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

December 19, 2011

Mr. T. A. Lynch
Vice President
Southern Nuclear Operating Company, Inc.
Joseph M. Farley Nuclear Plant
Farley Nuclear Plant
P.O. Drawer 470
BIN B500
Ashford, AL 36312

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT - NRC COMPONENT DESIGN BASES INSPECTION - INSPECTION REPORT 05000348/2011010 AND 05000364/2011010

Dear Mr. Lynch:

On, December 8, 2011, U. S. Nuclear Regulatory Commission (NRC) completed an inspection at your Joseph M. Farley Nuclear Plant, Units 1 and 2. The enclosed inspection report documents the inspection results, which were discussed on December 8, 2011, with Mr. Todd Youngblood and other members of your staff.

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

This report documents six NRC identified findings of very low safety significance (Green), which were determined to involve violations of NRC requirements. The NRC is treating these violations as non-cited violations (NCVs) consistent the NRC Enforcement Policy. If you contest 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 Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-001; with copies to the Regional Administrator Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Farley. Further, if you disagree with the cross-cutting aspect assigned to any finding 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 Farley. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

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 document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA by Binoy B. Desai For/

Rebecca Nease, Chief
Engineering Branch 1
Division of Reactor Safety

Enclosure: Inspection Report 05000348, 364/2011010,
w/Attachment: Supplemental Information

Docket No.: 50-348, 50-364
License No.: NPF-2, NPF-8

cc w/encl: (See page 3 and 4)

NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA by Binoy B. Desai For/

Rebecca Nease, Chief
Engineering Branch 1
Division of Reactor Safety

Enclosure: Inspection Report 05000348, 364/2011010,
w/Attachment: Supplemental Information

Docket No.: 50-348, 50-364
License No.: NPF-2, NPF-8

cc w/encl: (See page 3 and 4)

Distribution w/encl:
M. Miller, RII, DRP
T. Lighty, RII, DRP
G. Croon, RII, DRP
R. Musser, RII, DRP
J. Worosillo, RII, DRP
J. Dodson, RII, DRP
RIDSNRRDIRS
PUBLIC
RidsNrrPMFarley Resource

X PUBLICLY AVAILABLE ☐ NON-PUBLICLY AVAILABLE ☐ SENSITIVE **X** NON-SENSITIVE
ADAMS: **X** Yes ACCESSION NUMBER: ML#113530575 **X** SUNSI REVIEW COMPLETE

OFFICE	RII:DRS	RII:DRS	RII:DRS	RII:DRS	CONTRACTOR
SIGNATURE	/RA/	/RA/ via email	/RA/ via email	/RA/ via email	/RA/ via email
NAME	S SANDAL	J EARGLE	MAS-PENARANDA	D JONES	M SHLYAMBERG
DATE	12/14/2011	12/16 /2011	12/12/2011	12/14 /2011	12/14 /2011
E-MAIL COPY?	**YES** NO	**YES** NO	**YES** NO	**YES** NO	**YES** NO
OFFICE	CONTRACTOR	RII:DRS	RII:DRP		
SIGNATURE	/RA/ via email	/RA/B.Desai For	/RA/		
NAME	P WAGNER	R NEASE	J HICKEY		
DATE	12/ 14 /2011	12/19 /2011	12/ 14 /2011		
E-MAIL COPY?	**YES** NO	**YES** NO	**YES** NO	YES NO	YES NO

OFFICIAL RECORD COPY DOCUMENT NAME: S:\DRS\ENG BRANCH 1\BRANCH INSPECTION FILES\2011-2012-2013 CYCLE EB1 INSPECTION FOLDERS FOR ALL SITES\FARLEY\FARLEY CDBI (2011010)\FARLEY 2011010 CDBI IR (REV 1).DOCX

cc w/encl:
S. Kuczynski
Chairman, President and CEO
Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

J. L. Pemberton
SVP & General Counsel-Ops & SNC
Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

D. G. Bost
Chief Nuclear Officer
Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

T. A. Lynch
Vice President
Joseph M. Farley Nuclear Plant
Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

Dennis R. Madison
Vice President
Edwin I. Hatch Nuclear Plant
Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

Paula Marino
Vice President
Engineering
Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

T. E. Tynan
Site Vice President
Vogtle Electric Generating Plant
Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

M. J. Ajluni
Nuclear Licensing Director
Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

Mark Williams
Commissioner
Georgia Department of Natural Resources
Electronic Mail Distribution

B. D. McKinney, Jr.
Regulatory Response Manager
Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

D. W. Daughhetee
Licensing Engineer
Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

T. D. Honeycutt
Regulatory Response Supervisor
Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

L. Mike Stinson
Vice President
Fleet Operations Support
Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

N. J. Stringfellow
Licensing Manager
Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

L. P. Hill
Licensing Supervisor
Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

L. L. Crumpton
Administrative Assistant, Sr.
Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

William D. Oldfield
Principal Licensing Engineer
Joseph M. Farley Nuclear Plant
Electronic Mail Distribution

Todd L. Youngblood
Plant Manager
Joseph M. Farley Nuclear Plant
Electronic Mail Distribution

cc w/encl: continued on page 4

cc w/encl (continued)

John G. Horn
Site Support Manager
Joseph M. Farley Nuclear Plant
Southern Nuclear Operating Company, Inc.
Electronic Mail Distribution

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Joseph M. Farley Nuclear Plant
U.S. NRC
7388 N. State Highway 95
Columbia, AL 36319

Cynthia A. Sanders
Radioactive Materials Program Manager
Environmental Protection Division
Georgia Department of Natural Resources
Electronic Mail Distribution

James C. Hardeman
Environmental Radiation Program Manager
Environmental Protection Division
Georgia Department of Natural Resources
Electronic Mail Distribution

Mr. Mark Culver
Chairman
Houston County Commission
P. O. Box 6406
Dothan, AL 36302

James L. McNees, CHP
Director
Office of Radiation Control
Alabama Dept. of Public Health
P. O. Box 303017
Montgomery, AL 36130-3017

Donald E. Williamson
State Health Officer
Alabama Dept. of Public Health
RSA Tower - Administration
Suite 1552
P.O. Box 30317
Montgomery, AL 36130-3017

F. Allen Barnes
Director
Environmental Protection Division
Georgia Department of Natural Resources
Electronic Mail Distribution

U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.: 050000348, 05000364

License Nos.: NPF-2, NPF-8

Report Nos.: 05000348/2011010 and 05000364/2011010

Licensee: Southern Nuclear Operating Company, Inc.

Facility: Joseph M. Farley Nuclear Plant, Units 1 and 2

Location: Columbia, AL

Dates: August 29 – December 8, 2011

Inspectors: S. Sandal, Senior Reactor Inspector (Lead)
D. Jones, Senior Reactor Inspector
J. Eargle, Reactor Inspector
D. Mas Penaranda, Reactor Inspector
M. Shlyamberg, Accompanying Personnel
P. Wagner, Accompanying Personnel

Approved by: Rebecca Nease, Chief
Engineering Branch 1
Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000348/2011-010, 05000364/2011-010; 08/29/2011 – 12/08/2011; Joseph M. Farley Nuclear Plant, Units 1 and 2; Component Design Bases Inspection.

This inspection was conducted by a team of four NRC inspectors from the Region II office, and two NRC contract personnel. Six Green non-cited violations (NCV) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using the NRC Inspection Manual Chapter (IMC) 0609, “Significance Determination Process” (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” (ROP) Revision 4, dated December 2006.

NRC identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The team identified a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion III, “Design Control,” (with two examples) for the licensee’s failure to implement design control measures to verify the adequacy of design inputs, assumptions, or limiting plant conditions which were relied upon in the design basis analyses used to demonstrate the adequacy of Condensate Storage Tank (CST) design. The licensee entered these issues into their Corrective Action Program (CAP) as Condition Reports (CRs) 355226, 355293, and 355294. The licensee performed operability evaluations in support of current operability and implemented additional compensatory measures to ensure that CST level would be maintained above the condenser hotwell make-up elevation pending completion of proposed long term corrective actions which included a license amendment request to increase the minimum volume of water specified by the limiting condition for operation in Technical Specification (TS) 3.7.6.

  The failure to utilize conservative design inputs, assumptions, or limiting plant conditions when implementing design control measures to verify the adequacy of CST design was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the mitigating systems cornerstone attribute of design control and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the finding challenged the assurance that the CST contained an adequate volume of water to support its safety function to supply condensate to the Auxiliary Feedwater (AFW) system in response to design basis events. In accordance with NRC IMC 0609.04, “Initial Screening and Characterization of Findings,” the team used the mitigating systems column to perform a Phase 1 Significance Determination Process (SDP) screening and determined the finding to be of very low safety significance (Green) because it was not a design issue resulting in loss of function, did not represent an actual loss of a system safety function, did not result in exceeding a TS allowed outage time, and did not affect external event mitigation. This analysis was based on information contained in licensee operability determinations which demonstrated that, although the TS required minimum volume of 150,000 gallons

was non-conservative, reasonable assurance existed such that the volume of CST water below the condenser hotwell make-up elevation was sufficient for the tank to perform its safety function. A cross-cutting aspect was not identified because the design basis calculation associated with the finding was approved on March 25, 1999, and did not represent current licensee performance. [Section 1R21.2.3]

- Green. The team identified a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion III, "Design Control," for the failure to implement design control measures to verify the adequacy of design inputs, assumptions, or limiting plant conditions which were relied upon in the design basis analyses used to demonstrate the capability of the Auxiliary Feedwater (AFW) system to deliver the required flowrates to the Steam Generators (SGs). The licensee entered this issue into the Corrective Action Program (CAP) as Condition Reports (CRs) 352210, 353743, 355898, 363850, and 369676. Additionally, the licensee performed an operability determination which concluded that the AFW system remained capable of performing its safety function because actual AFW pump performance was not degraded as assumed in the accident analyses.

  The failure to conservatively model AFW system friction losses when implementing design control measures to verify the capability of the AFW system to deliver the flowrates required by accident analyses was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the mitigating systems cornerstone attribute of design control and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the finding challenged the assurance that the AFW system would be capable of delivering the required flow during worst case accident conditions due to non-conservative modeling of system friction losses. In accordance with NRC IMC 0609.04, "Initial Screening and Characterization of Findings," the team used the mitigating systems column to perform a Phase 1 Significance Determination Process (SDP) screening and determined the finding to be of very low safety significance (Green) because it was not a design issue resulting in loss of function, did not represent an actual loss of a system safety function, did not result in exceeding a Technical Specification (TS) allowed outage time, and did not affect external event mitigation. A cross-cutting aspect was not identified because the design basis calculation associated with the finding was approved on March 25, 1999, and did not represent current licensee performance. [Section 1R21.2.3]

- Green. The team identified a non-cited violation (NCV) of Technical Specification (TS) 5.4, "Procedures," for the licensee's failure to provide adequate procedural guidance for controlling steam generator (SG) and pressurizer level during loss of instrument air events and Chemical and Volume Control System (CVCS) malfunctions. Specifically, the licensee failed to evaluate the capability of motor-operated valves (MOVs) to be cycled as directed by abnormal operating procedures (AOPs). The licensee entered these issues into their Corrective Action Program (CAP) as Condition Reports (CRs) 355230, 355672 and 355695; performed DOEJ – FRSNC326893-E001, "Evaluate Cycling of Q1E21MOV8107, Q1E21MOV8107, and Q1E21MOV3764A through F"; and implemented a standing order (S-2011-12) that restricted the cycling the of the MOVs until the procedures were revised.

The failure to provide adequate procedural guidance for controlling SG and pressurizer level during loss of air events and CVCS malfunctions was a performance deficiency. The performance deficiency was more that minor because it was associated with the mitigating systems cornerstone attribute of equipment performance and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee directed the cycling of MOVs in AOPs without performing evaluations to provide assurance that the components would not fail as a result of the cycling operations and lead to a condition of inadequate SG and pressurizer level control. In accordance with NRC IMC 0609.04, "Initial Screening and Characterization of Findings," the team used the mitigating systems column to perform a Phase 1 Significance Determination Process (SDP) screening and determined the finding to be of very low safety significance (Green) because it was not a design issue resulting in loss of function, did not represent an actual loss of a system safety function, did not result in exceeding a Technical Specification (TS) allowed outage time, and did not affect external event mitigation. A cross-cutting aspect was not identified because the finding did not represent current performance. [Section 1R21.2.3]

- Green. The team identified a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion III, "Design Control," involving two examples. In the first example, the licensee failed to translate the minimum Component Cooling Water (CCW) flow for the Residual Heat Removal (RHR) seal coolers into Annunciator Response Procedures (ARPs). In the second example, the licensee failed to translate the Motor Driven Auxiliary Feedwater (MDAFW) and Turbine Driven Auxiliary Feedwater (TDAFW) pump minimum flow requirements into applicable ARPs. The licensee entered these issues into their Corrective Action Program (CAP) as Condition Reports (CRs) 348613 and 352485.

  The failure to correctly translate the applicable design bases information for the RHR pump seal coolers and the Auxiliary Feedwater (AFW) pumps into procedures was a performance deficiency. The finding was determined to be more than minor because it was associated with the procedure quality attribute of the mitigating system cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to translate the appropriate minimum flow requirements into ARPs adversely affected the quality of procedures used to respond to alarm conditions that are required by Regulatory Guide 1.33, "Quality Assurance Program Requirements." The inadequate procedures adversely affected the ability of operators to assess operability and to combat deficiencies associated with risk significant equipment. In accordance with NRC IMC 0609.04, "Initial Screening and Characterization of Findings," the team used the mitigating systems column to perform a Phase 1 Significance Determination Process (SDP) screening and determined the finding to be of very low safety significance (Green) because it was not a design issue resulting in loss of function, did not represent an actual loss of a system safety function, did not result in exceeding a Technical Specification (TS) allowed outage time, and did not affect external event mitigation. A cross-cutting aspect was not identified because the finding did not represent current performance. [Section 1R21.2.4]

- Green. The team identified a non-cited violation (NCV) of 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” for the licensee’s failure to perform condition monitoring or otherwise implement an appropriate preventive maintenance program for the 2C Diesel Generator (DG) A and B room exhaust fan louvers. The licensee entered this issue into their corrective action program (CAP) as condition reports (CRs) 351580, 349883, and 355130.

  The failure to perform condition monitoring or otherwise implement an appropriate preventive maintenance program for the 2C DG A and B exhaust fan louvers was a performance deficiency. This performance deficiency was more than minor because it was associated with equipment performance attribute of the mitigating systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to perform condition monitoring or otherwise implement an appropriate preventive maintenance program for the 2C DG A and B room exhaust fan louvers challenged the assurance that these components would remain capable of performing their intended functions. In accordance with NRC IMC 0609.04, “Initial Screening and Characterization of Findings,” the team used the mitigating systems column to perform a Phase 1 SDP screening and determined the finding to be of very low safety significance (Green) because it was not a design issue resulting in loss of function, did not represent an actual loss of a system safety function, did not result in exceeding a TS allowed outage time, and did not affect external event mitigation. Because the licensee had initiated CRs in 2008 and 2009 for the 2C DG room exhaust louvers, and repairs were not made in a timely manner to address the issue, this finding was assigned a cross-cutting aspect in the corrective action program component of the problem identification and resolution area [P.1(d)]. [Section 1R21.2.6]

- Green. The team identified a non-cited violation (NCV) of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the licensee’s failure to establish an adequate test procedure used to demonstrate that the Turbine Driven Auxiliary Feedwater (TDAFW) pump discharge check valves were capable of performing their design basis function. The test procedure was inadequate in providing assurance that the Auxiliary Feedwater (AFW) system was capable of providing the required design basis flow rates to the Steam Generators (SGs) with reverse flow into an idle TDAFW pump via the discharge check valves. This issue was entered into the licensee’s Corrective Action Program (CAP) as Condition Report (CR) 348795.

