UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
245 PEACHTREE CENTER AVENUE NE, SUITE 1200
ATLANTA, GEORGIA 30303-1257

June 28, 2013

Mr. William R. Gideon
Vice President
Progress Energy
H. B. Robinson Steam Electric Plant, Unit 2
3581 West Entrance Rd
Hartsville, SC 29550

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT- NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000261/2013007

Dear Mr. Gideon:

On May 16, 2013, U. S. Nuclear Regulatory Commission (NRC) completed an inspection at your H. B. Robinson Steam Electric Plant, Unit 2. The enclosed inspection report documents the inspection results, which were discussed on May 16, 2013, with you and other members of your staff. Additional inspection results were communicated on June 20, 2013, during a teleconference with you and other members of your staff, and June 27, 2013, with Mr. Hightower.

The 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 team reviewed selected procedures and records, observed activities, and interviewed personnel.

Five NRC identified findings of very low safety significance (Green) were identified during this inspection.

Four of these findings were determined to involve violations of NRC requirements. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2 of the Enforcement Policy.

If you contest the violations, or significance of these NCVs, 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 20555-001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the H. B. Robinson Steam Electric Plant.

If you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region II, and the NRC Resident Inspector at the H. B. Robinson Steam Electric Plant.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's Agencywide Document Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

RA

Rebecca Nease, Chief
Engineering Branch 1
Division of Reactor Safety

Docket No.: 05000261
License No.: DPR-23

Enclosure:
NRC Component Design Bases
Inspection Report 05000261/2013007
wAttachment: Supplemental Information

cc: (See page 3)

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DATE	6/26/2013	6/25/2013	6/26/2013	6/28/2013	6/25/2013	6/28/2013	6/25/2013
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U. S. NUCLEAR REGULATORY COMMISSION
REGION II

Docket No.:	05000261
License No.:	DPR-23
Report No.:	05000261/2013007
Licensee:	Carolina Power and Light Company.
Facility:	H. B. Robinson Steam Electric Plant, Unit 2
Location:	3581 West Entrance Road Hartsville, SC 29550
Dates:	April 8 – May 16, 2013
Inspectors:	G. Ottenberg, Senior Reactor Inspector (Lead) S. Walker, Senior Reactor Inspector J. Bartleman, Senior Construction Inspector A. Alen, Reactor Inspector M. Riley, Reactor Inspector P. Cooper, Reactor Inspector (Trainee) G. Nicely, Contractor (Electrical) H. Campbell, Contractor (Mechanical)
Approved by:	Rebecca Nease, Chief Engineering Branch 1 Division of Reactor Safety

SUMMARY

IR 05000261/2013007; 4/8/2013 – 5/16/2013; H. B. Robinson Steam Electric Plant, Unit 2; Component Design Bases Inspection.

This inspection was conducted by a team of six Nuclear Regulatory Commission (NRC) inspectors from Region II, and two NRC contract personnel. Four Green non-cited violations (NCVs), and one Green finding were identified. The significance of inspection findings is indicated by their color (Green, White, Yellow, Red) using the NRC Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," dated June 2, 2011. Cross cutting aspects are determined using IMC 0310, "Components Within the Cross Cutting Areas," dated October 28, 2011. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated January 28, 2013. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

<u>NRC identified and Self-Revealing Findings</u>

Cornerstone: Mitigating Systems

- <u>Green</u>. The team identified a Green finding for the licensee's failure to follow NRC Regulatory Guide 1.155, "Station Blackout," guidance (to which they are committed in the Updated Final Safety Analysis Report) for evaluating equipment needed to cope with a station blackout for the required duration for associated environmental conditions. This was a performance deficiency. The licensee entered the issue into their corrective action program as Nuclear Condition Report 600522, and established a calculation that determined the maximum expected temperature inside the compartment housing the dedicated shutdown diesel generator (DSDG) and evaluated the equipment to determine its capability to perform its function for the station blackout coping duration.

  The performance deficiency was more than minor because it affected the Mitigating Systems cornerstone attribute of Equipment Performance and adversely affected the cornerstone objective of ensuring reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the capability and reliability of the equipment located in the DSDG compartment was not ensured since a comparison of equipment temperature ratings and expected DSDG compartment temperatures was not performed. The finding was determined to be of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of a mitigating structure, system, or component, and the structure, system, or component maintained its functionality. No cross-cutting aspect was assigned to this finding because the team determined that the cause of the finding was not indicative of current licensee performance due to the age of the installation of the DSDG. (Section 1R21.2.13)

- <u>Green</u>. The team identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to have adequate analyses that supported safety-related load operation during a design basis accident while supplied by offsite power. This was a performance deficiency.

The licensee entered the issue into the corrective action program as Nuclear Condition Reports 601201 and 605969, and performed an evaluation that determined the capability of starting the safety-related motors at degraded voltage conditions, as well as the capability of the electrical loads during the degraded grid voltage relay (DGVR) time delay to ensure equipment function was preserved.

The performance deficiency was more than minor because it affected the Mitigating Systems cornerstone attribute of Equipment Performance and adversely affected the cornerstone objective of ensuring reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee did not ensure the capability of safety-related loads to respond to a design basis accident under degraded voltage conditions. Evaluations of the effects of starting motors at the DGVR voltage dropout setpoint and the equipment survivability during the DGVR time delay were not performed. The team determined the finding required a detailed risk analysis, because the finding was a deficiency affecting the design or qualification of a mitigating structure, system, or component, and the team assumed the performance deficiency represented a loss of operability or functionality of the equipment that could be lost during the DGVR time delay. This assumption was made to bound the risk of the finding, because the licensee was still investigating whether or not there would be a loss of function of any equipment during the DGVR time delay period as of the date of this inspection report issuance. The team assumed a recoverable loss of function of all 480V motor control centers and assumed a degraded voltage condition exposure time of one hour per year. The one hour per year assumption is conservative relative to actual plant data which indicated a degraded voltage condition exposure of 44 seconds over the past 3 operating years. The results of the detailed risk analysis indicated an increase in core damage frequency <1E-6/year, which is representative of a finding of very low safety significance (Green). No cross-cutting aspect was assigned to this finding because the team determined that the cause of the finding was not indicative of current licensee performance due to the age of the degraded voltage evaluation. (Section 1R21.2.15)

- Green. The team identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to verify the adequacy of the plant design during fast bus transfers. Specifically, the licensee failed to have an adequate analysis that ensured a successful fast bus transfer of the safety-related E1 bus feeder from the Unit Auxiliary Transformer to the Startup Transformer when required. This was a performance deficiency. The licensee entered the issue into the corrective action program as Nuclear Condition Reports 603357 and 605562, and performed an additional fast bus transfer evaluation of the E1 feeder breaker to ensure that the breaker would not trip under fast bus transfer conditions.

  The performance deficiency was more than minor because it affected the Mitigating Systems cornerstone attribute of Design Control and adversely affected the cornerstone objective of ensuring reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee did not ensure the capability of safety-related loads on the E1 bus because the licensee did not verify the E1 feeder breaker would not trip during a fast bus transfer. The finding was determined to be of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of a mitigating structure, system, or component (SSC), and the SSC maintained its operability and functionality. No cross-cutting aspect was assigned to this finding because the team determined that the cause of the finding was not indicative of current licensee performance due to the age of the fast bus transfer evaluation. (Section 1R21.2.16.1)

Green. The team identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to prescribe an adequate procedure that verified DGVR circuit operability following degraded voltage disable switch operation for reactor coolant pump (RCP) starts. This was a performance deficiency. The licensee entered the issue into the corrective action program as Nuclear Condition Report 602516, developed a test procedure, and verified the DGVR operability on both emergency buses.

The performance deficiency was more than minor because if left uncorrected, it could become a more significant safety concern. Specifically, by not properly testing the DGVR circuit to ensure continuity following switch manipulation for RCP starts, the circuit could unknowingly become inoperable and non-functional for an entire operating cycle. The finding was determined to be of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not represent a loss of system function, did not represent an actual loss of function of at least a single train for greater than its technical specification (TS) allowed outage time or two separate safety systems out-of-service for greater than its TS allowed outage time, and did not represent an actual loss of function of one or more non-TS trains. No cross-cutting aspect was assigned to this finding because the team determined that the cause of the finding was not indicative of current licensee performance due to the age of the modification that added the degraded voltage disable switches. (Section 1R21.2.16.2)

Cornerstone: Barrier Integrity

- Green. The team identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to account for instrument uncertainty on the containment bulk temperature instrumentation which was used to verify technical specification (TS) containment operability. This was a performance deficiency. The licensee entered this issue into their corrective action program as Nuclear Condition Report 603294 and performed an evaluation of the temperature instrumentation uncertainty. In addition, the licensee issued Standing Instruction 13-001 which specified the indicated containment temperature for entry into TS Limiting Condition for Operation 3.6.5 was to be 118 degrees Fahrenheit, a value that compensated for the temperature measurement uncertainty.

  The performance deficiency was more than minor because if left uncorrected, it would have the potential to lead to a more significant safety concern. Specifically, if the licensee did not account for the temperature measurement accuracy, containment temperature could unknowingly exceed the TS operability limit, and the licensee may not declare containment inoperable. The finding was determined to be of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment, containment isolation system, or heat removal components and did not involve a reduction in function of hydrogen igniters in the reactor containment. The cause of the finding was indicative of current licensee performance because the licensee failed to consider instrument uncertainty when they performed a containment re-analysis in 2013. The cause of the finding was directly related to the maintaining long term plant safety by maintenance of design margins cross-cutting aspect of the resources component in the area of human performance because when the containment re-analysis was performed, the licensee reduced margin between the analyzed value for containment starting temperature and the TS limit, making the instrument uncertainty of the temperature instruments more significant. [H.2(a)] (Section 1R21.2.9)

Licensee-Identified Violations

No findings were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R21 Component Design Bases Inspection (71111.21)

.1 Inspection Sample Selection Process

The team selected risk-significant components and related operator actions for review using information contained in the licensee's probabilistic risk assessment. In general, this included components and operator actions that had a risk achievement worth factor greater than 1.3 or Birnbaum value greater than 1 $X10^{-6}$. The sample included 17 components, of which three were associated with containment large early release frequency (LERF), and six operating experience (OE) items.

