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

December 12, 2000

EA-00-266

Duke Energy Corporation
ATTN: Mr. W. R. McCollum
Vice President
Oconee Site
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION - NRC INSPECTION REPORT NOS. 50-269/00-12, 50-270/00-12, AND 50-287/00-12

Dear Mr. McCollum:

On November 3, 2000, the NRC completed a Safety System Design Inspection at your Oconee Nuclear Station Units 1, 2, and 3. The enclosed report documents the inspection findings which were discussed with you and other members of your staff on November 2, 2000.

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. The inspection found that engineering activities generally supported the safe and reliable operation of the standby shutdown facility and the station auxiliary service water system.
Based on the results of this inspection, one issue was identified and is documented as a non-cited violation in accordance with Section VI.A.1 of the NRC Enforcement Policy. The issue did not enter the significance determination process (SDP) because it did not impact a cornerstone. Since it did not enter the SDP, the issue was assigned no color. This issue has been entered into your corrective action program and is discussed in the attached inspection report. If you deny this non-cited violation, you should provide a response with the basis for denial, within 30 days of the date of this inspection report, 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, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Oconee Nuclear Station.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure 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/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/
Charles R. Ogle, Chief
Engineering Branch
Division of Reactor Safety

Docket No: 50-269, 50-270, 50-287
License No: DPR-38, DPR-47, DPR-55

Enclosure: NRC Inspection Report w/Attachment

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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 Nos.:	50-269/00-12, 50-270/00-12, 50-287/00-12
Licensee:	Duke Energy Corporation
Facility:	Oconee Nuclear Station, Units 1, 2, and 3
Location:	7812B Rochester Highway Seneca, SC 29672
Dates:	October 16 - 20 and October 30 - November 3, 2000
Lead Inspector:	R. Schin, Senior Reactor Inspector Engineering Branch Division of Reactor Safety
Team Inspectors:	J. Lenahan, Senior Reactor Inspector P. Fillion, Reactor Inspector R. Moore, Reactor Inspector D. Billings, Resident Inspector
Approved By:	C. Ogle, Chief Engineering Branch Division of Reactor Safety

Enclosure

SUMMARY OF FINDINGS

Oconee Nuclear Station
NRC Inspection Report Nos. 50-269/00-12, 50-270/00-12, and 50-287/00-12

<u>ADAMS TEMPLATE</u>

IR 05000269-00-12, 05000270-00-12, and 05000287-00-12, on 10/16-20 and 10/30-11/3/00, Duke Energy Corporation, Oconee Nuclear Station. Engineering inspection of the standby shutdown facility and station auxiliary service water system. One finding (no color).

This Safety System Design Inspection was conducted by a regional team. The inspection found that engineering activities generally supported the safe and reliable operation of the standby shutdown facility and the station auxiliary service water system. The inspection identified one finding, which is documented as a non-cited violation in accordance with Section VI.A.1 of the NRC Enforcement Policy. The finding did not enter the significance determination process (SDP) because it did not impact a cornerstone. Since it did not enter the SDP, the finding was assigned no color. The Attachment to this report describes the NRC's Revised Reactor Oversight Process.

Cornerstone: Mitigating Systems

- No Color. The inspectors identified a non-cited violation for failure to update the Updated Final Safety Evaluation Report and Technical Specification Bases to include standby shutdown facility equipment interdependencies that affect operability. (Section 1R21.141)

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R21 Safety System Design and Performance Capability (71111.21)

.1 SYSTEM NEEDS

.11 Energy Source

.111 Diesel Fuel Oil

a. Inspection Scope

The team reviewed calculations for diesel fuel consumption, storage tank volume, and transfer pump capacity to verify that the fuel oil system capacity was adequate to provide the fuel requirements to operate the standby shutdown facility (SSF) emergency diesel generator (EDG) for the period of time required for SSF dependent events. The acceptance criteria for fuel oil quality was reviewed to verify these were consistent with vendor recommendations.

b. Findings

No findings of significance were identified.

.112 Starting Air

a. Inspection Scope

The team reviewed the storage capacity for starting air and the air start motor capability documented in vendor manuals, calculations, and drawings to verify the air start capacity and capability was consistent with the licensing base assumptions for SSF EDG starting.

b. Findings

No findings of significance were identified.

.113 Electrical Power Source

a. Inspection Scope

The team reviewed design basis specifications (DBS), calculations of record, and industry standards for sizing of the SSF 125 volt (V) normal and standby batteries, associated chargers, and inverters KSF1 and KSF2. The team reviewed the SSF EDG loadings to verify that the loadings did not exceed the EDG nameplate ratings. The team also reviewed sources of power for the SSF instrumentation and for the station auxiliary service water (ASW) pump to confirm that power would be available during design basis events.

b. Findings

No findings of significance were identified.

.12 Controls

a. Inspection Scope

The team reviewed the control circuits for the following 4160 V circuit breakers at the SSF switchgear: Bus B2T/Bus OTS1 tie breaker, 600 V load center, EDG breaker, and ASW pump. The team also reviewed DBDs, calculations, and drawings to verify that the control circuits correctly incorporated various features such as manual and automatic close and trip logic, necessary permissive contacts from synchronism check relay and EDG speed, protective relay trip and lockout, space heater energization, and annunciators. The team also reviewed the SSF control circuits for Steam Generator 3A Emergency Feedwater Control Valve 3CCW-268, Reactor Coolant Makeup Supply Penetration Isolation Valve SSF-3SF-97, and the reactor coolant makeup pumps.

b. Findings

No findings of significance were identified.