  The failure to develop an adequate test procedure which demonstrated that TDAFW pump discharge check valves were capable of performing their design basis function was a performance deficiency. This performance deficiency was more than minor because it adversely affected the equipment performance attribute of the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the acceptance criteria used in the test procedure was non-conservative when compared to the flow rates required by the accident analyses, and the test procedure was performed at lower system pressures (which were not representative of actual design conditions). In accordance with NRC IMC 0609.04, “Initial Screening and Characterization of Findings,” the team used the mitigating systems column to perform a Phase 1 Significance Determination Process (SDP) screening and determined the finding to be of very low safety significance (Green) because it was not a design issue resulting in loss of function, did not represent an actual loss of a system safety function, did not result in exceeding a Technical Specification (TS) allowed outage time, and did not affect external event mitigation. Because the test procedure did not contain complete, accurate, and up-to-date information consistent with the system design basis safety analysis, this finding is assigned a cross-cutting aspect in the resources component of the human performance area [H.2(c)]. [Section 1R21.2.7]

Licensee-Identified Violations

None

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 (PRA). 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 fifteen components, including one associated with containment large early release frequency (LERF), and four operating experience (OE) items.

The team performed a margin assessment and a detailed review of the selected risk-significant components to verify that the design bases had been correctly implemented and maintained. This margin assessment 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 (RIS) 05-020 (formerly Generic Letter (GL) 91-18) conditions, NRC resident inspector input of problem equipment, System Health Reports, industry OE, and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, 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 (15 Samples)

.2.1 Steam Generator (SG) Main Steam Isolation Valves (MSIV) - Q2N11HV3369(3370)

a. Inspection Scope

The team reviewed the updated Final Safety Analysis Report (UFSAR), TS, Functional System Description (FSD), and piping and instrumentation diagrams (P&IDs), applicable plant calculations, and drawings to identify the design bases requirements of the MSIVs. The team examined system health reports, records of surveillance testing and maintenance activities, and applicable corrective actions to verify that potential degradation or low margin design issues were being monitored, prevented and/or corrected. Additionally, the team reviewed station operating and off-normal response procedures to verify design bases requirements had been adequately translated into procedural instructions. The team performed a walkdown of the valve areas. The team reviewed design bases documentation, maintenance records, and drawings of the instrument air system to verify that the support function provided to the MSIVs was consistent with design requirements. Control panel indicators were observed and operating procedures reviewed to verify that component operation and alignments were consistent with design and licensing basis assumptions.

b. Findings

No findings were identified.

.2.2 Steam Generator Atmospheric Relief Valves (ARVs) - Q1N11PV3371A(B/C) - LERF

a. Inspection Scope

The team reviewed the UFSAR, TS, FSD, P&IDs, applicable plant calculations, and drawings to identify the design bases requirements of the ARVs. The team examined system health reports, records of surveillance testing and maintenance activities, and applicable corrective actions to verify that potential degradation or low margin design issues were being monitored, prevented and/or corrected. Additionally, the team reviewed station operating and off-normal response procedures to verify design bases requirements had been adequately translated into procedural instructions. The team performed a walkdown of the valve areas. The team reviewed design bases documentation, maintenance records, and drawings of the instrument air system to verify that the support function provided to the ARVs was consistent with design requirements. Control panel indicators were observed and operating procedures reviewed to verify that component operation and alignments were consistent with design and licensing basis assumptions. The team reviewed off-normal and emergency operating procedures to verify that adequate guidance exists for operators to respond to a design bases steam generator tube rupture event. The team observed a simulator scenario to verify the capability of the operators to mitigate a Steam Generator Tube Rupture (SGTR) event as described in the UFSAR. The team performed a walkdown of local manual actions associated with a SGTR event to verify the feasibility of the directed actions.

b. Findings

No findings were identified.

.2.3 Motor Driven Auxiliary Feedwater (MDAFW) Pump (Mechanical) - Q2N23P001B

a. Inspection Scope

The team reviewed the plant's TS, UFSAR, FSD, and P&IDs to establish an overall understanding of the design bases of the MDAFW. Design calculations and test data were reviewed to verify that design basis capability, and flow rates had been appropriately translated into these documents. The team concentrated its efforts on the pump's capability of performing its safety function (i.e., delivering the required flow rate to the steam generators at the prescribed design pressure). Records of surveillance testing and maintenance activities, and applicable corrective actions were examined to verify that potential degradation or low margin design issues were being monitored, prevented and/or corrected. MDAFW walkdowns were conducted to verify that the installed configurations would support its design basis function under accident conditions and had been maintained to be consistent with design assumptions and to visually inspect the material condition of the pumps. Control panel indicators were observed and operating procedures reviewed to verify that MDAFW operation and alignments were consistent with design and licensing basis assumptions. Vendor documentation, system health reports, preventive and corrective maintenance history, and corrective action system documents were reviewed in order to verify that potential degradation was monitored or prevented and the component replacement was consistent with inservice/equipment qualification life. The team reviewed off-normal procedures to verify that adequate guidance exists for operators to control pressurizer and SG levels during loss of air events. The team performed a walkdown of local manual actions associated with controlling pressurizer and SG levels to verify the feasibility of the directed actions.

b. .1 Findings

Introduction: A Green NRC identified NCV of 10 CFR 50, Appendix B, Criterion III, "Design Control," (with two examples) was identified for the failure to implement design control measures to verify the adequacy of design inputs, assumptions, or limiting plant conditions which were relied upon in the design basis analyses used to demonstrate the adequacy of Condensate Storage Tank (CST) design.

Description: The CST is a safety related, seismic category I tank that holds up to 500,000 gallons of water and is required by the TS 3.7.6 limiting condition for operation to be maintained at a minimum of 150,000 gallons for use by the AFW system under normal operation and in response to accident conditions. In order to ensure this requirement, the lower 13' 3-1/8" of the 46' inside diameter (ID) tank is designed to withstand the effects of tornado missiles. The CST has two 8" AFW suction pipes – one for the TDAFW pump and one for both MDAFW pumps. Both suction pipes open at 4" from the tanks' bottom facing down. The suction piping centers are approximately 1' 3" apart. The CST has an internal bladder that prevents introduction of air under normal operating conditions.

The team reviewed calculations BM-95-0961-001, "Verification of CST Sizing Basis", Rev. 4, dated March 25, 1999, and CBI-72-4859, "Condensate Storage Tank", Rev. 0 and identified two examples where the licensee had failed to implement design control measures to verify the adequacy of design inputs, assumptions, or limiting plant conditions which were relied upon in design basis analyses used to demonstrate that the CST would have a sufficient volume of water to perform its safety function. The following examples were identified:

Example 1 - Effect of +2% Calorimetric Error and 15 Megawatts Thermal (MWt) Heat Input from the Reactor Coolant Pumps (RCPs) – The team reviewed the applicable design basis analysis for the CST and noted that calculation BM-95-0961-001 used non-conservative design inputs and did not take into consideration +2% calorimetric error when establishing the initial assumed reactor thermal power as discussed in UFSAR accident analysis. Additionally, the team noted that the calculation used 10 MWt and not 15 MWt as a heat input from the RCPs as also discussed in UFSAR accident analysis. The team concluded that the use of non-conservative inputs in the CST design analyses adversely impacted the margin available in the TS 3.7.6 required CST volume. Based on the team's observations, the licensee entered the issue into the CAP as CR 355226 and documented their operability determination in PDO 0-11-06. The licensee's evaluation demonstrated that these non-conservative assumptions resulted in an increase in required water volume by 1,648 gallons in excess of the value originally calculated in BM-95-0961-001. The operability determination also concluded that sufficient water remained available below the condenser hotwell make-up elevation for the CST to be able to perform its safety function.

Example 2 - Effect of T_{avg} Assumed in the Accident Analysis on CST Volume Requirements – The team reviewed the applicable design basis analysis for the CST and noted that calculation BM-95-0961-001 did not address the CST volume requirements for the Main Feed Line Break (MFLB) case that TS Bases 3.7.6 described as the limiting event for the required condensate volume. The licensee entered this issue into the CAP as CR 355293. The licensee's evaluation of this observation identified that the volume required in the MFLB case was 416 gallons less than the loss of offsite power cooldown case volume. However, the team also noted that the licensee's evaluation was based on the Reactor Coolant System (RCS) sensible heat calculated for the no-load case where RCS T_{avg} is nominally 547 °F. As documented in UFSAR accident analysis, RCS T_{avg} for the MFLB at time zero was assumed to be higher than 547 °F at approximately 583 °F. Additionally, the team noted that UFSAR accident analysis assumed a vessel average temperature as high as 577.2 °F and stated that ± 6 °F steady-state T_{avg} error was considered in the analysis. From this information, the team concluded that the nominal use of 547 °F T_{avg} in the CST design basis analysis was not consistent with the assumptions stated in UFSAR accident analysis and was non-conservative in determining the most limiting CST required water volume. The team's evaluation of the effect of the increased T_{avg} on the required CST volume could be as high as an additional 5,565 gallons. The team concluded that the use of non-conservative T_{avg} as an input in the CST design basis analysis for RCS sensible heat load adversely impacted the margin available in the TS 3.7.6 required CST volume. Based on the team's observations, the licensee entered the issue into the CAP as CR 355294 to address the sensible heat concern.

The team reviewed applicable operability determinations completed by the licensee regarding the issues identified above and concluded that although the TS required minimum volume of 150,000 gallons was determined to be non-conservative, reasonable assurance existed such that the volume of CST water below the condenser hotwell make-up elevation was sufficient for the tank to remain capable of performing its safety function (at reduced margin). As a result of the team's observations, the licensee implemented additional compensatory measures to ensure that CST level would be maintained above the condenser hotwell make-up elevation pending completion of proposed long term corrective actions (including a license amendment to modify the TS required minimum volume). The team also reviewed longer term proposed licensee corrective actions to revise the applicable design basis calculations for the CST.

Analysis: The failure to utilize conservative design inputs, assumptions, or limiting plant conditions when implementing design control measures to verify the adequacy of CST design was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the mitigating systems cornerstone attribute of design control and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the finding challenged the assurance that the CST contained an adequate volume of water to support its safety function to supply condensate to the AFW system in response to design basis events. In accordance with NRC IMC 0609.04, "Initial Screening and Characterization of Findings," the team used the mitigating systems column to perform a Phase 1 SDP screening and determined the finding to be of very low safety significance (Green) because it was not a design issue resulting in loss of function, did not represent an actual loss of a system safety function, did not result in exceeding a TS allowed outage time, and did not affect external event mitigation. This analysis was based on information contained in licensee operability determinations which demonstrated that, although the TS required minimum volume of 150,000 gallons was non-conservative, reasonable assurance existed such that the volume of CST water below the condenser hotwell make-up elevation was sufficient for the tank to perform its safety function. A cross-cutting aspect was not identified because the design basis calculation associated with the finding was approved on March 25, 1999, and did not represent current licensee performance.

Enforcement: 10 CFR 50, Appendix B, Criterion III, "Design Control," states, in part, that measures shall provide for verifying or checking the adequacy of design. Contrary to the above, since March 25, 1999, the licensee failed to implement design control measures as described in the two examples above to verify the adequacy of design inputs, assumptions, or limiting plant conditions which were relied upon in design basis calculations used to demonstrate the adequacy of CST design. Because the violation was of very low safety significance and it was entered into the licensee's CAP, this violation is being treated as an NCV consistent with the NRC Enforcement Policy: NCV 05000348, 364/2011010-01, "Failure to Implement Design Control Measures to Verify the Adequacy of CST Design."

b. .2 Findings

Introduction: A Green NRC identified NCV of 10 CFR 50, Appendix B, Criterion III, "Design Control," was identified for the failure to implement design control measures to verify the adequacy of design inputs, assumptions, or limiting plant conditions which were relied upon in the design basis analyses used to demonstrate the capability of the AFW system to deliver the required flowrates to the Steam Generators (SGs).

Description: The AFW system is a safety related, seismic category I system that is credited to provide the required cooling water from the CST to each of the three Steam Generators (SGs). The AFW system design provides for redundancy by assuring that either a single TDAFW pump or two MDAFW pumps can deliver the required flows. The licensee established acceptability of the design based, in part, on the results of Unit 1 calculation 40.02, "Verification of AFW Flow Bases", Rev. 4, Unit 2 calculation 38.04, "Verification of AFW Flow Bases", Rev. 4, and calculation 11.13, "Available NPSH for Auxiliary Feedwater Pumps", Rev. 1. The AFW flow bases calculations identified a main feed water line break (MFLB) accident with the failure of the TDAFW pump as the most limiting case in terms of design flow margin to the other two steam generators. Calculation 40.02 identifies a limiting flow margin of 2.21 gallons per minute (gpm) per SG. Calculation 38.4 identifies a limiting flow margin of 0.82 gpm per SG. These margins are the flowrates available above the accident analysis required flowrate of 150 gpm.

The team reviewed the applicable design basis analyses for the MDAFW pumps and noted that calculations 40.02 and 38.04 used a non-conservative assumption in modeling friction losses of the AFW flow orifices (FO) 2861 A, B, C and FO 2862 A, B, C. The orifice resistance modeling did not take into account variation of the orifice and/or piping inside diameter (ID). Additionally, the analysis did not consider the effects of the change in elevation and resistance of piping internal to the steam generators and did not use the fluid temperature that would maximize friction losses. Based on the team's observations, the licensee entered this issue into the CAP as CR 352210, CR 353743, CR 355898, CR 363850, and CR 369676 and performed an operability determination to address the impact of the non-conservative assumptions on AFW design analyses. DOEJ-FRSNC326893-M005, Documentation of Engineering Judgment, "Hydraulic Evaluation of FNP Auxiliary Feedwater System to Support Operability Determinations," Ver. 1.0, determined that the orifice resistance is approximately 70% of the total AFW system resistance. Therefore, modeling of orifice resistance is critical to the validity of the AFW analysis. The licensee's evaluation demonstrated that once the model was corrected to account for these non-conservative assumptions, the predicted flow to the non-affected SGs for the MFLB case would have been below the required design basis flowrate of 150 gpm (assuming 5% degraded pump performance). The operability determination concluded that the AFW system remained operable because actual pump performance was not degraded, and therefore, would yield flows in excess of the required 150 gpm.

Analysis: The failure to conservatively model AFW system friction losses when implementing design control measures to verify the capability of the AFW system to deliver the flowrates required by accident analyses was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the mitigating systems cornerstone attribute of design control and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the finding challenged the assurance that the AFW system would be capable of delivering the required flow during worst case accident conditions due to non-conservative modeling of system friction losses. In accordance with NRC IMC 0609.04, "Initial Screening and Characterization of Findings," the team used the mitigating systems column to perform a Phase 1 SDP screening and determined the finding to be of very low safety significance (Green) because it was not a design issue resulting in loss of function, did not represent an actual loss of a system safety function, did not result in exceeding a TS allowed outage time, and did not affect external event mitigation. A cross-cutting aspect was not identified because the design basis calculation associated with the finding was approved on March 25, 1999, and did not represent current licensee performance.

