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

October 29, 2001

EA-01-125

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

SUBJECT: OCONEE NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 50-269/01-03, 50-270/01-03, AND 50-287/01-03 AND INDEPENDENT SPENT FUEL STORAGE INSTALLATION INSPECTION REPORT 72-40/01-01

Dear Mr. McCollum:

On September 29, 2001, the NRC completed inspections at your Oconee Nuclear Station. The enclosed report documents the inspection findings which were discussed on October 4, 2001, with you and other members of your staff.

The inspection examined activities conducted under your licenses as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your licenses. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the inspectors identified four issues of very low safety significance (Green). Two of the issues were determined to involve violations of NRC requirements. However, because of their very low safety significance and because they have been entered into your corrective action program, the NRC is treating these issues as non-cited violations, in accordance with Section VI.A.1 of the NRC's Enforcement Policy. If you deny these non-cited violations, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the United States Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, with copies to the Regional Administrator, Region II; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, D.C. 20555-0001; and the NRC Resident Inspector at the Oconee facility.

Since September 11, 2001, your staff has assumed a heightened level of security based on a series of threat advisories issued by the NRC. Although the NRC is not aware of any specific threat against nuclear facilities, the heightened level of security was recommended for all nuclear power plants and is being maintained due to the uncertainty about the possibility of additional terrorist attacks. The steps recommended by the NRC included increased patrols, augmented security forces and capabilities, additional security posts, heightened coordination with local law enforcement and military authorities, and limited access of personnel and vehicles to the site.

The NRC continues to interact with the Intelligence Community and to communicate information to you and your staff. In addition, the NRC has monitored maintenance and other activities which could relate to the site's security posture.

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/ADMAS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Robert C. Haag, Chief
Reactor Projects Branch 1
Division of Reactor Projects

Docket Nos: 50-269, 50-270, 50-287, 72-40
License Nos: DPR-38, DPR-47, DPR-55

Enclosure: NRC Integrated Inspection Report 50-269,270,287/01-03 and 72-40/01-01

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-269, 50-270, 50-287, 72-40
License Nos:	DPR-38, DPR-47, DPR-55
Report No:	50-269/01-03, 50-270/01-03, 50-287/01-03, 72-40/01-01
Licensee:	Duke Energy Corporation
Facility:	Oconee Nuclear Station, Units 1, 2, and 3
Location:	7800 Rochester Highway Seneca, SC 29672
Dates:	July 1, 2001 - September 29, 2001
Inspectors:	M. Shannon, Senior Resident Inspector D. Billings, Resident Inspector E. Christnot, Resident Inspector S. Freeman, Resident Inspector R. Chou, Reactor Inspector (Sections 1R02, 1R17 and 4OA5.1) R. Gibbs, Senior Reactor Inspector (Sections 1R02 and 1R17) C. Rapp, Senior Project Engineer (Sections 1R02 and 1R17) W. Sartor, Senior Emergency Preparedness Inspector (Sections 1EP2, 1EP3, 1EP4, 1EP5 and 4OA1.2) S. Vias, Senior Reactor Inspector (Section 1R12.2)
Approved by:	R. Haag, Chief Reactor Projects Branch 1 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000269,270,287/01-03, on 07/01/2001 - 09/29/2001, Duke Energy Corporation, Oconee Nuclear Station, Units 1, 2, and 3: Event Response, Flood Protection Measures, Operability Evaluations, and Surveillance Testing.

The inspection was conducted by resident inspectors and regional based inspectors. The inspection identified four Green findings, two of which were non-cited violations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using the Significance Determination Process (SDP) found in Inspection Manual Chapter 0609. Findings to which the SDP does not apply are indicated by "no color" or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

A. Inspector Identified Findings

Cornerstone: Initiating Events

- Green. A finding was identified for the lack of a detailed engineering review for the generator disconnect switch modification and for inadequate modification monitoring/ testing. This resulted in the failure of the Unit 1 generator disconnect switch and a subsequent complicated reactor trip. Although the manufacturer informed the licensee that this was the first disconnect switch designed and built for 33,000 amp use, the licensee had not considered potential heating effects related to the isolated phase system and did not identify any post modification monitoring activities necessary to ensure proper in-service operation.

  Because the failure of the switch resulted in a reactor trip with a loss of normal heat removal capability, the lack of a detailed design review and post modification monitoring had an actual impact on plant safety. Based on the proper operation of the mitigation systems, this issue was considered to be of very low safety significance (Section 4OA3).

Cornerstone: Mitigation Systems

- Green. A finding was identified for improper scaffold installation that blocked the closure path for two condenser waterbox outlet valves on Unit 3. The ability for these valves to close is part of the turbine building flood mitigation strategy. This finding was considered to have a credible impact on plant safety because these valves are credited to close for mitigation of a turbine building flood.

  Based on a phase 2 screening performed by the Region II senior reactor analyst, which considered the failure of both valves to close, this issue was determined to be of very low safety significance. The duration of the improper scaffold installation and the availability of mitigating systems to respond to a turbine building flood were key considerations in the review (Section 1R06).

- Green. The inspectors identified a non-cited violation for failure to follow an engineering procedure, which resulted in exceeding the licensed reactor thermal power on Unit 2 for approximately 14 hours. Based on operation for greater than 12 hours above licensed reactor thermal power and for operating at a power level which reduced the 2 percent uncertainty margin assumed in the accident analysis, this finding had a credible impact on safety.

The inspectors concluded that because the reactor thermal power operation had not exceeded the 102 percent power assumed in the accident analysis, this issue had very low safety significance (Section 1R15).

- Green. A non-cited violation was identified for failure to meet the Technical Specifications (TS) surveillance requirements of SR 3.8.1.9.a for testing of the Keowee hydro units. Due to an overshoot problem related to governor control, the TS required frequency of 57-63 cycles in less than 23 seconds could not be achieved. The potential damage to safety related equipment that could result from an over-frequency condition on the Keowee hydro units had a credible impact on plant safety.

  The inspectors concluded that redundancy in equipment not initially loaded onto the electrical busses and other mitigation systems unaffected by the overshoot, provided core damage protection. Consequently, this issue was considered to be of very low safety significance (Section 1R22.2).

