UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW, SUITE 23T85
ATLANTA, GEORGIA 30303-8931

April 21, 2008

Mr. David A. Baxter
Site Vice President
Duke Power Company, LLC
d/b/a Duke Energy Carolinas, LLC
Oconee Nuclear Station
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION - NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000269/2008006, 05000270/2008006, AND 05000287/2008006

Dear Mr. Baxter:

On March 13, 2008, the U. S. Nuclear Regulatory Commission (NRC) completed an inspection at your Oconee Nuclear Station Units 1, 2 and 3. The enclosed inspection report documents the inspection findings which were discussed on March 13, 2008, with you 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 license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the inspectors identified two findings of very low safety significance (Green). These two findings were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they are entered into your corrective action program, the NRC is treating these findings as Non-Cited Violations (NCVs) consistent with Section VI.A.1 of the NRC's Enforcement Policy. If you contest any of these NCVs you should provide a response within 30 days of the date of this inspection report, with the bases for your denial, to the United States Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, U. S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Oconee Nuclear Station.

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/

Binoy B. Desai, Chief
Engineering Branch 1
Division of Reactor Safety

Docket Nos.: 50-269, 50-270, 50-287
License Nos.: DPR-38, DPR-47, DPR-55

cc: See Page 3

Enclosure: Inspection Report 05000269/2008006, 05000270/2008006, and 05000287/2008006
w/Attachment: Supplemental Information

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.:	50-269, 50-270, 50-287
License Nos.:	DPR-38, DPR-47, DPR-55
Report No.:	05000269/2008006, 05000270/2008006, 05000287/2008006
Licensee:	Duke Power Company LLC
Facility:	Oconee Nuclear Station, Units 1, 2 & 3
Location:	7800 Rochester Highway Seneca, SC 29672
Dates:	February 11 through March 13, 2008
Inspectors:	R. Moore, Lead Inspector S. Kobylarz , Contractor D. Jones, Senior Reactor Inspector B. Sherbin, Contractor C. Peabody, Reactor Inspector D. Mas Penaranda, Reactor Inspector
Approved by:	Binoy B. Desai, Chief Engineering Branch 1 Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

IR 05000269/2008006, 05000270/2008006, 05000287/2008006; 02/11/08 - 02/15/08, 02/25/08 – 02/29/08, 03/10/08 – 03/13/08, Oconee Nuclear Station, Units 1, 2 and 3; Component Design Bases Inspection.

This inspection was conducted by a team of six NRC inspectors, which included two NRC contract inspectors. Two Green findings were identified during this inspection and were classified as non-cited violations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using 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.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance involving a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion III, Design Control. Specifically, the licensee failed to verify the applicability of design basis information, related to critical vortex height to assure adequate low pressure injection (LPI) pump suction conditions, before translating that information into the shutdown operations procedure for draining the reactor coolant system.

  This finding is greater than minor because if left uncorrected, the finding would become a more significant safety concern. In particular, the station routinely uses older calculations, test information, and analyses to establish operator action or alarm set points, support operability determinations, or change the design of the plant. If the applicability of that information is not verified for the system configuration and conditions under review, the quality of that engineering product could be compromised, resulting in a significant safety concern. The finding was determined to be of very low significance, via Manual Chapter (MC) 0609, Appendix G, Attachment 1, Shutdown Operations Significance Determination Process (SDP), Phase 1 because it did not significantly degrade the station capability to recover decay heat removal. The cause of the finding is related to the cross-cutting area of problem identification and resolution, specifically with respect to corrective action, because the licensee did not thoroughly evaluate the previous similar finding in the 2006 Oconee Component Design Bases Inspection (CDBI) such that the resolution adequately addressed causes and extent of condition (MC 0305, aspect P.1.c). [Section 1R21.2.2]

- Green. The inspectors identified a finding of very low safety significance involving a NCV of 10 CFR 50, Appendix B, Criterion III, Design Control. Specifically, the licensee failed to establish measures to verify the design capability for local manual handwheel operation of the emergency feedwater (EFW) flow control air operated

- valves (AOVs). Local manual operation was an alternate method of controlling EFW flow specified in station emergency procedures.

  The finding is more than minor because it is associated with the design control 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. The finding was of very low safety significance since it did not result in a loss of system safety function. Specifically, the licensee performed a technical evaluation during the inspection which demonstrated that a plant operator would be able to successfully cycle the valves using the manual handwheel. [Section 1R21.2.6]

B. Licensee-Identified Violations

None

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Mitigating Systems and Barrier Integrity

1R21 Component Design Bases Inspection (71111.21)

.1 Inspection Sample Selection Process

The team selected risk significant components and 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 two or Birnbaum value greater than 1 X10E-6. The components selected were located within the several plant systems including safe shutdown facility (SSF) support systems, low pressure injection (LPI) for mid loop operations, reactor vessel level indication for mid loop operation, and electrical distribution. In addition to design operation capability, the seismic capability of components was reviewed. The sample selection included 20 components, four operator actions, and five operating experience items. Additionally, the team reviewed two modifications by performing activities identified in IP 71111.17, "Permanent Plant Modifications," Section 02.02.a. and IP 71111.02, "Evaluations of Changes, Tests, or Experiments."

The team performed a margin assessment and detailed review of the selected risk-significant components to verify that the design bases have been correctly implemented and maintained. This design margin assessment considered original design issues, margin reductions due to modification, or margin reductions identified as a result of material condition issues. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as failed performance test results, significant corrective action, repeated maintenance, maintenance rule (a)1 status, Regulatory Issue Summary 05-020 (formerly Generic Letter 91-18) conditions, NRC resident inspector input of problem equipment, system health reports, industry operating experience 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 are included in the following sections of the report.

.2 Results of Detailed Reviews

.2.1 Pressurizer Safety Valves 3RC-67, -68

a. Inspection Scope

The team reviewed the design basis documents (DBDs), updated final safety analysis report (UFSAR), and applicable plant drawings to identify the design requirements for the valves. The team examined the machinery history of Pressurizer Safety Valves 3RC-67 and 3RC-68 to verify that design bases have been maintained. The team reviewed plant test procedures and results to verify that established acceptance criteria were met. The team examined records and test data for both corrective and preventative maintenance, in-service testing (IST) trending, as well as reviewed applicable corrective actions to verify that potential degradation was being monitored and/or prevented. The team reviewed seismic qualification records and modifications to verify that equipment installation is consistent with seismic requirements.

b. Findings

No findings of significance were identified.

.2.2 LPI Pump (reduced inventory/mid-loop operations)

a. Inspection Scope

The team reviewed the design basis documents, UFSAR, and applicable plant calculations, evaluations, procedures, and drawings to identify the design basis requirements of the LPI pumps during reduced inventory/mid-loop operations. This included the review of the adequacy of net positive suction head (NPSH) and determination of critical minimum levels established to prevent the onset of vortex conditions when in the mid-loop configuration. Set points for operator actions and alarms were reviewed to verify that NPSH and vortex considerations were appropriately incorporated into these values. Operator records from the 2006 Unit 3 refueling outage mid-loop operations were reviewed to verify that system conditions were monitored and maintained consistent with design requirements. The team also conducted a field walk down of the LPI pump area to verify that the installed configuration is consistent with the design basis and plant drawings, observe material conditions, and verify that the seismic evaluation of the pump mounting bolts was consistent with the field installation.