.13 Operator Actions

a. Inspection Scope

The team reviewed operating procedures for mitigating the consequences of a tornado, flooding of the turbine building, seismic, and station blackout (SBO) events to verify that the procedures specified appropriate operator actions and that those actions could be performed in a timely manner commensurate with the significance of the actions. The operator actions were also reviewed for consistency with the accidents described in the Updated Final Safety Analysis Report (UFSAR), DBDs, and licensing basis documents. This review included Abnormal Procedures (APs), Emergency Procedures (EPs), and Job Performance Measures (JPMs). The team performed a walkdown of operator actions delineated in the APs and EPs for locally starting the SSF diesel generator; actions taken to line up the reactor coolant makeup (RCMU) pump for reactor coolant pump (RCP) seal cooling; and starting the SSF ASW pump to provide alternate feedwater for heat removal. The team also performed a walkdown of the control room instrumentation and alarms to verify that the appropriate indications and controls were available and adequate for operators to make the necessary decisions during performance of the specific APs and EPs.

The team also reviewed requirements for the SSF submersible pump and associated equipment, inspected the equipment, and walked through the setup and installation. The team verified that the SSF submersible pump could provide sufficient water flow from the lake to the condenser circulating water (CCW) piping to assure a sufficient heat sink and suction source for the SSF ASW pump and SSF service water system.

b. Findings

No findings of significance were identified.

.14 Heat Removal

.141 SSF System Interdependencies

a. Inspection Scope

The team reviewed the UFSAR, Technical Specifications (TS), TS Bases, DBS, Safety Evaluation Report (SER), and Probabilistic Risk Assessment (PRA) descriptions of the SSF to understand the intended safety functions of the SSF and how the SSF systems were designed and operated to ensure that they could perform these safety functions. The team also reviewed sections of the licensee's Individual Plant Examination for External Events (IPEEE) Report applicable to the SSF, including the seismic analysis of the elevated water storage tank (EWST) which provides a backup water supply for some SSF design conditions.

b. Findings

The team noted that the DBS described dependencies of the SSF EDG and the SSF RCMU system on the SSF ASW system that were not included in the PRA, UFSAR, TS, or TS Bases. They were also not included in the NRC SER on the SSF. The dependencies were:

1) The SSF ASW system provides water to operate an SSF service water system suction pipe air ejector. The air ejector is needed to remove air from the high point in the suction pipe so that siphon flow is maintained. If siphon flow is lost, the water supply to the SSF ASW pump and SSF EDG cooling systems will be lost. The SSF ASW system support function of operating the SSF service water system suction pipe air ejector to support operability of the SSF EDG was not described in the UFSAR or TS Bases.

2) The SSF ASW system provides secondary cooling water to the once-through steam generators (OTSGs) in events where normal and emergency feedwater are lost and effectively reduces reactor coolant system (RCS) pressure to below the 2500 psig setpoint of the pressurizer code safety valves. The SSF RCMU system depends on this RCS pressure reduction to ensure that it can supply sufficient flow to the RCP seals to prevent seal damage. The SSF ASW system support function of reducing RCS pressure to support operability of the SSF RCMU system was not described in the UFSAR or TS Bases.

In response to NRC questions, the licensee initiated Problem Identification Process (PIP) Report O-00-03683 to review these dependencies and the need to revise the UFSAR, TS, and PRA. In addition, the licensee issued an engineering communication to operators describing the SSF interdependencies and how they should affect operator use of TS action statements. The engineering communication advised that when the SSF ASW system was inoperable, operators should also declare the SSF EDG and the SSF RCMU system inoperable.

Also, in response to NRC questions, the licensee initiated PIP O-00-03725 to review current operability of the SSF RCMU system in view of the facts that the DBS analysis had not accounted for allowable setpoint drift of the pressurizer code safety valves and the Unit 1 RCP seals required full RCMU flow within 10 minutes to prevent seal damage. The Units 2 and 3 RCP seals could withstand a 20 minute loss of RCP seal cooling, but the Unit 1 RCP seals could withstand only a 10 minute loss of RCP seal cooling. (The Unit 1 RCPs had older design seal assemblies that included O-rings that were not qualified for high temperatures and pressures. This older design was a subject of NRC Generic Safety Issue 23, Reactor Coolant Pump Seal Failures.) By procedure, operators were instructed to start the SSF RCMU system within 10 minutes and then start the SSF ASW flow to the OTSGs within 14 minutes. However, during SSF design basis events, the OTSGs would go dry in about seven minutes (resulting in an immediate RCS pressure increase to the pressurizer code safety valve lifting point).

The licensee's operability evaluation concluded that the SSF RCMU system was currently operable. The design of the RCMU pump discharge relief valve, 1HP-404, was such that it could weep at an RCS pressure slightly below 2500 psig (the nominal pressurizer code safety valve setpoint) and reduce the RCMU flow to the RCP seals to below the minimum required. However, the licensee's operability review found that recent surveillance test information indicated that 1HP-404 had not weeped at RCS pressures below 2575 psig. Also, recent test results indicated that pressurizer code safety valve lifting pressures had been below the allowable 2575 psig (2500 psig setpoint plus 3% allowable setpoint drift). The licensee's plan to ensure long term operability involved replacement of the Unit 1 RCP seals (which could withstand only a 10 minute loss of cooling) with the type of seals that were installed in Units 2 and 3 (which could withstand a 20 minute loss of cooling). The licensee planned to accomplish the Unit 1 RCP seal replacements during November and December of this year.

The team reviewed the operating procedures for the SSF and noted that they dealt with the dependencies adequately to support current operability. The procedures required operators to start the EDG and then the ASW pump, before starting the RCMU pump. The procedures did not direct operators to operate the EDG without the ASW pump. Except for the time between 10 to 14 minutes after loss of normal RCP seal cooling, the procedures did not direct operators to operate the RCMU pump without ASW flow to the steam generators.