Report Details

Summary of Plant Status:

Unit 1 began the inspection period at 100 percent power and remained there through September 12, 2001 (except for brief periods of power reduction for control rod and main turbine valve testing). On September 12 the unit experienced a turbine trip/reactor trip due to a faulty generator isolated phase disconnect switch. The unit was returned to service on September 15 and power was returned to 100 percent at 12:01p.m. on September 17. At 8:41 p.m. power was reduced to 50 percent due to temperature concerns with the generator isolated phase disconnect switch. Power was increased to 92 percent on September 18. On September 24 power was returned to 100 percent following modification of the isolated phase bus duct cooling system and remained there at the end of the inspection period.

Unit 2 began the inspection period at 100 percent power and remained at 100 percent through September 17 (except for brief periods of power reduction for control rod and main turbine valve testing). Power was reduced to approximately 89 percent at 11:33 p.m. on September 17 due to potential over heating of the generator isolated phase disconnect switch. Power was increased to 92 percent on September 18. On September 29 power was reduced to 15 percent and the turbine generator was taken off line to remove the generator disconnect switches. The unit remained at 15 percent through the end of the inspection period.

Unit 3 operated at 100 percent power throughout the inspection period (except for brief periods of power reduction for control rod and main turbine valve testing).

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R02 Evaluations of Changes, Tests, or Experiments

a. Inspection Scope

This inspection was conducted to review implementation of the licensee's program for 10 CFR 50.59, Evaluations of Changes, Tests, or Experiments. The inspection was conducted by review of a sample of completed 10 CFR 50.59 safety evaluations performed by the licensee. The sample selected included evaluations from all three Reactor Safety cornerstones, and included the most risk significant items from a list of evaluations provided by the licensee. The sample also included evaluations from all site groups performing evaluations, and consisted of evaluations of plant modifications, procedure revisions, changes to the Updated Final Safety Analysis Report (UFSAR), tests, and non-routine operating configurations. The evaluations were reviewed to verify that the changes could be conducted by the licensee under the provisions of 10 CFR 50.59, without prior NRC approval. The sample included a total of twenty five evaluations, fourteen of which were screen outs. The documents reviewed are included in the list at the end of this report.

In addition, the inspectors reviewed a sample of problem investigation process reports to confirm that the licensee was identifying issues and initiating actions to resolve concerns.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment

.1 Partial System Walkdown

a. Inspection Scope

The inspectors conducted partial equipment alignment walkdowns to evaluate the operability of selected redundant trains or backup systems with the other train or system inoperable or out of service. The walkdowns included reviews of plant procedures and other documents to determine correct system lineups, and verification of critical components to identify any discrepancies which could affect operability of the redundant train or backup system. The following systems were included in this review:

- Circulating Water System Siphon Seal Water A Header
- Unit 3 A Train of Low Pressure Injection (LPI) and Reactor Building Spray (RBS) Systems
- Emergency power system during maintenance on the Standby Busses

b. Findings

No findings of significance were identified.

.2 Complete Walkdown of Standby Shutdown Facility (SSF)

a. Inspection Scope

The inspectors conducted a system walkdown on accessible portions of the SSF Auxiliary Service Water System and Power System. The inspectors focused on verifying adequate material condition and correct system alignment. Documents reviewed included: OP/O/A/1600/005, SSF Normal Power; OP/O/A/1600/007, SSF Diesel Air System; OP/O/A/1600/009, SSF Auxiliary Service Water System; OP/O/A/1600/010, Operation of the SSF Diesel Generator ; Technical Specifications; UFSAR; drawings OFD-133A-2.5, OFD-137D-1.1, OFD-137D-1.2 and OFD-138A-1.1; and Second Quarter 2001 SSF System Health Report. The inspectors also held discussions with the system engineer on temporary modifications, future modifications, and operator workarounds, to ensure that impact on the equipment functionality was properly evaluated.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

.1 Monthly Fire Protection Inspection

a. Inspection Scope

The inspectors conducted tours of selected areas to verify that combustibles and ignition sources were properly controlled, and that fire detection and suppression capabilities were intact. The inspectors selected the areas based on a review of the licensee's safe shutdown analysis and the probabilistic risk assessment based sensitivity studies for fire-related core damage accident sequences. Inspection of the following areas were conducted during this inspection period:

- Emergency Power Systems transformers CT-1, CT-2, CT-3, and CT-4
- Units 1 and 2 Turbine Driven Emergency Feedwater (TDEFW) Pumps
- Units 1, 2, and 3 Auxiliary Building Ventilation Rooms
- Fire Protection Equipment in Turbine Building Basement
- Unit 1 West Penetration Room
- Units 1, 2, and 3 Equipment Rooms

b. Findings

No findings of significance were identified.

.2 Fire Brigade Drill Performance

a. Inspection Scope

The inspectors observed a fire brigade drill on July 13, 2001. The simulated fire was in the Unit 2 control battery room, a plant area important to safety. The inspectors observed the fire brigade performance in terms of the following: (1) protective clothing and self contained breathing apparatus equipment worn at the scene; (2) adequate fire hose available and properly laid out; (3) correct use and implementation of appropriate fire fighting technics, including simulated smoke removal operations; (4) sufficient fire fighting equipment at the scene to perform fire fighting duties; (5) fire brigade leader's command and control; (6) communications between fire brigade members and with plant operators; (7) checking for fire victims and fire propagation into other plant areas; (8) utilization of pre-plan fire fighting strategies; and (9) implementation of the drill scenario and the drill objectives acceptance criteria. The inspectors attended the post fire drill critique.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

On July 16, 2001, the inspectors observed scaffold poles blocking the closure path of two condenser waterbox outlet valves. The inspectors compared this condition against the licensee's turbine building flood mitigation plans to determine if the condition was consistent with design requirements and risk analysis assumptions.

b. Findings

The inspectors identified a green finding for improper scaffold installation that blocked the closure path for two condenser waterbox outlet valves on Unit 3. These valves must be able to be closed in order to mitigate the consequences of a turbine building flood.

The inspectors noted that scaffold poles erected for ongoing material condition improvement activities, had been installed such that they would physically block the closing of valves 3CCW-21 and 3CCW-22. These valves need to be able to close in order to meet the licensee's flood mitigation commitments. After being informed of the condition, the licensee corrected the situation by repositioning the scaffold poles. The licensee chose not to evaluate whether or not the force of the actuator would be sufficient to break the scaffold poles and close the valves. Consequently, as part of their corrective action review, the licensee assumed 3CCW-21 and 3CCW-22 would not close for mitigation of a turbine building flood during the period of July 9, 2001, to July 19, 2001. This was the period of time the licensee concluded that the scaffold poles could have been in place.