The team performed a margin assessment and a detailed review of the selected risk-significant components and operator actions to verify that the design bases had been correctly implemented and maintained. Where possible, this margin was determined by the review of the design basis and Updated Final Safety Analysis Report (UFSAR) response times associated with operator actions. This margin assessment also considered original design issues, margin reductions due to modifications, or margin reductions identified as a result of material condition issues. Equipment reliability issues were also considered in the selection of components for a detailed review. These reliability issues included items related to failed performance test results, significant corrective action, repeated maintenance, maintenance rule status, Regulatory Issue Summary 05-020 (formerly Generic Letter 91-18) conditions, NRC resident inspector input regarding problem equipment, system health reports, industry OE, and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, OE, and the available defense-in-depth margins. An overall summary of the reviews performed and the specific inspection findings identified is included in the following sections of the report.

.2 Component Reviews

.2.1 Main Steam Isolation Valves [MS-V1-3A, MS-V1-3B, MS-V1-3C] (LERF)

a. Inspection Scope

The inspection team reviewed the main steam isolation valves (MSIVs), MS-V1-3A, MS-V1-3B, and MS-V1-3C, at the Robinson Nuclear Plant (RNP) to verify the valves were capable of performing their design bases function. The team reviewed the licensee's calculations of MSIV operational margin and verified important inputs into the calculations were sufficiently conservative. The team examined system health reports and records of applicable corrective action documents to determine if potential degradation was being adequately monitored and appropriate actions were taken to correct any concerns. The team also reviewed completed surveillance and verification testing documentation that had been conducted under procedures EST-134 and OST-703-1 to ensure the MSIV valves and actuators operated properly and that they could perform their safety function. The team reviewed the maintenance history of the MSIVs to examine the mechanical condition and function of the components. The team verified that maintenance was performed in accordance with vendor instructions.

b. Findings

No findings were identified.

.2.2 Steam Generator Steam Line Power Operated Relief Valves [RV1-1, RV1-2, RV1-3] (LERF)

a. Inspection Scope

The inspection team reviewed the steam generator (SG) steam line power-operated relief valves (PORVs), RV1-1, RV1-2, and RV1-3, at RNP to verify they were capable of performing their design bases functions. The team reviewed the licensee's calculations of SG PORV operational margin and verified important inputs into the calculations were sufficiently conservative. The team examined system health reports and records of applicable corrective action documents to determine if potential degradation was being adequately monitored and appropriate actions were taken to correct any concerns. The team reviewed the maintenance history of the SG PORVs to examine the mechanical condition and function of the components. The team verified that maintenance was performed in accordance with vendor instructions.

b. Findings

No findings were identified.

.2.3 Steam Generator Steam Supply Valves to Steam Driven Auxiliary Feedwater Pump [MS-V1-8A, MS-V1-8B, MS-V1-8C]

a. Inspection Scope

The inspection team reviewed the steam driven auxiliary feedwater pump (SDAFP) steam supply motor-operated valves (MOVs), MS-V1-8A, MS-V1-8B, and MS-V1-8C, to verify these valves were capable of performing their design bases functions. The team reviewed the licensee's calculations of operational margin and verified important inputs into the calculations were sufficiently conservative. The team also verified that the infield diagnostic testing and setup of torque and limit switch settings for the valve actuators were within the setup windows assumed in design margin calculations, and verified that test equipment accuracies were considered.

The team reviewed the maintenance history of the valves and actuators, and engineering trending reports to examine mechanical condition and function of the components. The team verified that maintenance was performed in accordance with vendor instructions. The team reviewed the calculations that determined the degraded voltage at the MOV terminals, to ensure the proper voltage was utilized in calculating available motor output torque when determining margin. The team reviewed the calculations that establish control circuit voltage drop, short circuit, and protection/coordination including thermal overload sizing and application. Additionally motor control center (MCC) thermal overload testing programs were reviewed.

The team reviewed the licensee's initial evaluation and response to the 10 CFR Part 21 notification issued from Flowserve to the NRC, dated February 25, 2013, for a wedge pin failure of an Anchor/Darling Double Disk Gate Valve (AD-DDGV) at Browns Ferry Nuclear Plant Unit 1. The team reviewed the licensee's operability determination evaluation performed for the 41 safety-related AD-DDGVs used at RNP. This operability determination was completed under Nuclear Condition Report (NCR) 592717 to address the wedge pin failure outlined in this Part 21 notification. The licensee performed a Quick Cause Evaluation (QCE) Report, Form CAP-NGGC-0205-3-16, as required by the corrective actions identified in NCR 592717. This QCE described the analysis and immediate actions taken by the licensee to evaluate the wedge pin failure of an AD-DDGV. The team verified that the corrective actions outlined in the QCE, if implemented correctly, should adequately address this issue.

The team reviewed safe shutdown procedures, emergency operating instructions, abnormal operating instructions, and operator training material to verify that low margin time critical operator actions could be accomplished as relied upon in design assumptions. The team conducted a walkdown of a limiting safe shutdown procedure to assess if the time critical operator actions required to control the SDAFP during a Station Blackout (SBO) could be successfully accomplished. Equipment necessary to perform procedural steps was verified to be in the correct locations and available to the operators. The team interviewed individuals qualified to the task to ensure training was sufficient to accomplish the task. The team also reviewed the past results of exercises for this evolution to identify any past operator failures or challenges to accomplish this activity.

b. Findings

No findings were identified.

.2.4 Refueling Water Storage Tank (RWST) Discharge Valves [SI-864A, SI-864B]

a. Inspection Scope

The inspection team reviewed the RWST discharge MOVs, SI-864A and SI-864B, to verify they were capable of performing their design bases functions. The team reviewed the licensee's calculations of operational margin and verified important inputs into the calculations were sufficiently conservative. The team also verified that the infield diagnostic testing, and setup of torque and limit switch settings for the valve actuators were within the setup window assumed in design margin calculations, and verified that test equipment accuracies were considered.

The team reviewed the maintenance history of the valves and actuators, and engineering trending reports to examine mechanical condition and function of the components. The team verified that maintenance was performed in accordance with vendor instructions. The team reviewed the calculations that determined the degraded voltage at the MOV terminals, to ensure the proper voltage was utilized in calculating available motor output torque when determining margin. The team reviewed the calculations that establish control circuit voltage drop, short circuit, and protection/coordination including thermal overload sizing and application. Additionally, MCC thermal overload testing programs were reviewed. The team also reviewed the licensee's actions in response to the same 10 CFR Part 21 notification issued by Flowserve, dated February 25, 2013, as discussed above in section 1R21.2.3 as the actions were applicable to the SI-864A and SI-864B valves.

b. Findings

No findings were identified.

.2.5 Excess Letdown Heat Exchanger Relief Valve [CC-715]

a. Inspection Scope

The team reviewed the Component Cooling Water (CCW) design bases documents (DBDs), UFSAR, Technical Specifications (TS), and applicable plant drawings to identify the design bases requirements of the equipment. The team examined the corrective actions and maintenance history of the CCW Relief Valve, CC-715, to verify that design bases had been maintained. The team examined records and test data for corrective maintenance and engineering evaluations to verify potential degradation was being monitored, prevented, corrected, and/or justified. The team verified that the maintenance, testing, and inspections were being conducted in accordance with vendor recommendations and ASME code requirements. Lastly, the inspectors reviewed procedures for CC-715 for actions taken to mitigate the event scenario where the valve spuriously opened and failed to close.

b. Findings

No findings were identified.

.2.6 Residual Heat Removal (RHR) Pumps [RHR-PMP-A, RHR-PMP-B]

a. Inspection Scope

The team reviewed design basis documentation including original design specifications, the RHR and Safety Injection DBDs, portions of the UFSAR, and TS to identify the safety-related and functional requirements of the RHR pumps A and B. The RHR system engineer was interviewed to discuss current pump concerns and margin to design requirements. Further, a field walkdown with the system engineer was undertaken to evaluate the material condition and assess the pump's operating environment. Hydraulic calculations were reviewed to evaluate pump vortex and NPSH concerns, in addition to in-service test (IST) instrument uncertainty evaluations. Several years of IST surveillances, both Quarterly and Comprehensive, were reviewed to assess the capability of the pumps to perform their safety-related functions, as well as to evaluate potential long-term pump degradation. The licensee's evaluation of NRC Information Notice (IN) 97-90 and NRC Bulletin 88-04 were reviewed. As part of this OE review, the testing of the RHR system piping modification (1087, RHR Pumps Minimum Flow Recirculation Lines) was reviewed to ensure compliance with the concerns raised in Bulletin 88-04. The team reviewed calculations that establish voltage drop, protection and coordination, motor horsepower requirements, and short circuit for the motor power supply and feeder cable to verify that design bases and design assumptions have been appropriately translated into design calculations.

The team reviewed safe shutdown procedures, emergency operating instructions, abnormal operating instructions, and operator training material to verify that low margin time critical operator actions could be accomplished as relied upon in design assumptions. The team interviewed individuals qualified to the task to ensure training was sufficient to accomplish the task. The team also reviewed the past results of exercises to identify any past operator failures or challenges to accomplish this activity.

b. Findings

No findings were identified.

.2.7 High Head Safety Injection Pumps [SI-PMP-A, SI-PMP-B, SI-PMP-C]

a. Inspection Scope

The team reviewed design basis documentation including original design specifications, the Safety Injection system DBD, portions of the UFSAR, and TS to identify the safety-related and functional requirements of the high head safety injection pumps A, B and C. The team performed a walkdown of the pumps to assess the current material condition of the pumps and assess the pump's operating environment. Several years of IST surveillances, both Quarterly and Comprehensive, were reviewed to assess the capability of the pumps to perform their safety-related functions, as well as to evaluate potential long-term pump degradation. The licensee's evaluation of NRC IN 97-90 and NRC Bulletin 88-04 were reviewed. As part of this OE review, the team reviewed original and current pump vendor documents to ensure that the concerns identified in Bulletin 88-04, (i.e., minimum recirculation flow), were adequately addressed. The team reviewed calculations that establish voltage drop, protection and coordination, motor horsepower requirements, and short circuit for the motor power supply and feeder cable to verify that design bases and design assumptions have been appropriately translated into design calculations.

b. Findings

No findings were identified.

