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

November 29, 2005

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

SUBJECT: COLUMBIA GENERATING STATION - NRC BASELINE INSPECTION REPORT 05000397/2005009

Dear Mr. Parrish:

On October 21, 2005, the US Nuclear Regulatory Commission (NRC) completed an inspection at your Columbia Generating Station. The enclosed inspection report documents the inspection results, which were discussed on October 20, 2005, with you and other members of your staff.

This 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. The inspectors reviewed selected procedures and records, observed activities, and interviewed cognizant plant personnel.

Based on the results of this inspection, the NRC has identified six issues that were evaluated under the risk significance determination process as having very low safety significance (green). The NRC has also determined that violations are associated with these issues of very low safety significance which are being treated as noncited violations, consistent with Section VI.A of the Enforcement Policy. These noncited violations are described in the subject inspection report. If you contest the violation or significance of these noncited violations, 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 facility.

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Sincerely,

/RA/

Dwight D. Chamberlain, Director
Division of Reactor Safety

Docket: 50-397
License: NPF-21

Enclosures:
Inspection Report 05000397/2005009
w/Attachment: Supplemental Information

cc w/enclosures:
W. Scott Oxenford (Mail Drop PE04)
Vice President, Technical Services
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

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

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

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

Gregory V. Cullen (Mail Drop PE20)
Supervisor, Licensing
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

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

Dale K. Atkinson (Mail Drop PE08)
Vice President, Nuclear Generation
Energy Northwest
P.O. Box 968
Richland, WA 99352-0968

William A. Horin, Esq.
Winston & Strawn
1700 K Street, NW
Washington, DC 20006-3817

Matt Steuerwalt
Executive Policy Division
Office of the Governor
P.O. Box 43113
Olympia, WA 98504-3113

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

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-397
License:	NPF-21
Report No.:	05000397/2005-009
Licensee:	Energy Northwest
Facility:	Columbia Generating Station
Location:	Richland, Washington
Dates:	October 3 - 21, 2005
Team Leader:	C. J. Paulk, Senior Reactor Inspector Engineering Branch 1
Inspectors:	G. A. George, Reactor Inspector Engineering Branch 1 S. T. Graves, Reactor Inspector (NSPDP) Division of Reactor Safety S. P. Rutenkroger, PhD, Reactor Inspector (NSPDP) Division of Reactor Safety
Accompanying Person:	O. S. Mazzoni, PE, PhD, Contractor Beckman and Associates
Approved By:	Dwight D. Chamberlain, Director Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000397/2005-009; 10/3-21/2005; Columbia Generating Station; Evaluation of Changes, Tests, and Experiments; and Safety System Design and Performance Capability.

The report covered a 3-week period of inspection by a team of four regional inspectors and one contractor. Eight Green findings, of which seven were noncited violations, were identified. The significance of most findings is indicated by its 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.

A. NRC-Identified and Self Revealing Findings

Cornerstone: Mitigating Systems

- Severity Level IV. The team identified a finding of very low safety significance involving a noncited violation of 10 CFR 50.59, *Changes, Tests, and Experiments*, for the failure to obtain a license amendment pursuant to 10 CFR 50.90 for changes to the Updated Final Safety Analysis Report. In addition, licensee personnel did not create a written evaluation providing the bases for the determination that the change would not require a license amendment pursuant to paragraph (c)(2) of 10 CFR 50.59. In 2001, licensee personnel changed the Updated Final Safety Analysis Report by replacing allowable emergency diesel generator bus voltage minimums of 85, 81.9, and 80 percent with allowable minimum bus voltages of 75 percent.

  These changes to the Updated Final Safety Analysis Report result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the Updated Final Safety Analysis Report. By being a failure to adhere to a standard (10 CFR 50.59) that was reasonably within the licensee personnel's ability to foresee and correct, this issue qualifies as a performance deficiency and a finding. Since the issue has the potential to impact the NRC's ability to perform its regulatory function, this finding must be evaluated under traditional enforcement utilizing the Enforcement Manual with further NRC management review.

  This violation is more than minor because the change would have required NRC approval in accordance 10 CFR 50.59, as discussed previously. Since no changes were implemented in the plant, all systems designed to prevent or mitigate serious safety events were able to perform their intended safety functions. Therefore, since there was no actual loss of safety function, the finding screens as Green in Phase 1 of Manual Chapter 0609, Appendix A. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 2-05-08086, this violation is being treated as a Severity Level IV noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000397/2005009-001, Failure to Conduct an Evaluation and Obtain a License Amendment for FSAR Changes. (Section 1R02b.1)

- Green. The team identified a finding of very low safety significance involving a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, *Design Control*, in that, licensee personnel established a trip setpoint for the primary undervoltage relays outside of the technical specifications allowable range.

  This finding was more than minor because the lack of adequate control and quality of design basis calculations could result in undervoltage relay trip setpoints being set outside design basis allowable values. However, because of the very low safety significance, in that, it did not represent an actual loss of safety function, no instances were discovered where relay setpoints were actually set outside of allowable limits. Because this violation is of very low safety significance and was entered into the licensee's corrective action program as Condition Report 2-05-08128, this violation is being treated as a Green noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000397/2005009-003, Inadequate Loss-of-Voltage Relay Setpoint Calculation. (Section 1R21b.1)

- Green. The team identified a finding of very low safety significance involving a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, for failure to assure that Calculation E/I-02-91-1076, *Setting Range Determination for RHR-RLY-K70A*, Revision 2, incorporated temperature uncertainties.

  This finding is more than minor since the finding is similar to Example 3.j. of Appendix E to Manual Chapter 0612. The finding was of very low safety significance because it did not represent an actual loss-of-safety function. Because this violation is of very low safety significance and was entered into the licensee's corrective action program as Condition Report 2-05-07816, this violation is being treated as a Green noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000397/2005009-004, Failure to Incorporate Correct Design Basis Conditions in Design Basis Calculation. (Section 1R21b.2)

- Green. The team identified a finding of very low safety significance involving a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, for utilizing a nonconservative calculation in determining the usable volume in the diesel fuel oil storage tanks. Specifically, the licensee did not properly account for vortex prevention in the suction line in Calculation NE-02-87-20, *[Fuel] Oil Tank Capacity vs. Level*, Revision 1.