In addition, in response to NRC questions, the licensee reviewed the effect of these dependencies on the PRA and determined that they represented no increase in calculated core damage frequency. The PRA analysis was unaffected because of fundamental differences between PRA and design basis analyses. For example, PRA assumes that relief valves lift at nominal setpoints (not including allowable setpoint drift). Also, PRA credits methods that may not have been approved by the NRC or tested, such as the boiler-condenser mode of reactor core cooling and reverse gravity flow of lake water into the condenser circulating water (CCW) system.

Licensee engineers stated that these dependencies had been recognized in about the late 1980s, after the SSF had been approved by the NRC and placed in operation. The licensee had described the dependencies in engineering calculations and DBS, but had not described them in the UFSAR or TS Bases. 10 CFR 50.34 requires that the UFSAR description of systems be sufficient to permit understanding of the system designs and their relationship to safety evaluations. A knowledge of the support functions of the SSF ASW system, to support operability of the SSF EDG and SSF RCMU system, is necessary to permit understanding of the system designs and their relationship to safety evaluations. The team noted that other support functions necessary for SSF systems operability were described in the UFSAR and TS Bases; such as the portable pumping system supporting operability of the EDG and the ASW system, pressurizer heaters supporting operability of the ASW system, and air conditioning and ventilation systems supporting operability of the EDG.

The failure to describe these system design conditions in the UFSAR or TS Bases had a credible impact on safety in that it potentially could have lead to operability concerns or incorrect operator application of the TS. This issue did not enter the significance determination process (SDP) because it did not impact a cornerstone. Since it did not enter the SDP, the issue was assigned no color. However, there were related extenuating circumstances. This issue was related to an NRC Generic Safety Issue. Further, it could have impacted the NRC's ability to perform its regulatory function in that the information that was not provided to the NRC has some safety significance related to the design of the SSF. According to the PRA, the most important safety function of the SSF is to prevent RCP seal loss of coolant accidents (LOCAs) with the RCMU system. Since the SSF ASW system has approximately a 9% probability of failure (according to the PRA), reliance on it for RCMU system operability could potentially increase the probability of failure of the RCMU system.

10 CFR 50.71(e) requires that the UFSAR be updated to include the latest material developed. TS 5.5.15 requires that the licensee have a TS Bases control program that will ensure that the TS Bases are maintained consistent with the UFSAR. However, the licensee failed to update the UFSAR and the TS Bases to include the dependencies of the SSF EDG and the SSF RCMU system on the SSF ASW system. The licensee entered the issue into their corrective action system (PIPs O-00-03683 and O-00-03725) and took adequate immediate corrective actions. This Severity Level IV violation is being treated as an NCV, in accordance with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 50-269,270,287/00-12-01: Failure to Update the UFSAR and TS Bases to Include SSF Equipment Interdependencies That Affect Operability.

.142 SSF Internal Cooling

a. Inspection Scope

The team reviewed design documentation, equipment specifications, and system performance documentation to verify that the SSF diesel internal cooling system for the jacket water and lubricating oil were adequate to maintain the equipment operation within the vendor specifications. This included calculations for diesel heat load generation and specifications for engine heat exchangers.

b. Findings

No findings of significance were identified.

.143 SSF ASW System and Station ASW System

a. Inspection Scope

The team reviewed design documentation and original test reports to verify the water source adequacy and availability to provide the ultimate heat sink for steam generator feed and bleed used to achieve the SSF design function of plant hot standby conditions. Design documentation reviewed included decay heat calculations, system flow models and verification test information, net positive suction head (NPSH) analysis, and water source volume determinations.

b. Findings

No findings of significance were identified.

.144 Reactor Coolant Make-up System

a. Inspection Scope

The team reviewed calculations, analysis, drawings, and test documentation which verified the RCMU system adequacy and availability to meet the SSF design function of providing cooling water to the RCP seals during an SSF event.

b. Findings

No findings of significance were identified.

.145 Heating Ventilation and Air Conditioning (HVAC)

a. Inspection Scope

The team reviewed design documentation which identified ambient heat loads and related ventilation and cooling requirements for the SSF building and equipment. The vendor manuals and equipment specifications were reviewed to verify the ambient conditions were maintained consistent with the ranges specified for SSF equipment operation.

b. Findings

No findings of significance were identified.

.2 SYSTEM CONDITION AND CAPABILITY

.21 Installed Configuration

a. Inspection Scope

The team performed a general walk down inspection and examined electrical equipment in the SSF building to identify any degraded conditions, verify equipment labeling, and assess that the equipment was installed with a workable layout. A number of control switches, annunciator windows, and protective relay styles, and set points were compared to design drawings. The team also performed field walk downs of the SSF building equipment and the station ASW system to assess the material condition of mechanical equipment. Field verification included SSF diesel fuel oil and support systems to verify the configuration was consistent with drawings and design calculations. The team also reviewed seismic design requirements for the fire protection systems installed in the SSF and reviewed the sections of the licensee's Unresolved Safety Issue (USI) A46 Seismic Evaluation Report applicable to the SSF.

b. Findings

No findings of significance were identified.

.211 Instrument Set Points

a. Inspection Scope

The team reviewed the set points for the overcurrent relays at the SSF 4160 V switchgear. The review included the diesel generator backup relay (51V) and the ground fault protection. The accuracy of the current transformers used in the diesel generator differential scheme were reviewed to confirm that they matched. The team reviewed the overcurrent relay set points for the station ASW pump. The set points determined in the calculations were compared to the calibration procedures and verified by inspection of the relays where possible.

b. Findings

No findings of significance were identified.

.22 Operation

a. Inspection Scope

The team conducted a system walk down in the SSF building to verify that operation and system alignments of the SSF and associated interfacing and auxiliary support systems were consistent with the design and licensing basis (e.g., ASW, EDG fuel oil, lube oil, SSF building ventilation, and EDG starting air).

b. Findings

No findings of significance were identified.