The team reviewed the reliability, availability, and capability of the power supply to the LPI pump motor during mid-loop configuration. Also, the team reviewed the capability of the motor to support the design function of the pump. This included review of BHP requirements for the pump motor; testing of the motor; setting, testing and coordination of protection devices; vendor recommendations for motor maintenance; and material conditions. In addition, the team verified that the ambient conditions were consistence with the motor qualification. Also, the team review elementary diagrams to verify that the logic operations satisfied the design bases.

b. Findings

Introduction: The inspectors identified a finding of very low safety significance (Green) involving a non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion III, Design Control. Specifically, the licensee failed to verify the applicability of design basis information, related to critical vortex height to assure adequate low pressure injection (LPI) pump suction conditions, before translating that information into the shutdown operations procedure for draining the reactor coolant system.

Description: Operating procedure, OP/1A/1103/011, Draining and Nitrogen Purging RCS, Rev. 66, dated 5/18/2006, incorporated non-conservative information from Calculation OSC-4254, LPI NPSH Analysis of Saturated Conditions, dated 12/6/1991, to provide operators with pump flow versus reactor vessel level values that describe the onset of vortex conditions. At the onset of vortex conditions the reliability and capability of LPI pumps could be impacted. The use of non-conservative values did not provide assurance of the pumps' capability during RCS mid-loop conditions. The team concluded that the incorporation of the non-conservative information into the procedure was a design control deficiency in that it was an example of incorrect translation of the design basis into station operating procedures.

Calculation OSC-4254, provided two tables of LPI flow vs reactor vessel level to address the critical vortex height during mid-loop operation in which the suction source was the RCS hot leg. These tables were based on two different vortex correlations/equations. One was provided by the station Nuclear Steam Supply System Vendor, Babcock and Wilcox (B&W), based on test information for the Oconee Unit 3 mid-loop configuration. The other correlation was from Perry's Chemical Handbook, sixth edition, equation 5-157, credited to Norman G. McDuffie, based upon experiments for determination of vortex critical height in gas-liquid separators. The inspectors reviewed the results of the calculation, and determined that the critical submergence based on the McDuffie correlation was less conservative than the critical submergence based on the B&W correlation by approximately five inches of submergence for the flow rate of 3500 gallons per minute (gpm) applicable during mid-loop operation. The OSC-4254 calculation provided no justification for use of the McDuffie values to the Oconee mid-loop system conditions and configuration. In particular, there was no relationship established between the Oconee hot leg configuration and the configuration for a gas-liquid separator device; nor a relationship between the scale of the McDuffie physical model used to develop the correlation and the in-plant configuration scale. In contrast, the B&W test information was stated, in a reference used by the calculation, to be based on the in-plant configuration. The team concluded there was no justification established for use of the less conservative McDuffie vortex critical height values as design basis values and that based on the information provided in the calculation, the B&W values were appropriate for use as guidelines for pump protection.

A corrective action cross cutting aspect was identified with this design control finding. A similar issue was identified as a finding in the previous Oconee CDBI inspection and documented as NCV 05000269,270,287/2006005-01, Non-Conservative EOP Procedure Set Point for Operator Action to Accomplish BWST to RBES Swap Over on Low BWST Level. The previous performance deficiency resulted from the unjustified use of non-conservative values for the vortex critical height input in determination of operator action set points in station emergency procedures for emergency core cooling system (ECCS) pumps taking a suction from the borated water storage tank (BWST). The extent of condition review for the violation addressed only critical vortex determinations for pumps with tanks as a suction source. The adequacy of critical vortex analyses for RCS mid-loop configuration as a suction source for the LPI pump was not addressed. The corrective actions for the previous finding also did not address the apparent cause of the violation which was a design control deficiency in the use of unjustified design basis information in station procedures. The team also noted that the previous finding was identified in the licensee's corrective action program on 3/10/06 (PIP O-06-01374) and that the design control performance deficiency associated with the current finding occurred on 5/18/06.

In discussion, the licensee indicated that since the previous similar finding of the 2006 CDBI NCV was not identified as a safety concern and station testing demonstrated the emergency procedure setpoint values were adequate, the corrective action program did not require an extent of condition review. The team concluded that there was prior opportunity for the licensee to identify and address the translation of unjustified design basis information into the procedure used for mid-loop operation, OP/1/A/1103/011.

Analysis: The failure to verify the applicability of design basis information before translating that information into station operating procedures is identified as a performance deficiency. This finding is greater than minor because if left uncorrected, the finding would become a more significant safety concern. In particular, the station routinely uses older calculations, test information, and analyses to establish operator action or alarm set points, support operability determinations, or change the design of the plant. If the applicability of that information is not verified for the system configuration and conditions under review, the quality of that engineering product could be compromised resulting in a significant safety concern. This finding was reviewed for significance by use of Manual Chapter (MC) 0609, Appendix G, Attachment 1, Shutdown Operations Significance Determination Process (SDP), Phase 1, because the finding was associated with a procedure for shutdown operations. The finding was determined to be of very low significance because it did not significantly degrade the station capability to recover decay heat removal.

The cause of this finding was related to the cross-cutting area of problem identification and resolution, specifically with respect to corrective action, because the licensee did not thoroughly evaluate the previous similar finding in the 2006 Oconee CDBI such that the resolution adequately addressed causes and extent of condition (MC 0305, aspect P.1.c). This finding is entered into the licensee's corrective actions program as PIP 08-01265.

Enforcement: 10 CFR 50, Appendix B, Criterion III, Design Control, requires that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure appropriate quality standards are specified and included in design documents. Contrary to the above, the design basis was not correctly translated into procedure OP/1/A/1103/011, Draining and Nitrogen Purging RCS, Rev. 66, in that the critical vortex height values translated from OSC-4524, LPI NPSH Analysis at Saturated Conditions, dated 12/6/1991, were not verified as applicable to the Oconee reactor coolant system (RCS) mid-loop condition and configuration. The failure to verify the station specific applicability of the values was a result of inadequate quality standards in the development of design basis information (OSC-4524). Because this issue was of very low safety significance and it was entered into the licensee's corrective action program (PIP 08-01265), it is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000269,270,287/2008006-01, Inadequate Design Control in the Translation of Design Basis Information into Procedure for Draining and Nitrogen Purging the RCS.