.2.8 Containment Pressure Relief and Containment Vessel Post Accident Hydrogen Vent Isolation Valves [V12-10, V12-11] (LERF)

a. Inspection Scope

The team reviewed design basis documentation including original design specifications, the Generic Issue Document for Reactor Containment Isolation, TS, and portions of the UFSAR to identify the functional requirements of the containment pressure relief valves, V12-10 and V12-11. Further, the team reviewed system layout/piping diagrams, interviewed the system engineer, and performed a detailed walkdown of the accessible valve (exterior to containment) to assess the layout and current condition of V12-10.

The Appendix J calculation which evaluated and listed allowed and historical leakage rates for the valves was reviewed to ensure that the valve capabilities were consistent with functional requirements. Finally, work orders and IST stroke time and indication results over several years were reviewed to ensure that the valves were being maintained and operated satisfactorily.

b. Findings

No findings were identified.

.2.9 Containment Instrumentation

a. Inspection Scope

The team reviewed design basis documentation including portions of the UFSAR, DBD, TS, and related calculations to identify the safety-related and functional requirements of the containment instrumentation. Specific instruments included for review were those monitoring containment water level, pressure, and temperature. The team reviewed instrument uncertainty calculations for containment pressure and level instrumentation to ensure that acceptable instrumentation was being used to monitor the associated containment process parameters. The team reviewed work orders to verify that calibration checks of the containment temperature thermocouples were completed. The team reviewed condition reports and one operability evaluation addressing containment analyses. The team reviewed analyses evaluating the method used to determine the average bulk containment temperature, in addition to recent and historical containment bulk temperature data, to ensure that the TS limit had not been exceeded. Finally, the team reviewed system health reports to assess the current state and operability of the containment instrumentation.

b. Findings

Introduction: The team identified a Green non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to account for instrument uncertainty on containment bulk temperature instrumentation which was used to determine TS containment operability. This was a performance deficiency.

Description: In February 2013, the licensee identified errors in inputs to the original containment analysis (main steam line break and loss of coolant accident). The licensee performed a re-analysis of these design basis accidents and performed an operability determination. The analysis supporting the operability determination decreased the assumed accident initial containment temperature from 130 degrees F to 120 degrees F. The TS upper limit for containment temperature was 120 degrees F; therefore, following the containment re-analysis, there was no margin between the analyzed and TS allowed temperature limit. The station procedure guiding operations during hot weather conditions, PLP-118, did not instruct operators to declare containment inoperable at any indicated temperature below the TS required 120 degrees F. The team reviewed work orders that performed calibration checks on the containment thermocouples and found that no acceptance criteria were provided for the allowed tolerance between the temperature obtained by the test equipment standard and the actual containment instrumentation.

Based on the above, the team concluded that the licensee did not have adequate controls in place to ensure that containment would be declared inoperable at an actual bulk average containment temperature of 120 degrees F since neither the instrument used to perform the surveillance nor station procedures accounted for instrument uncertainty. Upon identification by the team, the licensee entered this issue into their corrective action program as NCRs 603294 and 606607. At the time of the inspection, the local ambient temperature and containment bulk temperature were significantly below the TS limit, thus no immediate operability concern was present. As an interim measure, the licensee performed an evaluation of the temperature instrumentation uncertainty, and issued Standing Instruction 13-001, which specified that the indicated containment temperature for entry into TS limiting condition for operation (LCO) 3.6.5 was to be 118 degrees F, a value that compensated for the temperature measurement uncertainty.

Analysis: The failure to account for instrument uncertainty on the containment bulk average temperature instrumentation, used to verify TS containment operability, was a performance deficiency. The performance deficiency was more than minor because if left uncorrected, it would have the potential to lead to a more significant safety concern. Specifically, the licensee did not account for the temperature measurement accuracy, containment temperature could exceed the TS operability limit. The team used IMC 0609, Att. 4, “Initial Characterization of Findings,” issued 6/19/12, for Barrier Integrity Cornerstone, and IMC 0609, App. A, “The Significance Determination Process (SDP) for Findings At-Power,” issued 6/19/12, and determined the finding to be of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment, containment isolation system, or heat removal components and did not involve a reduction in function of hydrogen igniters in the reactor containment. The team determined that the cause of the finding was indicative of current licensee performance because the licensee failed to consider instrument uncertainty when they performed a containment re-analysis in 2013. The cause of the finding was directly related to the maintaining long term plant safety by maintenance of design margins cross-cutting aspect of the resources component in the area of human performance because when the containment re-analysis was performed, the licensee reduced margin between the analyzed value for containment starting temperature and the TS limit, making the instrument uncertainty of the temperature instruments more significant. [H.2(a)]

Enforcement: Appendix B of 10 CFR Part 50, Criterion III, “Design Control,” required, in part, that design control measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into procedures. Contrary to the above, from early 2013 when the containment re-analysis was performed until the inspectors identified the issue, the licensee did not assure that the appropriate TS limit for containment temperature and the results of the their containment re-analysis were correctly translated into procedures for determining containment operability. This resulted in the potential to exceed the TS limit and not declare containment inoperable. As a result, the licensee issued a standing instruction to operators to ensure the TS limit would not be exceeded and re-performed the containment analysis to regain margin between the analyzed value for containment starting temperature and the TS limit. This violation is being treated as an NCV consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee’s corrective action program as NCR 603294. (NCV 05000261/2013007-01, Failure to Account for Containment Temperature Measurement Uncertainty)

.2.10 125V Direct Current (DC) Motor Control Center (MCC) "A" [MCC-A]

a. Inspection Scope

The team reviewed the battery profile and 125V DC voltage drop calculation to verify that the MCC had sufficient capacity to supply its loads under design basis events. The team also reviewed the protective device coordination between the 125V DC loads to verify that the protection scheme would isolate faults associated with the MCC and ensure availability of other safety-related components needed to respond in a design basis accident. The team reviewed system health reports, corrective action documents, and maintenance records to determine whether there were any adverse operating trends. The team performed a walkdown of the 125V DC safety buses to assess operability and condition. In addition, the team performed a non-intrusive visual inspection of the MCC to verify that it showed no signs of material degradation and vulnerability to hazards such as flooding, seismic interactions, and missiles.

The team reviewed safe shutdown procedures, emergency operating instructions, abnormal operating instructions, and operator training material to verify that low margin time critical operator actions could be accomplished as relied upon in design assumptions. The team interviewed individuals qualified to the task to ensure training was sufficient to accomplish the task. The team also reviewed the past results of exercises to identify any past operator failures or challenges to accomplish this activity.

b. Findings

No findings were identified.

.2.11 125V DC Distribution Panel "A" [Distribution Panel-A]

a. Inspection Scope

The team reviewed the plant's TS, UFSAR, system descriptions, and electrical drawings to establish an overall understanding of the licensee's 125V DC distribution system. The team reviewed DC voltage drop calculations and testing procedures to verify that Distribution Panel A was capable of supplying, and maintaining in an operable status, the required emergency loads. The team reviewed system health reports, corrective action documents, and maintenance records to determine whether there were any adverse operating trends. The team performed a walkdown of the 125V DC safety buses to assess operability and condition. The team also conducted interviews with responsible licensee personnel to answer questions that arose during the inspection pertaining to the control voltage at the 480V emergency buses.

The team reviewed safe shutdown procedures, emergency operating instructions, abnormal operating instructions, and operator training material to verify that low margin time critical operator actions could be accomplished as relied upon in design assumptions. The team conducted a walkdown of a limiting safe shutdown procedure to assess if the time critical operator actions required to use the spare battery charger could be successfully accomplished. Equipment necessary to perform procedural steps was verified to be in the correct locations and available to the operators. The team interviewed individuals qualified to the task to ensure training was sufficient to accomplish the task. The team also reviewed the past results of exercises for this evolution to identify any past operator failures or challenges to accomplish this activity.

b. Findings

No findings were identified.

.2.12 7.5 kVA Inverters "A" and "B"

a. Inspection Scope

The team reviewed the plant's TS, UFSAR, system descriptions, and electrical drawings to establish an overall understanding of the licensee's 120 VAC distribution system, which receives power from the 7.5kVA inverters. The team reviewed voltage drop calculations and testing procedures to verify that the inverters were capable of supplying, and maintaining in an operable status, the required emergency loads. The team reviewed system health reports, corrective action documents, and maintenance records to determine whether there were any adverse operating trends. The team performed a non-intrusive visual inspection of the inverters to verify that the inverters showed no signs of material degradation and that the inverters were operating within their required operating parameters for voltage and frequency.

b. Findings

No findings were identified.

.2.13 Dedicated Shutdown (DS) Diesel Bus [DS Bus]

a. Inspection Scope

The team inspected the DS Bus to verify its operational support role during Station Blackout (SBO) and Appendix R fire events. The team reviewed selected calculations for electrical distribution system load flow/voltage drop, short-circuit, and electrical protection and coordination. The adequacy and appropriateness of design assumptions and calculations were reviewed to verify that bus and circuit breaker capacity were not exceeded and bus voltages remained above minimum acceptable operating values. The protective device settings and breaker ratings were reviewed to ensure that selective coordination was adequate for protection of connected equipment during worst case short-circuit conditions. The team reviewed vendor manuals, preventive maintenance inspection and testing procedures to verify that breakers were maintained in accordance with industry and vendor recommendations. System health reports, component maintenance history and licensee corrective action program reports were reviewed to verify correction of potential degradation and deficiencies were appropriately identified and resolved. Finally, the team performed a visual non-intrusive inspection of observable portions of the DS bus and associated DS diesel generator (DSDG) to assess the installation configuration, material condition, and the potential vulnerability to hazards.

