 5.5.20 and 5.5.21 state that the purpose of the procedure is to vent the respective primary containment space through a disabled (de-energized) standby gas treatment system train. However, operators, following the detailed steps contained in each procedure, will de-energize a standby gas treatment system train but then align the system to vent the primary containment into the secondary containment without passing through the disabled standby gas treatment system train.

The implementation of the procedure as written would result in the venting of a combustible gas mixture from primary containment into secondary containment prior to being vented to the environment via the other operating (energized) standby gas treatment system train. The team noted that the actual system alignment resulted in combustible gases being discharged into the secondary containment at a location adjacent to energized equipment which could provide an ignition source. However, the engineers had not evaluated this condition to determine if there would be adequate dilution of the explosive mixture. In response, an engineer initiated Problem Evaluation Request 203-3561 to address this issue and place it in the corrective action program.

Analysis

The team considered this finding to be similar to Example 4.j of Appendix E of Manual Chapter 0612, and, therefore, more than minor. Specifically, the procedures were in a condition that could adversely affect the response to an emergency.

The performance deficiency associated with this finding is the issuance of inadequate procedures that have the potential to create an unanalyzed plant condition. The team considered the Barrier Integrity cornerstone affected because of the unanalyzed potential for both primary and secondary containment integrity being degraded by these conditions.

The cornerstone attribute of maintaining functionality of containment is challenged by the potential for combustion to occur. The finding was only of very low safety significance because it did not represent an actual loss of containment integrity. The specific accident conditions that could have challenged containment integrity have not existed.

The team found this issue resulted from a performance deficiency of very low safety significance. However, a contributing cause of this violation was considered to have cross-cutting implications in the problem identification and resolution area. That is, this finding was the direct result of the engineers failing to identify that the EOPs would not accomplish their intended functions. The team attributed this to the engineers not understanding the intent of the actions required by the EOPs. Therefore, the engineers did not develop procedures to accomplish the filtered release of a contaminated combustible gas mixture without igniting the mixture.

Enforcement

Technical Specification 5.4.1b. requires written procedures shall be established, implemented, and maintained covering activities, including emergency operating procedures, required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1, as stated in Generic Letter 82-33.

Criterion III of Appendix B to 10 CFR Part 50 states, in part, that "[m]easures shall be established to assure that applicable regulatory requirements and the design basis, as defined in §§50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions."

Contrary to the above, the Columbia Generating Station's staff failed to develop adequate procedures for emergency venting of the primary containment. As a result, the approved procedures would not accomplish the intended purpose of the procedures. Further, the approved procedures created a plant configuration and method of operation with potential adverse impact on containment integrity following an accident.

Because of the very low safety significance of the finding, and because the finding has been entered into the corrective action program, the team considered this to be a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000397/2003010-005)

b.6. Operability Testing of the Hydrogen Recombiners

Introduction

The team identified a finding of very low safety significance involving a noncited violation of Technical Specification 5.4, “Procedures”; Criterion V of Appendix B to 10 CFR Part 50, “Instructions, Procedures and Drawings”; and Criterion XI of Appendix B to 10 CFR Part 50, “Test Control.” The finding involved a failure to establish an acceptance criterion for satisfying Surveillance Requirement 3.6.3.1.1, that is, to “[p]erform a system functional test for each primary containment hydrogen recombiner.” Specifically, engineers failed to identify acceptance criteria for system flow as a critical parameter for testing to demonstrate recombiner operability.

Description

Procedures OSP-CAC-B701, “CAC-HR-1A Preheater Operability Test," Revision 6; and TSP-CAC-B701, “CAC-HR-1A Functional Test and Visual Examination,” Revision 7, are utilized to perform testing to satisfy Technical Specification SR 3.6.3.1.1. Procedure OSP-CAC-B701 has the additional purpose “to check for degradation in CAC blower capacity.” The engineering personnel did not consider the blower performance to be a system parameter necessary to demonstrate satisfactory performance for the system functional test in accordance with Surveillance Requirement 3.6.3.1.1.

Both Technical Specification Bases B 3.6.3.1, “Primary Containment Hydrogen Recombiners,” and Section 6.5.2.3, “Design Evaluation,” of the Final Safety Analysis Report, discuss the minimum flow required through a single recombiner to maintain the hydrogen and oxygen concentrations in primary containment below combustible limits. Inability to provide the minimum required flow through the recombiner under the atmospheric conditions predicted when the system operation is initiated 6 hours after a loss-of-coolant accident would prevent the recombiner from fulfilling its safety function of controlling the combustible gas mixture in primary containment. Therefore, the team found that the blower performance is a system parameter necessary to demonstrate satisfactory performance of the system functional test per Surveillance Requirement 3.6.3.1.1.

The acceptance criteria of Procedure OSP-CAC-B701 state, that “[t]his surveillance is satisfactorily completed when all steps preceded by a # have been initialed, all other steps have either been initialed or properly documented and the CRS/Shift Manager has reviewed and signed the cover sheet.” Detailed procedure steps measuring flow are not preceded by a number and, thus, are not considered critical for demonstrating satisfactory completion of the technical specification surveillance. No minimum test flow limit had been established under which the recombiner should be considered inoperable. If the flows is less than the administrative limit, the system engineer is required to initiate an evaluation.

Based on the current procedures, the potential exists for returning a recombiner to service with low flow to operation when it is not capable of performing its safety function. The determination that the recombiner should be declared inoperable and the performance of the appropriate action statement of Technical Specification 3.6.3.1 would be delayed for an unspecified period pending evaluation by the system engineer. In response, the system engineer initiated Problem Evaluation Requests 203-3388 and 203-3557 to address these issues and enter them into the corrective action program.