.23 Design

a. Inspection Scope

The team reviewed the electrical one-line diagrams related to the SSF and the station ASW pump; the ampacity of the 4160 V and 600 V cables associated with the SSF; the SSF 600 V molded-case circuit breaker sizing and interrupting rating; and the SSF 125 VDC system battery, charger output, and main bus incoming breaker sizes. The team also verified that the SSF RCMU system components were included in the Equipment Qualification program as required. In addition, the team verified that the voltage drop in the 600 V, 1250-foot-long cable for the portable submersible pump was not excessive. Also, the team reviewed the pressurizer heater ground detection circuitry (Bank 2, Group B).

a. Findings

No findings of significance were identified.

.24 Testing

a. Inspection Scope

The team reviewed the SSF 125 VDC battery performance tests to confirm that the batteries had the required capacity. The team also witnessed performance testing of the SSF submersible pump, including the backup pump; the station ASW pump; and the SSF RCMU pump.

b. Findings

No findings of significance were identified.

.3 SELECTED COMPONENTS

.31 Component Inspection

a. Inspection Scope

The team reviewed equipment history, testing, and preventive maintenance procedures to assess the licensee's actions to verify and maintain the design functions, reliability, and availability of selected components. The selected components included the RCMU pump, SSF ASW pump, station ASW pump, SSF diesel, SSF HVAC equipment, motor operated valves CCW-268 and 287, and relief valves HP-304 and 404.

b. Findings

No findings of significance were identified.

.32 Component Degradation

a. Inspection Scope

The team reviewed procedures used by the licensee to ensure that RCMU system components inside containment were not degraded by washdowns which take place during refueling outages. The team also reviewed the controls and procedures associated with use of single cell chargers on the SSF batteries.

b. Findings

No findings of significance were identified.

.33 Design Dhanges

a. Inspection Scope

The team reviewed design changes to SSF equipment and the station ASW pump that were accomplished by the licensee design change program to verify that the system and equipment design function was appropriately evaluated and maintained.

b. Findings

No findings of significance were identified.

.34 Operating Experience

a. Inspection Scope

The team reviewed the licensee's evaluation for selected Information Notices (INs) issued by the NRC which describe industry experience related to various systems or components installed in the SSF. The INs reviewed related to the diesel generator, fire protection system, circuit breakers, control switches, testing and/or maintenance problems, and operational issues.

b. Findings

No findings of significance were identified.

.4 IDENTIFICATION AND RESOLUTION OF PROBLEMS

a. Inspection Scope

The team reviewed twelve PIP reports related to the SSF that had been initiated by the licensee prior to this inspection and ten that were written by the licensee as a result of this inspection, as listed in the Appendix to this report.

b. Findings

No findings of significance were identified.

.5 OPEN ITEMS REVIEWED

.51 (Closed) VIO 50-269,270/98-268-01012: Failure to Meet Technical Specifications and 10 CFR 50.46 for Long Term Cooling

This item was related to the incorrect swapover set point for the borated water storage tank level and was previously addressed in NRC Inspection Reports (IRs) 50-269,270,287/98-10 and 99-07. The item was open pending the NRC review of a sample of safety-related, risk significant, historical calculations that had been evaluated by the licensee's calculation enhancement project. A sample of five calculations was reviewed by the team. The sample demonstrated that the focus of the project was on design inputs and assumptions which directly addressed the area of weakness identified by the violation and related corrective action. This item is closed.

.52 (Closed) URI 50-269,270,287/98-03-09: Licensing Basis Issues With Single Failure and Quality Assurance (QA) for Non-Safety Equipment Relied on to Mitigate Design Basis Events

This URI had been opened for further NRC review of two concerns: 1) a potential that there may be many single failure vulnerabilities in the three Oconee Units, which may represent unrecognized risks and may be contrary to NRC requirements and UFSAR descriptions; and 2) a lack of quality assurance for non-safety components that were relied upon to mitigate design basis events. Subsequent NRC inspection, documented in IR 50-269,270,287/98-08, confirmed that there were many installed components (both non-safety and safety-related) in various systems whose single failure could disable the safety function of a system. That inspection also confirmed that there were many installed non-safety components (that were not in a QA program) that were relied upon to mitigate design basis events.

Single failure vulnerabilities of the emergency feedwater (EFW) system were subsequently addressed in IRs 50-269,270,287/98-15, 99-10, and 99-13; at a meeting with the licensee at the NRC headquarters on February 8, 1999; in a letter from the NRC to the licensee dated February 24, 1999; in LER 50-269,270,287/99-01, Emergency Feedwater Outside Design Basis due to Deficient Documentation; in a licensee Single Failure Analysis of the EFW system and related PIP O-99-03909, which identified and documented 37 EFW system single failure vulnerabilities; at a predecisional enforcement conference on April 25, 2000; and in enforcement actions documented in an NRC letter to the licensee dated May 9, 2000.

At the predecisional enforcement conference, the licensee stated that they would conduct single failure analyses of other safety systems. During this inspection, the team verified that the licensee had planned and contracted for single failure analyses of eight systems, to be completed by the end of this year. The eight systems included: penetration room ventilation, control room ventilation, core flood, high pressure injection, low pressure injection, reactor building spray, reactor building cooling units, and low pressure service water (including emergency condenser circulating water). At the time of this inspection, the licensee had completed one of these analyses, of the penetration room ventilation system, and had identified five single failure vulnerabilities. The team verified that these five single failure vulnerabilities were appropriately documented in the corrective action program, in PIP O-00-03532. The team judged that the licensee’s two completed single failure analyses, of EFW and penetration room ventilation, were thorough and effective in identifying single failure vulnerabilities. Based on the enforcement actions documented in the NRC letter of May 9, 2000; the licensee’s demonstrated ability to conduct single failure analyses; and the licensee’s plans to complete seven additional single failure analyses by the end of this year; this concern is closed.

