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

April 30, 2001

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 50-269/00-08, 50-270/00-08, AND 50-287/00-08

Dear Mr. McCollum:

On March 31, 2001, the NRC completed inspections at your Oconee Nuclear Station. The enclosed report documents the inspection findings which were discussed on April 9, 2001, with Mr. M. Nazar 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.

No findings of significance were identified.

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 Haag, Chief
Reactor Projects Branch 1
Division of Reactor Projects

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

Enclosure: NRC Integrated Inspection Report 50-269,270,287/00-08, w/Attached NRC's Revised Reactor Oversight Process

cc w/encl:
Compliance Manager (ONS)
Duke Energy Corporation
Electronic Mail Distribution

Lisa Vaughn
Legal Department (PB05E)
Duke Energy Corporation
422 South Church Street
Charlotte, NC 28242

Rick N. Edwards
Framatome Technologies
Electronic Mail Distribution

Anne Cottingham
Winston and Strawn
Electronic Mail Distribution

Mel Fry, Director
Division of Radiation Protection
N. C. Department of Environmental
Health & Natural Resources
Electronic Mail Distribution

Henry J. Porter, Assistant Director
Div. of Radioactive Waste Mgmt.
S. C. Department of Health and
Environmental Control
Electronic Mail Distribution

R. Mike Gandy
Division of Radioactive Waste Mgmt.
S. C. Department of Health and
Environmental Control
Electronic Mail Distribution

County Supervisor of
Oconee County
415 S. Pine Street
Walhalla, SC 29691-2145

Lyle Graber, LIS
NUS Corporation
Electronic Mail Distribution

C. J. Thomas, Manager
Nuclear Regulatory Licensing
Duke Energy Corporation
526 S. Church Street
Charlotte, NC 28201-0006

Peggy Force
Assistant Attorney General
N. C. Department of Justice
Electronic Mail Distribution

<u>Distribution w/encl</u>:
D. LaBarge, NRR
NRR (RidsNrrDipmlipd)
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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-269, 50-270, 50-287
License Nos:	DPR-38, DPR-47, DPR-55
Report No:	50-269/00-08, 50-270/00-08, 50-287/00-08
Licensee:	Duke Energy Corporation
Facility:	Oconee Nuclear Station, Units 1, 2, and 3
Location:	7800 Rochester Highway Seneca, SC 29672
Dates:	December 31, 2000 - March 31, 2001
Inspectors:	M. Shannon, Senior Resident Inspector D. Billings, Resident Inspector E. Chrisnot, Resident Inspector S. Freeman, Resident Inspector E. Testa, Senior Health Physicist (Sections 2OS3, 2PS3, 4OA1.3, 4OA1.4, 4OA3.1) J. Kreh, Emergency Preparedness Inspector (Section 1EP4) K. VanDoorn, Sr. Reactor Inspector (Section 1RO2) R. Moore, Reactor Inspector (Section 1RO2) J. Blake, Senior Project Manager (in-office review, Sections 4OA3.2, 4OA3.3) W. Rogers, Senior Reactor Analyst (in-office review, Sections 4OA3.4, 4OA7) D. Thompson, Safeguards Inspector (Sections 3PP1, 3PP2 and 4OA1.5)
Approved by:	R. Haag, Chief Reactor Projects Branch 1 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000269,270,287/00-08, on 01/31/2001 - 03/31/2001, Duke Energy Corporation, Oconee Nuclear Station, Units 1, 2, and 3, Resident Quarterly Integrated Inspection Report.

The inspection was conducted by resident inspectors, as well as six regional inspectors who reviewed 10 CFR 50.59 implementation; health physics; emergency preparedness; security; and Licensee Event Reports (in-office reviews). 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**

No findings of significance were identified.

B. **Licensee Identified Violations**

A violation of very low significance which was identified by the licensee was reviewed by the inspectors. Corrective actions taken or planned by the licensee appear reasonable. The violation is listed in section 4OA7 of this report.

Report Details

Summary of Plant Status:

Unit 1 began the inspection period in a unit shutdown for the End-of Cycle (EOC) 19 refueling outage. The EOC 19 refueling outage officially ended on January 13, 2001, but the unit remained shutdown due to main turbine generator vibrations. Following repairs the unit was restarted and 100 percent power was achieved on January 19, 2001. The unit operated at 100 percent power throughout the rest of the inspection period except for brief periods of power reduction for control rod and main turbine valve testing, and a power reduction to 68 percent on March 18, 2001, to support troubleshooting for a direct current (DC) ground. The unit was returned to 100 percent power the same day.

Unit 2 operated at 100 percent power throughout the inspection period except for brief periods of power reduction for control rod and main turbine valve testing.

Unit 3 operated at or near 100 percent power during the early portions of the inspection period. Due to leakage from a code safety valve and the resulting high quench tank temperature, the unit was shutdown for a maintenance outage on February 17, 2001. While shutdown, an inspection of the reactor vessel head revealed control rod drive mechanism nozzle leakage. At the end of the report period the unit remained shutdown for control rod drive mechanism nozzle repairs.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1RO2 Evaluations of Changes, Tests, and Experiments

a. Inspection Scope

The inspectors reviewed a sample of the licensee's evaluations for changes made to the facility and to station procedures in the past year, to verify that these changes were reviewed and documented in accordance with regulatory requirements of 10 CFR 50.59. This was a risk informed sample. The sample included fifteen 10 CFR 50.59 screening evaluations for which the licensee determined that the changes did not impact the facility as described in the Updated Final Safety Analysis Report (UFSAR) and therefore, did not require evaluation for an unreviewed safety question (USQ). The sample also included eleven USQ evaluations for changes that did impact the facility as described in the UFSAR for which the licensee determined that the change could be implemented consistent with the requirements of 10 CFR 50.59 without prior approval of the NRC. The documents reviewed are listed at the end of the report.