.2.3 Letdown Storage Tank (LDST) (operating curves, pressure transmitters, tank seismic)

a. Inspection Scope

The team reviewed DBDs, UFSAR, and applicable plant calculations, evaluations, procedures, and drawings to identify the design basis requirements of the LDST. The seismic calculations and analysis for the tank were reviewed against the foundation drawings to verify the installation was consistent with the assumptions and design inputs to the seismic analysis. The operating pressure/temperature curves for the tank hydrogen (H2) overpressure were reviewed to verify the values were consistent with the applicable design criteria. The tank drawings were reviewed to verify that level instrument tap elevations were consistent with input values to calibration procedures and alarm set points determinations. The calibration documentation of the level transmitters and pressure transmitters was reviewed to verify the instrumentation accuracy was monitored and maintained. Equipment history for the instrumentation as indicated by calibration data, maintenance documentation was reviewed to verify that equipment problems were identified and resolved. Recent PIPs related to identified corrosion were reviewed to ensure seismic capability of the tanks was maintained.

b. Findings

No findings of significance were identified.

.2.4 SSF HVAC (chiller and air handling unit (AHU))

a. Inspection Scope

The team reviewed DBDs, UFSAR, and applicable plant drawings to identify the design requirements for the SSF heating venting and air conditioning (HVAC) equipment. The team examined the machinery history of the SSF Chiller and AHU to verify that design bases have been adequately implemented and maintained. The team examined records and test data for both corrective and preventative maintenance, as well as reviewed applicable corrective actions to verify that potential degradation was being monitored and/or prevented. The team reviewed vendor information for maintenance and operation to verify equipment design and operation was adequately meeting the design bases requirements. The team reviewed the seismic qualification and flood protection measures for these SSF components to verify that equipment will be protected in the event of design basis earthquake or flood events respectively. The team also conducted a field walk down of the SSF Chiller and AHU to verify that the installed configuration is consistent with the design basis and plant drawings, observe material conditions, and verify that required seismic and flood protection measures are in place.

b. Findings

No findings of significance were identified.

.2.5 Check Valves 3 FDW-346, -442 (SSF auxiliary service water inlet to the Steam Generators)

a. Inspection Scope

The team reviewed the DBDs, UFSAR, and applicable plant drawings to identify the design requirements for the check valves. The team examined the machinery history of valves to verify that the design bases have been adequately implemented and maintained. The team examined records and test data for both corrective and preventative maintenance, as well as reviewed applicable corrective actions to verify that potential degradation was being monitored and/or prevented and was in accordance with NRC and American Society of Mechanical Engineers (ASME) Code requirements. The team reviewed the seismic qualification of these check valves to verify that this equipment would be protected during a design-bases earthquake. The team also conducted a field walk down of 3-FDW-442 to verify that the installed configuration is consistent with the design bases and plant drawings.

b. Findings

No findings of significance were identified.

.2.6 AOVs FDW-315, 316 (EFW flow control valves)

a. Inspection Scope

The team reviewed the DBDs, UFSAR, drawings, and procedures to identify the design basis requirements for the valves. The valve testing procedures and valve specifications were reviewed to verify the design bases requirements, including worst case system and environmental conditions, were incorporated into the test acceptance criteria and equipment design. The adequacy and availability of the nitrogen back up source for valve operation was assessed. The equipment history of the valves, as indicated by related maintenance work documentation, system health reports, and PIPs, was reviewed to verify that the valves were adequately maintained and that identified equipment problems were resolved. The performance test documentation was reviewed to assure that the equipment capability was monitored and maintained. The seismic qualification reports for the valves and back up nitrogen supply system were reviewed. The team reviewed the capability for local manual hand-wheel operation of the valves. A walk down was performed to assess the observable material condition, to verify the valves were accessible for local manual operation, and to verify the installed configuration was consistent with the design bases and plant drawings.

b. Findings

Introduction: The inspectors identified a finding of very low safety significance (Green) involving a NCV of 10 CFR 50, Appendix B, Criterion III, Design Control. Specifically, the licensee failed to establish measures to verify the design capability for local manual handwheel operation of the EFW flow control AOVs. Local manual operation was an alternate method of controlling EFW flow specified in station emergency procedures.

Description: The team identified that the licensee did not have an analysis or a testing program to demonstrate the capability of an operator to take local manual control of EFW flow control valves, EFW-315 and 316, following a loss-of-instrument air event. The valves were designed to fail open upon a loss of air pressure. The action to take local manual control of these valves to maintain steam generator water level was procedurally driven by the Emergency Operating Procedures, described in the UFSAR and credited in the probabilistic risk assessment documents. In addition to the normal air supply, the valves have a back-up bottled nitrogen supply system designed to provide two hours of nitrogen for valve operation. However, after the backup nitrogen supply is depleted, local manual operator action was necessary to continue plant cooldown. This would require an operator to manually close the valves to prevent excess emergency feedwater flow to the steam generators, or to throttle the valves to allow control of steam generator level. During a walkdown, the team observed that the handwheel diameter seemed small (less than 9 inches) for a valve that required manual repositioning during an event. Typically, the diameter of a manual valve operator handwheel is significantly larger than those installed on the EFW 315 and 316 valves where there is a significant differential pressure that the valve needs to operate against.

The licensee performed a technical evaluation to determine the capability for local manual handwheel operation of the valve. The licensee determined that the rim force required to turn the handwheel could be applied by an operator during manual operation at worst case system pressure conditions. The team reviewed the licensee's evaluation and concluded that it established adequate verification for the capability of local manual handwheel operation of the EFW flow control valves.

Analysis: The performance deficiency associated with this finding was that the licensee did not establish adequate design control measures, via technical analysis or a testing program, to demonstrate the local manual handwheel operations of the EFW flow control AOVs. Local manual operation was an alternate EFW flow control method specified in the emergency procedures, including EP/1/A/1700/008, Loss of Control Room, Rev. 12. The finding is more than minor because it is associated with the design control attribute of the Mitigating System Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of the EFW system to perform its intended safety function in the local manual mode of operation. The team assessed this finding and determined that it was of very low safety significance (Green) since it did not result in a loss of system safety function. This finding was reviewed for cross-cutting aspects and none were identified since the performance deficiency is not indicative of current licensee performance.

Enforcement: 10 CFR 50 Appendix B, Criterion III, Design Control, requires, in part, that design control measures shall be provided for verifying or checking the adequacy of design via simplified calculation methods, or by the performance of a suitable testing program. Contrary to the above, the adequacy of the design function for manually operating the EFW 315 and 316 valves had not been verified or checked through simplified calculation methods or by the performance of a suitable testing program. Because the finding was of very low safety significance and has been entered into the licensee's corrective action program (PIP 08-01062), this violation is being treated as a non-cited violation (NCV), consistent with Section VI.A of the NRC Enforcement Policy: NCV05000269,270,287/2008006-02, Inadequate Verification of Local Manual Operating Capability for EFW Flow Control Valves.