Analysis

The performance deficiency associated with this finding is the failure to identify appropriate acceptance criteria in procedures for satisfying Surveillance Requirement 3.6.3.1.1, "Perform a system functional test for each primary containment hydrogen recombiner." The team considered the reactor safety barrier integrity cornerstone affected because of the potential for a recombiner to be incapable of preventing the buildup of a combustible gas mixture in primary containment after a loss-of-coolant accident.

The finding was more than minor because it was similar to Example 3.a. of Appendix E of Manual Chapter 0612. Specifically, procedures required revision to meet regulatory requirements. The cornerstone attribute of maintaining functionality of containment was challenged by the potential for a deflagration to occur. The finding was of very low safety significance because it did not represent an actual loss-of-containment integrity. The specific accident conditions that could have challenged containment integrity(e.g., a loss-of-coolant accident with the generation of a combustible mixture of hydrogen and oxygen) have not existed.

The team found this issue resulted from a performance deficiency of very low safety significance. However, the contributing cause of this violation was considered to have cross-cutting implications in the problem identification and resolution area. That is, this finding was the direct result from the failure of engineers to identify that the surveillance procedure did not have an acceptance criterion for the recombiner air flow. The team attributed this to the engineers failure to consider the capability of the blower to be a critical attribute for demonstrating the operability of the recombiners.

Enforcement

Technical Specification 5.4.1a requires written procedures shall be established, implemented, and maintained covering activities including applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978.

Criterion V of Appendix B to 10 CFR Part 50, "Instructions, Procedures, and Drawings," states, in part, that "[a]ctivities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished."

Criterion XI of Appendix B to 10 CFR Part 50, "Test Control," states, in part, that "[a] test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents."

Contrary to the above, engineers failed to relate the recombiner flow under test conditions to the design requirements necessary for satisfactory performance under post-accident conditions. In addition, the program developed to demonstrate that the hydrogen recombiners would perform satisfactorily failed to identify recombiner flow as a required parameter for determining satisfactory performance and establish appropriate acceptance limits.

Because of the very low safety significance of the finding and because the finding has been entered into the corrective action program, the team considered this to be a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000397/2003010-006)

b.7. Determination of Minimum Flow Requirements for Hydrogen Recombiners

Introduction

The team identified a finding of very low safety significance involving a noncited violation of Criterion III of Appendix B to 10 CFR Part 50. Specifically, the team identified a calculation to determine required hydrogen recombiner performance during an accident with assumed plant conditions was not consistent with those expected as documented in the Final Safety Analysis Report. Specifically, engineers did not consider the requirement for increased flow with increased suppression pool vapor temperatures. The team found that there was no justification in the calculation that the assumed conditions would yield conservative results compared to system performance under the Final Safety Analysis Report conditions.

Description

Engineers performed Calculation NE-02-92-39, "Calculation for CAC Performance With Reduced Capacity," Revision 1, to determine the minimum flow requirements for the containment atmospheric control system with varying initial temperatures in the suppression chamber free air space prior to a loss-of-coolant accident. Assumption 6 of the calculation states that “[t]he containment spray system (RHR A&B) cools the containment atmosphere to 150°F and 0 psig within 1 hour [sic]; the pressure and temperature in containment are assumed to remain constant for the duration of the LOCA analysis.”

Operators will place the recombiners into service 6 hours after a loss-of-coolant accident. The team noted that, from the analysis results in Final Safety Analysis Report, Section 6.2, "Containment Systems," both higher containment temperature and pressure would exist at that time. The team found no analysis comparing the recombiner performance and minimum required flow rate under the assumed conditions versus the Final Safety Analysis Report conditions. Therefore, the engineers had no basis for accepting the calculation results as providing satisfactory recombiner performance after a loss-of-coolant accident. In response, the system engineer initiated Problem Evaluation Request 203-3552 to address this issue and enter the finding into the corrective action program.

A discussion of this calculation is included in Section 6.2.5.2.4, “Design Evaluation,” of the Final Safety Analysis Report. This discussion states that “[t]he parametric analysis shows that with a wetwell air space temperature of up to 114°F, a blower capacity of at least 112 scfm is adequate and with a wetwell air space temperature of 130°F that a blower capacity of at least 120 scfm is adequate and with a wetwell air space temperature of 150°F, the limit for equipment qualification, a blower capacity of at least 138 scfm is sufficient.”

Section 6.2.5.2.4 of the Final Safety Analysis Report also states that “[t]he minimum input flow to the recombiner is 44.8 scfm (112 scfm total flow with 60% recycle).” Technical Specification Basis B 3.6.3.1, “Primary Containment Hydrogen Recombiners,” states that “[a]ir flows through the unit at ≥ 112 scfm (which includes 60% recycle flow), with a constant speed rotary lobe blower providing the motive force.”

In accordance with the calculation results, the minimum recombiner flow of 112 scfm would be required with an initial wetwell temperature of 114°F. However, plant operation with wetwell temperatures up to 150°F is permitted (Final Safety Analysis Report, Section 3.1, "Environmental Design of Mechanical and Electrical Equipment"), which requires a minimum recombiner flow of 138 scfm. Additionally, the system engineer issued Problem Evaluation Request 203-3388 to “[e]valuate the need to determine CAC blower flowrate to determine system operability.” This request states that “[i]n order for the CAC system to perform its required safety function the CAC blower flow capacity must be greater than 120 SCFM during accident conditions.” The team found this assumption to be incorrect on the basis that the design allows operation of the plant up to a wetwell temperature of 150°F. At that temperature, the Final Safety Analysis Report states that the minimum flow would be 138 scfm.