The team reviewed safe shutdown procedures, emergency operating instructions, abnormal operating instructions, and operator training material to verify that low margin time critical operator actions could be accomplished as relied upon in design assumptions. The team conducted a walkdown of a limiting safe shutdown procedure to assess if the time critical operator actions required to establish service water using the DS system could be successfully accomplished. Equipment necessary to perform procedural steps was verified to be in the correct locations and available to the operators. The team interviewed individuals qualified to the task to ensure training was sufficient to accomplish the task. The team also reviewed the past results of exercises for this evolution to identify any past operator failures or challenges to accomplish this activity.

b. Findings

Introduction: The team identified a Green finding for the licensee's failure to follow NRC Regulatory Guide (RG) 1.155, "Station Blackout," guidance (to which they are committed in the UFSAR) for evaluating equipment needed to cope with a station blackout for the required duration for associated environmental conditions. This was a performance deficiency.

Description: The DSDG is used for SBO mitigation and as part of the station's dedicated shutdown capability for 10 CFR 50, Appendix R fire mitigation. During a walkdown of the DSDG, the team questioned the temperature ratings of the equipment located in the DSDG compartment. This included equipment, such as the DSDG as well as necessary support loads and their corresponding MCC located in the room. The team determined, per Section 1.8 of the station's Updated Final Safety Analyses Report (UFSAR), that the licensee was committed to following the guidance in RG 1.155. Section 3.2.4 of RG 1.155, states in part, that the design adequacy and capability of equipment needed to cope with a SBO for the required duration and recovery period should be addressed and evaluated as appropriate for the associated environmental conditions. Upon investigation, it was discovered that the licensee had not performed a calculation that determined the maximum expected temperatures inside the compartment housing the DSDG and its associated support equipment. Furthermore, the licensee had not established that the equipment ratings were adequate to withstand the expected environmental ambient temperatures. Rather, the licensee was relying on a specification that was provided to the DSDG supplier. The team noted that the continuous duty ambient temperature ratings (104 degrees F) of some of the equipment were lower than the maximum outdoor ambient temperature (107 degrees F) the licensee expected to experience. Ventilation is supplied by fresh outdoor air which is circulated through the room. After identification by the team, the licensee performed a calculation that determined the maximum expected temperature inside the compartment housing the DSDG and evaluated the equipment to determine its capability to perform its function for the SBO coping duration. The licensee determined that when the DSDG is operating, the temperature in the compartment could be as high as 120 degrees F and that the DSDG and supporting equipment would be able to perform their functions for the SBO coping duration. The licensee generated NCR 600522 to address the issue.

Analysis: The licensee's failure to follow NRC RG 1.155, "Station Blackout," guidance (to which they are committed to in the UFSAR) for evaluating equipment needed to cope with a station blackout for the required duration for associated environmental conditions was a performance deficiency. The performance deficiency was more than minor because it affected the Mitigating Systems cornerstone attribute of Equipment Performance and adversely affected the cornerstone objective of ensuring reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the capability and reliability of the equipment located in the DSDG compartment was not ensured since a comparison of equipment temperature ratings and expected DSDG compartment temperatures was not performed. The team used IMC 0609, Att. 4, "Initial Characterization of Findings," issued 6/19/12, for Mitigating Systems, and IMC 0609, App. A, "The Significance Determination Process (SDP) for Findings At-Power," issued 6/19/12, and determined the finding to be of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of a mitigating SSC, and the SSC maintained its functionality. No cross-cutting aspect was assigned to this finding because the team determined that the cause of the finding was not indicative of current licensee performance because the performance deficiency existed since initial installation of the DSDG.

Enforcement: This finding does not involve enforcement action because no violation of a regulatory requirement was identified. Because this finding does not involve a violation and is of very low safety significance, it is identified as FIN 05000261/2013007-02, Failure to Evaluate SBO Coping Equipment for Environmental Conditions.

.2.14 4KV Bus "3" [Bus 3]

a. Inspection Scope

The team inspected the 4kV Bus 3 to verify its operational support role during design basis events. System health reports, component maintenance history and licensee corrective action program reports were reviewed to verify that potential degradation was monitored or prevented. The team reviewed selected calculations for electrical distribution system load flow/voltage drop, short-circuit, and electrical protection and coordination. This review was conducted to assess the adequacy and appropriateness of design assumptions, and to verify that bus capacity was not exceeded and bus voltages remained above minimum acceptable values to support transmission of power to downstream safety-related 480V Emergency Bus E2. Additionally, the switchgear's protective device settings and breaker ratings were reviewed to ensure that selective coordination was adequate for protection of connected equipment during worst case, short-circuit conditions to ensure continuity of power to downstream safety-related buses. To determine if breakers were maintained in accordance with industry and vendor recommendations, the team reviewed the preventive maintenance inspection and testing procedures. The team reviewed the loss of voltage protection scheme. Finally, the team performed a visual non-intrusive inspection of 4kV Bus 3 to assess the installation configuration, material condition, and potential vulnerability to hazards.

b. Findings

No findings were identified.

.2.15 <u>480V Motor Control Centers "5" and "6" [MCC-5 and MCC-6]</u>

a. <u>Inspection Scope</u>

The team reviewed selected calculations for electrical distribution system load flow/voltage drop, short-circuit, and electrical protection and coordination. The adequacy and appropriateness of design assumptions and calculations were reviewed to verify that bus and circuit breaker capacity was not exceeded and bus voltages remained above minimum acceptable values under design basis conditions. The MCC's protective device settings and breaker ratings were reviewed to ensure that selective coordination was adequate for protection of connected equipment during worst case short-circuit conditions. To ensure that breakers were maintained in accordance with industry and vendor recommendations, the team reviewed the vendor manuals, preventive maintenance inspection and testing procedures. System health reports, component maintenance history and licensee corrective action program reports were reviewed to verify correction of potential degradation, and to verify that deficiencies were appropriately identified and resolved. Finally, the team performed a visual non-intrusive inspection of observable portions of the safety related 480V AC MCC 5 and 6 to assess the installation configuration, material condition, and the potential vulnerability to hazards.

The team reviewed safe shutdown procedures, emergency operating instructions, abnormal operating instructions, and operator training material to verify that low margin time critical operator actions could be accomplished as relied upon in design assumptions. The team interviewed individuals qualified to the task to ensure training was sufficient to accomplish the task. The team also reviewed the past results of exercises to identify any past operator failures or challenges to accomplish this activity.

b. <u>Findings</u>

<u>Introduction</u>: The team identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion III, "Design Control," for the failure to have adequate analyses that supported safety-related load operation during a design basis accident while supplied by offsite power. This was a performance deficiency.

<u>Description</u>: The team identified two examples of deficiencies in electrical calculation RNP-E-8.002, "AC Auxiliary Electrical Distribution System Voltage/Load Flow/Fault Current Study," that contributed to the failure of the licensee to: 1) verify and assure adequate starting voltages to safety-related loads with offsite power available, and 2) ensure load survivability by properly analyzing the operation of protective devices to safety-related loads during a design basis event under degraded voltage conditions for the duration of the degraded grid voltage relay (DGVR) time delay and subsequent re-sequencing onto the emergency diesel generators (EDGs). The following examples contributed to the identified performance deficiency:

1. The licensee did not calculate voltages at the terminals of all safety-related equipment on the DGVR monitored bus while the bus is connected to offsite power and at the DGVR voltage dropout setting (less tolerances) to ensure adequate starting voltages during design basis accidents. TS 3.3.5 allows a voltage setpoint for the DGVRs of 430V +/-4V. The appropriateness of the setpoint was not adequately verified by design calculation RNP-E-8.002, because it failed to verify

that required Class 1E motors would have adequate motor starting voltages with 480V Buses E1 and E2 at the DGVR dropout setting.

The licensee performed additional analyses and determined that the affected equipment would have adequate starting voltages.

2. If the voltage on the E1/E2 buses drops below the DGVR voltage dropout setting, a timer is initiated that results in disconnection of the bus from offsite power after the timer “times out” and the loads are subsequently re-sequenced onto the EDG. During the time delay period, loads that were running may become stalled or motors that have received a start signal that do not have adequate voltages to accelerate may continue in a stall condition. Both of these conditions may result in tripping of the protective devices which would prevent the associated load from re-sequencing onto the EDG. TS 3.3.5 allows a time delay setpoint for the DGVRs of 10 seconds +/- 0.5, to allow some time for offsite power voltage to recover to avoid unnecessarily swapping power source to the EDG. The setpoint was not properly verified by design calculation RNP-E-8.002, because it failed to ensure the protective devices for the Class 1E loads would not trip during a design basis event concurrent with a degraded voltage at the E1 and E2 buses below the DGVR setting but above the Loss of Voltage setting for the duration of the DGVR time delay and subsequent re-sequencing onto the EDG. Additionally, the calculation did not evaluate control power circuits and their fuses for motors that are actuated during the design basis event to ensure that the fuses would not actuate if the control circuit starter stays in an inrush condition for the 10 seconds of the DGVR. The licensee initiated corrective actions to evaluate the ability of the equipment to withstand the specified time delay, which was ongoing as of June 28, 2013.

Analysis: The licensee’s failure to have adequate analyses that supported safety-related load operation during a design basis accident while supplied by offsite power was a performance deficiency. The performance deficiency was more than minor because it affected the Mitigating Systems cornerstone attribute of Equipment Performance and adversely affected the cornerstone objective of ensuring reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee did not ensure the capability of safety-related loads to respond to a design basis accident under degraded voltage conditions because evaluations of the effects of starting motors at the DGVR voltage dropout setpoint and the equipment survivability during the DGVR time delay were not performed. The team used IMC 0609, Att. 4, “Initial Characterization of Findings,” issued 6/19/12, for Mitigating Systems, and IMC 0609, App. A, “The Significance Determination Process (SDP) for Findings At-Power,” issued 6/19/12, and determined the finding required a detailed risk analysis, because the finding was a deficiency affecting the design or qualification of a mitigating SSC, and the team assumed the performance deficiency represented a loss of operability or functionality of the equipment that could be lost during the DGVR time delay. This assumption was made to bound the risk of the finding, since the licensee was still investigating whether or not there would be a loss of function of any equipment during the DGVR time delay period as of the date of this inspection report issuance. A detailed risk evaluation was performed by a Regional SRA in accordance with the guidance of NRC Inspection Manual Chapter 0609 Appendix A. A bounding analysis was performed using the latest NRC H.B. Robinson SPAR model. The major analysis assumptions included: a one hour exposure period, a loss of all 480V MCCs, and nominal manual recovery as surrogate for motor operated valves which could trip on thermal overload during the DGVR interval. The risk was mitigated by the short exposure period and the availability of alternate equipment and the recovery action. The analysis result was an increase in delta CDF <1E-6/year, a GREEN finding of very low safety significance. No cross-cutting aspect was assigned to this finding because the team determined that the cause of the finding was not indicative of current licensee performance due to the age of the degraded voltage evaluation.