The lack of quality assurance for non-safety components that were relied upon to mitigate accidents had been recognized by the NRC in the early 1990s, and a commitment for resolution had been made by the licensee. The resolution was to be a new Oconee Safety Related Designation Clarification (OSRDC) program. The OSRDC program was to identify all components relied upon to mitigate accidents and create a new quality assurance category of QA-5 for components that were not classified as safety-related (QA-1). The QA-5 components would then receive sufficient testing or maintenance to reasonably assure that they would function when called upon.

During this inspection, the team verified that the licensee had made substantial progress toward completing the OSRDC program. The licensee had made a computer listing of all components relied upon to mitigate accidents, including over 2000 that were not safety-related (and not in a QA program). Additionally, the licensee had completed the designation of all new QA-5 components and had new maintenance and testing procedures and schedules in place for some of the QA-5 components. The licensee planned to complete the new QA-5 maintenance and testing procedures by early next year. Based on the licensee’s demonstrated progress and plans to complete the OSRDC program by early next year, this concern is closed. Since both concerns are closed, URI 50-269,270,287/98-03-09 is closed.

4. OTHER ACTIVITIES

4OA6 Management Meetings

The Lead Inspector discussed the progress of the inspection with licensee representatives on a daily basis and presented the results to Mr. W. McCollum and other members of licensee management and staff on November 2, 2000. The licensee acknowledged the findings presented. The inspection was completed on November 3, 2000, with no additional results. Proprietary information is not included in this inspection report.

PARTIAL LIST OF PERSONS CONTACTED

Licensee:

L. Azzarello, Design Basis Engineering Manager
D. Brewer, Engineering Supervisor, PRA
E. Burchfield, Special Projects Engineering Supervisor
T. Curtis, Manager, Mechanical Systems/Equipment Engineering
J. Forbes, Manager, Oconee Nuclear Station
W. Foster, Manager, Safety Assurance
R. Freudenberger, Systems Engineering Supervisor
T. Geer, Manager, Civil, Electrical, Nuclear Engineering
K. Grayson, SSF System Engineer
W. McCollum, Vice-President, Oconee Nuclear Station
M. Nazar, Manager of Engineering
L. Nicholson, Manager, Regulatory Compliance
J. Weast, Senior Specialist, Regulatory Compliance

Other licensee employees contacted included engineers, operators, and administrative personnel.

NRC:

C. Casto, Director, Division of Reactor Safety
K. Landis, Chief, Reactor Projects Branch 5
M. Shannon, Senior Resident Inspector

ITEMS OPENED AND CLOSED

Opened

None

Opened and Closed

50-269,270,287/00-12-01	NCV	Failure to Update the UFSAR and TS Bases to Include SSF Equipment Interdependencies That Affect Operability (Section 1R21.141)

Closed

50-269,270/98-268-01012	VIO	Failure to Meet Technical Specifications and 10 CFR 50.46 for Long Term Cooling (Section 1R21.51)
50-269,270,287/98-03-09	URI	Licensing Basis Issues With Single Failure & Quality Assurance (QA) for Non-Safety Equipment Relied on to Mitigate Design Basis Events (Section 1R21.52)

APPENDIX

LIST OF DOCUMENTS REVIEWED

TECHNICAL SPECIFICATIONS (TS) AND SELECTED LICENSEE COMMITMENTS (SLC)

TS 3.10.1, Standby Shutdown Facility

TS Bases, Section B3.10.1, Standby Shutdown Facility

SLC 16.7.12, SSF Diesel Generator Air Start System Pressure Instrumentation

SLC 16.9.9, Auxiliary Service Water System and Main Steam Atmospheric Dump Valves

SLC 16.10.1, Condensate Inventory Requirements for Emergency Feedwater

SLC 16.10.3, Emergency Feedwater (EFW) Pump and Valve Testing

SLC 16.10.6, Emergency Feedwater Controls

SLC 16.10.7, Alternate Source of Emergency Feedwater (EFW)

SLC 16.13.1, Conduct of Operations, Minimum Station Manning

UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR)

UFSAR Section 3.2, Classification of Structures Components, and Systems

UFSAR Section 3.9.3.3, Design and Installation Details for Mounting of Pressure Relief Devices

UFSAR Section 9.2.3, Auxiliary Service Water System

UFSAR Section 9.6, Standby Shutdown Facility

UFSAR Section 10.4.7, Emergency Feedwater System

CALCULATIONS

OSC 1366, Relay Settings for SSF Facility and Related Equipment, Rev. 6, dated 5/28/99

OSC 2030, SSF HVAC Load Calculation, Rev. 10, dated 5/11/00

OSC 2218, SSF Diesel Engine Fuel Oil System - Tank and Vent Sizing, Rev. 5, dated 8/10/00

OSC 2236, SSF Diesel Engine Fuel Oil System Pumps, Piping, Filters, Strainers, and Relief Valves, Rev. 1, dated 9/23/94

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OSC 2284, Condenser Cooling Water System Piping Volume, Rev. 2, dated 8/22/95

OSC 2310, Minimum SSF ASW Flow Evaluation, Rev. 7, dated 7/31/95

OSC 2322, SSF Suction Supply Availability, Rev. 6, dated 8/4/99

OSC 2324, SSF Service Water Supply Submersible Pump Test, Rev. 2, dated 12/18/98

OSC 3233, SSF Service Water System Hydraulic Model - Analytical Model Type II, Rev. 7, dated 12/30/96

OSC 3767, Diesel Jacket Water Hx and Diesel Lube Oil Cooler Tube Plugging Evaluation, Rev. 1, dated 8/4/99

OSC 4171, SSF ASW Design Inputs, Rev. 19, dated 8/21/00

OSC 4535, SSF RC Makeup System Overpressure Protection Report (SSF) Type II, Rev. 9, dated 8/18/00