Enforcement: 10 CFR 50, Appendix B, Criterion III, "Design Control," states, in part, that measures shall provide for verifying or checking the adequacy of design. Contrary to the above, since March 25, 1999, the licensee had failed to implement design control measures to verify the adequacy of design inputs, assumptions, or limiting plant conditions which were relied upon in design basis calculations used to demonstrate the adequacy of AFW design to meet flowrates required by the accident analysis. Because the violation was of very low safety significance and it was entered into the licensee's CAP, this violation is being treated as an NCV consistent with the NRC Enforcement Policy: NCV 05000348, 364/2011010-02, "Failure to Implement Design Control Measures to Verify the Adequacy of AFW Design."

b. .3 Findings

Introduction: A Green, NRC identified NCV of TS 5.4, "Procedures," was identified for the licensee's failure to provide adequate procedural guidance for controlling SG and pressurizer level during loss of instrument air events and CVCS malfunctions. Specifically, the licensee failed to evaluate the capability of MOVs to be cycled as directed by AOPs.

Description: Pressurizer level and SG levels are normally controlled by air-operated flow control valves (FCVs). During malfunctions, such as a loss of instrument air, the FCVs fail in the full open position. To prevent overfill of the pressurizer and SGs with a failed open FCV, the operators are directed by procedure to cycle (close – open – close) normally open motor operated valves (MOV) in the respective flow paths to control levels. The licensee uses procedure FNP-1-AOP-6.0, "Loss of Instrument Air," to mitigate a loss of instrument air event. Step 7 of the procedure directs operators to cycle MOV 8107 or 8108 to maintain pressurizer level between 20 to 50 percent. Step 8 of the procedure directs operators to cycle MOVs 3764A – F to maintain SG narrow range levels between 35 to 69 percent. The licensee uses procedure FNP-1-AOP-16.0, "Chemical and Volume Control System (CVCS) Malfunction," to mitigate malfunctions of the charging and letdown portions of the CVCS. Step 22 of the procedure directs operators to cycle MOV 8107 or 8108 to maintain pressurizer level between 20 to 60 percent.

The team determined that the procedural guidance for controlling pressurizer and SG levels was inadequate because the licensee failed to evaluate the capability of the MOVs to be cycled as directed in the procedures. A MOV failure could occur as a result of the cycling due to the tripping of thermal overload devices or overheating of other electrical components. The failure of the MOVs would result in inadequate level control until local manual control was established by operators. During the inspection, the team verified the feasibility of the local manual actions. The licensee entered these issues into their CAP as CRs 355230, 355672 and 355695; performed DOEJ –FRSNC326893-E001, "Evaluate Cycling of Q1E21MOV8107, Q1E21MOV8107, and Q1E21MOV3764A through F"; and implemented a standing order (S-2011-12) that restricted the cycling of the MOVs until the procedures were revised. This issue is also applicable to Unit 2.

Analysis: The failure to provide adequate procedural guidance for controlling SG and pressurizer level during loss of air events and CVCS malfunctions was a performance deficiency. The performance deficiency was more that minor because it was associated with the mitigating systems cornerstone attribute of equipment performance and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee directed the cycling of MOVs in AOPs without performing evaluations to provide assurance that the components would not fail as a result of the cycling operations and lead to a condition of inadequate SG and pressurizer level control. In accordance with NRC IMC 0609.04, "Initial Screening and Characterization of Findings," the team used the mitigating systems column to perform a Phase 1 SDP screening and determined the finding to be of very low safety significance (Green) because it was not a design issue resulting in loss of function, did not represent an actual loss of a system safety function, did not result in exceeding a TS allowed outage time, and did not affect external event mitigation. A cross-cutting aspect was not identified because the finding did not represent current performance.

Enforcement: Technical Specification 5.4, Procedures, states, in part, that written procedures shall be established, implemented, and maintained covering the activities recommended in Appendix A of Regulatory Guide 1.33, Rev. 2, February 1978. The Regulatory Guide states in part that safety-related activities, such as combating emergencies and other significant events (i.e. loss of instrument air) should be covered by written procedures.

Contrary the above, since 1987, the licensee failed to maintain an adequate written procedure for combating emergencies and other significant events. Specifically, the licensee failed to provide adequate procedural guidance in procedures AOP-6.0, "Loss of Instrument Air" and AOP-16.0, "CVCS Malfunction", for controlling SG and pressurizer level during design bases events. Because the violation was of very low safety significance and it was entered into the licensee's CAP, this violation is being treated as an NCV consistent with the NRC Enforcement Policy: NCV 05000348, 364/2011010-03, "Failure to Provide Adequate Procedural Guidance for Controlling Steam Generator and Pressurizer Level During Loss of Air Events."

b. .4 Findings

Introduction: The team identified an unresolved item (URI) regarding the licensee's evaluation of the minimum required submergence for the AFW pumps given the potential for vortex formation in the CST.

Description: The CST is a safety related, seismic category I tank that holds up to 500,000 gallons of water and is required by the TS 3.7.6 limiting condition for operation to be maintained at a minimum of 150,000 gallons for use by the AFW system under normal operation and in response to accident conditions. In order to ensure this requirement, the lower 13' 3-1/8" of the 46' inside diameter (ID) tank is designed to withstand the effects of tornado missiles. The CST has two 8" AFW suction pipes – one for the TDAFW pump and one for both MDAFW pumps. Both suction pipes open at 4" from the tanks' bottom facing down. The suction piping centers are approximately 1' 3" apart. The CST has an internal bladder that prevents introduction of air under normal operating conditions.

The team reviewed calculations BM-95-0961-001, "Verification of CST Sizing Basis", Rev. 4, and CBI-72-4859, "Condensate Storage Tank", Rev. 0 and made the following observations regarding the design basis of the CST:

The CST tornado missile-protected height of 13' 3-1/8" is based on the elevation of the 24" condenser hotwell make-up line. The hotwell make-up line is not designed to withstand a design basis seismic event or damage from tornado missiles. The team computed the maximum protected volume (including the unusable lower 4" of the tank) to be approximately 164,841 gallons. The team noted that this volume did not take into account any CST fabrication tolerances. Calculation BM-95-0961-001 established that there was a margin of 4,300 gallons with respect to the TS 3.7.6 requirements for the CST. Although this calculation addressed the losses of CST inventory due to the line break, it did not analyze that this line break would create an air introduction path under the CST bladder, allowing a vortex to form, and adversely affect the usable volume of water in the CST. Additionally, the team noted that calculation BM-95-0961-001 did not evaluate the effects of tornado missile damage to the un-protected portion of the CST. Tornado missile damage to the tank could also create an air introduction path under the CST bladder which would allow a vortex to form.

Based on the team's observations, the licensee entered the issue into the CAP as CRs 351170, 353599 and 355457 and performed a prompt determination of operability (PDO) 0-11-06, "Prompt Determination of Operability," Rev. 2 which concluded that vortex formation could lead to an additional loss of required CST level of 5.8" or 6,021 gallons. Although the additional water required to account for CST vortexing exceeded the TS minimum required value, the licensee concluded that sufficient water remained available below the condenser hotwell make-up elevation for the CST to be able to perform its safety function. Additionally, the licensee implemented administrative measures to ensure that CST level was maintained above the level determined to be required by the licensee's evaluation. The PDO conclusions are supported, in part, by calculation SM-SNC335993-001, "CST AFW Pump Suction – Submergence Analysis", Ver. 1.0. This calculation utilizes a methodology based on Akalank K. Jain, "Air Entrainment in Radial Flow towards Intakes", ASCE Journal of Hydraulic Division, September 1978, to determine the minimum submergence water level in the tank to prevent vortexing.

Summary: The team determined that additional inspection and consultation with a vortexing subject matter expert at NRC headquarters would be warranted to evaluate the licensee's application of the methodology used for determining minimum AFW pump submergence. Additionally, the team concluded that additional evaluation of minimum required AFW pump submergence would be necessary to determine if this issue resulted in a more than minor performance deficiency. (URI 05000348, 364/2011010-04, "Evaluation of CST Vortex Effect on AFW Pump Minimum Submergence")

b. .5 Findings

Introduction: The team identified an URI regarding the use of non-conservative assumptions in design bases analyses used to demonstrate adequate available AFW pump net positive suction head (NPSH) and subsequent analysis of the impact of AFW system operation during a loss of instrument air event on available NPSH.

Description: The AFW system is a safety-related, seismic category I system that is credited to provide the required cooling water from the CST to each of the three Steam Generators (SG). The AFW system design provides for motive force redundancy by assuring that either a single TDAFW pump, or 2 MDAFW pumps can deliver the required flows. The licensee established acceptability of the design based, in part, on results of Unit 1 calculation 40.02, "Verification of AFW Flow Bases", Rev. 4, Unit 2 calculation 38.04, "Verification of AFW Flow Bases", Rev. 4, and calculation 11.13, "Available NPSH for Auxiliary Feedwater Pumps", Rev. 1. The AFW flow bases calculations identified a main feed water line break (MFLB) accident with the failure of the TDAFW pump as the most limiting case in terms of design flow margin to the other two steam generators. Calculation 40.02 identifies a flow margin of 2.21 gpm per SG for SG A and B, and calculation 38.4 identifies a flow margin of 0.82 gpm per SG for SG A and C vs. the required flow of 150 gpm. Additionally, NPSH calculation 11.13 established that for a bounding case (2 MDAFW pumps operating) the margin between available NPSH ($NPSH_A$) and required NPSH ($NPSH_R$) was less than 1 foot.

The team reviewed calculations 40.02, 38.04 and 11.13 and identified examples where non-conservative design inputs, assumptions, or limiting plant conditions were relied upon in design basis analyses used to demonstrate that the MDAFW pumps would have a sufficient capacity and head to perform their safety function. The following specific examples were identified:

- Calculation 11.13 AFW flowrates were based on the flow rates developed in calculation 40.02 for the MSLB case with all three AFW pumps operating. However, for the $NPSH_A$ calculation, a conservative assumption would have been a failure of TDAFW pump, since it would maximize the flow through the remaining MDAFW pumps.

- $NPSH_A$ value was based on a CST temperature of 100°F and not 110°F as specified in FSAR Section 9.2.6.3 as the maximum CST temperature. In response to this observation, Farley performed an evaluation that established that use of 110 °F temperature resulted in decrease of $NPSH_A$ by 0.75 feet.

Based on the team observations, the licensee entered this issue into the CAP as CR 355025 and CR 352168 and performed an operability determination to address the impact of the non-conservative assumptions on AFW safety function. The licensee's evaluation documented in IDO 355898 and DOEJ-FRSNC326893-M005 concluded that reasonable assurance existed that AFW safety function would not be adversely impacted by crediting operator actions to manually isolate the faulted SG after 30 minutes of operation (for MFLB) and by crediting the remaining CST level to add an approximately 9 foot increase in the $NPSH_A$ value in comparison to the one used in the calculation 11.13. This CST level increase leads to the corresponding $NPSH_A$ greater than the $NPSH_R$.

The team's review of IDO 355898 and DOEJ-FRSNC326893-M005 identified the following concerns and observations:

- DOEJ-FRSNC326893-M005 did not address $NPSH_A$ vs. $NPSH_R$ conditions at the lower CST levels. The team noted that abnormal operating procedures used to combat a loss of non-safety related instrument air allow cycling of AFW header MOVs in lieu of local-manual throttling of the AFW flow control valves. Since the pump flow rates will remain virtually the same every time the operators will open AFW isolation MOVs, the $NPSH_A$ advantage credited in the DOEJ-FRSNC326893-M005 will be decreasing until it will become negative prior to the CST becoming empty (since the 9 foot of added margin is less than the height of the 13 feet of protected volume).

- The licensee's analyses did not address the potential for long-term AFW flow restricting orifice erosion that could lead to increased AFW flowrates and decreased NPSHA. Because the actual performance of the AFW flow restricting orifices is not periodically compared to the performance assumed in the design basis analyses, the team did not have sufficient information to conclude that the orifice bores would not erode undetected resulting in degraded performance.

<u>Summary</u>: The team determined that additional information and/or evaluation by the licensee were required to determine if operation of the AFW system during a loss of instrument air event (as described above) was consistent with system design basis assumptions and operability determinations. Additionally, the team determined that additional information from the licensee regarding the condition or performance of the AFW flow restricting orifices would be necessary to establish that current component

performance remains consistent with design basis analyses and operability determinations. Additionally, the team concluded that these additional licensee evaluations would be necessary to determine if this issue resulted in a more than minor performance deficiency. (URI 05000348, 364/2011010-05, "Non-Conservative Assumptions Regarding AFW Net Positive Suction Head")

.2.4 Residual Heat Removal (RHR) Pump Seal Coolers - Q1E11P001

a. Inspection Scope

The team reviewed the plant's TS, UFSAR, Functional System Description (FSD), and piping and instrumentation diagrams (P&IDs) to establish an overall understanding of the design bases of the seal coolers. Design calculations were reviewed to verify that design basis heat removal requirements, capability, and flow rates had been appropriately translated into these documents. Component walkdowns were conducted to verify that the installed configurations would support their design basis function under accident conditions and had been maintained to be consistent with design assumptions. Control panel indicators were observed and operating procedures reviewed to verify that component operation and alignments were consistent with design and licensing basis assumptions. Vendor documentation, system health reports, preventive and corrective maintenance history, and corrective action system documents were reviewed in order to verify that potential degradation was monitored or prevented and the component replacement was consistent with inservice/equipment qualification life. Maintenance Rule (MR) information was reviewed to verify that the component was properly scoped, and that appropriate preventive maintenance was being performed to justify current MR status.

b. Findings

Introduction: A green NRC identified NCV of 10 CFR 50, Appendix B, Criterion III, "Design Control," was identified involving two examples. In the first example, the licensee failed to translate the minimum CCW flow for the RHR seal coolers into ARPs. In the second example, the licensee failed to translate the MDAFW and TDAFW pump minimum flow requirements into applicable ARPs.

Description: The team identified two examples of the licensee not translating the design of components into procedures required by Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

RHR Seal Cooler Design Flow Requirements – The RHR seal coolers utilize CCW flow to maintain the RHR pump seals below their design temperature limits. Low flow alarms are provided for the CCW flow on the return lines of the seal coolers. These alarms are received in the control room.

Calculation CN-96-0047, "Component Cooling Water System Evaluation – Power Uprate and Replacement Steam Generator," Rev. 8 establishes that the minimum CCW flow to maintain the RHR Pump Seal Cooler process fluid outlet temperature below its maximum temperature of 180ºF is 3.5 gpm. The team noted that the licensee established low CCW flow alarm setpoints of 3 gpm (+1/ -0 gpm). These setpoints were then translated into the following Unit 1 and Unit 2 procedures: FNP-1(2)-ARP-1.3, "Main

Control Board Annunciator Panel C", Versions 28.1 and 22. The team determined that the setpoints in these procedures were non-conservative (low) when compared to calculated minimum design requirements and if left uncorrected could result in (1) inadequate CCW flow to the RHR seal coolers without the operators receiving the alarm in the control room, and (2) the operators subsequently failing to start the standby train of RHR and secure the in-service train of RHR. The licensee entered this issue into their CAP as CR 348613.