Analysis

The performance deficiency associated with this finding is the failure to accurately incorporate design basis information into a calculation evaluating the ability of a system to perform its safety function. The team considered the reactor safety barrier integrity cornerstone affected because of the potential for a recombiner to not be capable of preventing the buildup of a combustible gas mixture in primary containment after a loss-of-coolant accident.

The finding was more than minor because it has the potential to permit the accumulation of an explosive gas mixture. Specifically, the failure to accurately determine the minimum flow required through the recombiners under all conditions could result in releases of radio nuclides after an ignition of an explosive gas mixture in the containment.

Subsequent review by engineering personnel revealed that the actual flow through the recombiners could not meet the minimum flow required with the wetwell temperature at 150°F; however, there would be sufficient air flow with the wetwell temperature below 130°F. Therefore, operations personnel were instructed to declare the hydrogen recombiners inoperable if the wetwell temperature reached 130°F.

The cornerstone attribute of maintaining functionality of containment is challenged by the potential for a deflagration to occur. The finding was only of very low safety significance because it does not represent an actual loss of containment integrity. The specific accident conditions that could have challenged containment integrity have not existed.

The team found this issue resulted from a performance deficiency of very low safety significance since the plant has never operated with suppression pool airspace temperatures greater than 130°F, and the recombiners provided sufficient flow for temperatures up to 130°F. However, the contributing cause of this violation was considered to have cross-cutting implications in the problem identification and resolution area. That is, this finding was the direct result of the engineers failure to identify that the calculations contained unsupported and incorrect assumptions. The team attributed this to a lack of understanding of the operation of the recombiners.

Enforcement

Criterion III of Appendix B to 10 CFR Part 50 states, in part, that "[m]easures shall be established to assure that applicable regulatory requirements and the design basis, as defined in §§50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions."

Contrary to the above, the measures established to assure that design conditions were correctly translated into specifications and procedures were inadequate in that incorrect assumptions were utilized in a calculation, resulting in an unsupported design value for the minimum flow requirement for the recombiners.

Because of the very low safety significance of the finding and because the licensee has entered the issue into their corrective action program, the team considered this to be a noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000397/2003010-007)

b.8 Other Issues

Introduction

The team determined that the following findings are examples of a failure to maintain complete and accurate design information and to appropriately translate design information into required procedures and calculations. Each of the following issues was determined to be minor issues, below the significance of that associated with green findings, and are not the subject of formal enforcement action.

The team determined that these findings are cross-cutting issues affecting mitigating systems and barrier integrity.

Discussion

I. The team found that the containment spray system was listed as a commitment to the NRC in the Final Safety Analysis Report and not a requirement. The Final Safety Analysis Report stated that containment spray was not credited in the Chapter 15 accident analysis. However, Criterion 38 of Appendix A to 10 CFR Part 50, "Containment Heat Removal," requires a containment spray system; therefore, the Final Safety Analysis Report is incorrect in stating that the containment spray system was a commitment to the NRC.

In addition, this error also existed in the design requirements document for the residual heat removal system. In this document, the author states that the containment spray system was classified as having no safety function, and was just a commitment to the NRC.

ii. The team also found that the Final Safety Analysis Report had conflicting statements concerning whether containment spray was needed for temperature control or not. Section 6.2.2.1 of the Final Safety Analysis Report states that the purpose of containment spray is to prevent excessive containment temperatures and pressures, thus maintaining containment integrity following a loss-of-coolant accident. Section 6.2.2.2 states that water is drawn from the suppression pool, pumped through one or both residual heat removal system heat exchangers and delivered to the vessel, the suppression pool, the drywell spray header, or the suppression pool vapor space spray header. Section 6.2.2.3 states that containment spray is not required for heat removal and section 6.2.1.1.8.3 states that drywell spray is needed for temperature control so mixing fans are not required.

The containment spray system has been maintained as a safety-related system. Plant personnel stated that this was because of a commitment to the NRC to do so, not because it is a regulatory requirement. This position was not contained in the Final Safety Analysis Report that was in existence at the time of licensing.

iii. The team found that the shutdown cooling mode of the residual heat removal system was also considered to be non-safety related since there were two alternate shutdown cooling trains using the safety valves discharging cold water into the suppression pool. In reviewing the Safety Evaluation Report, the team noted that the alternate shutdown cooling mode was simply a backup to the shutdown cooling mode of the residual heat removal system because some of the piping for the residual heat removal system was not redundant. The alternate shutdown cooling mode was considered by the NRC to meet the single-failure requirements that the residual heat removal system lacked because the suction line was not redundant.

The team reviewed the Final Safety Analysis Report, Section 7.4.2.2, which stated that portions of the residual heat removal system could be downgraded from safety-related. The team noted that ,at the time of the inspection, the shutdown cooling mode of the residual heat removal system was maintained as safety related.