In addition, the inspectors reviewed the self-assessment activity related to licensee performance of 10 CFR 50.59 evaluations to verify that findings and performance weaknesses were appropriately entered into the station corrective action program and resolved.

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, as appropriate, 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:

- Unit 3 low pressure service water (LPSW) and high pressure service water (HPSW) systems as a result of temporary modification ONTM-2112

- The electrical lineup (emergency and normal) during removal of the Keowee underground feeder to Unit 3

- Unit 2 Emergency Feedwater System

b. Findings

No findings of significance were identified.

.2 Complete Walkdown for Unit 1 and 2 LPSW System

a. Inspection Scope

The inspectors performed a full walkdown of the Unit 1 and Unit 2 LPSW system to verify that components were properly operating, labeled, and in good working condition. The Oconee Unit 1 and 2 LPSW system is a shared system. The walkdown involved components in the turbine building and the auxiliary building. The full LPSW inspection included reviews of operating procedures, maintenance procedures, instrument calibration procedures, system drawings, and outstanding work requests. The reviews also included corrective action program documents to verify that the licensee was properly identifying and correcting problems.

b. Findings

No findings of significance were identified.

.3 High Room Temperatures in the Low Pressure Injection and High Pressure Injection Rooms Following a Loss of Coolant Accident

a. Inspection Scope

The inspectors walked down the LPSW system, observed the conditions of various room coolers, and reviewed the low pressure injection/reactor building spray (LPI/RBS) and high pressure injection (HPI) room post accident environmental calculations contained in Oconee Station Calculation (OSC)-6667, Auxiliary Building and Turbine Building Loss of Coolant/Ventilation Analysis to determine if post accident room temperature would adversely impact equipment performance. In addition, the inspectors reviewed Problem Investigation Process reports (PIP) O-99-00193 and O-97-01681.

b. Findings

An unresolved item (URI) was identified for a failure to identify conditions adverse to quality following completion of room temperature calculations in 1996 and revisions to the calculations in 1999. Potential high room temperatures were identified in licensee calculations, however, it appeared that the licensee had not adequately evaluated the effects on the pumps and motors related to the post accident temperature conditions in the LPI/RBS and HPI pump rooms.

In January 1999, after reviewing room temperature calculations contained in OSC-6667, dated September 4, 1996, NRC inspectors questioned Oconee's operating basis for the Auxiliary Building Ventilation System (ABVS). Specifically, whether the ABVS was required to provide cooling for equipment in the HPI and LPI/RBS pump rooms during/following a design basis loss of cooling accident (LOCA). The licensee's response was that the ABVS was non-safety related and that there was no licensing basis for the need of the ABVS for accident mitigation. At that time, the inspectors had raised a concern of potential flashing in the LPI or HPI pump suctions due to high post accident temperatures contained in the licensee's calculation. Calculation OSC-6667 documented that the Unit 1 LPI/RBS pump rooms would heat up to 233-259 degrees F and that the combined Unit 1 and Unit 2 HPI pump room would heat up to 257 degrees F following a LOCA event.

Following the initial discussions on January 19, 1999, the licensee initiated PIP O-99-0193 to address the potential for flashing in the pump suctions and the potential requirements for ventilation to support the room equipment. The licensee noted in PIP O-99-00193 that this was not a current operability issue, the ABVS was non-safety related, and there was no licensing basis discussion on the need for ABVS for accident mitigation. Subsequently, the licensee recalculated the accident room temperatures for the LPI/RBS and HPI pump rooms. Revision 2 of OSC-6667 was completed on August 24, 1999, and it documented that the LPI/RBS pump room temperatures would reach 179 to 190 degrees F and that the combined HPI pump room temperature would reach 230 degrees F during a LOCA event.

In December 2000 the inspectors questioned the need for the LPI/BS pump room coolers to maintain room temperatures during post LOCA conditions. At that time the inspectors were informed that the room coolers were not needed. The inspectors then reviewed the calculations (Revision 2 to Calculation OSC-6667) that contained the room temperature profiles following a LOCA. After reviewing operating data on the LPI pumps and RBS pumps, the inspectors noted that with room temperatures at this level, the LPI pumps and RBS pumps and motor bearings would be above their design limits. In January 2001, this issue was discussed again with the licensee. The discussions and further review of the calculations indicated that the licensee's room temperature calculations appeared to be conservative and had not considered the effects of the room cooling from reestablishing the non-safety related ABVS.

Following the January 2001 discussions, the licensee again re-analyzed the post LOCA temperature conditions in the LPI/RBS rooms and this time credited reestablishment of the ABVS. However, the inspectors noted that the room temperatures still approached 160 degrees F, and concluded that some of the pumps and motors would still exceed their design temperature ratings for bearing temperatures. Subsequently, corrective action number 7 of PIP O-99-00193 was initiated by the licensee on March 6, 2001, to specifically consider elevated temperature effects on the HPI, LPI and RBS pumps/motors located in their respective rooms. In addition, the inspectors were informed that the motor manufacturer and the pump manufacturer had been contracted to review the design limits for the bearings to see if higher limits on temperature could be established. After discussions with the manufacturers, the licensee determined the HPI, LPI and RBS pumps/motors were operable based on their belief that the equipment would perform their required safety functions during accident conditions.

The inspectors concluded that the licensee had failed to identify a condition adverse to quality in that the calculated room temperatures clearly exceeded the operating limits for the air cooled motor and pump bearings for the LPI pump bearings, the HPI lower motor bearings, and the RBS pump bearings following completion of the original and revised OSC-6667 calculations. Since the safety significance has not been determined this issue is being identified as an URI 50-269,270,287/00-08-01: Risk Significance of High Temperatures in the Low Pressure Injection and High Pressure Injection Pump Rooms.