Because of the potential to prevent turbine building flood mitigation, the inspectors concluded the improper scaffold installation had a credible impact on safety and reviewed this finding using the SDP for operations at power. Given the relatively short duration of this condition and the availability of mitigating systems to respond to a turbine building flood, the inspectors concluded this finding to be of very low safety significance (Green). This finding was described in the licensee's corrective action program as Problem Investigation Process (PIP) O-01-02697.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed simulator training on September 11, 2001, for reactor operators and senior reactor operators. The inspectors observed an equipment failure drill, a reactor trip response drill, and a training exercise using a new revision to the emergency operating procedure (EOP). The EOP training exercise included different scenarios involving anticipated transients without a scram. The inspectors looked for any deficiencies or discrepancies in the training and evaluated the operators' performance. Following the simulator scenario, the inspector observed the critique conducted by the training instructors to assess their ability in identifying operator and/or simulator performance deficiencies.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule (MR) Implementation

.1 Routine MR Implementation

a. Inspection Scope

The inspectors sampled portions of selected structures, systems, and components (SSCs) listed below to assess the licensee's implementation of the maintenance rule (10 CFR 50.65) and to determine the effectiveness of maintenance efforts that apply to scoped SSCs. Reviews focused on: (1) maintenance rule scoping in accordance with 10 CFR 50.65; (2) characterization of failed SSCs; (3) safety significance classifications; (4) 10 CFR 50.65 (a)(1) or (a)(2) classifications; and (5) the appropriateness of performance criteria for SSCs classified as (a)(2) or goals and corrective actions for SSCs classified as (a)(1). The selected SSCs were as follows:

- Flood Door between Auxiliary Building and Turbine Building
- Condenser Waterbox Outlet Valves 3CCW-21 and 3CCW-22
- Maintenance of the Westinghouse Type DB-50 breaker, used in the Keowee Hydro-Station (KHS) safety related switchgear
- Availability of the KHS Unit 2 following a failure to start during a surveillance test
- Unit 1 and 2 spent fuel pool cooling pump A seal leakage
- PIP O-01-3342, Maintenance rule function not defined for unit main turbine frequency indication, as needed for AP/3/A/1700/34, Generator Voltage and Electrical Grid Disturbances, Rev 0,

b. Findings

No findings of significance were identified.

.2 Review of Periodic MR Assessment

a. Inspection Scope

The inspectors reviewed the licensee's third periodic assessment, "Maintenance Rule Periodic Assessment for Maintenance Rule Implementation, Oconee Nuclear Station," dated June 5, 2001, for the period of July 1, 1999 - December 21, 2000. The assessment report was issued to satisfy paragraph (a)(3) of the Maintenance Rule 10 CFR 50.65. The inspectors verified that the assessment was issued in accordance with the time requirements of the Maintenance Rule and also that the assessment included all required areas including balancing, reliability and unavailability, review of a(1) activities, review of a(2) activities, and consideration of industry operating experience. The inspectors held discussions with system engineers for the LPI systems, Borated Water Storage Tank (BWST) system and Lee Combustion Plant (LCP) to discuss the MR program and their role in the gathering, tracking and analysis of the data. The inspection included review of the following documents:

- Maintenance Rule Periodic Assessment for Maintenance Rule Implementation, Oconee Nuclear Station, dated June 5, 2001, for the period of July 1, 1999 - December 21, 2000
- Nuclear System Directive 310, Requirements for the Maintenance Rule, Rev 7
- Engineering Directives Manual 210, Engineering Responsibilities for the Maintenance Rule, Rev 12
- Engineering Directives Manual 201, Engineering Support Program, Rev 5
- Maintenance Rule (a)(1) lists for 2nd Quarter 2001
- Maintenance Rule System Health Reports and various PIPs associated with the following systems: LPI, LCP, Chilled Water System (WC), and Refueling System (RFS)
- Expert Panel Meeting minutes for 3/8/01 and 11/20/00
- Oconee Nuclear Site - Three Site Engineering Functional Area Assessment (SA-99-22), 11/2/99
- DPC Assessment Report - Maintenance Rule, 2-O-MSE-013-99
- Maintenance Rule Unavailability Charts for the Lee Combustion System
- PIP Reports: O-01-02173; O-01-01148; O-01-0112; O-01-01403; O-00-03138; O-01-01285; O-01-00742; O-00-04418; O-01-00069; O-01-01586; O-99-05158; G-01-00185; O-01-01163; O-01-01204; O-01-01086; O-01-01631; O-01-01403; O-01-01339; O-01-01300; O-01-01277; O-01-01258; O-01-01257; O-01-01251; O-01-01222; O-01-01216; O-01-01208; O-00-00497; O-00-03958; O-00-00084; O-00-00473; O-00-03720; O-01-00958; and O-99-04999

- PIP Functional Failures: O-99-00018; O-99-00306; O-00-00461; O-00-00933; O-00-02141; O-99-03313; O-00-00462; O-00-02383; and O-99-03943

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluations

a. Inspection Scope

The inspectors evaluated the activities listed below to determine if: (1) risk assessments performed before the activities were accurate, complete, and in accordance with 10 CFR 50.65(a)(4); (2) the management of risk was in accordance with licensee procedures and preserved key safety functions; (3) upon identification of an unforseen situation, were the necessary steps taken to plan and control the resulting emergent work activities; and (4) problems associated with risk assessment and emergent work were adequately identified and resolved.