In addition, the team reviewed the Design Specification Division 300, Section 311, "Residual Heat Removal System," Revision 3. The team found that this document stated that the shutdown cooling mode of the residual heat removal system was not safety related. The team determined that the shutdown cooling mode of the residual heat removal system is required to be safety related because it fulfills Criterion 34 of Appendix A to 10 CFR Part 50, "Residual Heat Removal."

iv. The team identified that the design specification, a design basis document, regarding the standby gas treatment system in Section I-064 incorrectly listed the design and operating pressures for the standby gas treatment system as positive numbers rather than negative numbers. Engineers introduced this error in a change dated November 3, 1983. For 20 years, the design specification has contained an obvious error.

v. The team identified that Calculations 9.23.00, "Heating Ventilation Air Containment - Standby Gas Treatment," Revision 1, with Calculation Modification Record 91-0247; and 9.49.33, "Building Volume and Air Change," Revision 0, had two different values for the reactor building volume. While researching this NRC-identified error, engineers found additional volumes in design documents. A third example was in Calculation NE-02-94-19, "Secondary Containment Draw Down Analysis," Revision 0, with Calculation Modification Records 95-0199, 95-0206, 96-0211, and Engineering Change 0000002505, as well as two different volumes reported in the Final Safety Analysis Report in Section 6.2.3.4 and in Table 6.2-12. The volumes identified in these documents ranged from 3,225,600 cubic feet to 3,900,000 cubic feet. These conflicts have existed for at least 12 years.

vi. The team found the Technical Specifications Basis, which is a design basis document at Columbia Generating Station, Section B.3.6.4.3, states that only a heater and fan are required for system operability. Engineering was processing a Licensing Document Change Notice Form TSB-03-052, which changed Section B.3.6.4.3 to read that a lead heater and a lead fan or a lag heater and a lag fan are required for system operability.

    The standby gas treatment system has two trains. Each train has two fans and two heaters. There is a discharge valve for each fan. One heater and fan that share the same division of electric power are arbitrarily designated the lead pair and the other the lag pair. The system the engineer did not consider the lead exhaust fan discharge valves (SGT 5A-1, SGT 5B-2) to be necessary for system operability. (These valves are normally closed and the lead valves receive a signal to open upon an accident signal.)

vii. The team found the design specification for the standby gas treatment system listed "General Electric Criteria Documents" and "Burns and Roe Criteria Documents." The engineering staff could not, during the course of the inspection, retrieve these documents. The team noted that the standby gas treatment system was not part of the General Electric scope of supply, but part of the Burns and Roe scope of supply.

<u>Analysis</u>

The team noted that, in response to an NRC request that licensee's inform the NRC of the adequacy and availability of design basis information, the response provided by the Columbia Generating Station's representative stated that the information available at the Columbia Generating Station was adequate and available. This response was based on the idea that the programs governing the control of changes were, and had been, operating effectively. The response contained information that a review and update of the Final Safety Analysis Report would be performed to provide further assurance that the assumption that the design basis information was both adequate and available.

The team found that an update program had been undertaken and completed. However, the errors and inconsistencies, identified previously in this report, were present at the time the review was performed. The personnel performing the review (system and design engineers, licensing engineers, operations and training personnel) failed to identify any of the issues identified by the team.

The team considered the review of the design basis information to have been ineffective. The lack of knowledge of the design basis and design requirements on the part of the personnel performing the review, along with the lack of adequate design basis information, contributed to a false sense of achievement. This resulted in inaccurate and inconsistent design bases and design requirements information being perpetuated.

In addition to the review of the Final Safety Analysis Report, the team noted that there was an audit of the Design Requirements Document for the residual heat removal system. The team found the results to be similarly flawed. The positions presented in this document demonstrated a lack of understanding of the design requirements of the residual heat removal system, not just in the engineering organizations, but across multiple organizations. The team found that this lack of understanding had dated back to at least 1993 when the Final Safety Analysis Report had been revised to insert the position that the shutdown cooling mode of the residual heat removal system was not safety related, and could be down graded.

The team was aware of a similar lack of understanding of design basis information on the part of personnel at the Columbia Generating Station when the reactor core isolation cooling system was down-graded in the 1990s. At that time, plant personnel did not recognize that the reactor core isolation cooling system had a safety function under certain design basis events. As a result, the reactor core isolation cooling system was erroneously down graded.

The inclusion of the words in the Final Safety Analysis Report regarding the idea that the residual heat removal system was troubling to the team. While the system was being maintained at the appropriate level of qualification, there were no assurances that it would continue to be so maintained.

These issues regarding the lack of knowledge and understanding of the design bases and design requirements are considered to be cross-cutting human performance issues.

Enforcement

Each of the above was identified as a minor issue. Therefore, they are not subject to either the significance determination process or traditional enforcement. They are being documented as additional examples of problem identification and resolution issues with cross-cutting aspects.

1RDS3: Changes, Tests, or Experiments

a. Inspection Scope

The inspection was also conducted to monitor the effectiveness of the licensee's implementation of changes to facility structures, systems, and components, risk-significant normal and emergency operating procedures; test programs; and the updated final safety analysis reports in accordance with 10 CFR 50.59, "Changes, Tests, and Experiments."

The sample size for this procedure is 5 evaluations and 10 screenings. The team reviewed three evaluations to verify that the licensee personnel had appropriately considered the conditions under which the licensee may make changes to the facility or procedures or conduct tests or experiments without prior NRC approval. The team selected these three evaluations from the time period covered by the previous inspection because there were no evaluations performed during the time since the last inspection. The team reviewed 12 screenings, in which the licensee personnel determined that evaluations were not required, to ensure that the exclusion of a full evaluation was consistent with the requirements of 10 CFR 50.59.

b. Findings

No findings of significance were identified.

## 4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems (71152)

The team identified a cross-cutting issue in the problem identification and resolution area during this inspection of safety system design and performance (see Section 1RDS1 Enclosure 1b., above). The team found that plant personnel from several organizations (i.e., system engineering, design engineering, licensing, regulatory compliance, operations, training, and maintenance) had failed to identify many conditions adverse to quality and take prompt corrective actions to correct the issues. As a result, there was a loss of a significant amount of design bases and design requirements information because of incorrect and inconsistent information being included in the collection of design documents and, in some cases, being incorporated into the plant (e.g., minimum flow for the recombiners, stroke time for the standby gas treatment system valves).