1R05 Fire Protection

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: all three units' east and west penetration rooms; the standby shutdown facility (SSF); all three units' equipment rooms; and the turbine building basement.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

.1 Flood Protection of Safety Related Equipment in the Auxiliary Building

a. Inspection Scope

The inspectors reviewed the design basis and risk assessment of flooding within the auxiliary building to ensure the licensee's flooding mitigation plans and equipment were adequate. The inspectors walked down the Units 1 and 2 HPI pump room, the Unit 3 HPI pump room, the Units 1 and 2 LPI/RBS pump rooms, and the Unit 3 LPI/RBS pump rooms in order to evaluate the licensee's methods of protecting risk important equipment contained in the rooms. The inspectors also walked down the cable spreading rooms and auxiliary building hallway for sources of potential flooding. The inspectors further walked down the fire protection/HPSW piping from the pumps to the auxiliary building and from the elevated water storage tank to the auxiliary building. The inspectors also reviewed: Design Study ONDS-0340; Auxiliary Building Flooding Design Study; PIPs O-96-00421, O-98-03017, O-99-01286, O-99-01671, O-00-02273, O-00-02821, and O-00-03098; various Final Safety Analysis Report (FSAR) sections, including FSAR Supplement 13; NRC safety evaluation reports (SER); Branch Technical Position (BTP) 9.5-1; Standard Review Plan (SRP) 3.6.1; and Design Basis Documents (DBD).

b. Findings

Brief Overview

An URI was identified regarding the risk significance of potential flooding problems from non-safety related fire protection/HPSW lines (one system serves both high pressure SW and fire protection functions) in the auxiliary building. Due to the adverse impact auxiliary building flooding could have on safety related equipment, compliance with 10 CFR 50.48, Fire Protection, will also be reviewed. Corrective actions taken in response to this issue, that was identified by the licensee in 1996, appeared to be untimely.

Background and System Design

The auxiliary building drain system collection tanks are the low activity waste tank (LAWT) and the high activity waste tank (HAWT) which are located in the basement of the building, in the same room as the Unit ½ and Unit 3 HPI pumps. For all three Units the LPI/RBS pump rooms are at the same elevation as the HPI pump rooms and are entered from above via spiral stairways from the auxiliary building first floor. During walkdowns, the inspectors found that each LPI/RBS pump rooms in Unit ½ are connected to the Unit ½ HPI pump room by multiple open piping penetrations, some located as low as 48 inches above the floor. The Unit 3 LPI/RBS pump rooms are connected to the Unit 3 HPI pump room by multiple penetrations, some located as low as 43 inches above the floor. Each LPI/RBS pump room is also connected to the adjacent LPI/RBS pump room by multiple penetrations, some as low as 55 inches off the floor. Also, the auxiliary building first floor is common to all three units. The LAWT & HAWT in both HPI pump rooms each contain two pumps which must be manually started from a panel on the first floor of the auxiliary building. The two LAWT pumps are rated at 100 gpm each and the two HAWT pumps are rated at 50 gpm each. All auxiliary building floor drains, including those in the HPI rooms, are piped to the respective LAWT. Equipment drains with the potential to carry high activity fluid, plus the discharge for the sumps in the LPI/RBS pump rooms are piped to the HAWT. The licensee determined that the maximum flow rate from the drain system into the LAWT could be approximately 340 gpm. This flow rate exceeded the pumping capacity of the LAWT and would result in the HPI pump room floor drains backing up. The inspectors also noted that any operator dispatched to start the LAWT pumps would potentially have to traverse a flooded auxiliary building first floor.

The inspectors determined that, due to the above design, water from any flood in the auxiliary building would find its way into the HPI pump room(s) via the drain system and any water on the auxiliary building first floor would find its way into the LPI/RBS pump rooms via the stairways. This was based, in part, on an August 3, 2000, event where a one-inch non-seismic drinking water pipe broke and spilled approximately 10,000 gallons of water into the Unit 2 east penetration room. The majority of this water traveled down the stairways to the auxiliary building first floor where drains carried it to the Unit ½ low activity waste tank (LAWT). Penetrations between HPI pump rooms and the LPI/RBS pump rooms increased the likelihood that a flood in one room would affect more than one train of emergency core cooling system (ECCS) pumps; thereby reducing the total amount of flooding necessary to affect all HPI and LPI/RBS pump rooms. Based on room sizes and mounting of equipment in each room, the licensee calculated that 48 inches of water (38,746 gallons in Unit ½ and 25,413 gallons in Unit 3) in the HPI pump rooms would enter the pump motors and disable them. Eighteen inches of water (9,000 gallons in Unit ½ and 8,250 gallons in Unit 3) in the LPI/RBS pump room would disable the RBS pump motors and 30 inches (13,787 gallons in Unit ½ and 13,439 gallons in Unit 3) would disable the LPI pump motors.

During plant walkdowns, the inspectors noted that the non-seismic 16-inch fire system header transited through the auxiliary building and posed a potential flooding problem to the auxiliary building should the piping rupture during a seismic event. Further review noted that the licensee had been requested to review the possibility of failure of non-seismic piping and any adverse affects on safety related equipment per a pre-licensing request by a NRC letter on September 26, 1972.

Problem Description

In the letter dated September 26, 1972, the NRC questioned whether the failure of any non-seismic equipment, particularly in the circulating water and fire protection systems, could adversely affect the performance of safety related equipment. The letter further requested plans and schedules for corrective action if the review determined that any safety related equipment could be adversely affected from a rupture of non-seismic piping. The licensee response, dated October 24, 1972, inappropriately stated that the fire protection headers in the auxiliary building were dry except when manually energized to fight a fire, and that 10 minutes were available for corrective action before safety related equipment was affected. Based on this response, the NRC believed that safety related equipment would not be adversely affected from a rupture of the fire system non-seismic piping; therefore, no corrective actions were deemed necessary.