- Removal of Keowee Overhead Power Path from service while digging the new trench for the Keowee Underground Power Path
- WR 98190694, Trouble shooting for a DC ground on the 2A main feed water pump trip system affecting the high pressure trip
- WO 98356959 and WO 98247975, Work on 4160 volt switch gear 2TC breaker 6 and 2TD breaker 14, changes in the scheduling of work based on risk insights
- WO 98400905, Removal from service, of both the 1A reactor building spray pump and 1C low pressure injection pump
- WO 98390112, Removal from service of both sump pumps for the SSF to replace dislodged pipe from Catch Basin CB-43
- WO-97081820, Removal from service of the makeup water supply to the control room chillers affecting the control rooms for all three units
- PIP-O-01-3069, Time delays for relay targets set at 2.0 amps instead of .2 amps for 3TE, cubicle 10, power to low pressure injection pump 3C
- WO’s-98428057 and 98428069, Removal from service of the PCB-21 and 24, Units 1 and 2 generator output breakers, due to low gas pressures and the issuance of PIP’s O-01-3449 and 3450

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-Routine Evolutions

a. Inspection Scope

The inspectors evaluated the adequacy of: (1) personnel performance during selected non-routine events and/or transient operations; and (2) operator response after reactor trips that required more than routine expected operator responses, or which involved operator errors. As appropriate, the inspectors: (1) reviewed operator logs, plant computer data, or strip charts to determine what occurred and how the operators responded; (2) determined if operator responses were in accordance with the response required by procedures and training; (3) evaluated the occurrence and subsequent personnel response using the SDP; and (4) confirmed that personnel performance deficiencies were captured in the licensee’s corrective action program. The non-routine evolutions reviewed during this inspection period included the following:

- Unit 1 reactor trip on September 12, 2001, and subsequent post trip review on September 14, 2001, resulting from the main generator disconnect switch failure (discussed in Section 4OA3)

- Unit 1 reactor start up on September 15, 2001, following repairs to the main generator disconnect switch

- Unit1 power reduction to less than 50 percent on September 17, 2001, due to overheating concerns with the repaired main generator disconnect switches

- Unit 2 power reduction to 15 percent on September 29, 2001, for replacement of main generator disconnect switch and repair of reactor coolant system resistance temperature detector (RED)

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed selected operability evaluations affecting risk significant systems to assess (1) the technical adequacy of the evaluations; (2) whether continued system operability was warranted; (3) whether other existing degraded conditions were considered; (4) if compensatory measures were involved, whether the compensatory measures were in place, would the compensatory measures work as intended, and were appropriately controlled; and (5) where continued operability was considered unjustified, the impact on TS Limiting Condition for Operation (LCO). The inspectors reviewed the operability evaluations described in the following PIPs:

- PIP O-01-00381, Transformer CT4, Keowee Hydro Units (KHU) underground feeder to Oconee standby bus, operable with the loss of both forced air and forced oil cooling

- PIP O-01-02738, Issuance of WR 98182711 and WR 98182713 for verification of fuse sizes for the Unit 2 and 3 SSF reactor coolant makeup pump motors

- PIP O-01-02811, Depressing air (Service Air) system affecting the cooling water supply to both of the KHU generators

- PIP O-01-02807, Past operability of KHU Unit 2 following the failure of the field supply breaker

- PIP O-01-03069, Operability of 3C LPI pump following the improper setting of the time delay targets on the 50-51 X and Y relays in switchgear 3TE cubicle 10

- PIP O-01-00455, Operability of HPI system suction from spent fuel pool

- PIP O-01-02127, Unit 2 Precision Calorimetric Exceeded Operator Aid Computer (OAC) Calorimetric

b. Findings

A Green Finding was identified and dispositioned as a non-cited violation (NCV) for failure to follow an engineering procedure which resulted in exceeding the licensed reactor thermal power on Unit 2 for approximately 14 hours.

On June 1, 2001, during power escalation following a Unit 2 refueling outage, engineering personnel performed Procedure PT/2/A/0325/011, Data Collection For Secondary to Primary Heat Balance Correction, Revision 10. This procedure calculated core thermal power independent from the OAC using separate and more accurate instrumentation and software. The calculated results were 73.3 percent of rated thermal power. At the time, the OAC indicated 73 percent rated thermal power. Because the difference was less than the two percent margin of the accident analysis and within the uncertainty of the OAC, engineering personnel recommended to operations that the power escalation continue to 100 percent of rated thermal power. At 11:29 p.m. on June 1, 2001, Unit 2 reached 100 percent rated thermal power as indicated by the OAC. Subsequent completion of PT/2/A/0325/011 determined power to be at 100.6 percent of rated thermal power. At 1:20 p.m. on June 2, 2001, operators reduced power to 99.3 percent of rated as indicated on the OAC. Engineering personnel attempted to reconcile the differences and determined thermal power calculated by PT/2/A/0325/011 should be reduced by 0.2 percent. Engineering was unable to find any other problems with either the calculation by procedure or the OAC and subsequently resolved the issue by adjusting the fouling factor on the OAC which normalized it to the value calculated by PT/2/A/0325/011.

The licensee's operability evaluation stated that actual power level was between 99.4 percent and 100.6 percent rated thermal power as indicated by the OAC and between 100.1 percent and 100.7 percent as measured by PT/2/A/0325/011. This determination was based on the uncertainty of each calculation. The evaluation concluded the results were acceptable because both calculations were within the two percent margin assumed in the accident analysis.

The inspectors noted that PT/2/A/0325/011 requires core thermal power be restricted if the PT calculations determines power is greater than OAC indication following the thermal power calculation at 73 percent. The procedure was not followed in that power was not restricted. When questioned about exceeding the licensed power level, the licensee indicated that both power indications were acceptable because they were considered to be within the uncertainty analysis for each calculation.

The inspectors concluded that Unit 2 exceeded licensed power because the unit was operated at 100.4 percent of rated power as measured by Procedure PT/2/A/0325/011 for 13 hours and 50 minutes. After reviewing OAC traces of power level for the time in question the inspectors noted that average power remained constant during the 13 hours and 50 minutes reviewed. Small deviations above licensed rated thermal power are allowed by a NRC letter issued August 22, 1980, however the letter provided guidance that average power level over the operating shift (12 hours) must not exceed 100 percent of rated power. Therefore, the inspectors concluded that core thermal power exceeded that allowed by the license.

The inspectors reviewed this finding using the SDP for operations at power. The finding had a credible impact on safety. Core thermal power level is an initial condition for several of the accidents analyzed in the UFSAR and operating above 102 percent (highest assumed power at the beginning of an accident) could adversely effect emergency core cooling systems from accomplishing their intended function. For this finding the reactor was operated at a power level which reduced with 2 percent uncertainty margin assumed in the accident analysis. The finding was considered to be of very low safety significance (green) because power though out the time period in question remained below 102 percent of rated power.