  This finding is a performance deficiency because the licensee did not properly evaluate and document the unusable volume of the diesel fuel oil storage tanks necessary to prevent vortexing and unreliable pumping. Through subsequent calculations and discussions, the licensee was able to demonstrate that there is sufficient margin in each tank capacity without affecting operability of emergency diesel generators. The issue is more than minor because it is similar to Example 3.i., of Manual Chapter 0612, *Power Reactor Inspection Reports*, September 30, 2005, Appendix E, since it was necessary to reperform the calculation to assure that the 7-day operating time for the emergency diesel generators was met. Since there is available margin in the tank capacity, this issue was confirmed not to involve a loss-of-safety function. Using Phase 1 of the significance determination process, the finding screens as having very low safety significance (Green). Because this violation is of very low safety significance and was entered into the licensee's corrective action program as Condition Report 2-05-07769, this violation is being treated as a Green noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000397/2005009-006, Diesel Fuel Oil Unusable Volume. (Section 1R21b.4)

Cornerstone: Initiating Events

- <u>Green</u>. The team identified a finding of very low safety significance involving a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, for failure to assure that Calculations 2.12.18, *Primary Undervoltage Relays on Buses SM-7 & 8*, Revision 3, and 2.12.24, *Primary Undervoltage Relays for Bus SM-4*, Revision 4, correctly incorporated design basis conditions. Specifically, temperature uncertainty effects were inappropriately omitted from the calculations.

  This finding is more than minor since the finding is similar to Example 3.j. of Appendix E to Manual Chapter 0612. The finding was of very low safety significance because it did not represent an actual loss-of-safety function. Because this violation is of very low safety significance and was entered into the licensee's corrective action program as Condition Report 2-05-08126, this violation is being treated as a Green noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000397/2005009-005, Failure to Utilize Design Basis Conditions into Design Basis Calculation. (Section 1R21b.3)

Cornerstone: Barrier Integrity

- <u>Severity Level IV</u>. The team identified a finding of very low safety significance involving a noncited violation of 10 CFR 50.59 for the failure to obtain a license amendment prior to implementing a new methodology for determining spent fuel pool heat loading.

  This finding is a performance deficiency because the licensee failed to meet requirements of 10 CFR 50.59. Specifically, licensee personnel failed to gain prior approval for a departure from an approved method to calculate spent fuel pool heat loading. Because violations of 10 CFR 50.59 are considered violations that impact the regulatory process, they are dispositioned using the traditional enforcement process. As stated in the NRC Enforcement Policy, the finding is more than minor because the departure from the method would require

Commission review and approval prior to implementation. Because this violation is of very low safety significance and was entered into the licensee's corrective action program as Condition Report 2-05-08137, this violation is being treated as a Severity Level IV noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000397/2005009-002, Failure to Obtain Prior Approval Prior to Implementing a New Methodology. (Section 1R02b.2)

B. Licensee-Identified Findings

None.

REPORT DETAILS

1 REACTOR SAFETY

1R02 Evaluation of Changes, Tests, or Experiments

a. Inspection Scope

The procedure requires a minimum sample size of five evaluations and 10 screenings. The team reviewed five licensee-performed safety evaluations to verify that the licensee 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 reviewed 11 licensee-performed screenings, in which a full evaluation had been excluded. The team did such to ensure consistency with the requirements of 10 CFR 50.59, *Changes, Tests, and Experiments*, in the exclusion of a full evaluation.

The team also reviewed changes made to the Updated Final Safety Analysis Report (USAR) and permanent plant modifications to determine if the requirements of 10 CFR 50.59 were properly implemented.

The inspectors reviewed a sample of 10 corrective action documents written by licensee personnel and a self-assessment that was performed since the last NRC inspection of safety evaluation-related activities to determine whether licensee personnel properly identified and subsequently resolved problems or deficiencies.

b. Findings

Two findings of very low safety significance were identified as noncited violations.

b.1 Failure to Conduct an Evaluation and Obtain a License Amendment for Updated Final Safety Analysis Report Changes

Introduction: The team identified a violation of very low safety significance for the failure to obtain a license amendment pursuant to 10 CFR 50.90. In addition, licensee personnel did not create a written evaluation to provide the bases for the determination that the change would not require a license amendment pursuant to paragraph (c)(2) of 10 CFR 50.59. In 2001, licensee personnel changed the USAR by replacing allowable Emergency Diesel Generator Divisions 1 and 2 bus voltage minimums of 85 percent, 81.9 percent, and 80 percent with allowable minimum bus voltages of 75 percent.

Description: Licensee personnel made these changes in response to an NRC violation documented in NRC Inspection Report 50-397/01-06, by generating Problem Evaluation Request 201-1790 and identifying multiple references to conflicting minimum bus voltages with the emergency diesel generators supplying the load. In response to this condition, licensee personnel changed all emergency diesel generator minimum bus voltages to a uniform 75 percent (and corresponding maximum 25 percent voltage dip).

Allowing the bus voltage to drop to as low as 75 percent has several impacts. First, the USAR states that the minimum starting voltage for Class 1E motors will be 80 percent, except where a deviation has been permitted with an analysis performed and documented to demonstrate that the motor can perform its safety function with the voltage supplied. No analysis was performed or documented for the Class 1E motors being supplied with a minimum voltage of 75 percent. In addition, all reviewed supporting design basis calculations assumed motor starting times based on at least 80 percent minimum bus voltages for Divisions 1 and 2. Therefore, under the current design basis, all Class 1E motors on Divisions 1 and 2 must be assumed incapable of fulfilling their safety function with the buses being supplied by the emergency diesel generators with the USAR stated minimum bus value of 75 percent.

In addition, the lower analytical limit for the relay for the start of the residual heat removal pump in the load sequencer assumes a minimum bus voltage of 85 percent to yield a low pressure core spray motor start time of 2.8 seconds. With a lower bus voltage, the low pressure core spray pump start time will be even longer. In Calculation E/I-02-91-1075, *Setting Range Determination for E-RLY-LPCS/62/1*, Revision 3, a bus voltage of 80 percent is given as an example and stated to yield a low pressure core spray motor start time of 3.7 seconds. Therefore, the low pressure core spray pump and residual heat removal pumps would be starting at the same time if the minimum bus voltage was too low. This finding affects both Divisions 1 and 2 and reduces the safety margin for both divisions of onsite power with minimum bus voltages less than 85 percent (and inoperable at some point below 80 percent). However, a review of actual emergency diesel generator start/load data demonstrates actual minimum bus voltages of at least 85 percent. Therefore, both divisions do have assurance of being, and having been, operable.

<u>Analysis</u>: The team determined that the failure to submit these changes to the NRC for review and approval and failure to conduct an evaluation pursuant to 10 CFR 50.59 was a performance deficiency. The Mitigation Systems Cornerstone was affected because the finding is associated with the operability, availability, reliability, or function of a system or train in a mitigating system, the Division 1 and 2 emergency diesel generators. Since the issue has the potential to impact the NRC's ability to perform its regulatory function, this finding must be evaluated under traditional enforcement.