Enforcement: Appendix B of 10 CFR Part 50, Criterion III, “Design Control,” required, in part, that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Contrary to the above, since 1995 when the licensee modified the DGVR design, until 2013 when the inspectors identified the issue, the licensee did not properly verify the adequacy of the DGVR design with respect to motor starting adequacy at the DVR voltage setpoint and load survivability during the DGVR time delay setpoint duration. This resulted in the potential for loss or unavailability of essential loads during a design basis event. When the issue was identified, the licensee performed an evaluation for both starting the motors as well as load survivability during the time delay. This violation is being treated as an NCV consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee’s corrective action program as NCRs 601201 and 605969. (NCV 05000261/20130007-03, Failure to Have Adequate Analyses Supporting the Degraded Voltage Relay Setpoints)

.2.16 480V Buses E1 and E2 Load Shed Circuitry

a. Inspection Scope

The team inspected the 480V Emergency Buses E1 and E2 load shed circuitry to verify its operational support role during design basis events. System health reports, drawings, component maintenance history and licensee corrective action program reports were reviewed to verify that potential degradation was monitored or prevented. The logic and operation of the load-shed circuitry as described in the UFSAR and design documents were also reviewed.

The team reviewed safe shutdown procedures, emergency operating instructions, abnormal operating instructions, and operator training material to verify that low margin time critical operator actions could be accomplished as relied upon in design assumptions. The team interviewed individuals qualified to the task to ensure training was sufficient to accomplish the task. The team also reviewed the past results of exercises to identify any past operator failures or challenges to accomplish this activity.

b. Findings

.1 Inadequate Fast Bus Transfer Evaluation

Introduction: The team identified a Green non-cited violation of 10 CFR 50, Appendix B, Criterion III, “Design Control,” for the licensee’s failure to verify the adequacy of the plant design during fast bus transfers. Specifically, the licensee failed to have an adequate analysis that ensured a successful fast bus transfer of the safety-related E1 bus feeder from the Unit Auxiliary Transformer to the Startup Transformer when required. This was a performance deficiency.

Description: Calculation RNP-E-7.016, Bus Transfer Study, provided an analysis for the 480V emergency bus E1 transfer from the UAT, supplied by the unit main generator output, to the SUT, supplied from offsite power, following a reactor/unit trip. This calculation determined that the E1 Bus main feeder breaker (52/18B) may experience as much as 14,200 amps during the bus transfer. Another calculation, RNP-E-2.010, Overcurrent Protection Emergency Bus E1 & E2, determined the setting and adequacy for the supply breaker for Emergency Bus E1. RNP-E-2.010 recommended a breaker (52/18B) short time pick-up setting of 16000 amps +/- 10%. At this setting, the breaker could trip at a current as low as 14,400 amps for a duration of approximately 380 msec. Calculation RNP-E-2.010 incorrectly determined that the maximum current the breaker could experience was 7,109 amps, based on a fully loaded E1 bus with the concurrent starting of other non-running safety related motors (i.e., the scenario assumed was less limiting than what was determined by the Bus Transfer Study). With conservatism for inrush currents added to bound the calculation, the calculation resulted in a margin of 5,200 amps between the breaker trip setpoint and the maximum current potentially experienced at the breaker. This as-calculated margin was incorrectly determined since calculation RNP-E-2.010 did not reference the Bus Transfer Study (RNP-E-7.016) or evaluate the inrush currents (approximately 14,200 amps) through breaker 52/18B during the fast bus transfer. The Bus Transfer Study calculation indicated the inrush currents quickly reduce within cycles, therefore it is likely that the protective device would not actuate due to its time delay setting of 380 msec. However, the team noted the Bus Transfer Study could potentially be non-conservative because the calculation had not been updated following several plant modifications, such as replacement of the UAT and Main Bank Transformers. Other potential non-conservatisms included operation at offsite power voltages less than nominal assumed, and the assumptions that had to be made for the loads in the 480V system when vendor motor/load torque curves were not available. Additionally, based on PM-402, "Inspection and Testing of CB for 480V Bus E1," the as-left setting for the short time pick-up could be as low as 14,400 amps in which the operation could be as low as 12,960 amps, which is less than the calculated inrush current of 14,200 amps. This could result in the feeder breaker tripping if the inrush current does not reduce as quickly as the Bus Transfer Study predicted. The licensee entered this issue into their corrective action program as NCRs 603357 and 605562 and performed an additional fast bus transfer evaluation of the E1 feeder breaker to verify that the breaker would not trip during a fast bus transfer.

Analysis: The licensee's failure to have an adequate analysis that ensured a successful fast bus transfer of the safety related E1 bus from the UAT to the SUT would occur when required was a performance deficiency. The performance deficiency was more than minor because it affected the Mitigating Systems cornerstone attribute of Design Control and adversely affected the cornerstone objective of ensuring reliability, availability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee did not ensure the capability of safety-related loads on the E1 bus because the licensee did not verify the E1 feeder breaker would not trip during a fast bus transfer. The team used IMC 0609, Att. 4, "Initial Characterization of Findings," issued 6/19/12, for Mitigating Systems, and IMC 0609, App. A, "The Significance Determination Process (SDP) for Findings At-Power," issued 6/19/12, and determined the finding to be of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of a mitigating SSC, and the SSC maintained its operability and functionality. No cross-cutting aspect was assigned to this finding because the team determined that the cause of the finding was not indicative of current licensee performance due to the age of the fast bus transfer evaluation.

Enforcement: Appendix B of 10 CFR Part 50, Criterion III, "Design Control," required in part that design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. Contrary to the above, from 1992 when the fast bus transfer evaluation was performed until 2013 when the inspectors identified the issue, the licensee did not verify the adequacy of the plant design for the E1 fast bus transfer. Specifically, the licensee did not verify the E1 bus feeder breaker would not trip due to high inrush currents resulting from the fast bus transfer from the UAT to the SUT. This resulted in the licensee not having a calculational basis for plant electrical system modifications. The licensee took immediate corrective actions to perform an additional fast bus transfer evaluation of the E1 feeder breaker. This violation is being treated as an NCV consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee's corrective action program as NCRs 603357 and 605562. (NCV 05000261/2013007-04, Failure to Have Adequate Analyses For the E1 Bus Fast Transfer)

.2 Inadequate Circuit Verification Following Degraded Voltage Disable Switch Operation

Introduction: The team identified a Green NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to prescribe an adequate procedure that verified DGVR circuit operability following degraded voltage disable switch operation for reactor coolant pump (RCP) starts. This was a performance deficiency.

Description: The H. B. Robinson Steam Electric Plant utilizes degraded voltage disable switches during RCP starts in order to prevent the resulting momentary voltage drops that occur on the safety-related 480V emergency buses E1 and E2 from actuating the degraded voltage relays. Actuation of these relays would cause the safety-related loads to be removed from the E1 and E2 buses, an automatic EDG start, and a subsequent re-sequencing of the loads back on to the buses after they are connected to the EDG source. The RCP starting operations are performed as part of a normal plant startup following refueling outages. The degraded voltage disable switches are placed into the disable position prior to RCP start per OP-101, "Reactor Coolant System and Reactor Coolant Pump Startup and Operation," and then returned to the normal (non-disabled) position following a successful start of the RCP. This was an activity affecting quality. The team determined that this procedure was not adequate to determine that the DGVR circuit had been restored to an operable and functional condition following the switch manipulation. Specifically, neither an operability test of the circuit was performed after the switch was returned to the normal position, nor was the circuit directly monitored by either an annunciator or other method to determine the relay had been successfully placed back into the circuit. The team noted that the licensee performed a circuit operability check following manipulation of the disable switch for DGVR testing performed during outages, however, the switch manipulation for RCP starting occurred after the circuit operability check was performed, and no subsequent test of the switch or relay circuit prior to reactor operation was performed.

Analysis: The licensee's failure to prescribe a procedure with appropriate acceptance criteria to verify DGVR circuit operability following degraded voltage disable switch operation for RCP starts was a performance deficiency. The performance deficiency was more than minor because if left uncorrected, it could become a more significant safety concern. Specifically, by not properly testing the DGVR circuit to ensure continuity following switch manipulation for RCP starts, the circuit could unknowingly become inoperable and non-functional for an entire operating cycle. The team used IMC 0609, Att. 4, "Initial Characterization of Findings," issued 6/19/12, for the Mitigating Systems Cornerstone, and IMC 0609, App. A, "The Significance Determination Process (SDP) for Findings At-Power," issued 6/19/12, and determined the finding to be of very low safety significance (Green) because the finding was not a design or qualification deficiency, did not represent a loss of system function, did not represent an actual loss of function of at least a single train for greater than its TS allowed outage time or two separate safety systems out-of-service for greater than its TS allowed outage time, and did not represent an actual loss of function of one or more non-TS trains. No cross-cutting aspect was assigned to this finding because the team determined that the cause of the finding was not indicative of current licensee performance due to the age of the modification that added the degraded voltage disable switches.

Enforcement: Appendix B of 10 CFR Part 50, Criterion V, "Instructions, Procedures, and Drawings," requires in part that activities affecting quality shall be prescribed by documented procedures of a type appropriate to the circumstances, and that the procedures shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Contrary to the above, from 1982 when the degraded voltage defeat switches were installed and utilized for RCP starts, until 2013 when the inspectors identified the issue, the licensee did not prescribe a procedure with appropriate acceptance criteria to determine that degraded voltage disable switch manipulation for RCP starts did not adversely affect the proper functioning of the degraded grid voltage relays. This resulted in the potential to operate an entire operating cycle with the DGVR circuit inoperable due to a failed switch contact. When the issue was identified, the licensee developed a test procedure and verified the DGVR operability on both emergency buses. This violation is being treated as an NCV consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee's corrective action program as NCR 602516. (NCV 05000261/2013007-05, Failure to Have Appropriate Procedure to Verify Degraded Voltage Relay Circuit Status)

.3 (Opened) Unresolved Item (URI): Questions Regarding License Basis Design Requirements for Degraded Voltage Relays

Introduction: The team identified an unresolved item (URI) regarding the degraded voltage relays. Specifically, the effect of system and transient harmonics on proper operation of degraded voltage relays was not analyzed.