.2.7 Motor Operated Valves (MOVs) 3SF-97 (spent fuel pool to reactor coolant makeup pump), 3V-186 (main condenser vacuum breaker), 3HP-23 (LDST outlet valve)

a. Inspection Scope

The team reviewed the DBDs, UFSAR, and applicable plant drawings to identify the design requirements for the MOVs. The team examined the machinery history of MOVs 3SF-97, 3V-186, and 3HP-23 to verify that design bases have been adequately implemented and maintained. The team reviewed plant test procedures and results to verify that established acceptance criteria were met. The team examined records and test data for both corrective and preventative maintenance, and reviewed applicable corrective actions to verify that potential degradation was being monitored and/or prevented. The team also conducted a field walkdown of these MOVs to verify that the installed configuration is consistent with the design basis and plant drawings.

The team reviewed design torque values for selected critical MOVs to verify that the safety functions (open/closed) were adequately tested assuming minimum voltage. Also, the team reviewed elementary diagrams to verify that the logic operations satisfied the licensing basis assumptions. The team reviewed testing, setting and coordination of protective devices, vendor recommendations for motor installation and maintenance, material conditions, environmental qualification, and a selected sample of corrective maintenance and installation work orders relates to protection devices.

b. Findings

No findings of significance were identified.

.2.8 Vital Battery Room Ventilation

a. Inspection Scope

The team reviewed the DBDs, UFSAR, and applicable plant drawings to identify the design requirements for the ventilation equipment. The team examined the machinery history of the vital battery room ventilation system to verify that design bases have been adequately implemented and maintained. The team examined records and test data for both corrective and preventative maintenance, and applicable corrective actions to verify that potential degradation was being monitored, prevented, and corrected. The team also conducted a field walkdown of the fans and dampers to verify that the installed configuration is consistent with the design bases and plant drawings.

b. Findings

No findings of significance were identified.

.2.9 Circulating Cooling Water (CCW) Condenser Coolers (seismic/flooding)

a. Inspection Scope

The team reviewed the as-built drawings, seismic qualification, and heat exchanger specifications to assess the structural stability of the condenser coolers and potential contribution to Turbine Building flooding. A walk down was performed to assess the observable material condition and verify that attached piping was mounted consistent with station drawings and specifications. Additionally the team reviewed the seismic qualification utility group (SQUG) documentation related to the coolers to verify that the seismic capability was properly maintained for the condenser cooler and attached piping.

b. Findings

No findings of significance were identified.

.2.10 SSF Seismic Capability – auxiliary service water pump, structure, HVAC, DC power distribution

a. Inspection Scope

The team reviewed the DBD, UFSAR, and system and component drawings to identify the seismic design basis requirements for the SSF structure and support equipment. Seismic design calculations were reviewed to ensure ground accelerations and equipment foundation loads were appropriately addressed. The structure and as-built equipment drawings were reviewed to identify seismic critical aspects of the construction and equipment foundation configuration and performed walkdowns to verify the as-built condition was consistent with the seismic requirements. Additionally, the team assessed the potential impact of non-seismically qualified structures and equipment on SSF support equipment.

b. Findings

No findings of significance were identified.

.2.11 SSF Below Grade Penetrations - Flood Withstand Capability

a. Inspection Scope

The team reviewed the SSF building structural drawings to identify building penetrations that would potentially provide below grade water intrusion paths. The installed penetration barriers were assessed to determine if the barriers were installed consistent with the design and that the design was adequate to provide a barrier to standing water or flood conditions. A building walkdown was performed to verify there were no water intrusion paths which were not identified in the drawings and to verify that installed barriers were intact and capable of preventing water intrusion. The operator rounds procedure was reviewed to ensure incorporation of periodic inspection of building flood barriers. SSF building health report and recent PIPs were reviewed to ensure potential flood concerns have been identified and addressed.

b. Findings

No findings of significance were identified.

.2.12 4160 VAC Feeder Breakers (N1, S1, E1) – maintenance

a. Inspection Scope

The team reviewed the DBD, UFSAR, and elementary diagrams to identify the design bases for the feeder breakers. The maintenance and testing documentation was reviewed to verify that the reliability and capability of the breakers was maintained. Test and calibration documentation was reviewed to verify that the protection functions were monitored and maintained. The short circuit calculations were reviewed to verify that system duty requirements did not exceed breaker ratings. The adequacy and availability of breaker DC control power was reviewed to assess the reliability of the breaker control functions. Also, the calculation “DC Power Trip Breaker Coil” was reviewed to verify the reliability of the breaker functions to trip and close under worst case conditions. Equipment history as indicated by related maintenance documentation and PIPs was reviewed to verify that identified equipment problems were resolved. The adequacy and availability of breaker DC control power was assessed. Industry experience, related to 4160 VAC breaker failures, was reviewed to assess the applicability and potential for common cause failure for these station breakers.

b. Findings

No findings of significance were identified.

.2.13 SSF DC Distribution (batteries, inverters, seismic)

a. Inspection Scope

The team reviewed the design basis specification, UFSAR, general design criteria, vendor manuals, and drawings to identify the design basis requirements for the SSF batteries and inverters. Design calculations were reviewed to verify the adequacy of sizing for the batteries, inverters, and selected feeder cables. Surveillance testing documentation was reviewed to verify that the battery service and capacity testing was consistent with vendor recommendations, technical specification acceptance criteria, and the most limiting design conditions. Periodic maintenance procedures for cell and connection maintenance were reviewed to verify that vendor recommendations were incorporated. Equipment history as indicated by corrective maintenance documentation and PIPs was reviewed to verify that equipment problems were resolved. A field walk down of the DC system equipment was performed to assess observable material conditions, to verify equipment installation was consistent with seismic requirements, and to verify local environmental conditions were consistent with vendor equipment recommendations.

b. Findings

No findings of significance were identified.

.2.14 Breaker 3TE-10 (LPI pump C bkr)

a. Inspection Scope

The team reviewed the DBD, UFSAR, system drawings and elementary diagrams to identify the design basis requirements for the breaker. The breaker specifications were reviewed to verify the breaker ratings and protective functions were consistent with system conditions and operational requirements. The adequacy and availability of breaker DC control power was reviewed to assess the reliability of the breaker control functions. Also, the calculation "DC Power Trip Breaker Coil" was reviewed to verify the reliability of the breaker functions to trip and close under worst case condition. Surveillance testing and maintenance documentation was reviewed to verify that vendor recommendations were incorporated and breaker degradation was adequately monitored. Equipment history as indicated by corrective maintenance documentation, and PIPs was reviewed to verify that equipment problems were resolved. An equipment walkdown was performed to assess observable material conditions. The seismic qualification users group (SQUG) seismic qualification requirements and assumptions were reviewed to verify that current installation was consistent with the originally qualified configuration.

b. Findings

No findings of significance were identified.

.2.15 Switchgear 3TC (vital 4160 VAC)

a. Inspection Scope

The team reviewed the DBD, UFSAR, and one line diagrams to identify the design basis requirements for the switchgear. The switchgear protection scheme and calibration procedures were reviewed to verify proper coordination and translation of set point information into procedures. The voltage drop calculations were reviewed to verify that adequate voltage was available to the bus under worst case voltage conditions. Short circuit calculations were reviewed to verify that the equipment duty requirements did not exceed the equipment specifications. The breaker test program documentation was reviewed to verify the trip and close capability was monitored. The maintenance program was reviewed to assess the incorporation of industry experience and vendor manual recommendations. PIPs related to the switchgear and components were reviewed to verify that identified equipment problems were resolved. The team performed a non-intrusive visual inspection of the switchgear to assess visible material condition and vulnerability to hazards (flooding, seismic interactions, and missiles). The team reviewed seismic requirements for the switchgear and verified that related SQUG modifications were consistent with SQUG analysis assumptions.

b. Findings

No findings of significance were identified.