According to the guidance contained within the Enforcement Manual and Enforcement Policy, the finding qualifies as more than minor because the change would have required NRC approval in accordance with 10 CFR 50.59, as discussed previously. Since no changes were implemented in the plant to reflect the USAR change to 75 percent minimum bus voltage, all systems designed to prevent or mitigate serious safety events were able to perform their intended safety functions. The finding screens as Green in Phase 1 of Manual Chapter 0609, Appendix A, since there was no actual loss-of-safety function. Being a failure to obtain prior Commission approval required by 10 CFR 50.59 for a change that was evaluated as having very low safety significance (i.e., green) by the significance determination process, this issue has been assigned as a Severity Level IV violation.

Enforcement: 10 CFR 50.59 states, in part, that a licensee shall obtain a license amendment pursuant to Section 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety previously evaluated in the USAR.

Contrary to the above, licensee personnel made changes to the USAR in 2001, which would result in more than a minimal increase in the likelihood of occurrence of a malfunction of the Division 1 and 2 emergency diesel generators and their supplied Class 1E motors without obtaining a license amendment. Because this violation is of very low safety significance and has been entered into the licensee's corrective action program as Condition Report 2-05-08086, this violation is being treated as a Severity Level IV noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000397/2005009-001, Failure to Conduct an Evaluation and Obtain a License Amendment for FSAR Changes.

b.2 Failure to Obtain NRC Approval Prior to Implementing a New Methodology

Introduction: The team identified a violation of very low safety significance for the failure to obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a new methodology for determining spent fuel pool heat loading.

Description: During a review of the licensee's safety evaluations, the team reviewed Evaluation 5059-05-0001. This safety evaluation evaluates a change of methods from Auxiliary Systems Branch Technical Position (ASBTP) 9-2 to the ORIGEN-ARP code. These methods determine decay heat loads of irradiated fuel bundles. According to USAR, Section 9.1.3.2.1, Position ASBTP 9-2 is used in the analysis of maximum spent fuel pool temperature. With the proposed change to methodologies, ORIGEN-ARP, which has been approved for use in the evaluation of independent spent fuel storage installations, would be used in the analysis of maximum spent fuel pool temperature.

Evaluation 5059-05-001 discussed the change in methodology in two parts: (1) implement a change from Position ASBTP 9-2 to ORIGEN-2, which is a previously approved methodology for the analysis of spent fuel pool heat loads; then, (2) implement a change from ORIGEN-2 to ORIGEN-ARP, which yields results that are "essentially the same" to ORIGEN-2. Using guidance from Nuclear Energy Institute (NEI) 96-07, *Guidelines for 10 CFR 50.59 Evaluations*, Revision 1, and the *USA 10 CFR 50.59, Resource Manual*, licensee personnel determined the change of methodologies was not a departure from a method of evaluation described in the USAR; therefore, licensee personnel concluded that prior Commission approval was not necessary.

However, the team concluded that licensee personnel misinterpreted the guidance of NEI 96-07 and the USA Resource Manual. In reaching this conclusion, the team identified that the phrase “essentially the same” only applies to changing inputs or elements of an approved methodology not changing to a new methodology, Position ASBTP 9-2 to ORIGEN-ARP. Therefore, the change was a departure from a method of evaluation described in the USAR, making prior Commission approval necessary.

From discussions with Nuclear Reactor Regulation (NRR), NRR concluded that this change would need prior Commission approval. Furthermore, Commission approval would likely have been given because ORIGEN-ARP is technically adequate for the application.

Analysis: This finding is a performance deficiency because licensee personnel failed to meet requirements of 10 CFR 50.59. Specifically, licensee personnel failed to gain prior approval for a departure from an approved method to calculate spent fuel pool heat loading. Because violations of 10 CFR 50.59 are considered violations that impact the regulatory process, they are dispositioned using the traditional enforcement process. As stated in the NRC Enforcement Policy, the finding is more than minor because the departure from the method would require Commission review and approval prior to implementation. Since the finding results in a condition of very low safety significance (Green), the finding results in a Severity Level IV violation.

Enforcement: 10 CFR 50.59 (c)(2)(viii) states, in part, that a licensee shall obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a proposed change if the change would result in a departure from a method of evaluation described in the USAR used in establishing design bases or in the safety analyses.

Contrary to the above, in 2005, licensee personnel failed to obtain a license amendment for a change in methodology to evaluate heat loads in the spent fuel pool. Because this violation is of very low safety significance and was entered into the licensee’s corrective action program as Condition Report 2-05-08137, this violation is being treated as a Severity Level IV noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000397/2005009-002, Failure to Obtain Prior Approval Prior to Implementing a New Methodology.

1R21 Safety System Design and Performance Capability

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 safety-related 4160 Vac system and the safety-related battery systems. The primary review prompted parallel review and examination of support systems, such as, fuel oil storage, and fuel and lubricating oil consumption.

The team assessed the adequacy of calculations, analyses, engineering processes, and engineering and operating practices that the licensee used for the selected safety systems 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 USAR, 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 two samples by reviewing the safety-related 4160 Vac system and the safety-related battery systems.

b. <u>Findings</u>

b.1 <u>Inadequate Loss-of-Voltage Relay Setpoint Calculation</u>

<u>Introduction</u>: The team identified a finding of very low safety significance involving a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, *Design Control*, in that, licensee personnel established a trip setpoint for the primary undervoltage relays outside of the technical specifications allowable range.

<u>Description</u>: Undervoltage protection is provided by loss-of-voltage relays, whose primary function is to detect and disconnect the Class 1E (Divisions 1, 2 and 3) buses from the preferred supply when voltage levels reach a setpoint below which safe operation of equipment cannot be assured. The trip setpoints are determined using detailed calculations and analysis and are a balance between ensuring availability of Class 1E systems and preventing damage to safety-related equipment.

The team noted that licensee personnel used nonconservative values in Calculations 2.12.18, *Primary Undervoltage Relays for Buses SM-7 and 8*, Revision 3; and 2.12.24, *Primary Undervoltage Relays for Bus SM-4*, Revision 4. Section 3.0 of the calculations, *Derivation of Maximum Calculated Allowable Value & Maximum Drop out Setting,* established a maximum allowable value of 3157 Vac, which is outside the range allowable by technical specifications (i.e. #3135 Vac and $2450 Vac).