In a letter dated December 31, 1976, the licensee submitted a report that compared the Oconee fire protection system with Appendix A to Branch Technical Position APCSB 9.5-1, Guidelines for Fire Protection for Nuclear Plants Docketed Prior to July 1, 1976. Position A.5 of that report again inappropriately stated that failure or inadvertent operation of an automatic fire suppression system would not incapacitate redundant safe shutdown systems or functions. In a letter dated August 11, 1978, the NRC issued the Fire Protection SER for Oconee. Based on the licensee's submittal, section 4.3.1.7 of that report accepted the licensee's position and inappropriately concluded that the effects of water from fire suppression system pipe breaks would have no effect on safety related equipment.

In PIP O-96-00421, dated February 29, 1996, the licensee questioned the statements in the flooding section of the Oconee seismic DBD regarding dry fire protection headers and the 10 minute response time. In PIP O-98-03017, dated June 9, 1998, the licensee acknowledged that the fire protection header in the auxiliary building was not dry, but had always been fully filled with water. The inspectors reviewed the original FSAR drawings for the fire protection system and, based on the drawings, confirmed that the fire protection header in the auxiliary building had always been energized.

In the evaluation of PIP O-98-03017, the licensee determined that all equipment remained operable because the SSF provided mitigating capability for an auxiliary building flood just as it would for a turbine building flood. However, the licensee acknowledged they were not in conformance with the Updated Final Safety Analysis Report (UFSAR) and initiated corrective action to change it. The UFSAR change was not completed because the safety evaluation for 10 CFR 50.59 revealed that the SSF was not licensed for an auxiliary building flood and changing the UFSAR would constitute an unreviewed safety question (USQ). The licensee then initiated a design study to resolve the issue. The study showed that pipe breaks in the 16-inch or four-inch fire protection headers in the auxiliary building would affect safety related equipment in the HPI and LPI/RBS pump rooms for all three units. Additionally, some rooms would be affected by pipe breaks in the recirculated cooling water (RCW) or non-seismic LPSW lines. The study also showed that 45 minutes would be necessary to identify and isolate a pipe break in the auxiliary building.

<u>Risk Significance</u>

The inspectors found several circumstances that could affect the risk associated with auxiliary building flooding. First of all, as part of its core damage assessment, the Oconee PRA had assumed a flood of greater than 150,000 gallons to disable the HPI pumps and assumed an independent failure of the component cooling (CC) system leading to a LOCA via the RCP seals. However, the licensee discovered that much less than 150,000 gallons was needed to disable the HPI and LPI pumps and the inspectors noted that flooding could potentially disable the CC system. During walk downs of the drain system the inspectors noted that the power supply breakers for all CC pumps

(Units 1, 2, 3) were located in MCCs on the first floor of the auxiliary building. Furthermore, the breakers were on the bottom row of the MCCs.

Secondly, the alarms for LAWT high level, which would be used to identify a flooding condition in the auxiliary building, had not been calibrated or tested since 1982-1983. In addition, the LPI/RBS room sump pump level switches and alarms were not in a routine calibration program. This issue was discussed with the licensee. The licensee indicated to the inspectors that these level instruments would be placed in a routine testing or calibration program.

Thirdly, the licensee did not have contingency plans or abnormal procedures for isolation of potential sources of auxiliary building flooding. There were no contingency plans or equipment staged to remove flood water from the LPI/RBS or HPI pump rooms and no designated procedures in place for replacement of any flooded equipment. In order to isolate a ruptured auxiliary building fire protection line the licensee would have to review various piping drawings to identify the applicable isolation valves and then send operators to close them. Reviewing the drawings would have been time consuming and closing the applicable isolation valves would have been difficult because of their location which would require the use of ladders. The inspectors noted that there was a PIP recommendation to install chain operators on some of these valves. During the August 3, 2000, drinking water pipe break in the auxiliary building, isolation of the system took greater than one hour, due to the piping being in the overhead area of the penetration room with limited access, the system piping diagrams lacked sufficient detail to identify an isolation point, and the lack of contingency plans to mitigate the consequences of a flooding event in the auxiliary building. At the end of the inspection period, the licensee was in the process of developing the contingency plan procedures for flooding in the auxiliary building. No other corrective actions have been implemented.

This issue was determined to have a credible impact on safety because of the potential to affect multiple trains of ECCS equipment on multiple units. Also because flooding could affect both the initiating events and mitigating systems cornerstones, further NRC analysis of the potential risk will be performed. Pending additional NRC review of the analysis of the flooding issues and compensatory measures in the auxiliary building, this issue is being identified as URI 50-269,270,287/00-08-02: Risk Significance of Potential Flooding Problem From Non-Safety Related Lines in the Auxiliary Building. In addition, compliance with 10 CFR 50.48 will be reviewed as part of this URI.

.2 Potential Flooding Issue from Use of Cable Room Sprinkler Systems

a. Inspection Scope

The inspectors walked down the cable spreading rooms and the equipment rooms for sources of potential flooding. The inspectors reviewed station procedures which included OP/1/A/6101/1003, Alarm Response Guide 1SA-03; ONS Station Fire Plan, Revision 1, Dated April 23, 1999; RP/0/B/1000/29, Fire Brigade Response, Revision 0; and ONS Fire Brigade Guide #3, revised February 27, 1997. The inspectors also reviewed Design Study ONDS-0340, Auxiliary Building Flooding Design Study; PIPs O-96-00421, O-98-03017, O-99-01286, O-99-01671, O-00-02273, O-00-02821, and

O-00-03098; various FSAR sections, including FSAR Supplement 13; NRC SERs; and BTP 9.5-1.

b. Findings

Brief Overview

An URI was identified regarding the risk significance of potential flooding problems from fire suppression systems in the cable spreading rooms. The inspectors concluded that the licensee had not been timely in resolving the cable spreading room flooding vulnerability from the open head sprinkler design of the fire suppression system. This issue was initially identified by the licensee in 1995 and the proposed corrective actions specified changing the fire suppression sprinkler heads.