.2.16 Switchyard Batteries

a. Inspection Scope

The team also reviewed design and licensing basis documents, drawings, and vendor manuals to identify the design requirements and capability for the station switchyard batteries and support equipment. The design calculations were reviewed to verify the battery sizing and the adequacy of voltage and sizing of feeder cables at selected loads. Battery room environmental conditions, including temperature control and hydrogen removal capability were reviewed. Maintenance and surveillance test documentation for service and capacity testing and for the battery cell and connectors was reviewed to verify incorporation of vendor and industry experience recommendations and the capability to identify battery degradation. Corrective maintenance work documentation and PIPs were reviewed to assess the resolution of identified battery and support equipment problems. A field walk down was performed to assess observable material conditions and verify installation of equipment was consistent with vendor recommendations and SQUG analysis assumptions.

b. Findings

No findings of significance were identified.

.2.17 Reactor vessel level indication (red. inventory/mid loop ops)

a. Inspection Scope

The team reviewed design and licensing basis documents, the licensee response to GL 88-17, Loss of Decay Heat Removal (non power operations), drawings, and vendor manuals to identify the design requirements and capability for the reactor vessel level indication used during reduced inventory/mid-loop operation. The electrical drawings and component instrument loop/detail drawings were reviewed to verify the independence of level indication systems. The team reviewed related design basis information, reactor coolant system drawings, instrument tap elevations, procedures, and calibration documentation to verify that level set points for operator actions and alarms were consistent with the LPI pump operating requirements. Additionally, the team reviewed the incorporation of mid-loop vortex and NPSH considerations for the LPI pumps into set point values. The team reviewed the calibration of the LT-5A, 5B and ultrasonic level indication instrument loops to verify instrument accuracy was monitored and maintained. Maintenance history as indicated by work orders and PIPs was reviewed to verify the equipment maintenance was consistent with vendor recommendations and identified problems were resolved. The team also reviewed the calibration procedures for the various sensing and signal processing components that were installed in the system to verify that instrument uncertainty had been included.

b. Findings

No findings of significance were identified.

.2.18 RCS temperature indication (reduced inventory/mid loop operations)

a. Inspection Scope

The team reviewed design and licensing basis documents, the licensee response to GL 88-17, Loss of Decay Heat Removal (non power operations), and station procedures to identify the design requirements for the RCS temperature instrumentation used during reduced inventory/mid-loop operation. The team reviewed the connection diagram and the power supply drawings to verify independence of the core exit thermocouple instrumentation used to provide RCS temperature indication. Calibration and surveillance documentation were reviewed to verify that instrument accuracy was monitored and maintained. Equipment history, as indicated by component related maintenance work order documentation and PIPs, was reviewed to verify that equipment reliability and capability were maintained.

b. Findings

No findings of significance were identified.

.2.19 Upper Surge Tank and Main Condenser Hotwell Level Instrumentation

a. Inspection Scope

The team reviewed design and licensing basis documents, drawings, and vendor manuals to identify the design requirements and capability for the upper surge tank and hotwell level indication instrumentation. The calibration and periodic test procedures were reviewed to verify incorporation of applicable set points, uncertainty, and instrument loop overlap. Electrical drawings were reviewed to verify adequacy and availability of electrical power to the instrumentation for all plant conditions in which level indication was required. Equipment history as indicated by maintenance documentation, calibration data sheets, and PIPs was reviewed to verify that instrument performance capability was monitored and maintained. A field walk down was performed to assess observable material conditions.

b. Findings

No findings of significance were identified.

.2.20 Emergency Feedwater Sources (Upper Surge Tank / Hotwell)

a. Inspection Scope

The team examined the capability of the Upper Surge Tank (UST) and the Condenser Hotwell to provide a source of Emergency Feedwater. The team reviewed the Feedwater DBDs, UFSAR, and applicable plant drawings to identify plant design bases. The team examined calculations of water volume available in each of these sources, to verify that they were in agreement with plant design bases. The team reviewed seismic qualification documents to verify that this equipment would be protected in the event of an earthquake. The team examined corrective actions to verify that potential degradation was being monitored, prevented, and corrected. The team also conducted a field walkdown of the UST and Hotwell to verify that the installed configuration is consistent with the design basis and plant drawings.

b. Findings

No findings of significance were identified.

.3 Review of Low Margin Operator Actions

a. Inspection Scope

The team performed a margin assessment and detailed review of a sample of risk significant and time critical operator actions. Where possible, margins were determined by the review of the assumed design bases and safety analysis report response times and performance times documented by job performance measure results within operator time critical task verification tests. For the selected operator actions, the team performed a walk through of associated emergency procedures (EPs), abnormal procedures (APs) , annunciator response procedures, and other operations procedures with appropriate plant operators and engineers to assess operator knowledge level, adequacy of procedures, availability of special equipment when required, and the conditions under which the procedures would be performed. Detailed reviews were also conducted with risk assessment engineers, engineering safety analysts, training department leadership, and through observation and utilization of a simulator training period to further understand and assess the procedural rationale and approach to meeting the design bases and safety analysis report response and performance times. The following operator actions were reviewed:

- Provide long term water source for emergency feedwater pumps (Break main condenser vacuum – local manual, V-186 valve)
- Shutdown loss of coolant accident (incl. mitigation to gravity fill Rx vessel from BWST)

- Loss of offsite power during reduced inventory/mid-loop operation
- Provide alternate power supply to high pressure injection pump (HPI) pump (Tornado event)

b. Findings

No findings of significance were identified.

.4 Review of Industry Operating Experience

a. Inspection Scope

The team reviewed selected OE issues from domestic and foreign nuclear facilities for applicability at the Oconee Nuclear Station to determine the need for a detailed review. The issues that received a detailed review by the team included:

- IN 06-24, Recent Operating Experience Associated with Pressurizer and Main Steam Safety/relief Valve Lift Set Points
- RIS 06-23 Post Tornado Operability of Ventilating and Air Conditioning Systems Housed in Emergency Diesel Generator Rooms
- Loss of 400 Kv Switchyard and Two SR Essential Electrical Trains Because of CCF
- IN 06-31, Inadvertant Fault Interruption Rating of Breakers
- IN 07-09, Equipment Operability Under Degraded Voltage Conditions

b. Findings

No findings of significance were identified.