<u>Analysis</u>. The team determined that a performance deficiency existed, in that, licensee personnel failed to use conservative values in a design basis calculation. Furthermore, the team determined that it was reasonably within licensee personnel control to have identified that calculated allowable values exceeded technical specification allowable values.

The team reviewed the examples described in Appendix E to Inspection Manual Chapter 0612, *Power Reactor Inspection Reports*, and determined that none the examples were applicable. Using the criteria in Appendix B of Inspection Manual Chapter 0612, the finding was determined to be more than minor because the finding was associated with the Mitigating System Cornerstone attribute of design control, and the finding was determined to affect the associated cornerstone objective of ensuring the availability and reliability of systems that respond to initiating events to prevent undesirable consequences. The team performed a Phase 1 screening of the finding using the significance determination process in accordance with the guidance in

Inspection Manual Chapter 0609, *Significance Determination Process*, Appendix A. The result of the screening indicated that the finding is of very low safety significance (Green) because the finding does not represent a loss-of-safety system function, it does not represent actual loss-of-safety function of a single train for longer than it's technical specification allowed outage time, it does not represent an actual loss-of-safety function of one or more non-technical specification-related trains of equipment designated as risk-significant in accordance with 10 CFR 50.65, for longer than 24 hours, and it does not screen as potentially risk-significant because of external events.

<u>Enforcement</u>: Criterion III of 10 CFR Part 50, Appendix B, requires, in part, that measures be established to assure that applicable regulatory requirements and the design basis, for those systems, structures and components for which this appendix applies, are correctly translated into specifications, drawings, procedures and instructions.

Contrary to the above, the measures established were inadequate to assure that design basis values were correctly translated into plant documents. Specifically, on February 22, 2001, licensee personnel approved Calculations 2.12.18 and 2.12.24, Revisions 3 and 4, respectively, which derived an allowable values that were outside of the technical specification allowable range for the trip setpoint of the primary undervoltage relays.

Because this violation is of very low safety significance and was entered into the licensee's corrective action program as Condition Report 2-05-08128, this violation is being treated as a Green noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000397/2005009-003, Inadequate Loss-of-Voltage Relay Setpoint Calculation.

b.2 <u>Failure to Include Temperature Uncertainties in Design Basis Calculation</u>

<u>Introduction</u>: The team identified a violation of very low safety significance for the failure to incorporate the correct design basis conditions in Calculations 2.12.18, Revision 3, and 2.12.24, Revision 4.

<u>Description</u>: The calculations established voltage setpoints for the primary undervoltage relays on safety-related Buses SM-4, 7, and 8, with respect to design basis and technical specification requirements. These relays shed their respective buses when the voltage on the associated bus drops below a certain point for a sufficient length of time. By not incorporating the temperature uncertainty effect because of the design basis temperatures, the potential for the relay to shed the bus unnecessarily is increased since the margin between the allowed setpoint and the technical specification and analyzed expected transient conditions is less than expected. In each case, the temperature uncertainty effect was neglected with the justification that the relay was physically located in a "mild environment." However, the design basis maximum temperature for the switchgear rooms is 120EF. Also, other relay calculations assume an 18EF heat rise in addition to design basis room temperatures, which yields a maximum temperature of 138EF. The vendor information states that the relays are

qualified for -20EC to 55EC (-4EF to 131EF). However, there is no indication as to the temperature induced uncertainty effect within this qualified range.

An interview of licensee personnel determined that the uncertainty effect could be bounded by as much as 1 percent per 10EC. This corresponds to a bounding of approximately 3.8 percent uncertainty at high temperatures. The team found that this result compared unfavorably with the included uncertainty effects, which total 3.256 percent. The team also found that, although not explicitly determined in the calculation, the trip setpoint, according to the normal trip setpoint range with as-calculated uncertainty added, yields a 3094 Vac bus voltage, which is less than the 3135 Vac technical specification value (i.e., conservative). By comparison, the trip setpoint according to the normal trip setpoint range with given uncertainty plus worst-case temperature uncertainty added yields a 3254 Vac bus voltage, exceeding the technical specification value (i.e., nonconservative). A licensee engineer initiated Condition Report 2-05-08126 to address this issue.

Analysis: The team determined that failure to incorporate correct design basis conditions into design basis calculations was a performance deficiency. The Initiating Events Cornerstone was affected because the finding is associated with an increase in the likelihood of an initiating event by unnecessarily shedding the safety related buses causing a plant transient. This finding is more than minor since the finding is similar to Example 3.j. of Appendix E to Manual Chapter 0612. The engineering staff had to revise the calculation and conduct operability evaluations in order to ensure the technical specifications were not exceeded. The finding screens as Green in Phase 1 of Manual Chapter 0609, Appendix A, for Initiating Events since it does not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment or functions will not be available.

Enforcement: Criterion III of Appendix B to 10 CFR Part 50 states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Measures shall also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components.

Contrary to the above, on February 22, 2001, the established measures were not adequate in that licensee personnel failed to assure that the design basis was correctly translated into Calculations 2.12.18 and 2.12.24. Because this violation is of very low safety significance and was entered into the licensee's corrective action program as Condition Report 2-05-08126, this violation is being treated as a Green noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000397/2005009-004, Failure to Incorporate Design Basis Conditions into Design Basis Calculation.

b.3 Use of Incorrect Design Basis Conditions in Design Basis Calculation

Introduction: The team identified a violation of very low safety significance for the failure to utilize the correct design basis conditions in Calculation E/I-02-91-1076, *Setting Range Determination for RHR-RLY-K70A,* Revision 2.

Description: Calculation E/I-02-91-1076 determines the total loop uncertainties, minimum and maximum settings, and upper and lower allowable values for the setpoints for the Residual Heat Removal Pump 2A and 2B loss-of-coolant accident/loss-of-offsite power time delay relays. The lower temperature uncertainty effect, TE-, is used to establish the safety margin between the low pressure core spray pump starting time and the residual heat removal pump starting time, which is a technical specification controlled value. The calculation incorrectly utilized the upper temperature limit and upper temperature range conversion factor in order to calculate the lower temperature uncertainty effect instead of the lower temperature limit and lower temperature range conversion factor. The combined effect of using both the incorrect temperature and the incorrect conversion factor was unable to be determined by the licensee's engineers. So, the impact upon the existing safety margin could not be analyzed directly. Therefore, licensee personnel revised the calculation utilizing vendor supplied uncertainty data which bounded and included the temperature induced uncertainty. As a result of this revision, licensee personnel raised the minimum setpoint of the relay from 3.14 to 3.34 seconds in order to maintain sufficient margin between the technical specification lower allowable value of 3.04 seconds. Licensee personnel initiated Condition Report CR 2-05-07816 to address this issue.