Unit 2 Renewed Facility Operating License DPR-47 Paragraph 3.A authorizes operations at a steady state power level of 2568 megawatts thermal. TS Section 1.1 defines 100 percent rated power as 2568 megawatts thermal. TS 5.4.1 requires that procedures be implemented for activities recommended by Regulatory Guide 1.33. Regulatory Guide 1.33, Section 8.b.(1) w, establishes a requirement for procedures to perform heat balance calibrations. Procedure PT/2/A/0325/011, Data Collection for Secondary to Primary Heat Balance Correction, Section 12.8, requires notification of the unit shift supervisor to limit reactor power to less than 100 percent power minus the calculated error. Contrary to TS 5.4.1, for procedures recommended by Regulatory Guide 1.33, the licensee failed to follow engineering procedure PT/2/A/0325/011, Section 12.8, in that on July 1, 2001, the unit shift supervisor was not notified of the procedure requirement to limit reactor power. This resulted in Unit 2 operating in excess of its licensed rated thermal power (2568 megawatts thermal) for a period of 13 hours 50 minutes on June 1 and 2, 2001. This issue is being treated as a NCV, consistent with Section VI.A.1 and is identified as NRC Enforcement Policy NCV 50-270/01-03-01: Failure to Follow an Engineering Procedure Results in Unit 2 Exceeding Licensed Power Level. This violation has been captured in the licensee's corrective action program as PIP O-01-02127.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors performed a review of existing operator workarounds and assessed their cumulative impact on plant safety. Specifically, the inspectors reviewed the PIPs associated with the workarounds, interviewed operations personnel and reviewed Nuclear System Directive 506, Operator Workarounds, Revision 0, to determine if the existing workarounds affected reliability and availability of risk significant systems, increased the probability of an initiating event, or affected the operators' ability to respond to plant transients and accidents.

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors evaluated 10 modifications in three cornerstone areas, to verify that the modified systems' designs had not been degraded, and that the modifications had not left the plant in an unsafe condition. The inspectors verified inspection procedure components such as: energy requirements could be supplied by supporting systems; materials/replacement components were compatible with physical interfaces; replacement components were seismically qualified for the application; code and safety classification of replacement system, structures, and components were consistent with design bases; modification design assumptions were appropriate; post-modification testing would establish operability; failure modes introduced by the modification were bounded by existing analyses; and that appropriate procedures or procedure changes had been initiated.

The sample of Nuclear Station Modifications (NSMs) reviewed are listed at the end of this report. The inspectors also reviewed additional information as necessary such as applicable sections of the UFSAR, Technical Specifications, and procedures.

In addition, the inspectors reviewed a sample of problem investigation process reports to confirm that the licensee was identifying issues and initiating actions to resolve concerns.

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (PMT)

a. Inspection Scope

The inspectors reviewed PMT procedures and/or test activities for selected risk significant components or systems to assess whether: (1) the effect of testing on the plant had been adequately addressed by control room and/or engineering personnel; (2) testing was adequate for the maintenance performed; (3) acceptance criteria were clear and adequately demonstrated operational readiness consistent with design and licensing basis documents; (4) test instrumentation had current calibrations, range, and accuracy consistent with the application; (5) tests were performed as written with applicable prerequisites satisfied; (6) jumpers installed or leads lifted were properly controlled; (7) test equipment was removed following testing; and (8) equipment was returned to the status required to perform its safety function. The inspectors observed testing and/or reviewed test results for the following items:

- Valve testing specified in IP/0/A/3001/001, Limitorque Preventive Maintenance, Rev. 60, following preventive maintenance on building spray system valve 2-BS-1 valve operator

- PT/0/A/0620/09, Monthly KHS Operation, Rev. 20, following completion of WO 98431113, test and install KHS Unit 2 field supply breaker

- Post modification testing of the AHU-15 cooling water control valve, documented in modification package OE-16026, following replacement of the valve
- TT/2/A/0251/93, Testing of Unit-1&2 LPSW Auto-Start Circuitry, Rev 0, PMT for modification NSM-23802 to LPSW pumps A, B, and C

- TN/2/B/2121/01, Testing for modification ONTM 2121, Rev 0, chill water to Unit 1 generator isolated phase bus duct cooler

- PMT following installation of plant modification NSM ON-13026, Installation of Unit 1 Isolated Phase Bus Disconnect Switch (Issue discussed in Section 4OA3 of this report)

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

.1 Routine Surveillance Observations

a. Inspection Scope

The inspectors witnessed surveillance tests and/or reviewed test data of the selected risk-significant SSCs listed below, to assess whether the SSCs met TS, UFSAR, and licensee procedure requirements. In addition, the inspectors determined if the testing effectively demonstrated that the SSCs were ready and capable of performing their intended safety functions.

- PT/2/A/0204/007, 2BS Reactor Building Spray Pump Test, Rev. 55

- PT/1/A/0600/013, 1B Motor Driven Emergency Feedwater Pump Test, Rev. 42

- PT/0/A/0400/011, SSF Diesel Generator Performance, Rev. 9 and PT/0/A/0400/005, SSF Auxiliary Service Water Pump Test, Rev. 38

- PT/3/A/0150/022M, 3FDW-315 and 3FDW-316 Stroke Test, Rev. 17

- PT/3/A/0110/004, 3A Penetration Room Ventilation Filter Test, Rev. 4 and PT/3/A/0170/005, 3A Penetration Room Ventilation Monthly Test, Rev. 24

- PT/3/A/0610/01C, Emergency Power Switching Logic Standby Bus 1 & 2 Voltage Sensing, Rev. 13

b. Findings

No findings of significance were identified.

.2 <u>(Closed)</u> Unresolved Item (URI) 50-269,270,287/00-06-05: Inadequate Surveillance Testing of KHU

Previous testing of the KHUs documented that although the KHUs reached the specified voltage and frequency within the required 23 seconds, there was an overshoot in frequency and subsequent operation within the frequency band was not regained within the specified 23 seconds. Therefore, TS surveillance SR 3.8.1.9.a was not being met. A notice of enforcement discretion was issued that temporarily removed the frequency limits associated with SR 3.8.1.9.a until engineering evaluations could be conducted to define the appropriate limits. The licensee subsequently completed the engineering evaluations and decided to modify the KHUs' such that the requirements in SR 3.8.1.9.a could be met. Discussions indicated that such modifications would include replacement of the KHU governors, changes to the gate position controls, and installation of over-frequency trip relays. These items were documented as recommended corrective actions in the licensee's corrective action program in PIP O-00-03229.

The potential damage to safety-related equipment that could result from an over-frequency condition on the KHUs has a credible impact on plant safety. The issue was processed through the SDP and was analyzed by the Region II Senior Risk Analyst. For the analysis, it was assumed that there was a failure of the equipment initially loaded onto the KHU following a loss of power (with/without a loss of coolant accident) due to the high frequency condition of the overshoot. The analysis concluded that redundancy in equipment not initially loaded onto the electrical busses and other mitigation systems unaffected by the overshoot, provided core damage protection. This issue was considered to be of very low safety significance (Green).