Background and Design

All of the sprinkler heads in the cable spreading rooms are an open design, in that, following a manual actuation of the sprinkler system for a fire, all of the heads would flow water into the cable spreading room at a flow rate in excess of 1100 gpm. A relatively short firefighting efforts of 20-30 minutes could deposit in excess of 22,000 to 33,000 gallons in the affected cable spreading room. In addition, if a fire hose with a 200 gpm capacity was also used during this period an additional 4,000 to 6,000 gallons would be added to the these totals. The inspectors noted that in a 1995 response to Generic Issue 57, Effects of Fire Protection System Actuation On Safety-Related Equipment (Rev.3), the licensee identified that fire zones were vulnerable to water migration to other zones and recommended that the open head sprinklers be replaced with a closed head design to limit water migration during a fire.

Problem Description

The inspectors noted that there were no floor drains in the cable spreading rooms. Since the Unit 1 and Unit 2 cable spreading rooms are connected via a fire door, actuation of the sprinkler system could result in the flooding of both cable rooms from the large quantity of fire suppression water. The cable room floors contain equipment hatches and multiple wiring penetrations leading to the equipment room below. During the Unit 2 August 3, 2000, potable water line break, water leaked through the equipment hatches onto the vital motor control centers (MCCs) in the Unit 2 equipment room below. Based on this flooding event and observations of wiring penetrations in the floor of the cable spreading rooms, the inspectors concluded that the water from manual actuation of the cable spreading room suppression system could leak from the cable rooms into the equipment rooms below, with potential to cause grounding/shorts in the MCCs and DC isolating diode panels. This condition could then result in a transient on a unit not initially affected by the cable spreading room fire.

Since the cable spreading rooms do not have floor drains to remove the suppression system water, the licensee stated that if the suppression system was actuated, the fire doors between the stairways and the cable rooms could be opened to allow the fire suppression water to drain down the stairs to the auxiliary building basement level. The inspectors noted that the water could then drain into the HPI and LPI/BS pump rooms.

From the Auxiliary Building Flooding Design Study, the licensee had previously calculated the critical flood volumes for affecting safety-related equipment. Based on these critical flood volumes, the inspectors noted that the large amount of water from the suppression system had the potential to significantly impact the LPI/RBS pumps if operated for even 20 minutes and could impact the HPI pumps if the suppression systems were needed for greater than 30 minutes.

The inspectors concluded that because actuation of the suppression system could result in a significant amount of water being injected into the cable spreading rooms, the lack of drainage could result in flooding of the cable spreading room or flooding into the auxiliary building basement if the doors were opened. In addition, because the Unit 1 and Unit 2 cable spreading rooms are connected via doorways, actuation of either Unit 1 or Unit 2 cable spreading room fire suppression systems could result in flooding in both Unit 1 and Unit 2 cable spreading rooms.

Subsequently, the IPEEE report, submitted on December 28, 1995, stated that walk downs associated with Generic Issue 57 had identified that fire zones were vulnerable to water migration to other zones. The walk down recommended that the open head sprinklers be replaced with a closed head design to limit water migration during a fire. The licensee did not enter this recommendation into their corrective action program until January 11, 1999, when PIP O-99-00062 was initiated. PIP O-99-00062 recommended replacement of the suppression system sprinkler heads. At the end of this inspection period the sprinkler heads had not been replaced. Because flooding from the cable spreading room fire suppression system could impact the operability of safe shutdown equipment, the inspectors concluded that the licensee's corrective actions have been untimely in correcting this issue.

The inspectors noted that the licensee did not have written contingency plans for removal of the water following actuation of the fire suppression sprinkler system. In addition, as stated in Section .1, the level detectors used to identify flooding in the auxiliary building have not been calibrated or tested since 1982-1983, and mitigation strategies have not been developed for the potentially flooded HPI and/or LPI/BS pump rooms. Because of the potential safety significance of affecting the LPI and HPI pumps in the mitigation cornerstone and the RBS pumps in the containment cornerstone, and the potential to lose multiple vital DC busses, further NRC analysis of the potential risk will be performed. Pending additional NRC review, this issue is being identified as URI 50-269,270,287/00-08-03: Risk Significance of Potential Flooding Problem From Fire Suppression Systems in the Cable Spreading Rooms.

.3 <u>Potential Flooding Issue from Main Feedwater (MFW) Line Break in the East Penetration Room</u>

b. <u>Inspection Scope</u>

The inspectors walked down the East and West Penetration Rooms for sources of potential flooding. The inspectors reviewed ONDS-0340, Auxiliary Building Flooding Design Study; OSC-2034, East Penetration Room Lower Blowout Panels Calculation; PIPs O-99-2433 and O-00-3730; various FSAR sections, including FSAR Supplement 13 and Chapter 15 Accident Analysis; NRC SERs; BTP 3.6-1; and the December 15, 1972, NRC letter concerning consequences of pipe breaks outside containment (i.e., the Giambusso letter).

c. Findings

Brief Overview

An URI was identified regarding the risk significance of an uncontrolled design change which bolted and sealed the east penetration lower blowout panels in place. The licensee had unknowingly modified the east penetration room blowout panels by adding bolts, a polymer sealer, fiberglass cloth, and silicone sealant to the blowout panels. The blowout panels are design to prevent flooding within the penetration rooms due to a rupture of piping in the room.

Background and System Design

The inspectors noted that a MFW line rupture would result in a significant amount of water being injected into the east penetration rooms and there did not appear to be an acceptable drainage pathway to prevent subsequent flooding of the auxiliary building. The inspectors previously noted that flooding in the penetration rooms could result in flooding in the LPI/BS and HPI pump rooms. This conclusion was based on the potable water line break in August 2000, where significant leakage was observed through the various penetration seals, wall seals, and under the doors of the penetration room. The inspectors noted that with the high temperature, high impingement pressure, and the large volume of water, the flooding consequences of a MFW line rupture would be more safety significant.