Analysis: The team determined that failing to utilize correct design basis conditions in design basis calculations was a performance deficiency. The Mitigation Systems Cornerstone was affected because the finding is associated with the operability, availability, reliability, or function of a system or train in a mitigating system, the load sequencing of both trains of residual heat removal on the Division 1 and 2 Safety Buses SM-75 and SM-85 during a loss-of-coolant accident event. In particular, reducing the margin between the starting of the low pressure core spray pump and the starting of the residual heat removal pump increases the likelihood that both pumps will be loaded onto the bus simultaneously, which exceeds the analyzed starting capacity of the Division 1 and 2 emergency diesel generators. The inspectors considered this finding to be more than minor since the finding is similar to Example 3.j. of Appendix E of Manual Chapter 0612. The engineering staff had to revise the calculation and conduct operability evaluations in order to ensure the technical specifications were not exceeded.

This finding screens as very low safety significance, i.e. Green, in Phase 1 of the significance determination process in the Mitigation Systems Cornerstone of Manual Chapter 0609, Appendix A, since no actual loss-of-safety function was identified. The nominal relay setpoints used in the plant were 5 seconds, ensuring sufficient safety margin. Licensee personnel determined the time delay relays and associated systems and equipment were operable based on relay settings and revised calculations.

Enforcement: Criterion III of Appendix B to 10 CFR Part 50 states, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Measures shall also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components.

Contrary to the above, on April 17, 2003, the measures established were not adequate in that they failed to ensure that the design basis was correctly translated into Calculation E/I-02-91-1076. Because this violation is of very low safety significance and was entered into the licensee's corrective action program as Condition Report 2-05-07816, this violation is being treated as a Green noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000397/2005009-005, Failure to Utilize Correct Design Basis Conditions in Design Basis Calculation.

b.4 Diesel Fuel Oil Unusable Volume

Introduction: The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III having very low safety significance (Green), for utilizing a nonconservative calculation in determining usable volume in the diesel fuel oil storage tanks. Specifically, the licensee did not properly account for vortex prevention in the suction line.

Description: The team reviewed Calculation NE-02-87-20,*[Fuel] Oil Tank Capacity vs. Level*, Revision 1. This calculation evaluated the usable volume for each of the three diesel fuel oil storage tanks. In evaluating the unusable volume, the calculation accounted for the volume of internal structures and the height of the diesel fuel oil transfer pump suction inlet above the bottom of the storage tank. However, the calculation did not address vortexing in the suction line, which could ingest air into the pump, or the level at which the pump would reliably pump.

After the team identified these conditions to licensee engineers, the engineers performed a calculation and found that the elevation of unusable volume in each tank be raised 1 inch. This 1 inch corresponds to a loss of 259 gallons of usable volume in Diesel Tanks 1-A and 1-B, and 202 gallons usable volume in Diesel Tank 2. In order to meet the technical specification minimum volume of 55,500 gallons for Tanks 1-A and 1-B, and 33,000 gallons for Tank 2, the licensee engineers raised minimum requirements for fuel oil elevation 1 inch. Therefore, it was not necessary for a change to technical specifications.

Analysis: This finding is a performance deficiency because the licensee engineers did not properly evaluate and document the unusable volume of the diesel fuel oil storage tanks necessary to prevent vortexing and unreliable pumping. Through subsequent calculations and discussions, the licensee engineers were able to demonstrate that there is sufficient margin in each tank capacity so that operability of emergency diesel generators would not be negatively impacted.

The issue is more than minor because it is similar to Example 3.i. of Inspection Manual Chapter 0609, Appendix E, since it was necessary to reperform the calculation to assure that the 7-day operating time for the emergency diesel generators was met. Because there is available margin in the tank capacity, this issue was confirmed not to involve a loss-of-safety function. Using Phase 1 of the significance determination process, the finding screens as having very low safety significance(Green).

Enforcement: Criterion III of Appendix B to 10 CFR Part 50 states, in part, that measures shall be established to assure that the applicable design bases are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents.

Contrary to the above, the measures established to assure the applicable design bases are correctly transferred in plant documents were not adequate. Specifically, licensee engineers failed to translate design requirements (i.e., vortex prevention) into Calculation NE-02-87-20, Revision 1. Because this violation is of very low safety significance and was entered into the licensee's corrective action program as Condition Report 2-05-07769, this violation is being treated as a Green noncited violation consistent with Section VI.A of the NRC Enforcement Policy:
NCV 05000397/2005009-006, Diesel Fuel Oil Unusable Volume.

b.5 Unendorsed Calculation Methodology

Introduction: The team identified an unresolved item that has generic implications with respect to the methodology used for performing calculations to determine setpoints.

Description: As noted above, the methodology used by the Columbia Generating Station engineers to determine setpoints is similar to Method 3, as identified in ISA-S67.04, Part II, *Recommended Practice - Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation*. [NOTE: ISA-S67.04 Part II is sometimes identified as ISA-RP67.04 Part II.]

The NRC has provided guidance to the nuclear industry for the establishment of setpoints in Regulatory Guide 1.105, *Instrument Setpoint*, Revision 1, 1976; *Instrument Setpoints for Safety-Related Systems*, Revision 2, 1986; and *Setpoints for Safety-Related Instrumentation*, Revision 3, 1999. In Revision 1, the agency was concerned with ". . . the drift of a measured parameter out of compliance with a technical specification . . . [because] the selection of a setpoint . . . [did] not allow a sufficient margin between the setpoint and the technical specification limit to account for inherent instrument inaccuracy . . . In some cases, the setpoint selected was numerically equal to the technical specification limit and stated as an absolute value, thus leaving no apparent margin for error."

In Regulatory Position C.1. of Revision 1, the NRC states that "[t]he setpoints should be established with sufficient margin between the technical specification limits for the process variable and the nominal trip setpoints to allow for (a) the inaccuracy of the instrument, (b) uncertainties in the calibration, and (c) the instrument drift that could occur during the interval between calibrations." In Regulatory Position C.4., the NRC states that "[t]he accuracy of all setpoints should be equal to or better than the accuracy assumed in the safety analysis." And in Regulatory Position C.6., the NRC states that "[t]he assumptions used in selecting the setpoint values in regulatory position 1 and the minimum margin with respect to the limiting safety system settings, setpoint rate of deviation (drift rate), and the relationship of drift rate to testing interval (if any) should be documented."