<u>AFW Pump Minimum Flow Requirements</u> – Unit 1 and Unit 2 procedures FNP-1(2)-ARP-1.9, Version 47 are the alarm response procedures for responding to MDAFW and TDAFW pump low suction flow conditions. The MDAFW pump alarm setpoint is 40 gpm (+/- 4.5 gpm) and the TDAFW pump alarm setpoint is 80 gpm (+/- 4.5 gpm).

In 2005, the licensee revised the alarm response procedures. The revision added a note that states that the minimum flow requirements for the MDAFW pump was 50 gpm and the TDAFW pumps was 100 gpm. The minimum flow requirements were established by the manufacturer of the pumps. The team noted the associated alarm setpoints were non-conservative because the MDAFW pump alarm setpoint is 40 gpm (+/- 4.5 gpm) which is less than the minimum required value of 50 gpm; and the TDAFW pump alarm setpoint is 80 gpm (+/- 4.5 gpm) which is less than the minimum required value of 100 gpm. The licensee entered this issue into their CAP as CR 352485.

<u>Analysis:</u> The failure to correctly translate the applicable design bases information for the RHR pump seal coolers and the AFW pumps into procedures was a performance deficiency. The finding was determined to be more than minor because it was associated with the procedure quality attribute of the mitigating system cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to translate the appropriate minimum flow requirements into ARPs adversely affected the quality of procedures used to respond to alarm conditions that are required by Regulatory Guide 1.33, "Quality Assurance Program Requirements." The inadequate procedures adversely affected the ability of operators to assess operability and to combat deficiencies associated with risk significant equipment. In accordance with NRC IMC 0609.04, "Initial Screening and Characterization of Findings," the team used the mitigating systems column to perform a Phase 1 SDP screening and determined the finding to be of very low safety significance (Green) because it was not a design issue resulting in loss of function, did not represent an actual loss of a system safety function, did not result in exceeding a TS allowed outage time, and did not affect external event mitigation. A cross-cutting aspect was not identified because the finding did not represent current performance.

<u>Enforcement:</u> 10 CFR 50, Appendix B, Criterion III, "Design Control," requires, in part, that "Measures shall be established to assure that the design basis is correctly translated into procedures." Contrary to the above, the licensee failed to correctly translate the applicable design bases information for the RHR pump seal coolers and AFW pumps into procedures. Specifically, since 2005 the licensee failed to translate the minimum CCW flow requirements for the RHR pump seal coolers and the minimum flow requirements for the AFW pumps into ARPs. Because the violation was of very low safety significance and it was entered into the licensee's CAP, this violation is being treated as an NCV consistent with the NRC Enforcement Policy: NCV 05000348, 364/2011010-06, "Failure to Correctly Translate the Design Basis into Procedures for

Minimum CCW Flow to the RHR Seal Coolers and Minimum Flow Requirements for the AFW Pumps."

.2.5 RHR Inlet Isolation Motor Operated Valves - Q1E11MOV8701(8702)

a. Inspection Scope

The team reviewed the plant TS, UFSAR, FSD, and P&IDs to establish an overall understanding of the design bases of the valves. Design calculations (i.e., differential pressure and required torque/thrust) were reviewed to verify that the design basis and design assumptions had been appropriately translated into these documents. The team reviewed calculations for degraded voltage at the MOV terminals to ensure worst-case voltage was used in calculating available motor output torque when determining margin. The team reviewed calculations that establish control circuit voltage drop and thermal overload sizing and testing to verify the capability of the valve to operate during design bases events. Component walkdowns were conducted to verify that the installed configurations would support their design basis function under accident conditions and had been maintained to be consistent with design assumptions. Control panel indicators were observed and operating procedures reviewed to verify that component operation and alignments were consistent with design and licensing basis assumptions. Test procedures and recent test results were reviewed against design basis documents to verify that acceptance criteria for tested parameters were supported by calculations or other engineering documents and that individual tests and analyses served to validate component operation under accident conditions. Vendor documentation, system health reports, preventive and corrective maintenance history, and corrective action system documents were reviewed in order to verify that potential degradation was monitored or prevented and the component replacement was consistent with inservice/equipment qualification life. Maintenance Rule (MR) information was reviewed to verify that the component was properly scoped, and that appropriate preventive maintenance was being performed to justify current MR status.

b. Findings

No findings were identified.

.2.6 2C Station Blackout (SBO) Diesel Generator (DG) - QSR43A504

a. Inspection Scope

The team reviewed the plant TS, UFSAR, FSD, and P&IDs to establish an overall understanding of the design bases of the air start system, fuel oil storage tank, and ventilation system. Design calculations and site procedures were reviewed to verify the design bases and design assumptions had been appropriately translated into these documents. The team reviewed system modifications over the life of the component to verify that the subject modifications did not degrade the component's performance capability and were appropriately incorporated into relevant drawings and procedures. Component walkdowns were conducted to verify that the installed configurations would support their design basis function under accident/event conditions and had been maintained to be consistent with design assumptions. Control panel indicators were observed and operating procedures reviewed to verify that component operation and alignments were consistent with design and licensing basis assumptions. Test procedures and results were reviewed against design basis documents to verify that acceptance criteria for tested parameters were supported by calculations or other engineering documents and that individual tests and/or analyses served to validate component operation under accident/event conditions. Vendor documentation, system health reports, preventive and corrective maintenance history, and corrective action system documents were reviewed in order to verify that potential degradation was monitored or prevented and the component replacement was consistent with inservice/equipment qualification life. Maintenance Rule (MR) information was reviewed to verify that the component was properly scoped, and that appropriate preventive maintenance was being performed to justify current MR status.

b. Findings

Introduction: A Green NRC identified NCV of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," was identified for the failure to perform condition monitoring or otherwise implement an effective preventive maintenance (PM) program for the 2C DG room A and B exhaust fan louvers.

Description: The 2C DG room ventilation system consists of three roof fans for exhausting heat during operation and shutdown of the SBO diesel generator. The A and B fans exhaust heat from the room during the operating cycle, and the C fan exhaust heat from the room during the shutdown cycle. One of the fans (A or B) and half of the intake air wall louvers are capable of maintaining the room temperature below shutdown (104 °F) and operating design room temperature (122 °F) limits. The C fan starts when the temperature inside the room reaches 75 ºF, the A fan starts when the temperature inside the room reaches 80 ºF, and the B fan starts when the temperature inside the room reaches 105 °F.

During a walkdown of the 2C DG room, the team noted that the A and C fan were running, the B fan was rotating backwards, and the louvers for the B fan were stuck partially open. The licensee followed up on the team's observation and noted that the louvers for the A fan were also stuck partially open when it was not running. This configuration would allow the running fan to short-cycle air through the open louver. Additionally, the non-running fan that was rotating backwards may not auto-start due to the tripping of thermal overloads. The licensee performed an operability/functionality assessment, and determined that with the A and C fan running, and the B fan spinning backwards, the 2C DG remained capable of performing its blackout function with reduced ventilation margin. The licensee's compensatory measures placed the control switch for the A and B fans in manual until repairs are performed. The licensee entered this issue into their CAP as CRs 351580, CR 349883, and CR 355130.

The team reviewed the preventive maintenance and corrective action history for the 2C DG room exhaust louvers and noted that the last annual PM performed on the exhaust louvers was completed in 2010. The team also noted that the work instructions for the annual PM included steps to clean and lubricate the exhaust fan louvers, but did not include steps to inspect the condition of the louvers for indications of degradation or otherwise assess their functional capability. The PM completed in 2010 did not identify evidence of louver degradation. However, the team noted that previous annual inspections of the same louvers resulted in CRs written in 2008 and 2009 to address deficient conditions, but no repairs had been made to correct those issues. Based on these observations, and the degraded condition of the louvers, the team concluded that the PM program for the 2C DG exhaust fan louvers had been ineffective in providing assurance that the components would remain capable of performing their intended function.

Analysis: The failure to perform condition monitoring or otherwise implement an appropriate preventive maintenance program for the 2C DG A and B exhaust fan louvers was a performance deficiency. This performance deficiency was more than minor because it was associated with equipment performance attribute of the mitigating systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to perform condition monitoring or otherwise implement an appropriate preventive maintenance program for the 2C DG A and B room exhaust fan louvers challenged the assurance that these components would remain capable of performing their intended functions. In accordance with NRC IMC 0609.04, “Initial Screening and Characterization of Findings,” the team used the mitigating systems column to perform a Phase 1 SDP screening and determined the finding to be of very low safety significance (Green) because it was not a design issue resulting in loss of function, did not represent an actual loss of a system safety function, did not result in exceeding a TS allowed outage time, and did not affect external event mitigation. Because the licensee had initiated CRs in 2008 and 2009 for the 2C DG room exhaust louvers, and repairs were not made in a timely manner to address the issue, this finding was assigned a cross-cutting aspect in the corrective action program component of the problem identification and resolution area [P.1(d)].

Enforcement: 10 CFR 50.65(a)(1) states, in part, that licensee’s shall monitor the performance or condition of structures, systems and components (SSCs) within the scope of the rule as defined by 10 CFR 50.65(b), against license established goals, in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended function.

10 CFR 50.65(a)(2) states, in part, that monitoring as specified in 10 CFR 50.65(a)(1) is not required where it has been demonstrated that the performance or condition of an SSC is being effectively controlled through the performance of appropriate preventive maintenance, such that the SSC remains capable of performing its intended function.

Contrary to the above, since September 2010, the licensee had failed to demonstrate that the performance or condition of the 2C DG A and B exhaust fan louvers had been effectively controlled through the performance of appropriate preventive maintenance and did not otherwise monitor performance against licensee established goals. Because the violation was of very low safety significance and it was entered into the licensee’s CAP, this violation is being treated as an NCV consistent with the NRC Enforcement Policy: NCV 05000348, 364/2011010-07, “Failure to Monitor or Perform Effective Preventive Maintenance on the 2C EDG Exhaust Fan Louvers.”

.2.7 Turbine Driven Auxiliary Feedwater (TDAFW) Pump - Q2N23P0002

a. Inspection Scope

The team reviewed the plant’s TS, UFSAR, FSD, and P&IDs to establish an overall understanding of the design bases of the TDAFW. Design calculations and test data were reviewed to verify that design basis capability, and flow rates had been appropriately translated into these documents. The team concentrated its efforts on the pump's capability of performing its safety function (i.e., delivering the required flow rate to the steam generators at the prescribed design pressure). Records of surveillance testing and maintenance activities, and applicable corrective actions were examined to verify that potential degradation or low margin design issues were being monitored, prevented and/or corrected. TDAFW walkdowns were conducted to verify that the installed configurations would support its design basis function under accident conditions and had been maintained to be consistent with design assumptions and to visually inspect the material condition of the pumps. Control panel indicators were observed and operating procedures reviewed to verify that TDAFW operation and alignments were consistent with design and licensing basis assumptions. Vendor documentation, system health reports, preventive and corrective maintenance history, and corrective action system documents were reviewed in order to verify that potential degradation was monitored or prevented and the component replacement was consistent with inservice/equipment qualification life. The team reviewed the licensee's severe weather procedure to verify that adequate guidance exists for operators to isolate the AFW minimum flow recirculation line during a postulated severance of the line as described in design bases calculations. The team performed a walkdown of local manual actions associated with isolation of the AFW minimum flow recirculation to verify the feasibility of the directed actions.

b. .1 Findings

Introduction: A green NRC identified NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was identified for the licensee's failure to establish an adequate test procedure used to demonstrate that the TDAFW pump discharge check valves were capable of performing their design basis function. The test procedure was inadequate in providing assurance that the AFW system was capable of providing the required design basis flowrates to the SGs with reverse flow into an idle TDAFW pump via the discharge check valves.

Description: Unit 1 and Unit 2 procedures, FNP-1/2-STP-22.30, "Auxiliary Feedwater Pump Discharge Check Valve Reverse Flow Closure Operability Test," Version 6.1 and 5, were used by the licensee to demonstrate the closure of the TDAFW pump discharge check valves Q1/2N23V002D, Q1/2N23V002F, Q1/2N23V002H, and Q1/2N23V003. The function of the check valves is to prevent reverse flow into the TDAFW pump when the pump is idle. The test acceptance criteria for check valve back leakage were less than 5 gallons per minute (gpm). Excessive back leakage through the pump discharge check valves would lower AFW system flowrate to the SGs below the flows required in the safety analysis for design basis events. The team identified the following deficiencies in the test procedure:

The team noted that the Unit 1 calculation 40.02, "Verification of AFW Flow Bases," Rev. 4, and the Unit 2 calculation 38.04, Verification Of AFW Flow Bases, Rev. 4 identified a main feed water line break accident with the failure of the TDAFW pump as the most limiting case in terms of design flow margin to the other two SGs. Calculation 40.02 identified a flow margin of 2.21 gpm per SG for SG A and B, and calculation 38.4 identified a flow margin of 0.82 gpm per SG for SG A and C. Based on this information, the team concluded that the total acceptable leakage through the TDAFW pump discharge check valves was 4.21 gpm for Unit 1 and 1.64 gpm for Unit 2.

The team also noted that the check valves are tested at demineralized water system pressure which is approximately 80 to 126 psig. The actual system operating pressure these check valves would see is approximately 1130 psia. The higher operational differential pressure across the check valves could reasonably result in back leakage of more than 5 gpm at the higher design pressures. The team concluded that the procedure for the TDAFW discharge check valves was inadequate in that the 5 gpm acceptance criteria was non conservative with respect to the safety analysis margins available and the acceptance criteria was non conservative with respect to system pressure. This issue was entered into the licensee's CAP as CR 348795.

The team reviewed the most recently completed test results for the TDAFW pump discharge check valves which indicated no evidence of back leakage (0 gpm observed). The team concluded that, at the time of the inspection, the function of the check valves was not degraded.

Analysis: The failure to develop an adequate test procedure which demonstrated that TDAFW pump discharge check valves were capable of performing their design basis function was a performance deficiency. This performance deficiency was more than minor because it adversely affected the equipment performance attribute of the mitigating systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the acceptance criteria used in the test procedure was non-conservative when compared to the flowrates required by the accident analyses, and the test procedure was performed at lower system pressures (which were not representative of actual design conditions) without modifying the acceptance criteria. In accordance with NRC IMC 0609.04, "Initial Screening and Characterization of Findings," the team used the mitigating systems column to perform a Phase 1 SDP screening and determined the finding to be of very low safety significance (Green) because it was not a design issue resulting in loss of function, did not represent an actual loss of a system safety function, did not result in exceeding a TS allowed outage time, and did not affect external event mitigation. Because the test procedure did not contain complete, accurate, and up-to-date information consistent with the system design basis safety analysis, this finding is assigned a cross-cutting aspect in the resources component of the human performance area [H.2(c)].