OSC 4652, USQ Review for NSM ON-52876, Replace ASW Pump Impeller, Rev. 1, dated 10/11/99

OSC 4993, ASW Overpressure Protection Analysis, Rev. 0, dated 11/10/92

OSC 5093, SSF Electrical Distribution System Load Flow, Voltage Adequacy and Fault Study, Attachment 5, Rev. 0, dated 3/27/95

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OSC 5484, SSF RC Letdown Line and RC Make up Bypass Line Minimum Flow Requirements, Rev. 5, dated 2/22/98

OSC 5505, Sizing of New SSF RC Letdown Orifice, Rev. 2, dated 3/17/98

OSC 5950, Oconee Units 1, 2, and 3, Decay Heat Predictions for 480 Day Cycles, Rev. 0, dated 3/30/95

OSC 6013, (Elect) Environmental Qualification of SSF Reactor Building Equipment Operating in SSF Head Vent Letdown and/or Quench Tank Disk Rupture Modes, Rev. 2, 10/18/99

OSC 6048, Miscellaneous IPEEE & Fire/Seismic Interaction Issues, Rev. 0, dated 12/18/97

OSC 6195, Oconee SSF DC System Voltage and Fault Current Analysis/Battery and Inverter Sizing, Rev. 1, 2/17/99

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OSC 6619, Maximum LPSW System Pressure, Rev. 1, dated 7/30/96

OSC 7299, High Energy Line Break, Rev. 0, dated 1/25/99

OSC 7460, Time Critical Operator Actions, Rev. 0, Dated September 25, 2000

OSC 4514, SSF Events, Rev. 07, dated 10/27/00

OSC 2262, Tornado Events, Rev. 03, dated 12/21/99

OSC 2820, Tornado Events, Rev. 19, dated 05/09/00

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KC Unit 1-2-2009, Keowee Hydro Units 1 and 2, WL System Water Hammer Evaluation, Rev. 0, dated 5/31/95

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AP/1/A/1700/011, Loss of Power, Rev. 25

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AP/1/A/1700/019, Loss of Main Feedwater, Rev. 12

AP/0/A/1700/025, Standby Shutdown Facility (SSF) Emergency Operating Procedure, Rev. 17

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EDM-103, Engineering Work Management, Rev. 1

EDM-130, Engineering Drawings, Rev. 8

EDM-170, Design Specifications, Rev. 8

EDM-190, Document Quality Guidelines, Rev. 2

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NSD-203, Operability, Rev. 14

NSD-208, Problem Investigation Process (PIP), Rev. 22

NSD-209, 10CFR50.59 Program Manual, Rev. 9

NSD-301, Nuclear Station Modifications, Rev. 20

<u>JOB PERFORMANCE MEASURES (JPM)</u>

JPM CRO-47, Activate the SSF, Rev. 10, dated 07/26/99

JPM CRO-48, Establish Flow to the Steam Generators from the SSF ASW System, Rev. 13, dated 04/23/99

JPM CRO-49, Establish Reactor Coolant Makeup Flow with the SSF RCMU System, Rev. 13, dated 04/23/99

JPM CRO-50, Emergency Start the SSF Diesel Generator and Power the SSF, Rev. 11, dated 04/23/99

JPM CRO-51, Activate the SSF to Include Establishing RCMU flow to the Reactor Coolant Pumps and ASW Flow to the Steam Generators, Rev. 08, dated 05/17/00

JPM CRO-52, Perform Required Actions in Preparation for Manning the SSF, Rev. 04, dated 04/23/99

JPM CRO-56, Establish RCMU Flow With SSF RCMU Pump and Maintain RCS Inventory, Rev. 01, dated 04/23/99

JPM NLO-001, Perform SSF Diesel Generator Post Startup Walkdown, Rev. 11, dated 15/17/00

JPM NLO-022, Align and Start the Station ASW Pump, Rev. 12, dated 05/17/00

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COMPLETED PROCEDURES AND TESTS

IP/O/A/0385/001J, Standby Shutdown Facility 125 VDC Battery Performance Test, performed on standby battery on 5/10/99

IP/O/A/0385/001J, Standby Shutdown Facility 125 VDC Battery Performance Test, performed on normal battery on 7/15/99

MP/0/A/5050/032, Preventive Maintenance on SSF Diesel 12 and 16 Cylinder Engines, Rev. 20, performed on 3/5-9/00

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PT/0/A/0400/015, SSF Submersible Pump Test, Rev. 8, performed on 6/3/98, and Rev. 10, performed on 10/17/00

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TT/2/A/0400/24, Flow Balance Test of SSF ASW Valves Using 2B Motor Driven EFW Pump, performed on11/9/94

TT/3/A/0400/24, Flow Balance Test of SSF ASW Valves Using 3B Motor Driven EFW Pump, performed on 7/8/94

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TT/3/A/0400/025, Unit 3 SSF RC Letdown Line Discharge Test, performed on 10/7/96

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TT/2/A/0400/28, SSF RC Make up Flow Distribution Test (Unit 2), Rev. 0, performed on 5/8/98

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TT/3/A/0600/13, B EFDW Header Flow Path to SSF ASW Pump Discharge Test, performed on 1/11/95

NUCLEAR STATION MODIFICATIONS

ONOE - 5573, PIP 093-0366 - Spent Fuel Pool Makeup for RCMU Pumps, dated 9/21/93

ONOE - 5989, Replace SSF Diesel Generator Fuel Oil Tank Level Gauge, dated 12/13/93