In Revision 2, the NRC found that ISA-S67.04-1982, *Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants*, established "requirements acceptable . . . for ensuring that instrument setpoints in safety-related systems are initially within and remain within the technical specification limits."

In Revision 3, the NRC found that conformance to ISA-S67.04-1994, *Setpoints for Nuclear Safety-Related Instrumentation*, with noted exceptions, would provide an acceptable method for satisfying regulations for ensuring that setpoints for safety-related instrumentation are established and maintained within technical specifications. However, this revision explicitly states that ISA-S67.04-1994, Part II, *Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation*, was not addressed by the regulatory guide.

Branch Technical Position HICB-12, *Guidance on Establishing and Maintaining Instrument Setpoints*, Revision 4, states that, "[w]hile IEEE Std 603 references ISA-67.04-1988 for setpoint methodology, the Staff has not endorsed this version of the standard. The Staff is endorsing ISA-S67.04, Part 1," in Regulatory Guide 1.105, Revision 3.

Analysis: While the NRC's position on the establishment and maintenance of safety-related setpoints has been well documented and available, many licensees have elected to use a methodology that has not been endorsed by the NRC. As a result, the NRC and the Nuclear Energy Institute (NEI) have been in discussions over the acceptability of the unendorsed methodology. Because this is a generic concern, we (Region IV) cannot take a regulatory position at this time. Therefore, we will consider this as an unresolved item (URI 050000397/2008009-007).

An unresolved item is an item that requires additional information, review, and assessment to determine if the item is a regulatory concern and the significance, if it is a concern. Once the review and assessment are complete, Region IV will document its actions, and bases for those actions, in an inspection report.

4. OTHER ACTIVITIES

4OA6 Meetings, Including Exit

On October 20, 2005, the team leader presented the inspection results to Mr. J. V. Parish, Chief Executive Officer, and other members of his staff who acknowledged the findings. The inspectors confirmed that proprietary information was examined during this inspection; however, no proprietary information is included in the report.

ATTACHMENT

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel

S. Belcher, Manager, Operations
S. Boynton, Manager, System Engineering
B. Boyum, Assistant General Manager, Engineering
D. Coleman, Manager, Regulatory Programs
K. Dittmer, Supervisor, Design Engineering
A. Khanpour, General Manager, Engineering
W. LaFrambois, Manager, Design Engineering
T. Lynch, Plant General Manager
W. Oxenford, Vice President, Technical Services
J. Parrish, Chief Executive Officer
S. Wood, Supervisor, System Engineering

NRC personnel

H. Chernoff, Senior Project Manager - NRR/DLPM/PDIII-1
R. Cohen, Resident Inspector
Z. Dunham, Senior Resident Inspector
N. O'Keefe, Acting Chief, Engineering Branch 1

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

OPENED AND CLOSED

05000397/2005009-001	NCV	Failure to Conduct an Evaluation and Obtain a License Amendment for FSAR Changes (Section 1R02b.1)
05000397/2005009-002	NCV	Failure to Obtain Prior Approval Prior to Implementing a New Methodology (Section 1R02b.2)
05000397/2005009-003	NCV	Inadequate Loss-of-Voltage Relay Setpoint Calculation (Section 1R21b.1)
05000397/2005009-004	NCV	Failure to Include Temperature Uncertainties in Design Basis Calculation (Section 1R21b.2)

OPENED AND CLOSED

05000397/2005009-005	NCV	Use of Incorrect Design Basis Conditions in Design Basis Calculation (Section 1R21b.3)
05000397/2005009-006	NCV	Diesel Fuel Oil Unusable Volume (Section 1R21b.4)

OPENED

05000397/2005009-007	URI	Unendorsed Calculation Methodology (Section 1R21b.5)

LIST OF DOCUMENTS REVIEWED

Section 1R02

50.59 Screens

03-0237	04-0082	04-0127	04-0129	04-0218
03-0277	04-0083	04-0128	04-0162	05-0210
04-0076				

50.59 Evaluations

05-0001
05-0003
05-0004
05-0006
05-0009

Condition Reports

2-05-07649	2-05-07675	2-05-07941	2-05-08058	2-05-08073
2-05-07769	2-05-07680	2-05-07942		2-05-08137
2-05-07694				

Section 1R21

Calculations:

NUMBER	TITLE	REVISION
019001-S	Seismic Qualification of Diesel Generator C1E and SRM Equipment	1
02.12.18	Calculation for Under Voltage Relay on SM-7 and 8,	8
2.12.24	Primary Undervoltage Relays for Bus SM-4	4
343005-S	Seismic Qualification of Diesel Generator C1E and SRM Equipment	1
5.43.02	Diesel Oil Tanks (Storage and Day Tanks) Capacity Verification	0
E/I 02-87-07	Calculation For Load Flow & Voltage Analysis For the Plant Main Buses in AC Distribution Systems	5
E/I-02-91-03	Div.1, Div.2, and Div.3 Diesel Generator Loading Calculation	8
E/I-02-91-1075	Setting range determination for E-RLY-LPCS/62/1	3
E/I-02-91-1076	Setting Range Determination for RHR-RLY-K70A	2 & 3
E/I-02-91-1076	Setting Range Determination for RHR-RLY-K70A	3
E/I-02-91-1137	Setpoint and Allowable Value Determination for Instrument Loops SW-RLY-62/P1A, SW-RLY-TDS/P1A, SW-RLY-62/P1B, SW-RLY-TDS/P1B	0
E/I-02-92-09	Calculation for Short Circuit 4.16 KV and 6.9 KV Buses	1, CMR 515
E/I-02-92-17	Medium Voltage (4.16kV & 6.9 kV) Electrical Distribution System (EDS) Phase Overcurrent Relay Settings	1
ME-02-87-89	DO-TK-1A, 1B Low Level Tech Spec Requirement (analytical limit)	0
ME 02-91-46	Sizing of Air Dryers for DG1A, DG1B, and HPCS Air Start Systems	0
ME-02-91-50	Sizing of DG 1A/B water reservoir tanks	1
ME-02-91-51	Diesel Generator Engine Lube Oil Sump Capacity	0

Calculations:

NUMBER	TITLE	REVISION
ME-02-92-234	On Site Diesel Fuel Storage for the Emergency Diesel Generators DG-1, DG-2, and DG-3	0
ME-02-94-44	Diesel Starting Air System Capabilities to Meet the # of Starts Requirements	1
ME-02-03-02	Diesel Generator Building Flooding Analysis	0
NE-02-87-20	Oil Tank Capacity vs. Level	1

Condition Reports:

2-04-01019	2-05-02528	2-05-07685	2-05-07816	2-05-08057
2-04-03362	2-05-04675	2-05-07724	2-05-07941	2-05-08086
2-04-06340	2-05-05368	2-05-07725	2-05-07986	2-05-08126
2-05-01878	2-05-05474	2-05-07741	2-05-08046	2-05-08129
2-05-02358	2-05-06550	2-05-07753	2-05-08049	

Drawings:

NUMBER	TITLE	REVISION
3084-3, Sht.1	24" - 900# Swing Check Valve w/ Spring Assisted Closing P.S., B.W.E., Air Operator	2
3084-3, Sht 2	24" - 900# Swing Check Valve w/ Spring Assisted Closing P.S., B.W.E., Air Operator	3
3084-3, Sht 3	24" - 900# Swing Check Valve w/ Spring Assisted Closing P.S., B.W.E., Air Operator	2
46E062	AC Electrical Distribution Systems, 4.16 KV SWGR E-SM-1 FDR BRKR E-CB-1/7 SH.1	15
46E078	AC Electrical Distribution Systems, 4.16 KV SWGR E-SM-4 FDR BRKR E-CB-4/2 SH.1	12
46E080	AC Electrical Distribution Systems, 4.16 KV SWGR E-SM-7 FDR BRKR E-CB-7/1 SH.1	17

Drawings:

NUMBER	TITLE	REVISION
46E084	AC Electrical Distribution Systems, 4.16 KV SWGR E-SM-8 FDR BRKR E-CB-8/3 SH.1	19
46E106	AC Electrical Distribution Systems 4.16 kV	15
DSA-2539-1	3/4" Drain From Air Start Skid	3
DSA-2538-1	3/4" Drain From Air Start Skid	3
E501-1	Electric Symbol List, One Line & Elementary Diagrams, Power, Grounding, & Lighting Plans	21
E502-1	Main One Line Diagram	40
E502-2	Main One Line Diagram	52
E504	Vital One Line Diagram	52
E505-1	DC One Line Diagram	84
E505-2	DC One Line Diagram	0
EWD-8E-001	Electrical Wiring Diagram Low Pressure Core	17
EWD-8E-001A	Electrical Wiring Diagram, Low Pressure Core Spray System, LPCS-P-1 (E21-C001)	8
EWD-8E-010	Electrical Wiring Diagram Elementary Low Pressure Core Spray System Misc Relay Circuits	22
EWD-9E-001	Electrical Wiring Diagram, Residual Heat Removal System, Pump RHR-P-2A (E12-C002A)	15
EWD-9E-002	Electrical Wiring Diagram Residual Heat Removal System Pump RHR-P-2A Breaker RHR-CB-P2A	13
EWD-9E-004	Electrical Wiring Diagram, Residual Heat Removal System, Pump RHR-P-2B Breaker RHR-CB-P2B	17
EWD-9E-093	Electrical Wiring Diagram Residual Heat Removal System Miscellaneous Relay Circuits (Div 1)	18

Drawings:

NUMBER	TITLE	REVISION
EWD-13E-001	Electrical Wiring Diagram, Control Rod Drive Systems, Pump CRD-P-1A (C12-C001A)	17
EWD-13E-005	Electrical Wiring Diagram, Control Rod Drive Systems, Pump CRD-P-1B (C12-C001B)	18
EWD-46E-049	Electrical Wiring Diagram, AC Electrical Distribution System, 4.16kV & 6.9kV Switchgear Circuit Breaker Details	2
EWD-46E-106A	Electrical Wiring Diagram AC Electrical Distribution Systems 4.16 kV SWGR SM-7 Crit Bus 7 Undervoltage	16
EWD-47E-002	Electrical Wiring Diagram, Standby AC Power System, Diesel Generator 1 Breaker E-CB-7/DG1	10
EWD-47E-002A	Electrical Wiring Diagram, Standby AC Power System, Diesel Generator 1 Breaker E-CB-7/DG1	1
EWD-47E-003	Electrical Wiring Diagram, Standby AC Power System, Diesel Generator Breaker E-CB-DG1/7	20
EWD-47E-003A	Electrical Wiring Diagram, Standby AC Power System, Diesel Generator 1 Breaker E-CB-DG1/7	10
EWD-47E-005	Electrical Wiring Diagram, Standby AC Power System, Diesel Generator 2 Breaker E-CB-8/DG2 and E-CB-DG2/8	26
EWD-47E-005A	Electrical Wiring Diagram, Standby AC Power System, Diesel Generator No. 2 Breaker E-CB-DG2/8	2
EWD-47E-006	Electrical Wiring Diagram, Standby AC Power System, Diesel Generator 2 Breaker E-CB-8/DG2	20
EWD-47E-006B	Electrical Wiring Diagram, Standby AC Power System, Diesel Generator 2 Breaker E-CB-8/DG2	1
EWD-47E-007	Electrical Wiring Diagram, Standby AC Power System, Diesel Generator 2 Breaker E-CB-DG2/8	16
EWD-47E-007A	Electrical Wiring Diagram, Standby AC Power System, Diesel Generator 2 Breaker E-CB-DG2/8	3

Drawings:

NUMBER	TITLE	REVISION
EWD-47E-008	Electrical Wiring Diagram, Standby AC Power System, Diesel Generator 2 Breaker E-CB-DG2/8	15
EWD-47E-008A	Electrical Wiring Diagram, Standby AC Power System, Diesel Generator 2 Breaker E-CB-DG2/8	9
EWD-47E-008B	Electrical Wiring Diagram, Standby AC Power System, Diesel Generator 2 Breaker E-CB-DG2/8	2
EWD-58E-001	Electrical Wiring Diagram, Standby Service Water System, SW-P-1A	18
EWD-58E-002	Electrical Wiring Diagram Standby Service Water System SW-P-1A Breaker SW-CB-P1A	21
EWD-58E-003	Electrical Wiring Diagram, Standby Service Water System, SW-P-1B	18
EWD-58E-004	Electrical Wiring Diagram, Standby Service Water System, SW-P-1B Breaker SW-CB-P1B	26

Miscellaneous:

NUMBER	TITLE/DESCRIPTION	REVISION/ DATE
Action Request 2395	Need permanent power instead of extension cords in DG bldg. Provide permanent power to devices that are currently powered by extension cords in the Diesel Generator Bldg.	November 10, 2001
CMR-0000000595	Calculation Modification Record	August 25, 2000
CMR-96-0347	Calculation Modification Record	February 3, 1997
CVI/02-02E22-08	Instruction Manual for Vertical Induction Motors Open Enclosures – Square Frames	Issue 1
CVI/02-02E22-07	Instruction Manual for HPCS Diesel Generator	Issue 2