Enforcement: 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that "activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances" and "Procedures shall include appropriate quantitative or qualitative acceptance criteria." Contrary to the above, since April 2010, the licensee (1) failed to establish a procedure for testing the leakage of the TDAFW discharge check valves that was appropriate to the circumstances, and (2) failed to use appropriate quantitative or qualitative acceptance criteria. Specifically, the procedure for testing the TDAFW discharge check valves was inadequate in that the acceptance criteria was non conservative with respect to the safety analysis margins available for flow and the acceptance criteria was non conservative with respect to system pressure. Because the violation was of very low safety significance and it was entered into the licensee's CAP, this violation is being treated as an NCV consistent with the NRC Enforcement Policy: NCV 05000348, 364/2011010-08, "Failure to Develop an Adequate Procedure to Test the TDAFW Pump Discharge Check Valves."

b. .2 Findings

Introduction: The team identified an URI regarding the licensee's identification and evaluation of corrective actions taken to address AFW pump suction check valve oscillations.

Description: In 2005 the licensee retained services of Kalsi Engineering to perform an analysis of the effects of partially open TDAFW pump suction check valves (2446, Kalsi Engineering, "Auxiliary Feedwater Check Valve Analysis for Farley Nuclear Plant", Rev. 0, January 24, 2006). The engineering analysis determined that for 100 gpm of flow the check valve opening was 0.7 degrees with a disc peak-to-peak oscillating angle of 1.89 degrees. For 400 gpm of flow the check valve opening was 10.4 degrees with a disc peak-to-peak oscillating angle of 7.58 degrees. The analysis concluded that for each flow condition oscillation would not result in significant hinge pin wear. The scope of this review was limited to the TDAFW pump suction check valves only and the MDAFW pump suction check valves were not included in the analysis. The team noted that the 6" MDAFW pump suction check valves are the same model as the 8" TDAFW pump suction check valves. The team's review of the operating conditions for the AFW pump suction check valves identified the following observations:

- The quarterly inservice testing of the MDAFW pumps is performed at or near minimum flow conditions. The flow was not measured during the test; only the pump's differential pressure was monitored. The AFW system functional description indicates that this flow was approximately 50 gpm. The team determined that at this MDAFW pump flowrate, the check valve flow velocity was approximately 0.56 ft/sec, which was less than the TDAFW approximate velocity of 0.64 ft/sec at 100 gpm. However, since MDAFW pump flow was not monitored during testing, any pump flow instability due to suction check valve oscillations may not be revealed during the current method of testing.
- Degradation (partial sticking) of the MDAFW pump suction check valves may not be identified by routine testing. If these valves are partially stuck open, then functional testing by either the quarterly surveillances or the comprehensive surveillance test will not reveal this condition, since higher-than-design-basis CST levels would 'mask' the valve being partially stuck open.
- On October 2, 2008, the licensee initiated CR 2008110018, which identified that while the TDAFW pump was running in a minimum flow alignment, the pump exhibited flow fluctuations at the flow rates below approximately 230 gpm. The licensee attributed this condition to a partially open TDAFW suction check valve (Q2N23V0006). During the subsequent troubleshooting, the licensee disassembled this check valve on October 28, 2008, and verified that the check valve was functioning properly without any abnormal wear indications. Proposed corrective actions considered replacement of the swing check valves with the in-line check valves. However, because of the reliability concerns and difficulty in performing maintenance inspections on that type of check valve, this modification was not pursued. Additionally, the team noted that the MDAFW pump suction check valves were not inspected.

Based on a review of the operating and corrective action history related to AFW check valve oscillations for this issue over the last 5 years, the team concluded that there was not sufficient evidence to conclude that the MDAFW pump suction check valves would not be subject to the same oscillation issue that was observed on the TDAFW pump. Additionally, the team noted that the check valve oscillating condition that had been previously evaluated for the TDAFW pumps, had not been evaluated for MDAFW pumps.

Summary: The team determined that additional information and/or evaluation from the licensee regarding the current condition of the MDAFW pump suction check valves would be required to confirm that the check valves were not adversely impacting the AFW system design basis function. Additionally, the team concluded that this additional evaluation of the current condition of MDAFW pump suction check valves would be required to determine if this issue resulted in a more than minor performance deficiency. (URI 05000348, 364/2011010-09, “Evaluation of MDAFW Pump Suction Check Valves”)

.2.8 MDAFW Pump (Electrical) - Q2N23M0001A

a. Inspection Scope

The team reviewed the plant’s FSAR, TS and FSD to establish an overall understanding of the design bases of the controls for the MDAFW pumps. Electrical drawings and site procedures were reviewed to verify that the design bases and design assumptions had been appropriately translated into these documents. Vendor documentation, system health reports, and problem history were reviewed in order to verify that the MDAFW pump controls were being properly maintained. The team reviewed the licensee’s alarm response procedures associated with AFW flow alarms to verify that setpoints were consistent with design bases documents. The team reviewed maintenance documentation to verify that the components were calibrated.

b. Findings

A violation was identified regarding AFW pump minimum flow alarm setpoints and is documented in section 1R21.2.4 of this report.

.2.9 Reactor Trip and Bypass Breakers - Q1C11E0004ART(BBY)

a. Inspection Scope

The team reviewed the plant’s FSAR, TS and FSD to establish an overall understanding of the design bases of the reactor trip circuit breakers operation and actuation. Electrical drawings and site procedures were reviewed to verify that the design bases and design assumptions had been appropriately translated into these documents. Test procedures and results of previous testing and refurbishment activities were reviewed against FSDs to verify that acceptance criteria for tested parameters were supported by the accident analyses. Vendor documentation, system health reports, and maintenance history were reviewed in order to verify that the trip circuit breakers were being properly maintained.

Since the reactor trip and bypass circuit breakers were the same model (Westinghouse Model DS-416) that had been the subject of an earlier NRC Information Notice, the team also reviewed the actions that had been taken by the licensee in response to Information Notice (IN) 1992-29, “Potential Breaker Mis-coordination Caused by Instantaneous Trip Circuitry.”

b. Findings

No findings were identified.

.2.10 'F' 600 Volt Load Center - Q1R16B0008-AB

a. Inspection Scope

The team reviewed the plant's FSAR, TS and FSD to establish an overall understanding of the design bases for the 1F 600 Volt Load Center. The team reviewed electrical drawings and site procedures to verify that the design bases and had been appropriately translated into these documents. The team reviewed the operation of the load center 1F key interlocks that were provided between the 4160 Volt supply breakers, the associated disconnects, and the 600 Volt feeder circuit breakers to verify that adequate guidance was provided to ensure correct alignment to one of the two 4160 Volt engineered safeguard busses 1F or 1G. The team also reviewed involved plant procedures to ensure that adequate guidance was provided for connecting power from load center 1F to one of the other 600 Volt load centers.

b. Findings

No findings were identified.

.2.11 1-2A Emergency Diesel Generator Voltage Regulator - QSR43A0501GENRG

a. Inspection Scope

The team reviewed the plant's FSAR, TS and FSD to establish an overall understanding of the design bases of the emergency diesel generator (EDG) voltage regulator system. Electrical drawings and site procedures were reviewed to verify that the design bases and design assumptions had been appropriately translated into these documents. The team reviewed system modifications over the life of the component to verify that the subject modifications did not degrade the component's performance capability and were appropriately incorporated into relevant drawings and procedures. Component walk downs of the 1-2A EDG were conducted to verify that the installed configurations would support their design bases function under accident/event conditions and had been maintained to be consistent with design assumptions. Control panel indicators were observed and operating procedures reviewed to verify that component operation and alignments were consistent with design and licensing bases assumptions. Test procedures and results were reviewed against FSDs to verify that acceptance criteria for tested parameters were supported by calculations or other engineering documents and that individual tests and/or analyses served to validate component operation under accident/event conditions. Vendor documentation, system health reports, preventive and corrective maintenance history, and corrective action system documents were reviewed in order to verify that potential degradation was monitored or prevented and that component replacement was consistent with equipment qualification life.

b. Findings

No findings were identified.

.2.12 Auxiliary Building Safety-Related Batteries - Q2R42E002A-A(B-B)

a. Inspection Scope

The team reviewed battery sizing and loading calculations to verify that loads do not exceed battery bank capacity. The team verified that the load profile bounded all accident scenarios. Also, the team reviewed short circuit calculations to verify that the duty cycle does not exceed the equipment protection ratings. The team reviewed performance tests to verify that the minimum voltage at the end of the test is the minimum voltage required by the most limiting component that has to actuate. In addition, a review of the service test was performed to verify that for the required current, the battery can provide the adequate voltage during an accident. The team reviewed equalizing procedures for the batteries to verify proper voltage. Selective one-line and schematic diagrams were reviewed to verify proper configuration of the 125 Volt Direct Current (VDC) electrical distribution system. The team performed a walkdown to verify material condition of the batteries and reviewed a sample of condition reports to confirm that the licensee adequately identifies, evaluates, and dispositions adverse conditions.

b. Findings

No findings were identified.

.2.13 Auxiliary Building Safety-Related Battery Chargers - Q2R42E0001A(B/C)

a. Inspection Scope

The team reviewed battery charger sizing calculations to verify that the chargers are capable of carrying the continuous load during a Design Basis Accident (DBA) and will charge the batteries to full capacity within required time. Also, the team reviewed the last two tests of the battery chargers to look for signs of degradation due to aging. A review of the ac voltage calculation was performed to assure satisfactory voltage to the chargers under worst-case conditions. In addition, the team verified that the ampere-hours returned to the battery were greater than the ampere hours removed plus the charging losses. The team performed a walkdown to verify material condition of the components and reviewed a sample of condition reports to confirm that the licensee adequately identifies, evaluates, and dispositions adverse conditions.

b. Findings

No findings were identified.

.2.14 SG Narrow Range (NR) Level Instrumentation - Q1C22LT0474-476(484-486/494-496)

a. Inspection Scope

The team reviewed instrument setpoint and uncertainty calculations, as well as calibration procedures and calibration test records to verify that the SG NR level instruments were in accordance with design bases documents. The last two completed calibration test records were reviewed to confirm that instrument setpoints were consistent with setpoint calculations. Also, the team reviewed a sample of condition reports to confirm that the licensee adequately identified and corrected adverse conditions. In addition, the team reviewed the maintenance history to verify actions were taken to correct and prevent problems.

b. Findings

No findings were identified.

.2.15 Reactor Coolant Pump Thermal Barrier Isolation Valves - Q2P17HV3184/3045

a. Inspection Scope

The team reviewed the isolation valves to verify their capability to perform the required design function. The review included the licensing and design basis of the valves, review of recent corrective actions, review of recent test procedures and test results, walkdowns of the valves and related instruments, and interviews conducted with responsible engineering personnel. The team reviewed the test procedures associated with the valves to verify the valves and instruments were being tested in accordance with the design bases. In addition, the team reviewed the maintenance history to verify actions were taken to correct and prevent problems. The team also conducted walkdowns of the valves and associated equipment to verify the material condition of the components.

b. Findings

No findings were identified.

.3 Operating Experience (4 Samples)

a. Inspection Scope

The team reviewed four operating experience issues for applicability at Farley Nuclear 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:

- Generic Letter 1996-05, "Periodic Verification of Design Basis Capability of Safety-Related Motor-Operated Valves"

- NRC Information Notice 1992-29, "Potential Breaker Miscoordination Caused by Instantaneous Trip Circuitry"

- Generic Letter 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients"

- Bulletin 88-04, "Potential Safety Related Pump Loss"

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA5 Other Activities

.1 Review of Degraded Voltage Protection Design and Licensing Bases

a. Inspection Scope

During the inspection period, the team reviewed the licensee's degraded voltage protection design and licensing bases. The team reviewed functional system descriptions, technical specifications, corrective action program documents, licensee self-assessments, and safety evaluation reports related to degraded voltage protection. The team evaluated the current degraded voltage design and licensing bases using the regulatory requirements specified in 10 CFR 50.55a(h)(2) and 10 CFR 50, Appendix A, General Design Criterion 17, Electric Power Systems. Additionally, the team evaluated the degraded voltage protection design using the staff positions provided in Standard Review Plan, NUREG-0800, (July 1981), and Branch Technical Positions (BTPs) of Appendix 8-A (PSB), containing BTP PSB-1, "Adequacy of Station Electric Distribution System Voltages."

b. Findings and Observations

Introduction: The team indentified an Unresolved Item (URI) regarding the licensee's use of administrative controls in lieu of automatic degraded voltage protection to assure adequate voltage to safety-related equipment during design basis events.

Description: The team noted that the degraded voltage protection system at Farley uses administrative controls to assure adequate voltage to safety-related equipment during design basis events. Farley's current system configuration, which relies on administrative actions, was recognized as a deviation from the guidance on degraded voltage protection provided in a NRC letter (dated June 2, 1977), but was accepted by the NRC in a Safety Evaluation Report (SER) (dated November 21, 1995). The licensee entered this issue into their corrective action program as CR 2011106624 on May 17, 2011.

This same issue is currently being assessed at plant Hatch, another Southern Company licensee, where the agency issued a backfit letter (Hatch Inspection Report 05000321/2011009 and 05000366/2011009, dated May 25, 2011). In the backfit letter, the staff concluded that the NRC was in error in accepting the use of administrative controls.

Summary: Because of the similarities of this issue for plants Farley and Hatch, this issue is unresolved pending completion of the appeal process that is afforded to Southern Company for plant Hatch. (URI 05000348, 364/2011010-10, “Administrative Controls in lieu of Automatic Actions for Degraded Voltage Protection”.