Description: The Robinson degraded voltage protection design features three ITE Type 27N relays for each 480V emergency bus E1 and E2, arranged in a two out of three tripping scheme. BBC Instruction Bulletin 7.4.1.7-7 states, the relay employs a peak voltage detector, and harmonic distortion on the AC waveform can have a noticeable effect on the relay operating point and the measuring instruments used to calibrate the relay. The bulletin also notes that the relay is available with an internal harmonic filter for applications where waveform distortion is a factor; however, harmonic filters are not installed on the degraded grid voltage relays based upon their model number and specification package. The inspectors questioned if persistent harmonics on the 480V system could cause the relays to fail to actuate at the set point specified in Technical Specifications 3.3.5, and if transient harmonics could cause the relays to spuriously reset during the time delay that occurs during an actual degraded voltage condition concurrent with a design basis accident. Persistent harmonics can be produced by factors external to the nuclear site or by internal phenomena. A typical internal source of harmonics at nuclear power plants is defects in rotating equipment. Persistent harmonics could cause dropout set point shift, and mask an actual degraded voltage condition.

Transient harmonics could cause the relays to spuriously reset during an actual degraded voltage event, thereby delaying the protective function beyond the nominal value stipulated in Technical Specifications 3.3.5. The relay is susceptible to this type of mal-operation because it features an instantaneous voltage sensor that could reset in less than two cycles in the presence of harmonics, thereby reinitiating the relay's internal timer. The licensee has entered this item into their corrective action program as NCR 601203.

This issue is unresolved pending inspector consultation with NRC headquarters technical staff for clarification of license basis design requirements of degraded voltage relays to withstand the effects of harmonics. This issue is identified as URI 05000261/2013007-07, Questions Regarding License Basis Design Requirements for Degraded Voltage Relays.

.2.17 Security Uninterruptible Power Supply (UPS)

a. Inspection Scope

The team reviewed the licensee's security plan and also conducted interviews with responsible licensee personnel to establish an overall understanding of the function of the security UPS. The team reviewed the modification to change the security UPS to verify that the modification did not adversely affect the function of the UPS to provide constant power supply to necessary security equipment loads. The team reviewed calculations to verify that the security UPS was adequately sized to supply the load. The team reviewed maintenance and testing procedures to verify that these operations did not place the security equipment in an adverse operational condition and that the testing procedures verified that the security UPS and standby diesel could meet its intended function. A field walk-down was conducted to assess the observable material condition to assess equipment degradation and to assess the presence of hazards.

b. Findings

No findings were identified.

.3 Operating Experience

a. Inspection Scope

The team reviewed six operating experience issues for applicability at the H. B. Robinson Steam Electric Plant. The team performed an independent review for these issues and where applicable, assessed the licensee's evaluation and dispositioning of each item. The issues that received a detailed review by the team included:

- NRC Information Notice 2012-14, “Motor Operated Valves Inoperable Due to Stem – Disc Separation”

- NRC Information Notice 2011-14, “Component Cooling Water System Gas Accumulation and Other Performance Issues”

- NRC Information Notice 2010-12, “Containment Liner Corrosion”

- NRC Information Notice 2007-09, “Equipment Operability Under Degraded Voltage Conditions”

- NRC Information Notice 2012-06, “Ineffective Use of Vendor Technical Recommendations”

- NRC Information Notice 1997-90, “Use of Non-conservative Acceptance Criteria in Safety Related Pump Surveillance Tests”

b. Findings

No findings were identified.

4. Other Activities

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

4OA5 Other Activities

.1 (Closed) Unresolved Item (URI) 05000261/2012005-03: Questions Regarding Whether GOTHIC is Sufficiently Qualified for Use in Operability Determinations (ML 13037A500)

a. Inspection Scope

In 2012, the inspectors verified, per Temporary Instruction (TI) 2515/177, that the licensee implemented or was in the process of implementing the commitments, modifications, and programmatically controlled actions described in the licensee’s response to NRC Generic Letter 2008-01, “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems.” During the inspection, the team questioned the acceptability of the licensee’s use of the GOTHIC computer code to support operability determinations with respect to the concerns identified in Information Notice 2011-17 which cautioned the use of computer models when evaluating the acceptability of voids in emergency core cooling systems. The inspectors opened a URI pending further inspection by a GOTHIC subject matter expert in the NRC’s Nuclear Reactor Regulation (NRR) office in order to evaluate and verify the licensee’s conclusions regarding the continued use of GOTHIC to support operability determinations. An NRR subject matter expert reviewed the facts of the subject URI as well as plant-specific information in order to evaluate acceptability of Carolina Power and Light Company (CP&L) use of GOTHIC at the H.B. Robinson Steam Electric plant to support operability determinations.

Robinson's current design basis with respect to voids in the subject systems is a water-solid, no gas condition. This means the subject systems must be water-solid when transitioning from an outage into power operation. Once the transition is complete, in recognition of the possibility that voids will form during operation, such voids are acceptable provided operability is reasonably maintained. Robinson has used GOTHIC to support past and continued operability assessments. GOTHIC is a multi-dimensional, multi-component computer code with the capability to model two-phase flow in nuclear power plant systems.

CP&L contracted with Nuclear Applications, Inc. (NAI) to apply GOTHIC to predict behavior associated with gas in the ECCS and RHR suction and discharge pipes at the H. B. Robinson plant. GOTHIC was compared to a broad range of test conditions and to tests that directly simulated aspects of behavior that may occur in plant piping. GOTHIC was then used to predict the behavior of gas trapped in specific RHR and ECCS locations.

Following review of additional information, the staff determined that GOTHIC was able to predict gas transport behavior for assessment of operability provided that (a) the pump inlet void fractions and volumes predicted by GOTHIC are shown to be acceptably conservative, (b) appropriate modeling methodologies are used in GOTHIC calculations and are consistent with testing modeling, and (c) GOTHIC predicted results are consistent with simplified methodologies, such as the Froude number, when those methodologies apply. The staff performed an in-depth evaluation of a reasonably bounding GOTHIC evaluation of the effect of trapped gas in an elevated 95-foot long 10-inch diameter pipe during initiation of flow from the containment sump during the emergency recirculation phase of operation. The GOTHIC evaluation was found to have acceptably determined gas volumes that would not jeopardize operability. The staff's review regarding Robinson's modeling detail support a conclusion that the licensee's use of GOTHIC to evaluate the potential movement of trapped gas with respect to operability was acceptable. Additionally, the licensee's use of GOTHIC was found to have acceptably assessed if there is a potential water hammer problem in a piping system and estimated the magnitude of potential pressure spikes. Based on this additional review, this URI is now closed.

b. Findings

No findings were identified.

.2 (Closed) URI 05000261/2012005-04: Questions Regarding the Adequacy of the Fill and Vent Procedure for the RHR Heat Exchangers (ML 13037A500)

a. Inspection Scope

In 2012, the inspectors verified, per TI 2515/177, that the licensee implemented or was in the process of implementing the commitments, modifications, and programmatically controlled actions described in the licensee's response to NRC Generic Letter 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems." During the inspection, the team identified a URI regarding the adequacy of the residual heat removal (RHR) system fill and vent procedure (OP-201-01, "RHR System Venting," Rev. 7) to ensure adequate fill and vent of the RHR heat exchangers (HXs). The inspectors reviewed the facts of the subject

URI as well as the licensee's evaluation of the RHR system fill and vent procedure (OP-201-01). The licensee's evaluation determined that it failed to evaluate the RHR system fill and vent procedure with respect to the HXs during the assessment of GL 2008-01 and therefore failed to incorporate adequate instructions to effectively vent the HXs. The inspectors reviewed a water hammer analysis for the RHR HXs, as a result of full flow testing the system (via OST-253, "Comprehensive Flow Test for the RHR Pumps," Rev 52) with a partially voided HX, to verify that the resulting loads where within the design limits and did not aversely challenge the system. The inspectors reviewed operating and test procedures to evaluate the sequence of system alignments following fill and vent evolutions to verify that the potential HX voids did not impact other portions of the system during its operation before and throughout the system flow test. Finally, the inspectors reviewed the status of the proposed corrective actions to verify they were being tracked to completion within the licensee's corrective action program.

b. Findings

No findings were identified. However, the inspectors did identify a minor performance deficiency and associated minor violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings." In accordance with IMC 0612, minor violations are not routinely documented in inspections reports. However, they may be documented to capture inspection activities and conclusions for closing a URI.

The inspectors determined that the licensee's failure to provide an adequate RHR fill and vent procedure to successfully vent the RHR HXs by establishing minimum flow rates and times necessary to dynamically flush the HXs was contrary to 10 CFR 50 Appendix B, Criterion V, and was a performance deficiency. Using IMC 0612 Appendix B, "Issue Screening," dated 9/7/12, the team determined this performance deficiency to be of minor significance because the HXs would have been effectively flushed during the post maintenance system flow test, prior to the system being placed in service, without any adverse impact to the system availability, reliability, or capability. Because this issue was entered into the licensee's corrective action program, as NCR 575346, and was of minor significance, the failure to comply with 10 CFR 50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," constitutes a minor violation that is not subject to enforcement action in accordance with NRC's Enforcement Policy.

This URI is now closed.

4OA6 Meetings, Including Exit

On May 16, 2013, the team presented the inspection results to Mr. Gideon and other members of the licensee's staff. Additional inspection results were communicated on June 20, 2013, on a teleconference with Mr. Gideon and other members of station staff, and on June 27, 2013 with Mr. Hightower. The inspectors verified that no proprietary information was documented in this report.