SR 3.8.1.9.a requires the licensee to verify that on a simulated emergency actuation signal that each KHU auto starts and achieves frequency of 57-63 hertz in less than 23 seconds. Due to an ongoing overshoot problem related to governor control, the licensee was not achieving the rated frequency of 57-63 cycles in less than 23 seconds. This is a violation of the testing requirements of SR 3.8.1.9.a. This violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 50-269,270,287/01-03-02: Failure to Meet the Surveillance Requirements of SR 3.8.1.9.a for Testing of the Keowee Hydro Units. This violation is in the licensee's corrective action program as PIP O-00-03229.

1R23 <u>Temporary Modifications</u>

a. <u>Inspection Scope</u>

The inspectors reviewed documents related to and/or observed portions of the installation of selected temporary modifications to determine if (1) the installation was consistent with the modification documents and was in accordance with the configuration control process; (2) adequate procedures and changes were made; and (3) post installation testing was adequate. The inspectors reviewed system design bases, the UFSAR, TS, System operability/availability evaluations, and 10 CFR 50.59 screening to assess the adequacy of the temporary modifications. The following temporary modifications were reviewed:

- Modification ONOE 16123: Remove the air supply to Unit 1 Alterex temperature control valve and fail the valve open, installed May 17, 2001.

- Modification ONTM 2121: Replace recirc cooling water supply to Unit 1 isolated phase bus coolers with chill water from the Unit 3 C and D chillers

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert and Notification System Testing

a. Inspection Scope

The inspector evaluated the alert and notification system (ANS) design and the testing program. The system consisted of 63 sirens within the 10-mile emergency planning zone. The siren testing program consisted of low growl tests, weekly silent tests, and quarterly full cycle tests. Individual siren coverage was being improved with the installation of new sirens (21 to date) scheduled to be complete mid-2002.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization Augmentation

a. Inspection Scope

The inspector reviewed the design of the emergency response organization augmentation system and evaluated the licensee's capability to staff emergency response facilities within stated timeliness goals.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

The inspector reviewed changes to the Emergency Plan and the emergency action levels (EALs) to determine whether any of the changes decreased the effectiveness of the Emergency Plan. The current Oconee Nuclear Plant Emergency Plan was Revision 2001-01 dated February 2001. The review was performed against the requirements of 10 CFR 50.54(q).

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

a. Inspection Scope

The inspector evaluated the efficacy of licensee programs that addressed weaknesses and deficiencies in emergency preparedness. Items reviewed included the 10 CFR 50.54(t) audit report and exercise/drill critique reports and the associated corrective actions.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation and Simulator Observations

a. Inspection Scope

The inspectors observed an emergency response organization drill and simulator scenarios conducted on July 17, 2001, to evaluate licensee performance in the area of emergency preparedness, and to assess the licensee's critique of those performances. The inspectors specifically verified the proper classification and notification of events and development of protective action recommendations during the simulations. These observations were made in the control room simulator, technical support center, operations support center, and in the applicable portions of the plant.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 Reactor Safety PI Verification

a. Inspection Scope

The inspectors conducted annual reviews of the following three Reactor Safety PIs, as submitted to the NRC by the licensee, for accuracy:

Cornerstone	PI
Initiating Events	Unplanned Scrams per 7,000 Critical Hours (All Units)

Cornerstone	PI
Initiating Events	Scrams with Loss of Normal Heat Removal (All Units)
Mitigating Systems	Safety System Unavailability for the Units 2, and 3 Emergency Feedwater Systems

This review was conducted for second quarter 2001 PI data submitted to the NRC on or about July 21, 2001. To verify the accuracy of PI data, the inspectors reviewed Licensee Event Reports, control room logs, surveillance records data reported to the NRC and PIPs. The inspectors verified samples of data since the last completion of these PI verifications.

b. Findings

No findings of significance were identified.

.2 Emergency Preparedness PI Verification

a. Inspection Scope

The inspector reviewed documentation, associated with the following areas of emergency preparedness, to evaluate the licensee's implementation of the PI verification process:

Emergency Response Organization (ERO) Drill/Exercise Performance: The inspector assessed the accuracy of the PI for ERO drill and exercise performance (DEP) through review of documentation. In addition, the inspectors reviewed and discussed the licensee's methodology for calculating the DEP PI, with emphasis on the opportunities provided for the control room communicators. The inspector reviewed data for the previous eight quarters ending June 2001 when verifying the accuracy of the reported PI value of 95.7 percent.

ERO Drill Participation: The inspector assessed the accuracy of the PI for ERO drill participation through review of source records for selected individuals. The inspector reviewed data for the previous eight quarters ending June 2001 when verifying the accuracy of the reported PI value of 93.8 percent.

Alert and Notification System Reliability: The inspector assessed the accuracy of the PI for the alert and notification system reliability through review of the licensee's records of the siren tests for the previous 12 months.

b. Findings

No findings of significance were identified.

4OA3 Event Followup

Unit 1 Turbine Trip/Reactor Trip on September 12, 2001

a. Inspection Scope

Following the reactor trip on September 12, the inspectors reviewed plant parameters and verified the status of mitigating systems and fission product barriers; evaluated performance of mitigating systems and licensee actions; confirm that the licensee properly classified and reported the event; subsequently communicated details of the event to the regional risk analyst; reviewed the post trip review; and reviewed the licensee's subsequent root cause evaluation and proposed corrective actions.

b. Findings

A Green finding was identified for the lack of a detailed engineering review for the generator disconnect switch modification and that adequate monitoring/ testing was not specified for the modification. Although the manufacturer noted that this was the first disconnect switch designed and built for 33,000 amp use, the licensee did not consider potential heating effects on the bus duct cooling system or identify any post modification monitoring activities necessary to ensure proper in-service operation.

At 6:13 p.m., on September 12, 2001, Unit 1 experienced a "Generator Lockout" which caused a turbine trip and reactor trip. All turbine system and reactor protection system trips functioned as designed. In addition, all mitigation systems responded as required. The plant did experience a loss of feedwater which was due to a temporary loss of power to the main feeder busses. This caused automatic starting of the emergency feedwater pumps. The temporary loss of the main feeder busses was not considered to be abnormal based on the location of the bus trip relays being on the same phase as the faulted disconnect switch.