.5 Review of Permanent Plant Modifications

a. Inspection Scope

The team reviewed two modifications related to the selected risk significant components in detail to verify that the design bases, licensing bases, and performance capability of the components have not been degraded through modifications. The adequacy of design and post modification testing of these modifications was reviewed by performing activities identified in IP 71111.17, Permanent Plant Modifications," Section 02.02.a. Additionally, the team reviewed the modifications in accordance IP 71111.02, "Evaluations of Changes, Tests, or Experiments," to verify the licensee had appropriately evaluated them for 10 CFR 50.59 applicability. The following modifications were reviewed:

- Installation of 600 VAC Safety Related Motor Control Center
- Replace 3FDW-346 with Item No. DMV-568

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4AO6 Meetings, Including Exit

Exit Meeting Summary

On March 13, 2008, the team presented the inspection results to Mr. Baxter, and other members of the licensee staff. The team returned all proprietary information examined to the licensee. No proprietary information is documented in the report.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

K. Alter, Mechanical Balance of Plant Engineering Supervisor
S. Batson, Engineering Manager
D. Baxter, Site Vice President
D. Brewer – Safety Assessments Manager
E. Burchfield – Reactor and Electrical Systems Engineering Manager
C. Curry – Mechanical Civil Engineering Manager
B. Edge – I & C Engineering Supervisor
R. Freudenberger – Safety Assurance Manager
M. Glover – Station Manager
S. Thomas - Safety Analysis Engineering Supervisor

NRC

D. Rich, Senior Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Open and Closed

05000269,270,287/2008006-01	NCV	Inadequate Design Control in the Translation of Design Basis Information into Procedure for Draining and Nitrogen Purging the RCS (Section 1R21.2.2)
05000269,270,287/2008006-02	NCV	Inadequate Verification of Local Manual Operating Capability for EFW Flow Control Valves (Section 1R21.2.6)

DOCUMENTS REVIEWED

Calculations

OM 245-0847, Seismic Analysis of 6-inch 630lb Carbon Steel Swing Check Valve, Anchor/Darling Valve Company 1/18/1983
OM 245-1554 001, Seismic Analysis of 6-Inch Class 630 Weld Ends Carbon Steel Tilt Disc Check Valve, Anchor/Darling Valve Company 4/25/1989
OSC-2030, Standby Shutdown Facility HVAC Load Calculations, Rev. 12
OSC-5244, Turbine Driven Emergency Pump NPSH Analysis With Suction From the Hotwell, Rev. 2
OSC-6040, Outlier Resolution for Pressurizer Power Relief Valve RCV A0066 and Pressurizer Relief Valves RCVA0067 & 68
OSC-7214, Unit 3 Motor Driven Emergency Feedwater Pump NPSHa From the Hotwell, Rev. 5
OSC-8156, SSF-HVAC Calculations Using RT3, Rev. 0
OSC-8443, SQUG: Post 12/6/1999 SEWS and NARE Evaluations, Rev. 3
OSC-4478, Steam Generator Emergency Range Level Uncertainty and EFW Low Level, Rev. 6, 10/21/04
OSC-4701, DC Power for Breaker Trip Coil, 10/28/99
OSC- 125VDC Vital Instrumentation and Control Load Profile, Battery Sizing, and Voltage Analysis, Rev. 0
OSC-2061, Oconee U3 Voltage and Load Study, Rev. 18
OSC-7435, Hydrogen Gas Generation in the Station Rooms, Rev. 01
OSC-6040, Anchorage for 4160V Switchgear 3TC, 3TD, & 3TE, Rev.0
OSC-4300, Protective Relaying Settings, Rev. 10
OSC-1732, Design Modifications to Mitigate the Consequences of a Turbine Building Flood, Rev. 11
OSC-1558.005, Qualification of Wall No. HO-771.3-1617-AhO (DWG. O-382) Under USNRC IEB 80-11, Rev. 2
OSC-4522, Oconee Nuclear Station Unit 1, September 7,1991 Loss of Decay Heat Removal Event Thermal Evaluation, Rev. 0
OSC-1291, Standby Shutdown Facility Miscellaneous Platforms and HVAC Plenum Support Frame Analysis and Design, Rev. 13
OSC-318, 230 KV Switchyard Relay House-Seismic Anchorage of Battery Racks, Rev. 0
OSC-1146, SSF Cable Tray Grid System Design, Rev. 1
OSC-6040, Sheets 401 to 411, Seismic Anchorage for 4160V Switchgear 3TC, 3TD, &3TE, June 15, 1994
OSC-6040, Sheets 1115 to 1136, Seismic Evaluation for the Condensate Coolers, February 28,1995
OSC-6040, Sheets 1137 to 1173, Seismic Evaluation of Letdown Storage Tank (All 3 Units), November 22, 1994
OSC-1293, Standby Shutdown Facility Seismic Bracing For Cable Tray Support, Rev. 10 59047-OC-05-01, Sheet 174, SSF Raceway (All Floors, All Areas), Revision 2
OSC-7256, External Flood/Groundwater Mitigation Requirements, Rev. 1
OSC-3653, Gravity Flow from BWST to RCS After Loss of Decay Heat Removal, Rev. 0
OSC 4254, LPI NPSH Analysis of Saturated Conditions, dated 12/6/1991
OSC-4505, Letdown Storage Tank Pressure Loop Accuracy Calculation PT-10 and PT-0228, Rev. 8
OSC-4506, Letdown Storage Tank Level Instrumentation Accuracy Calculation for 1,2,3HPILT0033P1, 33P2, Rev. 8

OSC-4606, LDST Operating Curve-Maximum Allowable Pressure vs. Indicated Level, Rev. 8
OSC-2820, Emergency Procedure Setpoints, Rev. 27
OSC-7641, Mechanical Design Inputs for NSM-x3076, Rev. 5
OSC-7866, System Review of the Operation of Manual Valves, Rev. 11
OSC-2406, Piping Analysis for SSF Sump Pump Piping: Piping Analysis Problem Number 4-13-02, Rev. 4
OSC-7561, Differential Pressure Calculations for Valves FDW-0315 and FDW-0316
OSC-7562, FDW-315 and 316 Capability Evaluation, Rev. 1
OSC-3770, EFW Isolation During 10CFR50 Appendix R Event, Rev. 0
OSC-4021, BWST Level Required for Gravity Flow, Rev. 0
OSC-8443, Pages 530-533, Screening Evaluation Worksheet for EQID No. 0ELPLDCSF, 125 VDC Panel Board DCSF, 12/3/04
OSC-8443, Pages 534-537, Screening Evaluation Worksheet for EQID No. 0ELPLDCSF1, 125 VDC Distribution Center, 12/3/04
KC-2026-022, Seismic Evaluation for PCB Junction Boxes and Distribution Panels nd SY-DC1 & 2, Rev. 0
KC-2078, Anchorage Evaluation for MCC's SY-DC1 & SY-DC2 in the 230 KV Relay House (Outlier Resolution), Rev. 0
OSC-8580, Electrical Design Inputs for NSM ON-53118, Type IV, Rev. 1
OSC-6195, Oconee SSF DC System Voltage and Fault Current Analysis / Battery and Inverter Sizing, Rev. 6
OSC-4458, 230 kV Switchyard Battery Voltage Calculation, Rev. 4
OSC-6038, Condenser Hotwell Level Uncertainty Calculation Loop – ON1,2,3C LT0019A and P, Rev. 2
OSC-2248, Uncertainty Estimation for UST Level, Rev. 7
OSC-5964, EFW Combined Inventory (EFDW), Rev. 6
OSC-8111, Electrical Input Calculation for NSM ON-33092 (600/208V AC Load Capacity) (Type IV), Rev. 2