ONOE - 7302, Replace Valve 1CCW-269, dated 6/19/95

ONOE - 7314, Replace Valve 1FDW-347, dated 6/19/95

ONOE - 7370, Replace 1CCW-269 Operator, dated 5/11/99

ONOE - 10964, SSF Fuel Oil Tank Lo Level Alarm, dated 10/2/97

ONOE - 13034, Part AL1, Replace Operator Motor for Valve 1HP-20, Change Power Source from 208VAC to 600VAC, dated 11/10/98

ONOE - 13133, Revise SSF Submersible Pump Design Limits, dated 12/18/98

ONOE - 22460, Increase RCMU Pump Design Capacity, dated 5/12/85

ONOE - 52764, Install New SSF Emergency HVAC Service Water Pump Piping in Parallel with Existing ASW Pumps, dated 7/8/88

ONOE - 52792,Part BM1, Modify the SSF Aux Service Water (ASW) Pump and/or Piping & Hangers as Required by Performance Tests, dated 9/13/88

ONOE - 52792,Part BL1, Electrical Portion of SSF ASW Modification (Replace Transmitter OCCWFT0071 and Install Local Indication of Flow), dated 9/13/88

ONOE - 52991(M) Part AL1, Replace SSF THKM 1200 Breakers, dated 9/13/99

DRAWINGS

0-320-V, Yard Area Tanks - SSF Diesel Fuel Oil Storage Tank, Rev. 3, dated 8/11/83

0-374, Yard Structures Elevated Raw Water Storage Tank Foundation Concrete and Reinforcing, Rev. 0, dated 5/26/67

0-447E, Piping Layout SSF Fuel Oil, Rev. 2, dated 1/17/82

O-702-A1, One Line Diagram 6900V & 4160 V Station Auxiliary System, Rev. 14, dated 6/23/99

O-0703-K, One Line Diagram 600 V and 208 V Essential Motor Control Centers Auxiliary Power Systems Standby Shutdown Facility, Rev. 45, dated 5/3/00

O-0706, One Line Diagram Essential SSF 125 VDC Auxiliary Power Systems, Rev. 9, dated 9/2/00

0-1526, Auxiliary Bldg. Units 1 and 2, Section at Elevation 771' and 0" Concrete, Rev. 15, dated 9/17/95

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OFD-101A-1.1, Flow Diagram of HPI System (Letdown Section), Rev. 33, dated 9/28/99

OFD 101A-1.4, Flow Diagram of HPI System (Charging Section), Rev. 27, dated 8/2/99

OFD 101A-1.5, Flow Diagram of HPI System (SSF Portion) Rev. 16, Dated 6/23/99

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OFD 133A-2.5, Flow Diagram of Condenser Circulating Water System (SSF ASW Service), Rev. 36, dated 11/19/98

OEE-117-11, Elementary Diagram 4160 V Switchgear - B1T Unit-10 Auxiliary Service Water Swgr, Rev. 2, dated 6/4/90

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OEE-117-92-OA, Elementary Diagram SSF 4160 V Switchgear OTS1 Comp. No. 1, 4 kV Feeder Breaker, Rev. 2, dated 11/10/93

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OEE-117-92-OC, Elementary Diagram 4160 V Switchgear OTS1, Emerg. Pwr. Switching Logic Interposing Relays, Rev. 2, dated 2/28/86

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OEE-117-97, Elementary Diagram SSF 4160 V Switchgear OTS1 Heater Circuit, Rev. 0, dated 4/11/83

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OEE-149-8, Elementary Diagram SSF Press. Htr. Group B bank 2, Rev. 18, dated 10/16/97

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OEE-351-39, Elementary Diagram Reactor Coolant Makeup Pump Mtr. Controls, Rev. 2, dated 6/15/98

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Chicago Bridge and Iron Company Drawings, 100 MG Water Sphere (Elevated Water Storage Tank), Foundation Plan - R1, and General Plan - R5

VENDOR MANUALS

OM 254-0230, Installation, Operation and Maintenance Instructions for Anderson-Greenwood Safety Related Relief Valves (HP-404), dated 9/25/95

OM 235-0345-001, SSF Self Contained Air Conditioning Unit Specification, OS-235I, dated 8/3/99

OM 208-0046-001, Instructions and Parts List for Installation, Operation, and Maintenance of LP and LPO Paper Stock Pumps (ASW pump) Ingersol Rand Model 8X17LPO, dated 11/3/69

OM 254-0281-001, Maintenance Manual D and DB Series Lonegan Relief Valves, dated 9/19/95

OM 351-0164-001, SSF Diesel Generator Instruction Manual, dated 3/14/00

Asea Brown Boveri (ABB) Instruction Leaflet 41-116.1B, Type COV Voltage Controlled Overcurrent Relay, effective September 1999

Asea Brown Boveri (ABB) Instruction Leaflet 41-348.11B, Type SA-1 Generator Differential Relay for Class 1E Applications, effective August 1986, and revision 11C effective November 1999

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PROBLEM INVESTIGATION PROCESS (PIP) REPORTS

O-95-01353 - Problems with SSF DC System

O-96-00261, Relief Valve Set Pressure Outside Allowable Range, dated 2/7/96

O-97-03365, HP-304 Set Pressure Outside Allowable Range, dated 10/6/97

O-98-01585, Inadequate Cooling of SSF HVAC Due to Leaking Valve, dated 3/28/98

O-98-02521, SSF RCMU Pump Discharge Pressure Low, dated 5/9/98

O-98-04744, SSF HVAC Discharge Pressure High, dated 10/12/98

O-99-01122, RCMU Pump PT Acceptance Criteria Low for Unit 1, dated 3/24/99

O-99-01455, SSF HVAC Compressor #2 Cycling, dated 4/16/99

O-99-01569, SSF HVAC Compressor #1 High Suction Pressure, dated 4/24/99

O-99-03315, SSF HVAC Tripped During Maintenance, dated 8/14/99

O-99-03626, Inconsistent Values for Fuel Oil Quantities in EOPs and DBD, dated 9/8/99