The inspectors reviewed the 1972 Giambusso letter, December 15, 1972, which was sent to the licensee to address 10 CFR 50, Appendix A, General Design Criteria (GDC) 4, Environmental and dynamic effects design bases. The letter specifically addressed high energy line breaks in the turbine and auxiliary buildings and asked the licensee to evaluate the potential consequences of pipe whip and building overpressurization due to piping ruptures in the auxiliary building. The letter documented the applicable revision of GDC 4 and stated that “A nuclear plant should be designed so that a reactor can be shutdown and maintained in a safe shutdown condition in the event of a postulated rupture, outside containment, of a pipe containing a high energy fluid, including the double ended rupture of the largest pipe in the main steam and feedwater systems” and that “Plant structures, systems, and components important to safety should be designed and located in the facility to accommodate the effects of such a postulated pipe failure to the extent necessary that a safe shutdown condition of the reactor can be accomplished.” The letter also stated “If the results of your review indicate that changes in the design of structures, systems, or components are necessary to assure safe reactor shutdown in the event this postulated accident situation should occur, please provide information on your plans to revise the design of your facility.”

Problem Description

The inspectors reviewed the licensee’s response to the 1972 Giambusso letter which directly related to the acceptability of the main steam and MFW lines being routed through the auxiliary building penetration rooms. To address the potential adverse effects from a main steam or MFW line break, the licensee subsequently installed blow out panels in the penetration rooms to prevent overpressurization of the auxiliary building and allow flooding from the postulated break to be directed outside the auxiliary building. Because the majority of each penetration room was enclosed with brick and/or concrete constructed walls, it became critical that a blowout panel opening of approximately 4 foot by 5 foot be opened during a MFW line break event in order to prevent flooding of the auxiliary building.

The inspectors identified that the licensee had bolted the lower blowout panels in place, had used RTV to seal the blowout panels, and had coated the inside of the blowout panels with a hardened fiberglass fiber matting and polymer coating that went from the panel to the adjoining structural supports. Since these panels had originally been designed to open with approximately 1 pound per square inch of internal pressure, the inspectors questioned whether the modifications to the lower blowout panel would prevent the panels from opening. The licensee noted that this work was performed to better seal the panels to ensure proper operation of the respective penetration room ventilation system and the changes had not considered the effect on the opening of the blowout panels during a MFW line rupture.

The inspectors noted that the personnel access doors to the Unit 2 and Unit 3 east penetration rooms from the stairways and elevator areas open outward, so only the door latches would be available to keep the door closed against penetration room pressure. The inspectors concluded that if placed under pressure during a feedwater line rupture event, the latches would likely fail which would result in water flooding down the stairs to the lower levels of the auxiliary building. The Unit 1 penetration room contained the service elevator with double doors whose failure would allow water to reach the lower levels of the auxiliary building very rapidly. This issue was discussed with the licensee and subsequently PIP O-01-00815 was initiated to review the configuration of the doors and door frames. An approximate 3 inch gap exists between the containment wall and the east penetration room floor. This area was filled with a foam and/or cork type of material. From below the floor this material can be seen falling out and some of it was missing. The inspectors noted that following a MFW line rupture, the impingement of the MFW would likely result in significant leakage through this gap to the lower levels of the auxiliary building.

Additionally, in front of the blowout panels is a 6 inch dam that would prevent all the water in the penetration from exiting via the blowout panels. Based on the observations of the access door configurations and the gap between the containment building and the east penetration room floor, the inspectors concluded that even if the lower blowout panel opened as designed, a significant amount of water would still reach the lower levels of the auxiliary building and cause flooding of the LPI/BS and HPI pump rooms.

The inspectors noted that there appears to be a potential for large pipe whip of the MFW lines during a rupture due to the lines only being supported with struts and/or struts and spring cans. The licensee's analysis documented that a MFW line rupture in the auxiliary building east penetration room would result in thrust of up to 174,900 pounds. Without solid supports on the MFW lines this would result in significant pipe whip. The inspectors concluded that this pipe whip had the potential of rupturing other safety related systems that are in close proximity of the MFW lines. Based on system layout, this type of failure could potentially result in loss of function of the various trains of systems. These systems include LPSW, LPI, HPI, Auxiliary Feedwater, BS, SSF auxiliary feedwater, etc. The failure of these systems would add to the total flood potential.

The inspectors noted that the Giambusso letter required the licensee to provide a discussion “of the potential for flooding of safety related equipment in the event of failure of a feedwater line or any other line carrying high energy fluid.” Although in response to the Giambusso letter, the licensee provided a short discussion of flooding in the east penetration room, the discussion did not include the potential flooding due to failure of penetration room doors, failure of the building gap material, or failure of other safety related systems due to pipe whip. At the end of this inspection period these issues were still under review by the licensee.

The licensee initially estimated that 196,000 gallons of water would result from a MFW piping rupture. The inspectors noted that this amount of water and any additional water from other ruptured systems could result in significant flooding of the LPI/BS and HPI rooms in all three units. The change in the design of the blowout panels will be reviewed for compliance with 10 CFR 50, Appendix B, Criterion III to determine if the changes were properly controlled. Pending further review of the effects that the changes had on blowout panel operation, the potential consequences on important equipment, and compliance with 10 CFR Appendix B, Criteria III, this issue is being identified as URI 50-269,270,287/00-08-04: Risk Significance of Uncontrolled Design Changes to Penetration Room Blowout Panels.