4OA6 Management Meetings

Exit Meeting Summary

On September 25, 2003, the team leader presented the inspection results to Mr. J. V. Parrish, and other members of licensee management and staff who acknowledged the findings. The team leader confirmed that proprietary information was presented to the team for review. However, no proprietary information is contained within this report.

**ATTACHMENT**

PARTIAL LIST OF PERSONS CONTACTED

Licensee:

P. Ankrum, Licensing Engineer
D. Atkinson, Vice President, Technical Support
J. Dobson, Mechanical Design Engineer
D. Feldman, Acting Plant General Manager
D. Gireaux, System Engineer, Standby Gas Treatment System
M. Holle, System Engineer, Containment Atmospheric Control System
T. Hoyle, Component Engineering Supervisor
M. Humphries, Engineering Manager
P. Inserra, Plant Engineering Manager
W. LaFramboise, Design Engineering Supervisor
V. Parrish, Chief Executive Officer/Chief Nuclear Officer
C. Perino, Licensing Manager
M. Schmitz, System Engineer, Residual Heat Removal System

NRC

G. Reploggle, Senior Resident Inspector
B. Benney, Project Manager, Columbia Generating Station
P. Elkman, Emergency Preparedness Inspector

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

| | | |
|---|---|---|
| 05000397/2003010-001 | NCV | Failure to Demonstrate Acceptability of As-Built Configuration of Safety/Relief Valve Piping (Section 1RDS1b.1.) |
| 05000397/2003010-002 | NCV | Inadequate Calculation (Section 1RDS1b.2.) |
| 05000397/2003010-003 | NCV | Failure to Include Design Requirements in Procedures (Section 1RDS1b.3.) |
| 05000397/2003010-004 | NCV | Failure to Use Appropriate Stroke Times into the Inservice Testing Program (Section 1RDS1b.4.) |
| 05000397/2003010-005 | NCV | Emergency Operating Procedures Would not Accomplish the Intent of the Procedures (Section 1RDS1b.5.) |
| 05000397/2003010-006 | NCV | Inadequate Acceptance Criteria for Recombiner Performance Surveillance (Section 1RDS1b.6.) |
| 05000397/2003010-007 | NCV | Use of Unsupported Assumptions in Performance of Calculation (1RDS1b.7.) |

DOCUMENTS REVIEWED

Calculations

| NUMBER | DESCRIPTION | REVISION |
|---|---|---|
| 5.09.04 | Iodine Content in Containment Atmosphere Following a Loss of Coolant Accident | 0 |
| 5.17.05 | “RHR System: Water Leg Pumps, Head, NPSH, RCIC, LPCS, and HPCS | 0 |
| 5.32.19 | “Velocity of Service Water Flow Through RHR Heat Exchanger” | 0 |
| 6.19.42 | Reactor Building Roof Plan | 2 |
| 9.22.24 | Containment Head Exhaust Fan (CRA-FN-4A & 4B) Pressure Drop Calculations | 1 |
| 9.23.00 | Heating Ventilation Air Containment - Standby Gas Treatment | 1 with Calculation Modification Record 91-0247 |
| 9.23.05 | Operating and Design Conditions for Standby Gas Piping | 0 |
| 9.49.33 | Building Volume and Air Change | 0 |
| CMR-94-0007 | Change to NE-02-92-39 Rev. 1 | 0 |
| CMR-94-0008 | Change to NE-02-92-39 Rev. 1 | 0 |
| CMR-94-0286 | Change to NE-02-92-39 Rev. 1 | 0 |
| CMR-97-0250 | Change to NE-02-92-39 Rev. 1 | 0 |
| E/I-02-92-1024 | Setpoint and Allowable Value Determination for Instrument Loops | 2 |

Calculations

| NUMBER | DESCRIPTION | REVISION |
|---|---|---|
| E/I-02-91-1055 | Setting Range Determination for Instrument Loops SGTR-FT-1A1, 1A2, 1B1, and 1B2 | 0 with Calculation Modification Record 92-0393 |
| E/I-02-91-1065 | Setting Range Determination for Instrument Loop SGT-DPIC-1A1, 1A2, 1B1, AND 1B2 | 1 |
| E/I-02-91-1064 | Setting Range Determination for Instrument Loop RES-DPT-1A1, 1A2, 1A3, 1A4 | 0 |
| EQ-02-85-286021-1 | Seismic and Hydrodynamic Qualification of RHR-R0-6A, B, C, anhd RHR-R0-7A, B, C | 0 |
| ME-02-83-24 | RHR Pump Minimum Flow Calculation | 0 |
| ME-02-85-70 | Seal Coolers for RHR-P-2A, 2B, and 2C | 0 |
| ME-02-89-09 | Reactor building Pressure Transient for LOOP/LOCA | 1 |
| ME-02-92-23 | Main Steam Line Flooding | 0 |
| ME-02-92-41 | Ultimate Heat Sink Analysis | 5 with Calculation Modification Records 00-2449 and 00-2479 |
| ME-02-92-43 | Room Temperature Calculation for Diesel Generator, Reactor Building, Radwaste Building and Service Water Pumphouse Under Design Basis Accident Conditions | 7 |
| ME-02-94-75 | Potential Loss of Containment Pressure Suppression From CAC Test Switch | 0 |
| ME-02-99-19 | MOV Pressure Locking Evaluation for CAC-2, 6, 8, 11, 13, 15 & 17 | 0 |
| ME-02-99-20 | Standby Gas Treatment Filter Unit Carbon Bed Face Velocity | 0 |