Operating Procedures

EP/3/A/1800/001, Blackout, Rev. 36
EP/1/A/1800/001, Emergency Operating Procedure, Rev. 36
EP/3/A/1800/001, Emergency Operating Procedure, Rev. 36
AP/2/A/1700/026, Loss of Decay Heat, Rev. 17
AP/1/A/1700/026, Loss of Decay Heat, Rev. 26
OP/1/A/1103/011, Draining and Nitrogen Purging RCS, Rev. 66
EP/1/A/1800/001, EOP-IMAs and SAs, Rev. 36
AP/1/A/1700/008, Loss of Control Room, Rev. 12

Procedures

SD 1.3.5, Shutdown Protection Plan, Rev. 22
PT/0/B/0610/009, Emergency Lighting, Rev. 13
OP/0/A/1102/026, Pre-Job Briefings, Rev. 15
OP/1/A/1103/011, Draining and Nitrogen Purging RCS, Rev. 70
EM/0/A/0050/001, Procedure to Provide Emergency Power to an HPI Pump Motor from the ASW Switchgear, Rev. 3
MP/0/A/3007/019, Air Handling Unit – SSF – Air Conditioning, Preventative Maintenance Safety Related System, Rev. 029

RE-3.03 Relaying – Motor Control Center Breaker and Overload Heater Selection – Oconee, Rev. 4
IP/0/A/3000/004, Instrumentation and Control Battery Equalizing Charge, Rev. 18
PT/3/A/0610/001 L, Load Shed Channel Verification, Rev. 8, Completed 10/28/07
PT/3/A/0610/017, Operability Test of 4160V Breakers, Rev. 23, Completed 02/04/08, 01/07/02
PT/3/A/0610/025, Electrical System Weekly Surveillance Unit 3, Rev. 19 Completed 01/24/08, 01/31/08
IP/O/A/2001/003C, Inspection and Maintenance of ITE Type HK Metal-Clad Switchgear, Associated Bus and Disconnects. Rev. 15
MP/0/A/1400/033, Flood Door-SSF Building-Presray-Semi-Annual And 5 Year-PM and Repair, Rev. 7
OP/0/A/1108/001, Curves and General Information, Rev. 73
OP/2/ A/1102/020 D, SSF and Outside Rounds, Rev. 25
PT/1/A/0600/001, Periodic Instrument Surveillance, Enclosure 13.13, Rev. 293
IP/0/A/3000/026, Battery Cell Connection Resistance Test, Rev. 028
IP/0/A/3000/15, 230 kV Switchyard Battery Service Test And Annual Surveillance, Rev. 036
IP/0/A/3000/023 A, 230 kV Switchyard Battery Performance Test, Rev. 019
IP/0/B/0200/027 A, Reactor Level Instrumentation and Calibration, Rev. 024
IP/0/A/0275/010 B, Condensate System Hotwell System Instrument Calibration, Rev. 042
IP/0/A/0275/010 L, Upper Surge Tank Level Instrument Calibration, Rev. 35

Completed Procedures

PT/3/A/0251/032, ASW Switchgear to HPI Pump Motor Power Path Test, Rev. 0, dated 12/2/07
PT/3/A/0251/024, HPI Full Flow Test, Rev. 30, dated 12/13/07
PT/0/A/0400/006, SSF HVAC Service Water Pump Test, Rev. 032, and performances dated 10/3/2007, 11/6/2007, and 11/7/2007
PT/0/A/0400/016, SSF HVAC System Flow Test, Rev. 010, and performance dated 4/4/2006 and 11/9/2006
PT/0/A/0400/016, SSF HVAC System Flow Test, Rev. 012, and performances dated 11/8/2007
TT/0/A/0170/022, SSF Air Conditioning Unit Performance Test, Rev. 001, and performances dated 7/11/2007, 9/18/2007, and 9/30/2007
PT/3/A/0152/011, High Pressure Injection System Valve Stroke Test, Rev. 021
PT/3/A/0152/019, Vacuum System Valve Stroke Test, Rev. 007
PT/3/A/0400/007, SSF RC Makeup Pump Test, Rev. 057
Wyle Laboratories – 1115, Testing of Dresser Model 31700 Safety Valves at Elevated Temperature for Duke Energy Company – Oconee Nuclear Station, Rev. D
PT/3/A/0600/018, Emergency Feedwater Train Operability Test, Rev 9, performed 5/27/06
PT/3/A/0150/022 M, 3FDW-315 and 3FDW316 Stroke Test, Rev. 31, performed 1/15/08
PT/3/A/0251/014, Feedwater Check Valve Functional Test, Rev. 7, performed 12/10/07
PT/3/A/0251/024, HPI Full Flow Test, Rev. 30, performed 12/1/07
PT/3/A/0600/029, 3FDW-315 and 316 Nitrogen Supply Leakage Test, Rev. 4, performed 8/27/07

Operations Training Related Documents

RCS Drain (CP-RCD) Lesson Plan, Rev.6
Loss of Decay Heat Removal (TA-DHR) Lesson Plan, Rev.6
OP-OC-EAP-LOSCM, Attachment 7 – Extended EFDW Operation, Rev. 3
OP-OC-CP-RCD, RCS Drain Lesson Plan, Rev. 6

Problem Investigation Program Reports (PIPs)