4OA6 Meetings, Including Exit

On September 30, 2011, the team discussed the status of the inspection with Mr. Tom Lynch and other members of the licensee’s staff. On December 8, 2011, the team presented the inspection results to Mr. Todd Youngblood and other members of the licensee’s staff. Proprietary information that was reviewed during the inspection was returned to the licensee or destroyed in accordance with prescribed controls.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee personnel:
B. Oldfield, Licensing
B. Nobles, Site Design
M. Byrd, Design Engineering Supervisor

NRC personnel
R. Nease, Chief, Engineering Branch Chief 1, Division of Reactor Safety, RII
E. Crowe, Senior Resident Inspector, Division of Reactor Projects, Farley Resident Office

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000348, 364/2011010-01	NCV	Failure to Implement Design Control Measures to Verify the Adequacy of CST Design (Section 1R21.2.3)
05000348, 364/2011010-02	NCV	Failure to Implement Design Control Measures to Verify the Adequacy of AFW Design (Section 1R21.2.3)
05000348, 364/2011010-03	NCV	Failure to Provide Adequate Procedural Guidance for Controlling Steam Generator and Pressurizer Level During Loss of Air Events (Section 1R21.2.3)
05000348, 364/2011010-06	NCV	Failure to Correctly Translate the Design Basis into Procedures for Minimum CCW Flow to the RHR Seal Coolers and Minimum Flow Requirements for the AFW Pumps (Section 1R21.2.4)
05000348, 364/2011010-07	NCV	Failure to Monitor or Perform Effective Preventive Maintenance on the 2C EDG Exhaust Fan Louvers (Section 1R21.2.6)
05000348, 364/2011010-08	NCV	Failure to Develop an Adequate Procedure to Test the Turbine Driven Auxiliary Feedwater Pump Discharge Check Valves (Section 1R21.2.7)

Opened

05000348, 364/2011010-04	URI	Evaluation of CST Vortex Effect on AFW Pump Minimum Submergence (Section 1R21.2.3)
05000348, 364/2011010-05	URI	Non-Conservative Assumptions Regarding AFW Net Positive Suction Head (Section 1R21.2.3)
05000348, 364/2011010-09	URI	Evaluation of MDAFW Pump Suction Check Valves (Section 1R21.2.7)
05000348, 364/2011010-10	URI	Administrative Controls in lieu of Automatic Actions for Degraded Voltage Protection (Section 4OA5.1)

LIST OF DOCUMENTS REVIEWED

Calculations

1.10, Condensate Storage/Decay Heat Removal, Rev. 1 including the following change:
CN-96-030, Verify Water Supply in Missile Protected Portion of CST at LOSP to Cool RCS, Rev. A
11.13, Available NPSH for Auxiliary Feedwater Pumps, Rev. 1
36.09, Time Required to Hydrogen Concentration in Auxiliary Building Battery Rooms to Reach 4 Volume Percent Following Failure of Battery Room Exhaust Fan, Ver. 2
37.4, CCW Heat Exchanger Models and Heat Removal Capacity Calculation, Rev. 0
CN-96-0047, Component Cooling Water System Evaluation – Power Uprate and Replacement Steam Generator, Rev. 8
38.04, Verification of AFW Flow Bases Unit 2, Rev. 4
40.02, Verification of AFW Flow Bases Unit 1, Rev. 4 including the following changes:
CN-00-0169, Incorporate MSLB case (1 faulted SG (B) at 14.7 psia, 550 psia in intact SGs, 3 pumps operating), Rev. A
CN-96-0026, AFW Flow Bases Verification for Power Uprate, Rev. A
CN-98-0001, AFW Flow Bases Verification of Steam Generator Tube Rupture, Rev. A
CN-98-0043, Documenting Superseded Calculations, Rev. A
CN-98-0124, AFW System Evaluation, Rev. A
8-13, Determine an Acceptable Restricting Orifice Tolerance for FO 2862 A,B,C, Rev. 0
A-350972, Criteria for Selection/Evaluation of Thermal Overload Heaters and Recommendations for Selection of Magnetic Breaker Setpoints for the Telemacanique (ITE-GOULD) Motor Control Center Starters Controlling the Motor Operated Valve Actuators, Rev. 1
BM-95-0961-001, Verification of CST Sizing Basis, Rev. 4
CBI-72-4859, Condensate Storage Tank, Rev. 0
E-082, Plant Electrical Distribution System Coordination Study, Ver. 10
E-095, Auxiliary Building Battery Capacity and Voltage Evaluation, Ver. 12
E-114, Sizing Breakers and Cables in the 120V vital and regulated AC Power Supply System, Rev. 0
E-115, Auxiliary Building Battery DC Control Circuit Lengths, Ver. 5
E-126, Short Circuit Availability at the Auxiliary Building 125V DC Switchgear, Distribution Panels, Diesel Generator Local Control Panels, Ver. 4
E-130, Verification of the Sizing of the Battery Chargers for Unit Auxiliary Building Batteries 1A, 1B, 2A and 2B, Ver. 0
E-143, Voltage Drop 120V Vital AC Distribution System, Ver. 3
E-144, Auxiliary Building Battery Voltage Margins at Safety-Related Components for Various Load Profiles, Ver. 7
E-42, Steady State Diesel Generator Loading Calculation for LOSP, SI and SBO, Ver. 19
E-98, Minimum Available DC Voltage and Permissible Control Circuit Lengths for Existing Battery Load Profile per E-95, Ver. 4
MC-HVAC-007, Diesel Generator Building Heating and Ventilation System, Rev. 3
SE-02-9834-001, Battery Capacity Calculation for TDAFW-UPS, Ver. 4
SE-90-1845-2-PE, Large, Small and SBO Diesel Dynamic Study, Ver. 6
SE-94-0-0378-001, MOV Combination Starter Component Sizes and Settings, 07/08/1994
SE-94-0470-001, Unit 1 As-Built Load Study, Ver. 7
SE-94-0470-004, Unit 1 Load Study Summary, Rev. 4
SE-94-0470-005, Unit 2 Load Study Summary, Rev. 5
SE-94-0470-007, Unit 2 As-Built Load Study, Ver. 6

SE-99-0-2010-001, Verification Package for Computer Software Used to Calculate MOV Thermal Overload Heater Sizes, Rev. 0
SH-07-00097663, J.M. Farley Main Steam Isolation Valve – Updated Actuator Opening & Closing Force Margin Calculations, Rev. 3
SJ-97-1407-004, Calculation to establish the total loop uncertainty for loops Q1/2P17FISH-3045, Rev. 0
SJ-97-1407-005, Calculation to establish the setpoint uncertainty for the RCP thermal barrier return line pressure interlock loops N1/2P17PSH-3184A, B and C, Rev. 0
SM-04-0148-001, AOV Setpoint Review for Q2P17HV3184, Ver. 2
SM-04-4801-005, AOV Setpoint Review for Q1N11PV3371A, Ver. 1
SM-04-4801-006, AOV Setpoint Review for Q1N11PV3371B, Ver. 1
SM-04-4801-007, AOV Setpoint Review for Q1N11PV3371C, Ver. 1
SM-1009959901-002, Diesel Generator Fuel Oil Consumption and Storage Capacity, Rev. 1
SM-2053014901-001, AOV Setpoint Review for Q2P17HV3045, Ver. 1
SM-2072448001-001, Verification of MSIV Accumulator Size for 12" Air Cylinders, Ver. 3
SM-90-1653-001, MOV Thrust Requirements for Gate & Globe Valves, Rev. 12
SM-90-1653-002, Reduced Voltage Torque/Thrust capability for both opening/closing and stall thrust at reduced and over-voltage conditions, Ver. 18
SM-90-1653-003, Design Basis Differential Pressure for the MOV Program, Rev. 15
SM-90-1905-001, Diesel Generator Fuel Oil Pump NPSH, Rev. 1
SM-92-2216-01, Verification of Diesel Generator Building Ventilation & Heating System, Rev. 1
SM-92-2216-03, Determine The Expected Average Room Temperature Inside Diesel Generator Building During Normal And LOSP Operating Conditions, 6/6/1992
SM-96-1059-001, Minimum Emergency Diesel Generator Fuel Oil Storage Capacity, Rev. 2
SM-C081865601-001, Heat Exchanger Tube Plugging Criteria, Rev. 6
SM-ES-89-1499-007, Service Water System Flow Balance Evaluation, Rev. 6
SM-SNC335993-001, CST AFW Pump Suction – Submergence Analysis, Ver. 1.0

Completed Procedures
FNP-1-STP-213.1A, S/G 1A Level Q1C22LT0474 Loop Calibration (1070678201), 04/08/2009
FNP-1-STP-213.1A, S/G 1A Level Q1C22LT0474 Loop Calibration (1090390001), 11/01/2010
FNP-1-STP-213.2A, S/G 1A Level Q1C22LT0475 (1070743201), 04/08/2009
FNP-1-STP-213.2A, S/G 1B Level Q1C22LT0475 Loop Calibration (1090390901), 11/01/2010
FNP-1-STP-213.3A, S/G 1A Level Q1C22LT0476 Loop Calibration (1070679001), 04/08/2009
FNP-1-STP-213.3A, S/G 1A Level Q1C22LT0476 Loop Calibration (1090389701), 11/01/2010
FNP-1-STP-213.4A, S/G 1B Level Q1C22LT0484 Loop Calibration (1070742801), 04/08/2009
FNP-1-STP-213.4A, S/G 1B Level Q1C22LT0484 Loop Calibration (1090390701), 11/02/2010
FNP-1-STP-213.5A, S/G 1B Level (1070380901), 04/09/2009
FNP-1-STP-213.5A, S/G 1B Level Q1C22LT0485 Loop Calibration (1090216101), 11/02/2010
FNP-1-STP-213.6A, S/G 1B Level Q1C22LT0486 Loop Calibration (1070744101), 04/09/2009
FNP-1-STP-213.6A, S/G 1B Level Q1C22LT0486 Loop Calibration (1090390501), 11/02/2010
FNP-1-STP-213.7A, S/G 1C Level Q1C22LT0494 Loop Calibration (1070679501), 04/19/2009
FNP-1-STP-213.7A, S/G 1C Level Q1C22LT0494 Loop Calibration (1090390201), 11/03/2010
FNP-1-STP-213.8A, S/G 1B Level Q1C22LT0495 Loop Calibration (1090390301), 11/03/2010
FNP-1-STP-213.8A, S/G Level Q1C22LT0495 Loop Calibration (1070744601), 04/19/2009
FNP-1-STP-213.9A, S/G Level Q1C22LT0496 Loop Calibration (1070677501), 04/19/2009
FNP-1-STP-213.9A, S/G Level Q1C22LT0496 Loop Calibration (1090389901), 11/03/2010
FNP-1-STP-24.16, CTMT CLR and RCP MTR AIR CLR Service Water Valves In-service Test (1092504401), 04/23/2011

FNP-1-STP-45.1, CVCS Cold Shutdown Valves In-service Test (1090398501), 11/08/2010
FNP-1-STP-45.12, Seal Return and B Train ECCS MOVs Refueling Outage Valves In-service Test (1090398201), 11/06/2010
FNP-1-STP-45.15, MS ARV and Emergency Air Compressor Cold Shutdown Test, 11/05/07
FNP-1-STP-45.15, MS ARV and Emergency Air Compressor Cold Shutdown Test, 10/26/10
FNP-1-STP-45.4, ECCS Refueling Outage Valves In-service Test (1090398401), 10/31/2010
FNP-1-STP-45.9, CCW Refueling Outage Valves In-service Test (1070675001), 04/28/2009
FNP-1-STP-45.9, CCW Refueling Outage Valves In-service Test (1090406601), 10/12/2010
FNP-1-STP-45.9, CCW Refueling Outage Valves In-service Test (2070078801), 11/12/2008
FNP-1-STP-45.9, CCW Refueling Outage Valves In-service Test (2082296001), 05/12/2010
FNP-1-STP-47.0, Miscellaneous Valves In-service Test (1092280001), 03/31/2011
FNP-1-STP-608.0, Main Steam Safety Valve Operational Test, 04/05/06
FNP-1-STP-608.0, Main Steam Safety Valve Operational Test, 10/12/10
FNP-1-STP-627, Local Leak Rate Test, 04/13/2009
FNP-1-STP-627, Local Leak Rate Test, 04/18/2006
FNP-1-STP-627, Local Leak Rate Test, 04/22/2007
FNP-1-STP-627, Local Leak Rate Test, 10/12/2007
FNP-1-STP-627, Local Leak Rate Test, 10/30/2008
FNP-2-STP-22.16, Turbine Driven Auxiliary Feedwater Pump Quarterly Inservice Test, 07/29/10
FNP-2-STP-22.16, Turbine Driven Auxiliary Feedwater Pump Quarterly Inservice Test, 10/13/10
FNP-2-STP-22.16, Turbine Driven Auxiliary Feedwater Pump Quarterly Inservice Test, 01/17/11
FNP-2-STP-24.16, CTMT CLR and RCP MTR Air CLR Service Water Valves In-service Test (2101679001), 06/17/2011
FNP-2-STP-45.1, CVCS Cold Shutdown Valves In-service Test (2082284301), 04/28/2010
FNP-2-STP-45.12, Seal Return and B Train ECCS MOVs Refueling Outrage Valves In-service Test (2082285601), 04/22/2010
FNP-2-STP-45.4, ECCS Refueling Outage Valves In-service Test (2082284201), 04/15/2010
FNP-2-STP-45.7, MSIV and Bypass Valves CSD Valves Inservice Test, 10/05/07
FNP-2-STP-45.7, MSIV and Bypass Valves CSD Valves Inservice Test, 11/17/08
FNP-2-STP-45.7, MSIV and Bypass Valves CSD Valves Inservice Test, 05/08/10
FNP-2-STP-47.0, Miscellaneous Valve In-service Test (2101668601), 05/30/2011
FNP-2-STP-905.0, "A" Train Auxiliary Building Battery Inspection Q2R42E0002A (2063404001), 11/09/2008
FNP-2-STP-905.0, "A" Train Auxiliary Building Battery Inspection (2082298201), 04/29/2010
FNP-2-STP-905.0, 2B Train Auxiliary Building Battery Inspection (2070934701), 10/26/2008
FNP-2-STP-905.0, 2B Train Auxiliary Building Battery Inspection (2082298301), 04/18/2010
FNP-2-STP-905.1, "A" Train Auxiliary Building Battery Service Test (2082233901), 05/09/2010
FNP-2-STP-905.1, "B" Train Auxiliary Building Battery Service Test (2080429101), 05/09/2010
FNP-2-STP-905.1, 2A Auxiliary Building Battery Service Test (2071084101), 11/10/2008
FNP-2-STP-905.1, B Train Auxiliary Building Battery Service Test (2070645701), 11/03/2008
FNP-2-STP-905.2, 2B Perform Battery Performance Test (2060143601), 04/19/2007
FNP-2-STP-905.2, Battery Performance Test Q2R42E0002A (2070645801), 04/29/2010

Completed Work Orders
1081343601, Replace 1A MDAFW Pump Motor Rotor, dtd 11/06/09
1090806201, Refurbish Unit 1 Reactor Trip Breaker B, dtd 3/4/11
1091121001, Refurbish Unit 1 Reactor Trip Breaker A, dtd 4/28/11
2101284601, Replace 2B MDAFW Pump Motor, dtd 4/6/10
1101040801, Tan Delta testing of cable 1DBDJ03P, 03/11/2010