ATTACHMENT: SUPPLEMENTARY INFORMATION

SUPPLEMENTARY INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:
M. Connelly, Senior Regulatory Affairs Engineer
W. Hightower, Supervisor, Licensing/Regulatory Programs
J. Kunzmann, Supervisor Nuclear Rapid Response
A. Zimmerman, Lead Licensing Engineer

NRC personnel
J. Hickey, Senior Resident Inspector, Division of Reactor Projects (DRP), Robinson Resident Office
C. Scott, Resident Inspector, DRP, Robinson Resident Official
W. Lyon, Office of Nuclear Reactor Regulation, Division of Safety Systems
G. MacDonald, DRP, Region II
G. Hopper, Chief, Projects Branch 4, DRP, Region II
R. Nease, Chief, Engineering Branch Chief 1, Division of Reactor Safety, Region II

LIST OF ITEMS

Opened and Closed

05000261/2013007-01	NCV	Failure to Account for Containment Temperature Measurement Uncertainty [Section 1R21.2.9]
05000261/2013007-02	FIN	Failure to Evaluate SBO Coping Equipment for Environmental Conditions [Section 1R21.2.13]
05000261/2013007-03	NCV	Failure to Have Adequate Analyses Supporting the Degraded Voltage Relay Setpoints [Section 1R21.2.15]
05000261/2013007-04	NCV	Failure to Have Adequate Analyses For the E1 Bus Fast Transfer [Section 1R21.2.16.1]
05000261/2013007-05	NCV	Failure to Have Appropriate Procedure to Verify Degraded Voltage Relay Circuit Status [Section 1R21.2.16.2]

Opened

05000261/2013007-07	URI	Questions Regarding License Basis Design Requirements for Degraded Voltage Relays [Section 1R12.2.16.3]

<u>Closed</u>

05000261/2012005-03	URI	Questions Regarding Whether GOTHIC is Sufficiently Qualified for Use in Operability Determinations [Section 4OA5.1]
05000261/2012005-04	URI	Questions Regarding the Adequacy of the Fill and Vent Procedure for the RHR Heat Exchanger [Section 4OA5.2]

LIST OF DOCUMENTS REVIEWED

Calculations

DP-002-SIS, Design Basis Differential Pressure Report for the Motor-Operated Valves (MOVs) in the Safety Injection System (SIS), Rev. 8
PCHG-EVAL, Engineering Change, 83689R2, Rev. 2
PCHG-DESG, Engineering Change, 0000072699R1, Evaluation of Containment Liner, Insulation, Sheathing, and Coatings
PCHG-DESG, Engineering Change, 55265R0, Rev. 0
RNP-C/CONT-1002, Determination of Containment Heat Sink, Rev. 3
RNP-C/EQ-1334, Weak Link Analysis MOV SI-860 A/B CV Sump Recirculation; SI-862 A/B RWST to RHR Loop Isolation; SI-864 A/B RWST Discharge Isolation, Rev. 4
RNP-C/EQ-1364, USI A-46 Seismic Qualification Records, Rev. 1
RNP-C/STRU-1128, Minimum Allowable Containment Liner Thickness, Rev. 6
RNP-C/STRU-1130, Analysis of Containment Liner Bulge, Rev. 2
RNP-E-1.003, Instrument Bus Channel Loading, Rev. 6
RNP-E-1.025, 120 Vac Instrument Bus Coordination, Rev. 4
RNP-E-2.005, OC Protection for RHR Pumps B & C Motors, Rev. 2
RNP-E-2.006, OC Protection for Safety Injection Pumps A, B, & C Motors, Rev. 4
RNP-E-2.009, OC Protection for Emergency Bus E1 & E2 Emergency Supply, Rev. 3
RNP-E-2.010, OC Protection for Emergency Bus E1 & E2 Normal Supply, Rev. 3
RNP-E-2.011, OC Protection/Coordination for Feeder Breaker to MCC 5 & 16, Rev. 4
RNP-E-2.012, OC Protection/Coordination for Feeder Breaker to MCC 6, Rev. 3
RNP-E-5.018, Ampacity Evaluation of 125 Vdc and 120 Vac Power Cables, Rev. 8
RNP-E-5.043, Class 1E MCCs Control Loop Analysis, Rev. 3
RNP-E-6.004, DC Short Circuit Study, Rev. 5
RNP-E-6.005, Overcurrent Protection and Coordination of 125 Vdc Distribution System, Rev. 1
RNP-E-6.018, DC Control Circuit Loop Analysis, Rev. 3
RNP-E-6.018.A001.17B, Panel A, Circuit 1, Compt 17B
RNP-E-6.022, DC Voltage Profile, Rev. 4
RNP-E-8.002, AC Auxiliary Electrical Distribution System Study, Rev. 8C
RNP-E-8.042, AC MOV Evaluation, Rev. 4
RNP-E-8.054, Load Flow/Short circuit analysis, 640kW/800kVA Security Diesel Generator, 9/26/11
RNP-E-8.059, Security UPS Sizing and Loading, Rev. 1
RNP-E-8.060, Security Battery and Charger Sizing, Rev. 0
RNP-F/NFSA-0052, RNP Plant Data for LOCA Analysis, Rev. 2
RNP-F/NFSA-0188, RNP Cycle 28 Plant Parameters Document, Rev. 3
RNP-I/INST-1010, Emergency Bus – Degraded Grid Voltage Relay, Rev. 3
RNP-I/INST-1023, Refueling Water Storage Tank Uncertainty and Scaling Calculation, Rev. 4
RNP-I/INST-1044, Containment Pressure Loop Uncertainty and Scaling Calculation, Rev. 3
RNP-I/INST-1057, Containment Pressure Instrument Uncertainty Calculation, (PT-956, 957), Rev. 1
RNP-I/INST-1058, Containment Water Level Uncertainty Calculation, Rev. 4
RNP-I/INST-1067, RHR Flow Instrument Uncertainty Calculation, Rev. 1
RNP-I/INST-1108, Misc Flows EOP Setpoint Parameters, Rev. 1
RNP-I/INST-1109, Containment EOP Setpoint Parameters, Rev. 3
RNP-M/HVAC-1061, Volume Weighted Containment Bulk Average Temperature, Rev. 2
RNP-M/HVAC-1065, Transfer of RNP-M/HVAC-1061 to ERFIS, Rev. 0
RNP-M/MECH-1114, Residual Heat Removal Parameters and Component Flow and Resistance Coefficients, Rev. 3
RNP-M/MECH-1202, RWST Percent Level VS Inventory – RNP Curve Book 8.11, Rev. 3
RNP-M/MECH-1435, Set-up Calculation for MOV SI-864A, Rev. 10
RNP-M/MECH-1436, Set-up Calculation for MOV SI-864B, Revs. 7, 8, and 9

RNP-M/MECH-1515, Set-up Calculation for MOV AFW-V2-14A, Rev. 8
RNP-M/MECH-1599, RHR Pump NPSH, Rev. 0
RNP-M/MECH-1621, Containment Isolation Valves 10CFR50 Appendix J Allowable Leakage Rates, Rev. 5
RNP-M/MECH-1637, CS/SI/RHR System Hydraulic Model, Rev. 9
RNP-M/MECH-1651, Containment Analysis Inputs, Rev. 12
RNP-M/MECH-1716, Air Operated Valve Required Thrust and Actuator Capabilities for RVI-1, Rev. 1
RNP-M/MECH-1717, Air Operated Valve Required Thrust and Actuator Capabilities for RVI-2, Rev. 0
RNP-M/MECH-1718, Air Operated Valve Required Thrust and Actuator Capabilities for RVI-3, Rev. 1
RNP-M/MECH-1734, Basis for AOV Calculations, Rev. 0
RNP-M/MECH-1802, Safety Related Pump Minimum Performance Requirements, Rev. 3
WES 1543, 88-10, Resolution to Generic Letter, Rev. 0, dated 6/24/88

Completed Procedures
EST-134, Main Steam Isolation Valves Air Leakage Test (Refueling), Rev. 10, dated 3/9/12
EST-137-6, Local Leak Rate Test of Post-Accident Venting and Pressure Relief Valves (P-41), Rev. 0, dated 7/5/10 and 3/12/12
JPM IP-005, Locally Establish AFW Flow to “A”, “B”, and “C” S/G’s from the SDAFW Pump and Control S/G Levels and Pressures IAW EPP-1 and EPP-1 Attachment 1, Rev. 8, dated 1/25/07 and 1/13/05
JPM IP-005, Locally Establish AFW Flow to “A”, “B”, and “C” S/G’s from the SDAFW Pump and Control S/G Levels and Pressures IAW EPP-1 and EPP-1 Attachment 1, Rev. 9, dated 6/20/12 and 1/27/10
OST-022, Inverter A and B Weekly Surveillances, Rev. 18, dated 4/12/13 and 4/19/13
OST-151-1, Safety Injection System Components Test – Pump A, Rev. 36, dated 1/8/13
OST-151-2, Safety Injection System Components Test – Pump B, Rev. 33, dated 1/18/13
OST-151-3, Safety Injection System Components Test – Pump C, Rev. 36, dated 1/22/13
OST-151-4, Comprehensive Flow Test For Safety Injection Pump A, Rev. 17, dated 5/6/11
OST-151-5, Comprehensive Flow Test For Safety Injection Pump B, Rev. 17, dated 9/16/11
OST-151-6, Comprehensive Flow Test For Safety Injection Pump C, Rev. 17, dated 7/3/12
OST-251-1, RHR Pump A and Components Test, Rev. 28, dated 3/21/13
OST-251-2, RHR Pump B and Components Test, Rev. 29, dated 1/31/13
OST-253, Comprehensive Flow Test For The RHR Pumps, Rev. 52, dated 2/27/12
OST-405, TSC/EOF/Security Diesel Generator (Semiannual), Rev. 35 dated 2/7/13, and Rev. 37 dated 7/22/13
OST-701-8, V12-10 and V12-11 Inservice Valve Test, Rev. 11, dated 2/8/11 and 1/1/12
OST-702-1, Secondary Side Inservice Valve Test for MSIV’s, Rev. 4, dated 7/6/10
OST-702-1, “Secondary Side Inservice Valve Test for MSIV’s,” Rev. 6, dated 3/18/12
OST-707-8, Inservice Valve Position Indicator Verification, Rev. 4, dated 9/9/09, and Rev. 5, dated 3/9/12
TMM-035, Attachment 10.10, Static Post Test Evaluation of GL 89-10 Program Rising Stem MOVs, for Valve No. MOV MS-V1-8A, Test No. 05290002, Rev. 21, dated 10/17/05
TMM-035, Attachment 10.10, Static Post Test Evaluation of GL 89-10 Program Rising Stem MOVs, for Valve No. MOV MS-V1-8B, Test No. 04120002, Rev. 21, dated 4/29/04
TMM-035, Attachment 10.10, Static Post Test Evaluation of GL 89-10 Program Rising Stem MOVs, for Valve No. MOV MS-V1-8C, Test No. 04113004, Rev. 21, dated 4/22/04
TMM-035, Attachment 10.10, Static Post Test Evaluation of GL 89-10 Program Rising Stem MOVs, for Valve No. MOV SI-864A, Test No. 07108004 / 05, Rev. 21, dated 4/18/07