O-07-03773, ON3 SF VA 00503SF-50 has had recurring problems with the chain operator binding during operation
O-01-00294, SA-01-29(ON)(RA) Assessment of aspects of Oconee pressurizer code safety relief valve leakage
O-01-00796, Pressurizer Safety Valve 3RC-67 and 3RC-68 leakage increased such that a forst outage was required to replace them
O-01-01289, Discharge piping for 3RC-68 moved such that snubbers on this piping could not be reinstalled
O-02-03134, It can not be confirmed that Rotork NA1 electric actuators with Add On Packs (AOP) manufactured between 1978 and Oct. 2001 are the same AOP specifications as those originally tested and qualified at Wyle in 1978
O-03-02831, 3HP-23 Failed to Open During Stroke Test
O-04-03714, There is a potential concern with the Rotork AN1 electric valve actuators primary switch mechanism.
O-05-00731, NRC interpretation of tornado risk as described in the Oconee UFSAR doesn't align with Duke's interpretation
O-05-03088, Water Pumped from Unit 1 CST to Unit 1 UST at a Temperature of 165 F
O-05-03691, The following items were identified as not meeting the requirements of the SQUG GIP as a result of the SQUG Phase II walkdown effort
O-05-03770, This PIP tracks recommended actions from the Oconee SSF Risk Reduction Review
O-06-07876, UST Project –DELAY – Upper Surge Dome Tank Flange Face Connection "C" was Rejected for Pitting by QC During Surface Inspection
O-06-07681, AHU 2-31 Not Operating Properly – No Air Flow from Discharge of AHU
O-06-08735, ONS Evaluation of Regulatory Issue Summary (RIS) 2006-23, "Post tornado operability of ventilating and air-conditioning systems housed in emergency diesel generator rooms."
O-07-00776, Unit 2 Control Battery Room Exhaust Fan Damper Presenting Safety Hazard
O-07-04837, AHU 2-31 Failure – U2 Control Battery Room
O-07-04959, U3 RMCU Pump Suction Pressure
O-07-05069, Lead SSF HVAC Compressor #2 found off with lag compressor running – second occurrence
O-07-05200, AHU 2-31 Vibrating Excessively
O-07-06464, FME Found in 3B UST
O-07-05711, Potential Margin Issues with the Unit 3 AC electrical distribution System, 10/17/07
O-07-02605, 62BN2/2 time delay relay found with normal open contact terminal 1 to terminal 10 in closed position
O-07-02703, Breaker would not close from control room during TT/2/A/0610/03
O-07-00468, NRC Information Notice 21006-31
O-07-02141, NRC Information Notice 07-09 Equipment Operability Under Degraded Voltage Conditions
05-03370, SSF Risk Reduction
06-00529, Approximately one-half of Nitrogen Bottles to EFW Control Valves Leak
06-05522, I/P to 3FDW-315 Needs to be Calibrated
06-05375, MRFF on SSF Flood System
06-04101, Buried Portion of SSF ASW Pipe Tornado Missile Protection Defeated
06-00740, SSF Sewage System Vent Line Too Low

06-07569, PT Exams on Letdown Storage Tank Support Legs Revealed Rejectable Indications on 3 of 4 Legs
07-07326, Increased Stroke Time for 3FDW-316 from Closed to Open
07-00343, Seismic Qualification of Barrels at SSF Questioned by NRC Resident
07-02608, U2 Letdown Storage Tank Supports
O-06-08667, Loss of 400-kV switchyard and two safety-related electrical trains because of a common cause failure
O-07-03456, Security received low voltage alarm on DCSF
O-07-03211, Distribution center low voltage alarm
O-07-06529, Unexpected status alarms at SSF
O-06-00372, Received status alarms SA-17/B-3 and SA-5/A-1 (battery trouble)
O-06-02241, During Planned Testing, Cell 56 of Battery SY-2 Degraded to 1 Vdc
O-07-01416, Received status alarms SA-5 A-1 (SY-1 Batt Trouble) and SA-5 B-4 (SY-2 Batt Trouble) unexpectedly
O-07-01823, ICCM train A trouble status alarm alarming intermittently

Work Orders

91100490 01, Rep/Stop Leak Past Flange of 3FDW-346, 12/10/1991
92020780 01, Replace 3FDW-346 With Item No. DMV-568, 4/2/1992
95028842 01, PM Check Valve 3FDW-346, 4/24/1995
96036523 01, PM Check Valve 3FDW-442, 6/25/1996
98545829 01, PM Check Valve 3FDW-442, 5/24/2005
98661120 01, 3FDW-346 PM/Disassemble, Inspect, Verify Disc Movement, 10/25/2004
98684748 01, PM Check Valve 3FDW-442, 11/11/2004
98748455-01, PM Check Valve 3FDW-442, 5/7/2006
01634119-01, PM U3 Control Battery Rm. Exhaust Fan, 11/1/2004
01637722-01, U3, I/R Exhaust Fan Control Battery R, 10/13/2004
01663263-01, PM U3 Control Battery Rm. Exhaust Fan, 5/13/2006
01741564-01, PM Check Valve 3FDW-442, 11/5/2007
98748767, U-3, Emergency S/G Level Control Calibration, 05/22/06
98748768, U-3, Emergency S/G Level Transmitter Calibration, 05/19/06
98748769, U-3, Emergency FDW Level Control Valve Calibration, 05/19/06
01741535, U-3, Emergency S/G Level Control Calibration, 11/22/07
01741536, U-3, Emergency S/G Level Transmitter Calibration, 11/14/07
01741537, U-3, Emergency FDW Level Control Valve Calibration, 11/16/07
01640388, Test & Inspect 3C-LPI Pump Motor, 07/14/06
98194831, MOV 3SF-097 Performance Test, 05/07/00
98278308, MOV 3SF-097 Performance Test, 10/31/2004
01708452, 3FDW-CV-0315 Perform Diagnostic Test, 08/30/06
01646312, 3SF-97 Repair Electrical Connector on Operator,
98660344, PM 3TC-1 4160V Breaker & Relays (Bus 1), 10/28/04
98208614, PM Relays in Switchgear 3TC 1 (Bus 1), 08/15/00
98660374, PM 3TC-14 4160V Breaker & Relays (Bus 1), 10/28/04
98208620, PM Relays in Switchgear 3TC 14 (Bus 1), 09/18/00
01741255, PM/Test EPSLP #1 Relays, 12/02/07
98748101, PM Test EPSLP #1 Relays, 11/05/06
01741256, PM/Test EPSLP #2 Relays, 11/23/07
98748103, PM Test EPSLP #2 Relays, 05/13/06
01647461, I/R 3HP-07 Seat Leak, 06/03/07

01708452, 3FDW-315 Perform Diagnostic Test, 08/29/2006
01787208, 3FDW-316 I/R Increased Stroke Time
98660012, U3 HPI LDST Level Instrument Calibration, 2/15/05
98665577, U3 HPI LDST Temperature Instrument Calibration, 5/29/05
98619142 04, U0, Performance Test on Battery DCSFS, 9/20/05
98619143 04, U0, Performance Test on Battery DCSF, 4/10/05
01707878 01, U0, SSF DCSF Annual Battery Service Test, 7/17/07
98754911 01, U0, SSF DCSF Annual Battery Service Test, 7/3/06
86761042 01, U0, SSF DCSFS Annual Battery Service Test, 8/16/06
01711508 03, U0, SSF DCSFS Annual Battery Service Test, 10/22/07

Drawings

OP-OC-EL-EPD-2, Electrical Distribution, dated 8/7/04
2844, Auxiliary Building Lighting Layout Penetration Room, Rev. 20
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08-01054	Seismic qualification of 4160 VAC swgr area (doors and breaker lift truck)
08-01062	EFW flow control valves (FDW-315,-316) manual operation capability
08-01032	N2 consumption calculation for FDW 315, 316 discrepancies

08-01061	Seismic discrepancies swgr 3TC, 3TD, 3T3 door fasteners loose
08-01168	SQUG of LDST not include slosh factor in seismic analysis
08-01265	LPI pump mid loop suction conditions – vortex
08-01273	4 KV DBD incorrect reference to EQ manual
08-01265	LPI pump suction conditions for mid-loop