Calculations

| NUMBER | DESCRIPTION | REVISION |
|---|---|---|
| NE-02-02-04 | Post-Accident Iodine Mass Deposited on Standby Gas Treatment Charcoal Filter | 0 |
| ME-02-03-12 | Suppression Pool Temperatures during PFSS w/o Cooling and With Cooling Initiated After Reactor Blowdown | 0 |
| NE-02-82-10 | Relative Humidity in Reactor Building | 0 |
| NE-02-85-12 | Secondary Containment Bypass Leakage Limit | 1 |
| NE-02-90-53 | RHR System Description and Analysis in Support of WNP-2 IPE/PSA | 1 |
| NE-02-91-01 | Standby Gas Treatment Humidity | 0 |
| NE-02-91-34 | Volume and Mass of Carbon in Standby Gas Treatment | 0 |
| NE-02-92-06 | Standby Gas Treatment Annubar Flow Meter Correction Factors | 0 |
| NE-02-92-39 | Calculation for CAC Performance With Reduced Capacilty | 1 |
| NE-02-94-14 | Determination of Containment $L_a$ | 1 |
| NE-02-94-19 | Secondary Containment Draw Down Analysis | 0 with Calculation Modification Record 95-0199, 95-0206, 96-0211, and Engineering Change 0000002505 |
| NE-02-94-25 | Calculation for Hydrogen Mixing Study for Containment Following Design Basis Accident | 0 |
| NE-02-97-16 | EOP/SAG Appendix C Calculations | 1 |

Drawings

| NUMBER | DESCRIPTION | REVISION |
|---|---|---|
| 796E379 | Residual Heat Removal System, Sheet 1 | 7 |
| 796E379AA | Residual Heat Removal System, Sheet 1 | 5 |
| E505-1 | DC One Line Diagram | 83 |
| EWD-46E-239D | Electrical Wiring Diagram - AC Electrical Distribution System Power Panel E-PP-7AE Circuit Details | 0 |
| EWD-46E-251D | Electrical Wiring Diagram - AC Electrical Distribution System Power Panel E-PP-8AE Circuit Details | 0 |
| M543-1 | Flow Diagram - Reactor Building Primary Containment Cooling and Purge System | 80 |
| M543-2 | Flow Diagram - Reactor Building Primary Containment Cooling and Purge System | 8 |
| M543-3 | Flow Diagram - Reactor Building Primary Containment Cooling and Purge System | 1 |
| M544 | Flow Diagram - HVAC - Standby Gas Treatment System - Reactor Building | 67 |
| M545-3 | Flow Diagram - Heating, Ventilation and Air Conditioning Reactor Building | 20 |
| M554 | Flow Diagram - H. & V. Containment Atmospheric Control System Reactor Building | 57 |
| M620-544-1 | Control Logic Diagram-Standby Gas Treatment System | 9 |
| M620-544-2 | Control Logic Diagram-Standby Gas Treatment System | 9 |
| M620-544-3 | Control Logic Diagram-Standby Gas Treatment System | 10 |

Drawings

| NUMBER | DESCRIPTION | REVISION |
|---|---|---|
| M620-544-4 | Control Logic Diagram-Standby Gas Treatment System | 9 |
| M620-544-5 | Control Logic Diagram-Standby Gas Treatment System | 3 |
| M620-544-6 | Control Logic Diagram-Standby Gas Treatment System | 5 |
| M620-544-7 | Control Logic Diagram-Standby Gas Treatment System | 5 |
| M620-544-8 | Control Logic Diagram-Standby Gas Treatment System | 4 |
| M620-544-9 | Control Logic Diagram-Standby Gas Treatment System | 4 |
| M620-544-12 | Control Logic Diagram-Standby Gas Treatment System | 5 |
| M783 | Flow Diagram - Primary Containment N2 Inerting System | 37 |

Miscellaneous Documents

| NUMBER | DESCRIPTION | REVISION |
|---|---|---|
| | 10CFR50.59 Resource Manual | 2 |
| | Design Specification for Division 1 Section 1E Mechanical Engineering Criteria | 1 |
| | Design Specification for Division 1 Section 1F Nuclear Power Engineering Design Criteria | 1 |
| | Design Specification Section I Process Piping and Pipe Support ASME Section III Design Specification | 10 |

Miscellaneous Documents

| NUMBER | DESCRIPTION | REVISION |
|---|---|---|
| | Design Specification for Division 300 Section 329 Standby Gas Treatment System | Draft |
| | Flanders Certificate of Conformance for Purchaser Order 00312497 | November 27, 2002 |
| | Interim POC Meeting Minutes/Activities 97-37.02 | September 30, 1997 |
| 39 | Standby Gas Treatment System Health Report | August 14, 2003 |
| 2808-18 | Standby Gas Treatment and Miscellaneous HEPA Filter Units | |
| Amendment No. 176 to NPF-21 and Safety Evaluation | Letter concerning Columbia Generating Station - Issuance of Amendment RE: (TAC NO. MB1777) | June 19, 2002 |
| Amendment No. 137 to NPF-21 and Safety Evaluation | Letter Concerning Issuance of Amendment for the Washington Public Power Supply System Nuclear Project No. 2 (TAC Nos. M87076 and M88625) | May 2, 1995 |
| B-1300-41F | Farr Vendor Bulletin | |
| BRWP-79-411 | Emergency Core Cooling System Pump Room Flooding and Reactor Building Transients | August 15, 1979 |
| BWROG EPGs/SAGs PC-12 | PC/G Monitor and control primary containment hydrogen and oxygen concentrations | 1 |
| CGS EPGs/SAGs PC-7 | PC/G Monitor and control primary containment hydrogen and oxygen concentrations | 1 |
| CVI No. 02-71-00 | Instruction Manual for Post-LOCA Hydrogen Recombiner System | 1 |
| GE-NE-208-17-0993 | Residual Heat Removal System | 1 |
| LO000144 | System Description Standby Gas Treatment System | 11 |