During subsequent review of this trip, the licensee identified that the "Z" phase of the generator disconnect switch, which had been installed during the last unit outage, had failed. Observations of the disconnect switch indicated that it had failed due to excessive temperature. The switch was replaced with the assistance of the switch manufacturer. Subsequent operation found that the switch temperatures became excessive when the plant reached 100 percent power. Both Unit 1 and Unit 2 reduced power to approximately 92 percent until the over temperature condition could be resolved. For Unit 1, a temporary modification to install chill water to the bus duct cooling system was installed and operation was returned to 100 percent. For Unit 2, the unit was taken off line and the generator disconnect switch was removed. During the removal, the licensee identified that one phase of the Unit 2 disconnect switch had experienced excessive temperature and the disconnect switch housing had been warped.

During the licensee's review, it was identified that the potential heating effects had not been considered and no monitoring of the modification following installation had been identified. Because the failure of the switch resulted in a reactor trip with a loss of secondary, the inspectors considered the lack of detailed design review and lack of post modification monitoring to have an actual impact on plant safety. Based on the review, and the recognition that mitigation systems operated properly the inspectors concluded this finding to be of very low safety significance (Green). This finding was described in the licensee's corrective action program as PIP O-01-03367.

4OA5 Other Activities

.1 Observation of Dry Cask Loading for Unit 3 (IP 60855)

a. Inspection Scope

The inspectors observed the following activities: loading of spent fuel assemblies into canister number 60; automatic welding of the inner and outer canister cover plates; monitoring of hydrogen concentration inside the top air space of the cask during the welding; various quality control (QC) inspections and nondestructive examinations; the penetrant examinations on the root weld (inner cover) and final pass welds (inner and outer covers); drying of the canister; helium leakage testing; sealing the vent and siphon ports; transportation of the cask from the spent fuel building decontamination area to the storage pad; and the insertion of the canister into the Horizontal Storage Module (HSM). The inspectors reviewed procedure MP/O/A/1500/016, Independent Spent Fuel Storage Installation Phase III Dry Storage Canister Loading and Storage, Revision 11, to verify activities were performed in accordance with procedural requirements.

The inspectors observed and verified that eight spent fuel assemblies were removed from the correct locations of the spent fuel pool and inserted into the designated locations of the canister. The inspectors observed transporter speed and security control. The inspectors observed radiation protection controls and monitoring.

The inspectors verified that the boron samples were taken within timeliness requirements and concentration acceptance criteria were met. The inspectors verified the hydrogen and helium purity met acceptance criteria. The inspectors reviewed the crane operator and quality control inspector training certificates, qualification and medical records. The inspectors reviewed the required records and data contained in the working copy of the procedure. The inspectors reviewed ONEI 0400-154 which contained the description and limits of the spent fuel assemblies to be placed in the canister.

b. Findings

No findings of significance were identified.

.2 (Closed) Apparent Violation (AV) 50-269,270,287/01-08-06: Failure to Promptly Correct the Inability to Align Station Auxiliary Service Water Within 40 Minutes of a Tornado Event

In a letter dated July 18, 2001, subsequent to the licensee's decline for a Regulatory Conference, the NRC informed the licensee of its final significance determination for AV 50-269,270,287/01-08-06. Specifically, the licensee was told that the issue described in the AV was a finding of low to moderate safety significance, which also represented a violation of TS 5.4.1 and 10 CFR 50, Appendix B, Criterion XVI. As such, the letter issued a Notice of Violation associated with a “White” SDP finding (EA-01-125). Accordingly, the AV is administratively closed, and for tracking purposes the recognized violation (VIO) and associated White finding will be identified as VIO 50-269,270,287/01-03-03: Failure to Promptly Correct Tornado Mitigation Procedures to Ensure the Station Auxiliary Service Water Pump Could be Aligned Within 40 Minutes of a Design Basis Tornado.

4OA6 Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. W. Foster, Safety Assurance Manager, and other members of licensee management at the conclusion of the inspection on October 4, 2001. No proprietary information was identified.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

C. Boyd, Work Control Superintendent
T. Coutu, Superintendent of Operations
T. Curtis, Mechanical System/Equipment Engineering Manager
W. Foster, Safety Assurance Manager
B. Hamilton, Engineering Manager
D. Hubbard, Modifications Manager
R. Jones, Plant Manager
C. Little, Civil, Electrical & Nuclear Systems Engineering Manager
W. McCollum Site Vice President, Oconee Nuclear Station
B. Medlin, Superintendent of Maintenance
L. Nicholson, Regulatory Compliance Manager
M. Thorne, Emergency Preparedness Manager
J. Twiggs, Manager, Radiation Protection
J. Weast, Regulatory Compliance

NRC

D. LaBarge, Project Manager, NRR
R. Haag, Chief Branch 1, Division of Reactor Projects, Region II
M. Widmann, Acting Chief Branch 1, Division of Reactor Projects, Region II

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-269,270,287/01-03-03	VIO	Failure to Promptly Correct Tornado Mitigation Procedures to Ensure the Station Auxiliary Service Water Pump Could be Aligned Within 40 Minutes of a Design Basis Tornado (Section 4OA5.2)

Opened and Closed

50-270/01-03-01	NCV	Failure to Follow an Engineering Procedure Results in Unit 2 Exceeding Licensed Reactor Power Level (Section 1R15)
50-269,270,287/01-03-02	NCV	Failure to Meet the Surveillance Requirements of SR 3.8.1.9.a for Testing of the Keowee Hydro Units. (Section 1R22.2)

Previous Items Closed

50-269,270,287/00-06-05	URI	Inadequate Surveillance Testing of Keowee Hydro Units (Section 1R22.2)
50-269,270,287/01-08-06	AV	Failure to Promptly Correct the Inability to Align Station Auxiliary Service Water Within 40 Minutes of a Tornado Event (4OA5.2)