TMM-035, Attachment 10.10, Static Post Test Evaluation of GL 89-10 Program Rising Stem MOVs, for Valve No. MOV SI-864A, Test No. I0125C09/11, Rev. 21, dated 5/5/10
TMM-035, Attachment 10.10, Static Post Test Evaluation of GL 89-10 Program Rising Stem MOVs, for Valve No. MOV SI-864B, Test No. 05271001, Rev. 21, dated 9/28/05
TMM-035, Attachment 10.10, Static Post Test Evaluation of GL 89-10 Program Rising Stem MOVs, for Valve No. MOV SI-864B, Test No. 05279007, Rev 21, dated 10/6/05
TMM-035, Attachment 10.10, Static Post Test Evaluation of GL 89-10 Program Rising Stem MOVs, for Valve No. MOV SI-864B, Test No. I0130C11/13, Rev. 21, dated 5/10/10
TMM-127, Attachment 10.1, Analysis of Category 1 AOV Diagnostic Data, Sliding Stem Valve with Spring/Diaphragm Actuator Spring to Close Configuration, for Valve No. RV1-1, Rev. 3, dated 5/7/07
TMM-127, Attachment 10.1, Analysis of Category 1 AOV Diagnostic Data, Sliding Stem Valve with Spring/Diaphragm Actuator Spring to Close Configuration, for Valve No. RV1-1, Rev. 5, dated 2/18/12
TMM-127, Attachment 10.1, Analysis of Category 1 AOV Diagnostic Data, Sliding Stem Valve with Spring/Diaphragm Actuator Spring to Close Configuration, for Valve No. RV1-2, Rev. 3, dated 5/7/07
TMM-127, Attachment 10.1, Analysis of Category 1 AOV Diagnostic Data, Sliding Stem Valve with Spring/Diaphragm Actuator Spring to Close Configuration, for Valve No. RV1-2, Rev. 5, dated 2/23/12
TMM-127, Attachment 10.1, Analysis of Category 1 AOV Diagnostic Data, Sliding Stem Valve with Spring/Diaphragm Actuator Spring to Close Configuration, for Valve No. RV1-3, Rev. 1, dated 10/13/05
TMM-127, Attachment 10.1, Analysis of Category 1 AOV Diagnostic Data, Sliding Stem Valve with Spring/Diaphragm Actuator Spring to Close Configuration, for Valve No. RV1-3, Rev. 5, dated 5/20/10

Completed Work Orders
WO 00067292 01, Test Distribution Panel A Breakers, dated 4/10/01
WO 00067404 01, Perform Inspection and Cleaning of MCCA, MCCB, Distribution Panel A and Distribution Panel B, dated 4/21/07
WO 00115872 01, Test Distribution Panel A Breakers, dated 4/10/01
WO 00756179 01, Test Distribution Panel A Breakers, dated 3/04/07
WO 00774394 01, Diagnostic Test (SI-864A-MO), dated 12/11/09
WO 00774548 01, Disassemble/Inspect/Replace Operator Diaphragm in RV1-3-AO, dated 4/17/10
WO 00774548 05, Disassemble/Inspect/Replace RV1-3-AO - Post-Maintenance Diagnostic Test, dated 4/17/10
WO 00774548 06, Disassemble/Inspect/Replace RV1-3-AO - Pre-Maintenance Diagnostic Test, dated 4/17/10
WO 00774599 01, Disassemble/Inspect/Repair RV1-3, dated 4/17/10
WO 00774599 06, Disassemble/Inspect/Repair RV1-3 – Pre-Maintenance Diagnostic Test, dated 4/17/10
WO 00774599 07, Disassemble/Inspect/Repair RV1-3 – Post-Maintenance Diagnostic Test, dated 4/17/10
WO 00774798 01, Perform Thermal Overload Test on MCC -5 (10J) (SI-864A), dated 1/12/10
WO 01348233 22, Perform Continuity Test on MS-V1-3A-AO IAW EC-66327, dated 2/11/12
WO 01348233 24, MS-V1-3A Rewire Light/Remove Jumper, dated 2/22/12
WO 01348233 25, MS-V1-3B Rewire Light/Remove Jumper, dated 2/22/12
WO 01348233 26, MS-V1-3C Rewire Light/Remove Jumper, dated 2/22/12
WO 01446513 01, Perform MST-921 on “A” Station Batt, dated 4/17/10

WO 01446513 01, Station Battery A Service Test, dated 5/26/10
WO 01485420 01, Diagnostic Test (SI-864B-MO), dated 12/11/09
WO 01485584 01, Check Valve Inspection on MS-V1-3A, dated 2/23/10
WO 01486079, Perform Calibration Check of Containment Average Temperature, dated 5/18/10
WO 01486091 01, Test Distribution Panel A Breakers, dated 1/13/10
WO 01537339 01, Limitorque Inspection SI-864A, dated 12/11/09
WO 01537340 01, Limitorque Inspection SI-864B, dated 12/11/09
WO 01584252 01, Limitorque Grease Inspection of Valve MS-V1-8B-MO, dated 10/20/10
WO 01687088 01, MS-V1-3A Accumulator Tank and Support, dated 8/29/11
WO 01687089 01, MS-V1-3B Accumulator Tank and Support, dated 8/29/11
WO 01687090 01, MS-V1-3C Accumulator Tank and Support, dated 8/29/11
WO 01710648 01, Matrix PMT for Valves MS-V1-3A, -3B, and -3C to Perform OST-702-1, dated 4/5/10
WO 01749670 69, UPS-Batt/UPS-1 operability testing, dated 7/23/11
WO 01757318 01, SI-864B-MO Needs Diagnostic Test Failed LLRT, dated 5/15/10
WO 01758805 03, Perform Valve Overhaul on SI-864B, dated 5/25/10
WO 01784639 01, Limitorque Grease Inspection of Valve MS-V1-8A-MO, dated 9/6/11
WO 01800149 03, Disassemble/Inspect/Replace RV1-2-AO - Pre-Maintenance Diagnostic Test, dated 2/1/11
WO 01800149 01, Disassemble/Inspect/Replace Operator Diaphragm in RV1-2-AO, dated 1/27/11
WO 01800149 04, Disassemble/Inspect/Replace RV1-2-AO - Post-Maintenance Diagnostic Test, dated 2/1/11
WO 01800153 01, Disassemble/Inspect/Replace Operator Diaphragm in RV1-1-AO, dated 1/27/11
WO 01800153 03, Disassemble/Inspect/Replace RV1-1-AO - Post-Maintenance Diagnostic Test, dated 2/1/11
WO 01800153 04, Disassemble/Inspect/Replace RV1-1-AO - Pre-Maintenance Diagnostic Test, dated 2/1/11
WO 01800327 01, Disassemble/Inspect/Repair RV1-1, dated 1/27/11
WO 01800327 04, Disassemble/Inspect/Repair RV1-1 – Pre-Maintenance Diagnostic Test IAW PM-477, dated 2/3/11
WO 01800327 06, Disassemble/Inspect/Repair RV1-1 – Post-Maintenance Diagnostic Test, dated 2/3/11
WO 01800329 01, Disassemble/Inspect/Repair RV1-2, dated 1/27/11
WO 01800329 07, Disassemble/Inspect/Repair RV1-2 – Post-Maintenance Diagnostic Test, dated 2/1/11
WO 01800329 08, Disassemble/Inspect/Repair RV1-2 – Pre-Maintenance Diagnostic Test, dated 8/23/11
WO 01800488, Disassemble, Inspect and Replace Seaport Obstruction, (RO27), dated 2/15/12
WO 01800603, Calibrate RHR Flow Transmitter, FT-605, and Perform Loop Check, dated 2/10/12 and 10/4/12
WO 01820971-01, Inspect and Repair Selected CV Liner, dated 11/15/11
WO 01820971-04, Inspect and Repair Selected CV Liner, dated 02/10/12
WO 01977400 01, Perform Security Battery Inspection, dated 3/5/12
WO 02041474 01, MCC-6 (11J)-42/C Coil Failed Pick Up Time for SI-864B, dated 2/14/12
WO 02053755 01, SG “B” Steam PORV Drain Line Sheared on RV1-2, dated 5/2/12
WO 02067295 01, Check Security and EFRIS Batteries, dated 4/17/12

<u>Corrective Action Program Documents</u>

Action Requests:

151422, Evaluate Operation with DVR’s Bypassed
230929-07, IN 2007-09 Equip Op Under Degraded Volt
243370-08, Investigation for improvement to recover margin for E1/E2 buses
274999, Reduced Voltage Testing of Control Circuit Voltages
393805, Security UPS Battery PM Revision
411609-07, NRC Information Notice 2010-12
422184, Evaluation of In-Service DB Breakers
478605, Information Notice 2011-14
500790-08, Existing Supports in Exhaust Piping (10-MS-24) for the SDAFP Turbine are Degraded and Non-Conforming
520845, SI-864A: Apparent Issue with Seats and Seal Welds (RO-27)
522848, SG "B" STEAM LINE PORV DRAIN LINE SHEARED
531773, A, B and C MSIV Fail to Shut
534216, EPP-010 Does not match CWD regarding RHR-759A
563249-18, Main Steam Isolation valves Fail to Close at Shearon Harris Nuclear Power Plant
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