2041697701, Breaker clean, inspect, adjust and lubricate, 09/09/2005
2090641201, Auxiliary Building Battery Equalization, 06/29/2010
2090782801, Auxiliary Building Battery Equalization, 08/31/2010
2090896501, LB18 clean, inspect & lubricate breaker per FNP-0-EMP-1340.01, 04/11/2010
2091178801, Auxiliary Building Battery Equalization, 02/09/2011
2091452601, Auxiliary Building Battery Equalization, 02/14/2011
2091839501, 2B Perform Battery Performance Test, 12/09/2010
1081549001, Q1P17FISH3045 Calibration, 02/11/2010
1070535801, Q1P17FISH3045 Calibration, 06/21/2008
2081252401, Q1P17FISH3045 Calibration, 02/25/2010
2070610501, Q1P17FISH3045 Calibration, 06/17/2008
M300357801, PI3184A Replaced due to Indicator Needle being Damaged, 05/23/2003
20M0303501, Gauge needs Replacing, needle is bent, 04/14/2000
M300357901, PI3184B Replaced due to Indicator Needle being Damaged, 05/23/2003
20M0399201, Replace Bent Pointer or Replace the Gauge, 05/13/2000
M300358001, PI3184C Replaced due to Indicator Needle being Damaged, 05/23/2003
0W48608801, Perform FNP-2-IMP-0.11.2 on Q2P17HV3184, 04/08/1998
0W61783301, Perform FNP-2-IMP-0.11.2 on Q2P17HV3045, 11/01/1999
0W61783801, Perform FNP-2-IMP-0.11.2 on Q2P17HV3184, 11/01/1999
0W64706301, Perform FNP-2-IMP-0.11.2 on Q2P17HV3045, 03/08/2001
0W64706601, Perform FNP-2-IMP-0.11.2 on Q2P17HV3184, 02/26/2001
1101040801, Test all three phases of cable 1DBDJ03P, 03/11/2010
2050551501, Auxiliary Building Battery Charger 2A Inspection, 02/28/2008
2052055101, 2A Auxiliary Building Battery Charger Load Test, 10/11/2006
2062327801, 2A Auxiliary Building Battery Charger Load Test, 02/26/2008
2091452701, Auxiliary Building Battery Charger 2A Inspection, 07/17/2011
2071377801, Auxiliary Building Battery Charger 2A Inspection, 05/20/2009
2071511501, 2A Auxiliary Building Battery Charger Load Test, 05/21/2009
1090394401, Perform FNP-1-STP-45.5, 11/1/2010
2071392301, Calibrate Q2P17FISL3062A, 1/22/2009
2092507901, Perform FNP-STP-22.24 on B Train Only, 5/3/11
2101074401, Perform FNP-STP-22.24 on A Train Only, 517/11
2092855001, Perform FNP-2-STP-22.30, 4/8/2010
1092855301, Perform FNP-1-STP-22.30, 10/15/2010
S092082701, Perform FNP-0-STP-154.3, 2/10/2004
S092509801, Perform FNP-0-STP-154.3, 5/5/2011
S071905701, D/G 2C Lube Oil Heat Exchanger Eddy Current Test, 8/13/2010
1090394401, Perform FNP-1-STP-45.5, 11/1/2010
1090855901, MOV8701A Design Basis Diagnostic Test, 3/10/2011
97M0730201, FI-3045 Zero Flow when Normal Indication Approximately 150 GPM, 09/25/1997
S080674101, HVAC Roof Ventilator Fan Bearings Lubrication, 9/14/2010
M300650301, Replace the 1B battery due to degradation from age, temperature, and low float conditions. DCP-03-1-9878 has been initiated to replace the current ring of NCN-25 cells with the new LCU-27 cells, 09/15/2003
S091300001, Temporary Modification to DG 2C Speed Signal Generator, dtd 7/9/09
S080674101, 2C Fan B HVAC Roof Ventilator Fan Bearings Lubrication, 9/14/2010
S080673901, 2C Fan A HVAC Roof Ventilator Fan Bearings Lubrication, 9/14/2010
S080675001, 2C Fan C HVAC Roof Ventilator Fan Bearings Lubrication, 9/14/2010
S092173001, Replacement of DG 2C Excitation Shutdown Relay, dtd 8/4/09

0000708161, Clean, inspect & lubricate breaker per FNP-0-EMP-1340.01, 03/23/2004
0000708163, Clean, inspect & lubricate breaker per FNP-0-EMP-1340.01, 03/26/2004
0000708167, Clean, inspect & lubricate breaker per FNP-0-EMP-1340.01, 06/26/2003
1049005502, DCP-S-04-1-00055 1A AUX building battery cell replacement, 01/21/2005
1072601701, Calibrate 1A RCP CCW Return Line Pressure Switch PSH3184A, 04/22/2009
1072601801, Calibrate 1C RCP CCW Return Line Pressure Switch PSH3184C, 04/22/2009
1072601901, Calibrate 1B RCP CCW Return Line Pressure Switch PSH3184B, 04/22/2009
1091121101, Refurbish Unit 1 Reactor Trip Bypass Breaker A, dtd 2/5/11
1091376801, Refurbish Unit 1 Reactor Trip Bypass Breaker B, dtd 3/4/11
1091389701, Calibrate 1A RCP CCW Return Line Pressure Switch PSH3184A, 10/21/2010
1091389801, Calibrate 1C RCP CCW Return Line Pressure Switch PSH3184C, 10/31/2010
1091389901, Calibrate 1B RCP CCW Return Line Pressure Switch PSH3184B, 10/31/2010
1100294801, Perform Calibration of Air Alarm Pressure Switch Q1P17PSL3184D, 10/15/2010
1102077501, Perform Setpoint Verification and Seat Leakage Testing Per FNP-1-STP-628.25, 1/26/11
2041488601, Charging Pump Discharge Header Isolation Completed FNP-0-EMP-15001.17, 11/6/2005
2052823106, Install old limits or replace open and close limits as needed, 11/21/2005
2052937901, MOVATS reported the following valves flex pull loose from valves, 11/12/2005
2052943901, Perform an “As found” FlowScan, replace the actuator spring, rebuild the actuator and perform an “As Left” setup of Q2P17HV3184 using the FlowScanner to the values listed in PDMS during the next 2R18 outage. The actuator spring part number is 6148407, 11/07/2005
2061167401, Calibrate 2B RCP CCW Return Line Pressure Switch PSH3184B, 04/12/2007
2061167501, Calibrate 2A RCP CCW Return Line Pressure Switch PSH3184A, 04/12/2007
2061167601, Calibrate 2C RCP CCW Return Line Pressure Switch PSH3184C, 04/12/2007
2070863201, Calibrate 2B RCP CCW Return Line Pressure Switch PSH3184B, 11/07/2008
2070863301, Calibrate 2A RCP CCW Return Line Pressure Switch PSH3184A, 11/09/2008
2070863401, Calibrate 2C RCP CCW Return Line Pressure Switch PSH3184C, 11/07/2008
2071377701, Clean, inspect & lubricate breaker per FNP-0-EMP-1340.01, 12/21/2009
2071453803, Replace 2A Battery bank during 2R20 per Farley LRP PS-03-2382, 11/02/2009
2071453901, Replace 2B Battery bank during 2R19 per Farley LRP PS-03-2382, 06/14/2007
2071802301, Clean, inspect and lubricate 2B battery charger supply breaker, 03/05/2011
2072915201, Bypass The Open Torque Switch Setting For Q2E11MOV8702A/B and Q2E11MOV8701A/B, 2/11/2008
2080243501, Clean, inspect, test and adjust GE AK-2A Circuit Breakers, 12/19/2010
2091156901, 2B Auxiliary Building Battery Quarterly Verification, 11/17/2010
2091316401, 2A Auxiliary Building Battery Quarterly Verification, 11/24/2010
2091900501, Clean, inspect & lubricate breaker per FNP-0-EMP-1340.01, 01/08/2011
2092081801, 2B Auxiliary Building Battery Quarterly Verification, 02/10/2011
2092081901, 2A Auxiliary Building Battery Quarterly Verification, 02/16/2011
S080532001, Louvers for DG 2C Room Roof Exhaust Ventilator Fan B Need Repair, 2/25/2008

Corrective Action Documents
2008101522, 1B Diesel Generator Room B Exhaust Fan Damper Stuck Open
2008101857, Louvers for DG 2C Room Roof Exhaust Ventilator Fan B Need Repair
2008108681, During the running of STP-22.2 on 2B MDAFW pump, the union on the balancing line under the pump casing had a leak of approx. 1 drop every 3 minutes

2008110018, During a run of the TDAFW Pump for Flow
2008110594, Minor, dry, chromate residue was noted on N2P17FE3045, N2P17FE3044, Q2P17HV3045 and Q2P17HV3067
2009103820, Gage needle bent due to pegging high
2009105765, While performing walkdown of containment, PI 3184C (0-200 psi) gauge front glass was missing and the indicator bent
2009101539, NRC CDBI 2009 - Emergency Air Issue
2009100845, No Performance Criterion for Monitoring Function R43-F03 in FNP-0-M-87
2009100037, Started 1B MDAFW Pump per STP-22.2 and Immediately Received Annunciator
2009102020, CCW from RCP pressure indicators have been repetitively found over-range
2009102135, Service Water Backup Service to AFW
2009102148, Work Order 2070521701 Left the Equalize at 55.2 VDC
2009102616, Deficiencies Noted with Operation and Maintenance of EAC
2009103343, Louvers for DG 2C Room Roof Exhaust Ventilator Fan B Need Repair
2009202038, Determine if Function R43-F03 is Risk Significant
2009204983, Generate and Complete an RER to Investigate and Evaluate Hardware
2010101698, Tracking CR, during calibration of Q2P17FISH3045 under work order 2081252401 as found level was found to be out of admin tolerance low as per data sheet
2010105432, Moderate flow of instrument air from vent hole of filter in line to Q2P17HV3184
2010112466, Louver for the D/G Day Tank Room Exhaust Ventilator Fan B Needs To Be Repaired
2010114366, Q1E11MOV8701A Went Closed When Breaker FU-TS Closed
2010200701, Evaluation of Postulated Degraded Voltage Event for Diablo Canyon
2011337450, Oil Level OOS High In 2C D/G Governor
2011351580, 2C DG B Fan Discharge Louver Is Slightly Open
336547, Corrective Actions for AFW Recirculation Line Cracking Were Not Effective in Preventing Recurrence

<u>Drawings</u>
11870036, Electrical Schematic Rectifier Assembly, Rev. 7
3D20701, AMSAC Interconnection Diagram, Sh. 2, Rev. 2
3D20701, AMSAC Interconnection Diagram, Sh. 4, Rev. 2
9034518910, Basler Voltage Regulator AP-20339
94207D, Schematic Diagram for MOV from U-214885, Rev. A2
B-177556, MCC Schedules – 208V, MCC-1A, Sh. 1D, Rev, 21.0
B-205810, Logic Diagram MDAFW Pumps, Rev. 5
C- 177118, Interlock Schematic Sta. Service Transformer 1F, Rev. 7
D-107001, Single Line Electrical Auxiliary System, Rev. 20
D-107089, Elementary Diagram LC Circuit Breakers, Rev. 10
D-11870036, DG Schematic and Interconnection Diagram, Sheet 3, Rev. 8
D-170060, P&ID Diesel Generator Fuel Oil System, Rev. 15
D-170337, Diesel Generator Building – HVAC – Roof Plan, Rev. 6
D-170339, Diesel Generator Building – HVAC – Sections Details & Design Data, Rev. 13
D-170801, P&ID – Lube Oil System for Diesel Generator 2C, Rev. 15
D-170807, P&ID – Air Start System for Diesel Generator 2C, Rev. 18
D-170809, P&ID – Fuel Oil System for Diesel Generator 2C, Rev. 9
D-172675, Elementary Diagram HVAC Diesel Generator And Oil Storage rooms 2C, Rev. 7
D-172775, Elementary Diagram EDG 1-2A Exciter and Misc. Controls, Rev. 18.0
D-173001, Low Voltage Switchyard, Rev. 12.0

D-173096, Unit 1 Electrical Loads Diagram, Sh. 1, Rev. 40
D-173096, Unit 1 Electrical Loads Diagram, Sh. 2, Rev. 10 & 17
D-175002, P&ID – Aux Feedwater System, Rev. 31
D-175002, P&ID – Component Cooling Water System, Rev. 48
D-175007, P&ID Aux. Feedwater System, Sheet 1, Ver. 31
D-175032, P&ID Component Cooling Water System, Sheet 1, Ver. 48
D-175032, P&ID Component Cooling Water System, Sheet 2, Ver. 26
D-175032, P&ID Component Cooling Water System, Sheet 3, Ver. 12
D-175033, P&ID Main Steam and Aux. Steam System, Sheet 1, Ver. 38
D-175033, P&ID Main Steam and Aux. Steam System, Sheet 2, Ver. 25
D-175034, P&ID Instrument Air, Sheet 1, Ver. 35
D-175034, P&ID Instrument Air, Sheet 2, Ver. 18
D-175034, P&ID Instrument Air, Sheet 3, Ver. 12
D-175035, P&ID Service Air, Sheet 2, Ver. 9
D-175038, P&ID Residual Heat Removal System, Sheet 1, Ver. 18
D-175038, P&ID Safety Injection System, Sheet 1, Ver. 42
D-175038, P&ID Safety Injection System, Sheet 2, Ver. 22
D-175039, P&ID Chem. & Vol. Control System, Sheet 1, Ver. 24
D-175039, P&ID Chem. & Vol. Control System, Sheet 2, Ver. 39
D-175039, P&ID Chem. & Vol. Control System, Sheet 3, Ver. 17
D-175039, P&ID Chem. & Vol. Control System, Sheet 4, Ver. 34
D-175039, P&ID Chem. & Vol. Control System, Sheet 5, Ver. 4
D-175039, P&ID Chem. & Vol. Control System, Sheet 6, Ver. 9
D-175039, P&ID Chem. & Vol. Control System, Sheet 7, Ver. 9
D-175041, P&ID – Residual Heat Removal System, Rev. 18
D-177012, Single Line Protection & Metering LC 1F, Rev. 10
D-177024, Single Line 120Vac Vital & Regulated System Train “A”, Rev. 35
D-177025, Single Line 120Vac Vital & Regulated System Train “B”, Rev. 30
D-177089, Elementary Diagram 600V Load Center Breakers, Rev. 10
D-177118, Interlock Schematic LC 1F Feeder Breakers, Rev. 7
D-177155, Elementary Diagram Bus 1F Feeder from SAT 1A, Rev. 20.0
D-177161, Elementary Diagram Bus 1F Feeder from SAR 1B, Rev. 20.0
D-177168, Elementary Diagram Bus 1G Feeder from SAT 1A, Rev. 16.0
D-177170, Elementary Diagram Bus 1F Differential Protection, Rev. 4.0
D-177653, Elementary Diagram Sequencer B1F Load Shedding, Rev. 22.0
D-177654, Elementary Diagram Sequencer B1G Load Shedding, Rev. 16.0
D-177944, Single Line Diagram, AMSAC - UPS, Sh. 2, Rev. 1
D-177944, Single Line Diagram, TDAFW Pump UPS, Sh. 1, Rev. 5.0
D-181701, Reactor Trip Switchgear Connection Diagram, Sh. 1, Rev. 4.0
D-200012, Unit 2, P&ID – Demineralized Water from Demineralizer to Storage Tank, Rev. 13
D-204623, Connection Diagram Distribution Panel 1J-n, Rev. 22
D-205002, Component Cooling Water System, Sheet 2, Rev. 19
D-205002, P&ID – Component Cooling Water System, Rev. 31
D-205007, P&ID Aux. Feedwater System, Sheet 1, Ver. 23
D-205033, Unit 2, P&ID – Main Steam and Auxiliary Steam System, Rev. 23.0
D-207001, Single Line Electrical Auxiliary System, Rev. 19.0
D-207012, Single Line Protection & Metering LC 2F, Rev. 4
D-207082, Single Line DC Distribution System 2A, Unit 2, Ver. 25
D-207083, Single Line DC Distribution System 2B, Unit 2, Ver. 30

D-207089, Elementary Diagram LC Circuit Breakers, Rev. 10
D-207118, Interlock Schematic LC 2F, Rev. 2
D-207186, Elem. Diag. AFWP 4160V No. 2A, Rev. 15.0
D-207198, Elem. Diag. Control Motor Generator Set 2A & 2B, Sh. 1, Rev. 9.0
D-207198, Elem. Diag. Reactor Trip Switchgear, Sh. 2, Rev. 3
D-207198, Elem. Diag. Reactor Trip Switchgear, Sh. 3, Rev. 4
D-207229, Elementary Diagram HHSI & AFW Fan Motors, Rev. 11.0
D-207653, Elementary Diagram Sequencer B2F Load Shedding, Rev. 17.0
D-207654, Elementary Diagram Sequencer B2G Load Shedding, Rev. 11.0
D-273001, Low Voltage Switchyard, Rev. 14.0
D-273096, Unit 2 Electrical Loads Diagram, Sh. 1, Rev. 28
D-273096, Unit 2 Electrical Loads Diagram, Sh. 2, Rev. 10
U-161693, Unit 1, Condensate Storage Tank General Plan, Rev. 1.0
U-166235, Function Diagrams Primary Coolant System Trip Signals, Rev. 1.0
U-209212, Outline and Dimensional Drawing for HV-3045, Ver. 2
U-213481, Unit 2, Condensate Storage Tank General Plan, Rev. 1.0
U-611272, Valve Assembly with D-100-160 PA Actuator, Ver. 3