Miscellaneous Documents

| NUMBER | DESCRIPTION | REVISION |
|---|---|---|
| N-621 | RHR-P-2B Pump Curve | 0 |
| N-622 | RHR-P-2A Pump Curve | 0 |
| N-624 | RHR-P-2C Pump Curve | 0 |
| NEI 97-04 | Design Basis Program Guidelines, Appendix B | 1 |
| PMR 91-0333-0 | CAC-FN-1A Discharge/Discharge Piping Modification | |
| TM 102 | Containment Hydrogen Control | November 8, 1971 |
| TM 197 | Post Accident Containment Hydrogen Control System | February 15, 1972 |
| TM 264 | Post Accident Containment Hydrogen Control System | April 26, 1972 |
| TM 407 | Containment Nitrogen Inerting and Make-up Systems | 1 |
| TM 532 | Post-LOCA Hydrogen Recombiners Specification 2808-71 | June 19, 1973 |
| TM-614 | Air Leakage Study-Reactor Building | |
| TM 1076 | Hydrogen Concentration in Primary Containment | March 13, 1978 |
| TM 1255 | Post LOCA Oxygen Generation and Control Inside Primary Containment | March 24, 1982 |
| TM 2065 | Requirements for Containment Mixing Fans | July 15, 1994 |
| TM 2120 | WNP-2 EOP/SAG Technical Document | 0 |
| TM-2099 | Secondary Containment Bypass Leakage | 0 |
| TSB-03-052 | Licensing Document Change Notice | Draft |

Miscellaneous Documents

| NUMBER | DESCRIPTION | REVISION |
|---|---|---|
| USNRC letter GI2-97-039 | Issuance of Amendment for the Washington Public Power Supply System Nuclear Project No. 2 (TAC NO.M 94226) | March 4, 1997 |
| USNRC letter | Safety Evaluation of “BWR Owner’s Group - Emergency Procedure Guidelines, Revision 4," NEDO-31331, March 1987 | September 12, 1988 |
| WPPSS letter GO2-95-265 | WNP-2, Operating License NPF-21, Request for Amendment to Technical Specifications | December 8, 1995 |

Problem Evaluation Requests

| | | | | |
|---|---|---|---|---|
| 200-1138 | 201-1866 | 203-0075 | 203-2966 | 203-3415 |
| 200-1234 | 201-2111 | 203-0078 | 203-2984 | 203-3416 |
| 200-1570 | 201-2169 | 203-0133 | 203-3036 | 203-3449 |
| 201-0045 | 201-2546 | 203-0815 | 203-3036 | 203-3467 |
| 201-0064 | 201-2596 | 203-0864 | 203-3294 | 203-3525 |
| 201-0117 | 201-2910 | 203-0903 | 203-3310 | 203-3532 |
| 201-0405 | 202-0040 | 203-1003 | 203-3323 | 203-3544 |
| 201-0582 | 202-0518 | 203-1049 | 203-3354 | 203-3549 |
| 201-1171 | 202-0681 | 203-1145 | 203-3370 | 203-3552 |
| 201-1181 | 202-0812 | 203-1181 | 203-3372 | 203-3556 |
| 201-1337 | 202-0978 | 203-1417 | 203-3372 | 203-3557 |
| 201-1445 | 202-1596 | 203-2308 | 203-3373 | 203-3560 |
| 201-1463 | 202-2024 | 203-2350 | 203-3379 | 203-3561 |
| 201-1485 | 202-2030 | 203-2411 | 203-3388 | 203-3767 |
| 201-1656 | 202-2063 | 203-2961 | 203-3389 | |
| 201-1861 | 202-2338 | | | |

Procedures

| NUMBER | DESCRIPTION | REVISION |
|---|---|---|
| 1.17.4 | Master Equipment List Update, Control, Use and Authority | 7 |
| 2.3.3A | Containment Atmospheric Control (Div I) | 13 |
| 2.3.5 | Standby Gas Treatment Systems | 24 |

Procedures

| NUMBER | DESCRIPTION | REVISION |
|---|---|---|
| 4.827.K1 | 827.K1 Annunciator Panel Alarms | 7 |
| 4.827.K2 | 827.K1 Annunciator Panel Alarms | 6 |
| 5.0.10 | Flowchart Training Manual | 6 |
| 5.2.1 | Primary Containment Control (Flow Chart) | 13 |
| 5.5.14 | Emergency Wetwell Venting | 5 |
| 5.5.15 | Emergency Drywell Venting | 4 |
| 5.5.16 | Emergency Drywell and Wetwell Purging | 6 |
| 5.5.20 | Emergency Wetwell Venting With High Hydrogen and Oxygen Concentrations | 5 |
| 5.5.21 | Emergency Drywell Venting With High Hydrogen and Oxygen Concentrations | 5 |
| 5.8.1 | Post-LOCA Hydrogen/Oxygen Monitoring | 1 |
| 7.4.6.5.1.2 | Standby Gas Treatment Functional Test | June 25, 1985 |
| ABN-RHR-SDC-ALT | Residual Heat Removal Alternate Shutdown Cooling | 2 |
| ISP-CMS-Q302 | Accident Monitoring Instruments Containment Hydrogen/Oxygen Analyzer - Div I - CC | 7 |
| MSP-SGT-B101 | Standby Gas Treatment System Unit A HEPA Filter Test | September 4, 2003<br>November 6, 1997<br>November 2, 2001 |
| MSP-SGT-B102 | Standby Gas Treatment System Unit B HEPA Filter Test | April 16, 1997<br>November 2, 2000<br>December 3, 2002<br>October 9, 1998 |