O-00-00623 - Powering SSF from Unit 2 During SSF Design Event May Not be Viable Option for SSF Diesel Generator

ACTION ITEMS INITIATED BY LICENSEE AS A RESULT OF THIS INSPECTION

PIP O-00-03605 - Electrical Drawing Errors on O-0702-B

PIP O-00-03662 - Housekeeping/Material Condition Discrepancies in SSF Pump Room

PIP O-00-03667 - SSF ASW DBD Incorrectly Includes an Open Item That Should Have Been Removed

PIP O-00-03673 - SSF CO2 System Seismic Documentation Discrepancies

PIP O-00-03682 - SSF Submersible Pump Testing Procedure Adherence

PIP O-00-03683 - Interdependencies of SSF Equipment Not Described in UFSAR or TS

PIP O-00-03725 - Errors in RCS Pressure Used as Input to Calculation OSC-4535 to Verify Setpoint for 1HP-404

PIP O-00-03778 - Electrical Drawing Error on OEE-117-93-OB

PIP O-00-03815 - UFSAR Error: Incorrect Statement that ESV System was Not Installed on Unit 1

PIP O-00-03855 - Equipment Data Base Errors: Incorrect QA Condition for EFW Flow Control Valves FDW-315 & 316 Valve Operators, Manual Loaders, & E/P Converters

Work Request (WR) 98-152725 - Repair Broken Emergency Light in SSF

PRA Change Form ONS-SSF-R3-1 - Add a “Start” Failure Mode for the SSF #1 HVAC Compressor Train

PRA Change Form ONS-SSF-R3-2 - Add a Failure Mode for 1HP-304: “Relief Valve Fails to Reseat”

PRA Change Form ONS-SSF-R3-3 - Review the Need to Add Conditional Failure Modes for the Submersible Pump and the ASW Suction Pipe Air Ejector When Lake Levels are Too Low to Support Reverse Flow Into the CCW System

PRA Change Form ONS-SSF-R3-4 - Document Basis for Differences Between Several Conservative Design Basis Assumptions and the More Realistic Assumptions Used in the PRA

OPERATING EXPERIENCE FEEDBACK DOCUMENTS - LICENSEE'S EVALUATION OF NRC INFORMATION NOTICES

Evaluation of NRC Information Notice 96-23: Fires in Emergency Diesel Generators Exciters During Operation Following Undetected Fuse Blowing

Evaluation of NRC Information Notice 96-67: Vulnerability of Emergency Diesel Generators to Fuel Oil / Lube Oil Incompatibility

Evaluation of NRC Information Notice 97-08: Potential Failures of General Electric Magne-Blast Circuit Breaker Subcomponents

Evaluation of NRC Information Notice 97-21: Availability of Alternate AC Power Source Designed for Station Blackout Event

Evaluation of NRC Information Notice 97-41: Potentially Undersized Emergency Diesel Generators (EDG) Oil Coolers

Evaluation of NRC Information Notice 97-71: Inappropriate Use of 10 CFR 50.59 Regarding Reduced Seismic Criteria for Temporary Conditions

Evaluation of NRC Information Notice 97-81: Deficiencies in Failure Modes and Effects Analyses for Instrument and Control Systems

Evaluation of NRC Information Notice 98-03: Inadequate Verification of Overcurrent Trip Setpoints in Metal-Clad, Low-Voltage Circuit Breakers

Evaluation of NRC Information Notice 98-20: Problems with Emergency Preparedness Respiratory Protection Programs

Evaluation of NRC Information Notice 98-21: Potential Deficiency of Electrical Cable/Connection Systems

Evaluation of NRC Information Notice 98-38: Metal-Clad Circuit Breaker Maintenance Issues Identified by NRC Inspections

Evaluation of NRC Information Notice 98-43: Leaks in Emergency Diesel Generator Lubricating Oil and Jacket Cooling Water Piping

Evaluation of NRC Information Notice 99-05: Inadvertent Discharge of Carbon Dioxide Fire Protection System and Gas Migration

Evaluation of NRC Information Notice 99-07: Failed Fire Protection Deluge Valve and Potential Testing Deficiencies in Preaction Sprinkler Systems

Evaluation of NRC Information Notice 99-13: Insights from NRC Inspections of Low- and Medium-Voltage Circuit Breaker Maintenance Programs

MISCELLANEOUS DOCUMENTS

Specification OSS-0254.00-00-4016, Design Basis Specification for Flooding from External Sources, Rev. 0, dated 06/29/94

Specification OSS-0254.00-00-4010, Design Basis Specification for Seismic Design, Rev. 0, dated 06 /1/93

Specification OSS.0254.00-0004, Oconee Relief Valve Specification, Rev. 0

Generic Letter 89-13, Service Water System Problems Affecting Safety Related Equipment, dated July 18, 1989

Duke Power Initial and Supplemental Response Letters to the NRC related to GL 89-13, dated 1/26/90, 11/19/90, 5/13/92, 7/14/93, 3/16/94, and 9/30/96

Service Water System Program Manual, dated 8/16/96

Design Guide DG-3.12, Cable Ampacity Design Criteria, Rev. 0, dated 7/9/87

National Technical Systems Test Report 60446-95N, Report of the Flow Testing of Orifices for Duke Power Company, dated 6/29/94

Various cable installation data sheets

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Law Engineering Report titled “Seismic Fragility Curves for Jocassee Dam and Oconee Dikes,” dated 6/5/81

Structural Mechanics Associates report titled “Conditional Probabilities of Seismic Induced Failures for Structures and Components for Oconee Unit 3,” dated 9/81

Letter from the NRC to Duke Power Co. dated December 27, 1984, pertaining to an exemption related to the requirement for emergency lighting

Oconee Nuclear Station PRA Revision 2 Summary Report, dated December, 1996