Miscellaneous:

NUMBER	TITLE/DESCRIPTION	REVISION/ DATE
RD 317	Design Specification forDivision 300 Section 317 AC/DC Electrical Distribution System	6
E-mail from Richard A. Hermann to James R. Zimmerschied, Paul T. Hand	DG1 HNES info, with photo attachments	October 5, 2005
E-mail: Powell to Wood	Subject: FW: Lube Oil Consumption	October 20, 2005
E-mail: Ferek to Richey	Subject: RE: Origen ARP	October 18, 2005
E-mail: Chiang to Wood	Subject: HPCS Question Responses	October 5, 2005
EC 2646	Engineering Change	June 2, 2004
EC 2647	Engineering Change	June 2, 2004
EC 3065	Engineering Change	May 3, 2004
EC 4140	Wrong EPN on EWD-47E-008; This ADOC was created to close CR 2-05-02795 & PTL 226784	August 16, 2005
EC 4208	Some SM-7 & SM-8 EWD's show old style breaker internals. This ADOC was created to close CR 2-05-05368 & PTL 229762	September 08, 2005
FAO for PER 203-3198	Follow-up Assessment of Operability of TOC contacts in Various DHP-VR Breakers	September 30, 2003
Field Change Request 99-0140-0-01	Control Schematic for DHP-VR is added to EWD-46E-049	April 16, 2001
LDCN-FSAR-97-126	Licensing Document Change Notice Form	July 30, 1997
LDCN/181449	Licensing Document Change Notice Form 01-064	October 1, 2001

Miscellaneous:

NUMBER	TITLE/DESCRIPTION	REVISION/ DATE
LDCN/183675	Licensing Document Change Notice Form 01-077	January 17, 2002
LDCN/183736	Licensing Document Change Notice Form 01-079	December 10, 2001
LDCN/184483	Licensing Document Change Notice Form 02-002	January 7, 2002
LDCN/187327	Licensing Document Change Notice Form 01-063	April 2, 2002
LDCN/187906	Licensing Document Change Notice Form 02-017	April 23, 2002
LDCN/188899	Licensing Document Change Notice Form 02-025	May 17, 2002
LDCN/189742	Licensing Document Change Notice Form 02-038	June 24, 2002
LDCN/194049	Licensing Document Change Notice Form 02-071	November 15, 2002
LDCN/194771	Licensing Document Change Notice Form 02-046	August 14, 2002
LDCN/198496	Licensing Document Change Notice Form 03-002	January 31, 2003
LDCN/199557	Licensing Document Change Notice Form 02-000	October 30, 2003
NEDO-10905	High-Pressure Core Spray System Power Supply Unit	May 1973
NEI 96-07	Guidelines for 10 CFR 50.59 Implementation	1
Operating Data	8, 12, 16, 20-645E4 Turbocharged Engines	
OSP-HPCS/IST-Q701, Pt. 892, 893, 900, 901	Test and Trend Data HPCS-V-1 and HPCS-V-14	

Miscellaneous:

NUMBER	TITLE/DESCRIPTION	REVISION/ DATE
QID Worksheet 213032	Effect of Motor Space Heaters on Qualified Life of ECCS Pump Motors, Vol. 1	
TMR 03-24	Temporary Modification Request to jumper TOC switch contacts in Safety-Related 4.16 kV Breakers	May 23, 2005
TSP-DG1/LOCA-B501	Div 1 EDG LOOP and LOOP/LOCA Test Data	June 11, 2001 June 7, 2003 August 2, 2005
TSP-DG2/LOCA-B501	Div 2 EDG LOOP and LOOP/LOCA Test Data	June 21, 2001 May 21, 2003 August 2, 2005
TSP-DG3/LOCA-B501	Div 3 EDG LOOP and LOOP/LOCA Test Data	June 20, 2001 June 2, 2003 August 2, 2005
204-0653	Problem Evaluation Request	March 24, 2004
295-0231	Problem Evaluation Request	April 21, 1995
299-2475	Problem Evaluation Request	November 5, 1999
2001-002	Technical Specification Implementing Notes	October 23, 2001

<u>Problem Evaluation Requests</u>:

294-0024	202-0492	203-3334
203-3693	204-1019	205-0024
205-0271		

Procedures:

NUMBER	TITLE	REVISION
ABN-ELEC-SM1/SM7	SM-7, SM-75, SM-72, SL-71, SL-73 & SL-11 Distribution System Failures	2
CSP-DO-C101	Diesel Generator New Fuel Test	5
OSP-DO/IST-Q701	DO-P-1A Operability	4
OSP-ELEC-Q101	DO-TK-1A	0
OSP-ELEC-W102	Electrical Distribution Subsystem Breaker Alignment and Power Availability Verification	15
PPM 1.3.9	Temporary Modifications	39
PPM 2.7.13	AC Electrical Breaker Racking	29
PPM 2.7.1A	6900 Volt and 4160 Volt AC Electrical Power Distribution System	15
PPM 8.3.418	Westinghouse 50DHP-VR350 Circuit Breaker Implementation Test	0
PPM 10.25.9	Insulation Resistance and Polarization Index Tests	11
PPM 10.25.13A	4.16kV Vacuum Breaker Maintenance with Stored Energy Mechanism	6
PPM 10.25.16	General Electric 4160v Circuit Breaker	20
PPM 10.25.54	Cable Pulling Instruction and Inspection	14
PPM 10.25.105	Motor Control Center and Switch Gear Maintenance	22
SOP-DG2-START	Emergency Diesel Generator (DIV2) Start	8
SWP-DES-01	Plant Modifications & Configuration Control	6
SWP-LIC-02	Licensing Basis Impact Determinations	5
SWP-MAI-02	Station Materiel Condition Inspection Program	10

Vendor Documents:

NUMBER	TITLE	DATE
GEH-1802V	Metal-clad Switchgear, Types M26 and M36, For Magne-Blast Air Circuit Breaker, Types AM-4.16 AND AM-13.8	N/A
GEK-7320F	Magne-Blast Circuit Breaker AM-4.16-350-2	N/A
IL 6352C57H03	Instructions for Performing the CloSure™ [SURE CLOSE] Test on Cutler-Hammer Medium Voltage Circuit Breakers	June, 2001
IB 6513C80D	Instructions for Installation, Operation and Maintenance of Type DHP-VR Vacuum Replacement Circuit Breakers for DHP Switchgear	July, 2000

Work Orders:

01040194	01098299	01082714
01065768	01060347	01102103