Procedures

| NUMBER | DESCRIPTION | REVISION |
|---|---|---|
| MSP-SGT-B103 | Standby Gas Treatment Filtration System Unit A Carbon Adsorber Test | November 10, 1997<br>November 30, 2001<br>December 15, 1999 |
| MSP-SGT-B104 | Standby Gas Treatment Filtration System Unit B Carbon Adsorber Test | April 16, 1997<br>September 6, 2000<br>November 5, 1998<br>December 18, 2002 |
| OSP-CAC-B701 | CAC-HR-1A Preheater Operability Test | 6 |
| OSP-INST-H101 | Shift and Daily Insrtument Checks (Modes 1, 2, 3) | 42 |
| OSP-RHR/IST-R701 | RHR A Check Valve Operability - Refueling Shutdown | 1, 3, 5 |
| OSP-RHR/IST-R702 | RHR B Check Valve Operability - Refueling Shutdown | 1, 2, 4 |
| OSP-RHR/IST-R703 | RHR C Check Valve Operability - Refueling Shutdown | 1, 2 |
| OSP-RHR/IST-Q702 | RHR Loop A Operability Test | 12, 13, 14 |
| OSP-RHR/IST-Q703 | RHR Loop B Operability Test | 12, 13 |
| OSP-RHR/IST-Q704 | RHR Loop C Operability Test | 10, 11 |
| OSP-SGT/IST-Q701 | SGT Valve Operability (System A) | June 3, 2003<br>September 17, 2003 |
| OSP-SGT/IST-Q702 | SGT Valve Operability (System B) | August 7, 2003<br>September 17, 2003 |

Procedures

| NUMBER | DESCRIPTION | REVISION |
| --- | --- | --- |
| OSP-SGT-B701 | Standby Gas Treatment System A - Manual Initiation, Bypass Damper, and Heater Test | July 15, 2002 |
| OSP-SGT-B702 | Standby Gas Treatment System B - Manual Initiation, Bypass Damper, and Heater Test | September 9, 2002 |
| OSP-SGT-M701 | Standby Gas Treatment System A Operability | June 3, 2003<br>August 7, 2003 |
| OSP-SGT-M702 | Standby Gas Treatment System B Operability | August 20, 2003 |
| OSP-SGT-SGT/IST-Q701 | Standby Gas Treatment Valve Operability (System A) | 1 |
| OSP-SGT-SGT/IST-Q702 | Standby Gas Treatment Valve Operability (System B) | 1 |
| Section 311 | Design Specification for Division 300 Section 311 Residual Heat Removal System | 0 and 3 |
| SOP-CN-CONT-VENT | Containment Vent, Deinert, Purge, and Ventilating | 1 |
| SOP-CN-START/OPS | Containment Nitrogen Startup and Operations | 2 |
| SPES-2 | The MEL Users Guide | October 7, 2002 |
| SWP-LIC-02 | Licensing Basis Impact Determinations | 3 |
| TSP-BOP/ISOL-B501 | Balance of Plant Isolation Logic System Functional Test | June 13, 2003 |
| TSP-CAC-B701 | CAC-HR-1A Functional Test and Visual Examination | 7 |
| TSP-CONT-R801 | Containment Isolation Valve and Penetration Leak test Program | June 24, 2003 |
| TSP-RB-B501 | Reactor Builing (Secondary Containment) Drawdown/Leakage Function Test | June 25, 2003 |

Procedures

| NUMBER | DESCRIPTION | REVISION |
|---|---|---|
| TSP-SGT-B101 | Standby Gas Treatment - System A - Flow and Filter Pressure Drop Test | November 2, 2001 |
| TSP-SGT-B102 | Standby Gas Treatment - System B - Flow and Filter Pressure Drop Test | December 29, 2001 |
| TSP-SGT-B501 | Standby Gas Treatment System Functional Test | March 10, 2003 |

Safety Evaluations

| NUMBER | DESCRIPTION |
|---|---|
| SE-01-0018 | Evaluation and justification for use of new mode of RCIC system operation to provide increased control of reactor pressure vessel water level |
| SE-01-0024 | Revised test requirements for containment atmosphere and suppression pool flow check valves |
| SE-99-0033 | FSAR revision to remove commitment to vent non-seismic portion of CRD system to secondary containment during Type A containment leakage rate testing |

Safety Evaluation Sceenings

| NUMBER | DESCRIPTION |
|---|---|
| 01-0040 | Fuse and thermal overload change for MOV RHR-V-40 |
| 01-0045 | RCIC-V-90 closure test revision |
| 01-0051 | AC electric breaker racking procedure revision |
| 01-0058 | Additional response to RRC Delta T annunciator |
| 01-0089 | Revised stroke times for FDR-V-222 |
| 01-0095 | Revision of GE reactor decay heat calculations |
| 01-0122 | Drawing revisions to more accurately reflect plant configuration |

Safety Evaluation Sceenings

| NUMBER | DESCRIPTION |
|---|---|
| 02-0065 | RPS room temperature limit increase to 104°F |
| 02-0269 | Lead blankets found installed on 4" EDR pipe without required evaluation |
| 03-0011 | Plant Design Change 0000001321, deactivation of electro-hydraulic operators associated with the RCC Temperature Control Valves |
| 03-0119 | Commitment to examine all circumferential butt welds within the break exclusion region (BER) |
| 03-0120 | ECCS time delay relay setpoint calculation revision due to new drift analysis and updated references |