LIST OF ACRONYMS USED

ALARA	-	As Low As Reasonably Achievable
AC	-	Alternating Current
ACB	-	Air Circuit Breaker
AHU	-	Air Handling Unit
AP	-	Abnormal Procedure
ASME	-	American Society of Mechanical Engineers
BTP	-	Branch Technical Position
BWST	-	Boron Water Storage Tank
CCW	-	Circulating Water
CFR	-	Code of Federal Regulations
CRDM	-	Control Rod Drive Mechanism
DBD	-	Design Basis Document
DC	-	Direct Current
DHR	-	Decay Heat Removal
ECCS	-	Emergency Core Cooling System
EOP	-	Emergency Operating Procedure
F	-	Fahrenheit
FSAR	-	Final Safety Analysis Report
GPM	-	Gallons Per Minute
HPI	-	High Pressure Injection
HPSW	-	High Pressure Service Water
INPO	-	Institute of Nuclear Power Operations
IP	-	Inspection Procedure
IR	-	Inspection Report
KHS	-	Keowee Hydro Station
KHU	-	Keowee Hydro Unit
LCO	-	Limiting Condition for Operation

LCP	-	Lee Combustion Plant
LER	-	Licensee Event Report
LPI	-	Low Pressure Injection
LPSW	-	Low Pressure Service Water
MCC	-	Motor Control Center
MDEFW	-	Motor Driven Emergency Feedwater
MR	-	Maintenance Rule
NCV	-	Non-Cited Violation
NDE	-	Non-Destructive Examination
NRC	-	Nuclear Regulatory Commission
NRR	-	Nuclear Reactor Regulation
OAC	-	Operator Aid Computer
PCB	-	Power Circuit Breakers
PI	-	Performance Indicator
PIP	-	Problem Investigation Process
PMT	-	Post-Maintenance Testing
RBS	-	Reactor Building Spray
RCP	-	Reactor Coolant Pump
RCS	-	Reactor Coolant System
RFS	-	Refueling System
RPS	-	Reactor Protection System
RTD	-	Resistance Temperature Detector
RWP	-	Radiation Work Permit
SDP	-	Significance Determination Process
SLC	-	Selected Licensee Commitment
SR	-	Surveillance Requirement
SSC	-	Structure, System and Component
SSF	-	Standby Shutdown Facility
TDEFW	-	Turbine Driven Emergency Feedwater
TS	-	Technical Specification
UFSAR	-	Updated Final Safety Analysis Report
URI	-	Unresolved Item
WC	-	Chilled Water System
WO	-	Work Order
WR	-	Work Request

List of Documents Reviewed

Documents Reviewed in Section 1R17 - Permanent Plant Modifications

Nuclear Station Modifications:

NSM ON-12147	2NIV-2 REPLACE VALVE
NSM ON-15763	BUILDING SPRAY DBD CHANGES
NSM ON-52991AL1	REPLACE SSF BREAKERS
NSM ON-12998/0AL1	REPLACE BATTERIES AND RACKS
NSM ON-13041	REPLACE RBCU DAMPERS
NSM ON-13054	MS STEAM HAMMER AND SEISMIC HANGER MOD
NSM ON-13066	INSTALLATION OF CARTRIDGE TYPE SEAL PACKAGES AND NEW SEAL LEAKOFF INSTRUMENTATION ON ALL 4 RCPs
NSM 22865AL1	REPLACEMENT OF RTDs
NSM 23032BM1	REPLACE REMAINING TYPE A WITH TYPE C CRDMs
NSM 23056AK1	MDEFWP ARC VALVE STRAINER

Documents reviewed in Section 1R02 - Evaluations of Changes, Tests, or Experiments

10 CFR 50.59s:

NSM 23032BM1	REPLACE REMAINING TYPE A WITH TYPE C CRDMs
NSM 22865AL1	REPLACEMENT OF RTDs
NSM ON-13066	INSTALLATION OF CARTRIDGE TYPE SEAL PACKAGES AND NEW SEAL LEAKOFF INSTRUMENTATION ON ALL 4 RCPs
NSM ON-52991AL1	REPLACE SSF BREAKERS
NSM ON-15763	BUILDING SPRAY DBD CHANGES
NSM ON-10385	SSF ASW TRANSMITTERS
NSM ON-11489	REPLACE HP CONTROL VALVE
NSM ON-13719	REPLACE HP CONTROL VALVE
NSM ON-13991	SSF ASW PUMP MINIMUM FLOW
NSM ON-15526	KEOWEE GOV. SETPOINTS
OP/3/A/1104/004, REV 95	LOW PRESSURE INJECTION

10 CFR 50.59 screens:

NSM 23056AK1	MDEFWP ARC VALVE STRAINER
NSM ON-13054	MS STEAM HAMMER AND SEISMIC HANGER MOD
NSM ON-13041	REPLACE RBCU DAMPERS
NSM ON-12998/0AL1	REPLACE BATTERIES AND RACKS
NSM ON-12147	2NIV-2 REPLACE VALVE
NSM ON-14514	CCW DESIGN CRITERIA
AP/2/A/1700/010, REV 4	UNCONTROLLED FLOODING OF TURBINE BUILDING
EP/1/A/1800/001, REV 29E	CHANGE HOLD EMERGENCY OPERATING PROCEDURE
OP/3/A/1106/006, REV 74	EMERGENCY FDW SYSTEM
OP/3/A/1107/010, REV 23	OPERATION OF BATTERIES AND BATTERY CHARGERS
PT/0/A/0150/031, REV 17	HYDROGEN ANALYZER AND POST ACCIDENT SAMPLE PANEL POST MAINTENANCE LEAK RATE TEST
PT/0/A/0400/015, REV12	SSF SUBMERSIBLE PUMP TEST

IP/0/A/0275/019B, REV 001	EMERGENCY FEEDWATER SYSTEM EMERGENCY STEAM GENERATOR LEVEL TRANSMITTER CALIBRATION
MP/0/A/1300/010, REV 17	PUMP-PACKING AND ADJUSTING PACKING

<u>Problem Investigation Process Reports:</u>

0-00-00349	PERFORMANCE ASSESSMENT SA-00-08(ON)(PA), ADEQUACY OF 50.59s.
0-00-02039	UFSAR CHAPTER 7 REVISION REQUIRES 50.59 PREPARATION.
0-00-01191	WRONG PIPING CLASS ASSIGNED DURING PREPARATION OF MINOR MODIFICATION.
0-00-02472	ASSESSMENT WCG0002P1, IDENTIFICATION AND CONTROL OF MAINTENANCE PROCEDURES EFFECTED BY MODIFICATIONS.
0-00-03689	OBSERVATION RELATED TO MINOR MODIFICATIONS BASED ON REWORK ANALYSIS.
0-01-02724	MODIFICATION PROCESS NEEDS CLARIFYING IN NSD-221.