Modifications
1061013101, 1B Unit Auxiliary Transformer Removal, Rev. 1.1
S081236201, Temporary Modification to 2C DG Signal Generator, dtd 6/4/09
S092173001, Modification of 2C DG Excitation Shutdown Relay, dtd 8/6/09

Miscellaneous
0-11-06, Prompt Determination of Operability, Rev. 2
024-5-015, Diesel Generator 2C Pre-Operational Test, 4/19/1977
05000454/2011010, 05000455/2011010, Inspection Report Byron Station Units 1 and 2
A181001, Service Water System, Rev. 56.0
A181010, Auxiliary Feedwater System, Rev. 22.0
A-2062-1, Unit 1 RHR Pump Seal Coolers, Rev. 3
AP-20339, Bechtel to Southern Co. Letter, Re. IN 92-29, 11/6/1992
AP-21052, Letter from Bechtel to Farley, Verification of Minimum Required Condensate Storage Tank (CST) Volume (REI 93-0063), Dated March 29, 1994
AP-21364, Letter from Bechtel to Farley, Verification of CST Design Basis (REI 95-0961), Dated April 1, 1996
AP-21415, Letter from Bechtel to Farley, Verification of CST Design Basis, Dated June 7, 1996
AP-2837, Letter from Bechtel to Farley, Auxiliary Feedwater System, Dated July 27, 1977
Basler Electric Letter, 2/16/1990
Basler SBSR Voltage Regulator Instructions
C091647301, Clean/Pure Water HX Long Range Plan, Sequence No. 3
Colt Industries Letter, 3/2/1990
Colt Industries Letter, 3/23/1990
DOEJ-FRSNC326893-M001, Documentation of Engineering Judgment, Condensate Storage Tank Submergence, Ver. 1.0
DOEJ-FRSNC326893-M002, Documentation of Engineering Judgment, Evaluation of Heat Tracing Heat Input on AFW Pump Suction Piping, Ver. 1.0
DOEJ-FRSNC326893-M003, Documentation of Engineering Judgment, Evaluation of Tornado Generated Missile Debris in the Condensate Storage Tank and Auxiliary Feedwater System, Ver. 1.0

DOEJ-FRSNC326893-M004, Documentation of Engineering Judgment, Evaluation of Condensate Storage Tank Volume, Ver. 1.0
DOEJ-FRSNC326893-M005, Documentation of Engineering Judgment, Hydraulic Evaluation of FNP Auxiliary Feedwater System to Support Operability Determinations, Ver. 1.0
Farley U1 and U2 Degraded Grid Protection, Screening Level Estimate, 6/3/2011
Final Safety Evaluation on Joint Owners' Group Program on Motor-Operated Valve Periodic Verification, 9/25/2006
FNP-93-0447, Farley Generation Dept. Memo, Re. IN 92-29, 2/11/1993
IOM-Heliflow-0507, Heliflow Installation, Operation and Maintenance Manual
Joseph M. Farley Nuclear Plant 180 – Day Response to GL 96-05, 3/14/1997
Joseph M. Farley Nuclear Plant Additional Information Regarding GL 96-05 Program at Joseph M. Farley Nuclear Plant Units 1& 2, 9/14/1999
Joseph M. Farley Nuclear Plant GL 96-05 MOV Periodic Verification Program JOG Supplemental Submittal, 6/10/1998
Joseph M. Farley Nuclear Plant Response to Revised Request For Additional Information Regarding GL 96-05 Program At Joseph M. Farley Nuclear Plant Units 1 and 2, 5/28/1999
Joseph M. Farley Nuclear Plant Units 1 & 2, Motor Operated Valve Information, 3/7/1989
JPM SO610A, Isolate the TDAFW Steam Supply from B SG at the Shutdown Panel
Maintenance Rule Documentation, D/G Building Heating, Ventilation and Air Conditioning
Maintenance Rule Documentation, Residual Heat Removal System
Minimum Flow Evaluation for TDAFW Pumps, Rev. 1
Minimum Flow evaluation, Auxiliary Feedwater Pumps, Rev. 1, 6/27/1989
NL-07-0950, Response to NRC Generic Level 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients," 05/04/2007
NRC Bulletin 88-04, "Safety-Related Pump Loss"
NRC Letter, License Amendments related to the Issuance of ITS, 11/30/1999
NRC letter, NRC Inspection Report Nos. 50-348/95-18 and 50-364/95-18, Notice of Violation, and Safety Evaluation Report, 11/21/1995
NSD-TB-92-06-RO, W Technical Bulletin Type LS and LSG Amptectors, 6/16/1992
Operations Training Simulator Exam Scenario, 2011 NRC CDBI, 9/07/2011
Portec Voltage Regulator Manual
Procedure Request form, FNP-1-ARP-1.9, Main control Board Annunciator J, Rev. 40
Procedure Request form, FNP-2-ARP-1.9, Main control Board Annunciator J, Rev. 27
PS-08-0168, RER C071213001-002 Transmittal, 3/5/2008
REA 97-1638, Steam Generator Tube Rupture Analysis Summary Report, 11/21/2000
REA 98-1731, Engineering Report, RTS and ESFAS Relay Uncertainty, 2/8/2001
RER 2082181801, Evaluate Unit 1 and 2 TDAFWP Suction Check Valve
Southern Company Letter, Steam Generator Replacement Related Technical Specification Change Request, 12/01/1998
Specific Background Document for FNP-1/2-EEP-3, Steam Generator Tube Rupture, Ver. 2
System Health Reports, Residual Heat Removal, 2nd Quarter 2008 – 2nd Quarter 2011
U-176261, Unit 1 Battery Rack 1A and 1B Layout of 60 Cells (Heavy Seismic Resistance), 07/03/2007
U184804, Diesel Engine Generators 1C and 2C Operation and Maintenance Manual, Rev. 3
U-184852, DG 1-2A, 1B & 2B Operation & Maintenance Manual, Rev. 26.0
U-206441, Unit 2 MDAFW Pumps General Arrangement, Rev. 1.0
U-211521, 125V Station Battery Auxiliary Building Rack 2A/2B Layout, Ver. 2
U-214885, Auxiliary Feedwater Pump & Motor Instruction Manual, Rev. 4.0
U-215471, Reactor Trip Switchgear Instruction Book, Rev. L

U-260232, Reactor Trip Switchgear Instruction Book, Rev. L
U-277549, Instruction Manual HD valves SS1102-36, Unit 2, Ver. 2
U-279795, Environmental and Seismic Qualification Report of 125 Volt DC Station Service Batteries 2A & 2B Auxiliary Building, 05/12/1993
U-407308, Qualified life based on activation energy, component replacement and surveillance ASCO catalog NP-1 valves, 03/08/1983
U418156, Instruction Manual RHR Pumps, Rev. 4
U-732587, Instruction and Operating Manual for 600A Auxiliary Building Battery Chargers, 07/27/2009
U-734319, Commercial Grade Dedication for Rectifier, Rev.0
WCAP-15097, Replacement Steam Generator Program NSSS Engineering Report, Book 1, 3/2001
WOGRTSDS416-01, Maintenance Manual for DS416 Reactor Trip Switchgear, Rev. 0

Procedures
FNP-0-AP-74, Development and Maintenance of Emergency Response Procedures, Ver. 14
FNP-0-ARP-19.2, 2850 KW 1C and 2C Local Alarm Panels, Rev. 26
FNP-0-EMP-1320.02, 600V LC Hot Bus Transfer, Rev.35.0
FNP-0-EMP-1322.03, W DS-206 & -416 CB Extended Maintenance, Rev. 11.0
FNP-0-EMP-1322.10, Maintenance and Cleaning of W Switchgear, Rev. 5.0
FNP-0-EMP-1402.01, Reactor Trip & Bypass Circuit Breakers, Rev. 20.0
FNP-0-SOP-43, Diesel Generator Building HVAC, Rev. 20
FNP-0-STP-154.3, 2C Diesel Generator Air Dryer Operability Verification, Rev. 23
FNP-0-STP-80.17, Diesel Generator 2C Operability Test, Rev. 36
FNP-1-AOP-16.0, CVCS Malfunction, Rev. 17
FNP-1-AOP-21.0, Severe Weather, Rev. 31
FNP-1-AOP-6.0, Loss of Instrument Air, Ver. 39
FNP-1-ARP-1.3, Main Control Board Annunciator Panel C, Rev. 28.1
FNP-1-ARP-1.9, Main Control Board Annunciator J, Ver. 47
FNP-1-ECP-0.0, Loss of All AC Power, Rev. 24
FNP-1-ECP-0.0, Loss of All AC Power, Rev. 24
FNP-1-EEP-0, Reactor Trip or Safety Injection, Rev. 42
FNP-1-EEP-3, Steam Generator Tube Rupture
FNP-1-EMP-1320.02, 600V LC Hot Transfer Backfeed to 1G from 1P, Rev. 4.0
FNP-1-SOP-22.0, Auxiliary Feedwater System, Ver. 66
FNP-1-SOP-36.3, Electrical Distribution System, Appendix 18, Rev. 62.0
FNP-1-SOP-41.0, Control Rod Drive and Position Indication, Rev. 30.0
FNP-1-SOP-7.0, Residual Heat Removal System, Rev. 99
FNP-1-STP-22.16, Turbine Driven Auxiliary Feedwater Pump Quarterly Inservice Test with Preservice Test Appendix, Rev. 57.1
FNP-1-STP-256.12, Turbine Trip-Reactor Trip Response Time Test, Rev. 24.0
FNP-1-STP-256.12A, Surveillance Test Procedure Data Package, Rev. 24.0
FNP-1-STP-256.12B, Surveillance Test Procedure Data Package, Rev. 24.0
FNP-1-STP-45.15, MS ARV and Emergency Air Compressor Cold Shutdown Test, Rev. 5.0
FNP-1-STP-45.5, RHR Cold Shutdown Valves Inservice Test, Rev. 10
FNP-1-STP-608.0, Main Steam Safety Valve Operational Test, Rev. 35.0
FNP-1-UOP-2.4, Planned Reactor Shutdown and Cooldown to Cold Shutdown, Rev. 9.0
FNP-1-UOP-2.4, Planned Reactor Shutdown and Cooldown to Cold Shutdown, Rev. 9.0
FNP-2-ARP-1.10, Annunciator Response Procedure, Rev. 34.0
FNP-2-ARP-1.3, Main Control Board Annunciator Panel C, Rev. 22
FNP-2-ARP-1.9, Annunciator Response Procedure, Rev. 34.0

FNP-2-EMP-1320.02, 600V LC Hot Transfer Backfeed to 2G from 2P, Rev. 5.0
FNP-2-SOP-22.0, Auxiliary Feedwater System, Rev. 69.0
FNP-2-SOP-41.0, Control Rod Drive and Position Indication, Rev. 31.0
FNP-2-STP-22.13, Turbine Driven Auxiliary Feedwater Pump Check Valve Flow Verification, Rev. 21
FNP-2-STP-22.16, Turbine Driven Auxiliary Feedwater Pump Quarterly Inservice Test, Rev. 62.0
FNP-2-STP-22.16, Turbine Driven Auxiliary Feedwater Pump Quarterly Inservice Test, Shutdown Inservice Test & Preservice Test, Rev. 17.0
FNP-2-STP-22.2, 2B Auxiliary Feedwater Pump Quarterly Inservice Test, Rev. 22.0
FNP-2-STP-22.30, Auxiliary Feedwater Pump Discharge Check Valve Reverse Flow Closure Operability Test, Rev. 5
NMP-ES-017-004, MOV Diagnostic Procedure for Gate & Globe Valves, Rev. 6
NMP-ES-051, Cable Monitoring Program, Rev. 3
PR-2380-0005-119, Final Inspection of Rectifier Assembly, Rev. 0
PR-2380-0005-99, Special Tests and Inspections of Rectifier Assembly, Rev. 0

Condition Reports Initiated Due to CDBI Activity
CR 348298, Generate Work Order (WO) for Battery Sample Tubes
CR 348303, Generate WO for 2A Battery
CR 348558, Potential Error on P&ID D-175002
CR 348613, Component Cooling Water (CCW) Flow Design Setpoint Discrepancy
CR 348681, FNP-0-EMP-1402.01 Procedure Change
CR 348721, Spare Cables not Tagged
CR 348789, Generate WO for 1A Battery
CR 348795, TDAFW Pump Discharge Check Valve Acceptance Criteria Non-conservative
CR 349883, Discharge Louver for 2C DG ‘A’ Ventilator Stuck Partially Open
CR 349888, Rubber Gasket Material on Frame for 2C DG Wall Louvers
CR 351170, CST Submergence Level Calculation Discrepancy
CR 351259, Lower Equipment Room unsecured equipment
CR 351314, Flex Conduit Opened
CR 351504, Traverse Holes in Duct Work not Sealed
CR 351580, 2C DG 'B' Fan Discharge Louver is Slightly Open
CR 351620, Partially Crimped Air Supply to HV-3045
CR 351817, No PM's Noted for RHR Seal Water Heat Exchanger during CDBI Review
CR 352168, Non-conservatism in Calculation 11.13
CR 352210, AFW calculation 40.02 has non-conservatisms
CR 352226, Verification of CST Sizing Basis
CR 352485, Non-conservative AFW Pump Suction Low Flow Alarm
CR 353014, WO1091121001 Test Equipment was not Properly Documented
CR 353599, Reevaluate CST TORMIS analysis
CR 353743, Cumulative Issues on AFW DBA Hydraulic Analyses
CR 355007, CDBI Question Minimum Voltage to Flash Field of EDG
CR 355015, AOP-21 Actions to Isolate CST after Tornado
CR 355017, Recommendation for Gap Analysis – Loss of Air Scenario
CR 355025, Calculation 11.13 does not Evaluate Highest AFW Pump Flow
CR 355040, Q2E21MOV8132A could potentially have a Negative Margin to Close
CR 355124, Calculation SE-90-1845-2-PE Needs Clarification
CR 355130, CDBI Question for 2C Diesel Generator Degraded Louver Issue

CR 355230, Evaluation to Cycle MOVs in Procedure AOP-6.0
CR 355293, Evaluate Assumptions for CST Volume in Technical Specification Bases
CR 355294, Verify Conservatism of Assumptions in BM-95-0961-001 and CN-96-030
CR 355457, CST Vortex Consideration Impacts TS Required Volume
CR 355667, Evaluation and Clarification Needed for FSAR Related to SGTR
CR 355672, Steam Generator Level Control with MOVs
CR 355695, Risk Impact of Using MOVs for Level Control
CR 355823, RHR Seal Cooler Heat Exchanger Velocities
CR 355835, No Evaluation for Potential SW Blockage of AFW Pump Restrictions
CR 355897, TDAWFP Check Valve Oscillating
CR 355898, AFW Orifice Resistance
CR 357172, Use of FSD for Design Inputs
CR 361514, AFW Pump Documentation
CR 363850, QA Records for AFW Flow Orifice(s)
CR 369676, AFW Pump Impeller Documentation Non-availability
CR 370045, CST Capacity Curves Are Inaccurate