1R07 Heat Sink Performance

b. Inspection Scope

The inspectors reviewed the 3A Reactor Building Cooling Unit (RBCU) heat exchanger preventive maintenance and testing data to ensure that the heat exchanger would be able to supply the necessary cooling as described in the Final Safety Analysis Report (FSAR). The inspection focused on deficiencies that could mask degraded performance of the heat exchangers, could result in common cause heat sink performance problems, and ensure that the licensee has adequately identified and resolved heat sink performance problems that could affect multiple heat exchangers in mitigating systems.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors observed simulator training, on March 12, 2001, for reactor operators and senior reactor operators. The inspectors observed a loss of Main Feedwater Pump scenario with a failure of a feedwater control valve. The inspectors evaluated the crew’s performance in terms of communications; ability to take timely actions in the safe direction; prioritizing, interpreting, and verifying alarms; correct use and implementation of procedures, including the alarm response procedures; timely control board operation and manipulation, including high-risk operator actions; and oversight and direction provided by the shift supervisor, including the ability to identify and implement appropriate TS actions, reportability determinations, emergency plan actions, and notifications. The inspectors also attended the evaluators critique.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule 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, as appropriate, 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:

- Standby Shutdown Facility standby battery chargers
- Unit 1 600 volt switchgear breakers 1X1-5B, 1X3-6C, and 1X4-5A
- Condensate booster pumps
- High Pressure Service Water Valve -25
- Turbine Building Flood Doors
- LPI Room Sumps

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluations

a. Inspection Scope

The inspectors evaluated, as appropriate for the selected SSCs listed below: (1) the effectiveness of the risk assessments performed before maintenance activities were conducted; (2) the management of risk; (3) that, upon identification of an unforseen situation, necessary steps were taken to plan and control the resulting emergent work activities; and (4) that maintenance risk assessments and emergent work problems were adequately identified and resolved. The following items were reviewed under this inspection procedure:

- Unit 1 bank 2 pressurizer heaters breaker not resetting required that MCC 1XSF be de-energize to repair, affecting power to containment isolation valves, with problem investigation Process (PIP) O-01-0264 being issued
- Removing transformer 2X4 from service due to low gas pressure resulting in the Unit 2 LPI/HPI flow path and LPI discharge header cross connection being inoperable
- Trip of the Unit 3 bank 2 pressurizer heaters breaker during testing, required to be operable due to code safety valve leakage, with PIP O-01-0283 being issued
- Unit 1 direct current ground on 1CIA bus resulting in an abnormal DC system line up on Units 2 and 3, the operating units
- Keowee air circuit breaker (ACB) increased resistance when operating ACB manually during preventive maintenance
- Removal of breaker 1TD-14, power feed from main feeder bus 2 to switchgear 1TD, for replacement
- Failure of the 1A Chiller due to breaker problem
- Removal of lifting lugs from Circulating Water Screen rests resulting in inability to clean screens, PIP O-01-00783

b. <u>Findings</u>

No findings of significance were identified.

1R14 <u>Personnel Performance During Nonroutine Plant Evolutions</u>

a. <u>Inspection Scope</u>

The inspectors reviewed personnel performance during selected non-routine events and/or transient operations. 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 Startup and Power Escalation on January 18, 2001, following Refueling and Forced Outage

- Unit 3 Forced Shutdown for Excessive Pressurizer Relief Valve Leakage on February 17, 2001

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed selected operability evaluations affecting the risk significant systems listed below, to assess, as appropriate: (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 work as intended, and were appropriately controlled; and (5) where continued operability was considered unjustified, the impact on TS limiting conditions for operations (LCO). The inspectors reviewed the operability evaluations described in the following PIPs:

- PIP O-01-0076, control fuse for Unit 1 reactor coolant makeup pump motor being the wrong size

- PIP O-01-0157, high pressure injection trains and pumps operable for small break loss of coolant and no loss of offsite power scenario

- PIP O-01-00281, Reactor Building Hydrogen Buildup from Pressurizer Relief valve leakage

- PIP O-01-00815, Evaluation for Penetration Room Blowout Panels for High Energy Line Break

- PIP O-01-00786, Reactor Building Cooling Unit 3B motor high temperature

- PIP O-01-00810, Unit 2 Core Flood Tank leak

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing

.1 Routine Post Maintenance Testing

a. Inspection Scope

The inspectors reviewed post-maintenance testing (PMT) procedures and/or test activities, as appropriate, for selected risk significant mitigating 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 the results of the following tests:

- OP/0/A/2000/01, Keowee Hydro-Station (KHS) Main Step-up Transformer, Revision 9
- OP/0/A/2000/13, KHS Unit 1 Generator Operation, Revision 9
- OP/0/A/2000/19, KHS Unit 1 Battery Bank Number 1, Revision 7
- OP/0/A/2000/41, KHS Modes of Operation, Revision 22
- PT/0/A/0610/24 KHS Emergency Start for Troubleshooting and Post Maintenance Checkouts, Revision 4
- PT/1/A/2200/19, KHS Unit 1 Turbine Sump Pump IST, Revision 6, post replacement of check valves 1TS-2 and 1TS-4
- MP/0/A/3009/17, Visual PM Inspection and Electrical Motor Test, Revision 5, change out of RBCU 3B
- PT/3/A/0160/08, Reactor Building Cooling Unit Fan Operational Test, Revision 1, post change out of RBCU 3B
- PT/0/A/0160/02, Reactor Building Cooling Unit Air Flow Test, Revision 8, post change out of RBCU 3B

b. Findings

No findings of significance were identified.

.2 Unit 1 RCP Seal Replacement

a. Inspection Scope

The inspectors reviewed the post modification testing results of the Unit 1 RCP seals to verify the functional capability of the new seals and instrumentation. The inspectors also walked down the Unit 1 reactor building during reactor coolant system (RCS) heatup to independently check for leaks from the new seals. The inspectors reviewed the following specific procedures:

- TN/1/A/3066/00/AK1, Installation Procedure for Unit 1 Reactor Coolant Pump Seal Cartridge Instrumentation, Revision 0

- TN/1/A/3066/00/AM1, Procedure for the Verification & Documentation of NSM 13066AM1, Revision 0
- MP/1/A/1310/052, RCP Seal - Unit 1 -Bingham Type R CRW 950B-3 - Static Fitness Test, Revision 0
- MP/0/A/1720/016, System/Component Pressure Test Controlling Procedure, Revision 22
- MP/0/A/1200/010A, Relief Valve Set Pressure Testing and Adjustment, Revision 10
- IP/0/A/0200/024A, Reactor Coolant Pump Temperature Instruments, Revision 0
- IP/1/B/0202/001C, High Pressure Injection System RC Pump Seal Flow Instrument Calibration, Revision 36
- IP/0/B/0200/024, Reactor Coolant Pump Pressure and Flow Instruments, Revisions 0 & 1
- AP/3/A/1700/016, Abnormal Reactor Coolant Pump Operation, Revision 9

b. Findings

No findings of significance were identified.

1R20 Routine Outage Observations

.1 Refueling Outage Inspections

a. Inspection Scope

The inspectors reviewed the following activities that occurred during the latter portions of the Unit 1 refueling outage for conformance to the applicable procedures and witnessed selected activities. The Unit 1 refueling outage extended past the end of the last reporting period. Surveillance tests were reviewed to ascertain completeness within the TS required specifications. Preparations and initial outage related activities were reviewed by the inspectors and are documented in NRC Inspection Report 50-269/270/287/00-07.

- Mode changes from Mode 6 - Refueling, to Mode 1- Power Operation
- Reactor startup
- Zero power physics testing
- Control rod worth testing
- Power escalation

- Outage-related procedures and tests:
  - PT/0/A/0105/07, Update of OAC Power Tilt Monitor Application, Revision 01
  - PT/1/A/0711/01, Zero Power Physics Testing, Revision 35
  - OP/1/A/1102/01, Controlling Procedure for Unit Startup, Revision 232
  - PT/0/A/0205/05, Thermal Power And RC Flow Calculations, Revision 20
  - PT/1/A/0630/01, MODE Change Verification, Revision 2
  - PT/1/A/1103/15, Reactivity Balance Procedure (Unit 1), Revision 52
  - PT/0/A/0811/01, Power Escalation Testing, Revision 25

b. Findings

No findings of significance were identified.

.2 Unit 3 Forced Outage

a. Inspection Scope

The inspectors observed selected activities and reviewed associated documentation related to the Unit 3 forced outage to verify conformance to applicable procedures. Surveillance tests were reviewed to ascertain completeness within the TS required specifications. Activities observed included the following:

- Reactor shutdown
- Reactor cooldown and initiation of decay heat removal (DHR)
- Calibration and operation of the low temperature overpressure (LTOP) reactor protective function
- Electrical power alignments and testing during major outage activities
- Containment closure

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors witnessed surveillance tests and/or reviewed test data of the selected risk-significant SSCs listed below, to assess, as appropriate, 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. The following testing was observed and/or reviewed:

- PT/1/A/0152/009, Unit 1 Feedwater System Valve Stroke Test, Revision 6
- PT/3/A/0251/01A, 3A LPSW Pump Test, Revision 61
- PT/3/A/0251/01E, 3A LPSW Pump Check Valve Test, Revision 61
- PT/3/A/0151/19, Penetration 19 Leak Rate Test (LRT), Revision 7
- PT/3/A/0151/20, Penetration 20 LRT, Revision 5
- IP/1/A/0270/004, Main Steam Line Break Online Analog Functional Test, Revision 0
- PT/1/A/0152/03, Condensate System Valve Stroke Test, Revision 3.

b. Findings

No findings of significance were identified.

1R23 Temporary Modifications

Active Temporary Modifications

a. Inspection Scope

The inspectors reviewed documents related to and/or observed portions of the installation of selected temporary modifications. Among the documents reviewed were system design bases, the UFSAR, TS, system operability/availability evaluations, and 10 CFR 50.59 screening. The inspectors observed, as appropriate, that the installation was consistent with the modification documents, was in accordance with the configuration control process, adequate procedures and changes were made, and post installation testing was adequate. The following items were reviewed:

- Modification ONTM-1200 and 1201: Install a desktop CPU, keyboard, and interface unit to monitor and record negative sequence currents from the Unit 1, Unit 2 and Unit 3 main generators.

- Modification ONTM-1207: Install a desktop CPU, keyboard, and interface unit to monitor and record negative sequence currents from the Unit 1 and Unit 2 main generators.

- Modification ONTM-1201: Jumper and disable the feedwater pressure switches in all three units for the reactor protective and emergency feedwater systems by disconnecting the switches and splicing the field cables.

- Modification ONTM-2112: Provide temporary additional cooling to the Unit 3 quench tank cooling recirculation line by connecting temporary heat exchangers to the Unit 3 CS, HPSW, and LPSW systems.

- Modification ONTM-2114: Provide temporary method to pump forward the Unit 2 moister separator re-heater drain tank by providing reverse control logic on valve 1HD-45, installed February 7, 2001.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP4 Emergency Action Level (EAL) and Emergency Plan Changes

a. Inspection Scope

The inspector conducted an in-office review of changes to the Emergency Plan, as contained in Revisions 00-1, 00-2, and 00-3, against the requirements of 10 CFR 50.54(q) to determine whether any of those changes decreased Plan effectiveness. Revision 00-2 included minor editorial (nonsubstantive) modifications to the EALs. Changes made via the other two revisions were primarily editorial in nature and did not involve modifications to the EALs.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspectors reviewed radiological procedures, problem evaluation reports, calibration data files, interviewed health physics technicians, health physics shift supervisors, and health physics section supervisors and managers to evaluate compliance with the

Radioactive Material Control Program, Updated Final Safety Analysis Report (UFSAR), Technical Specifications (TSs), Selected Licensee Commitments (SLCs) and 10 CFR Part 20 requirements. In addition the inspectors accompanied and observed a technician performing operational checks on tool monitors, personnel contamination monitors and portal monitors.

Procedures reviewed included the following:

- HP/O/1003/016 Revision 013, Calibration of Automated Personnel Monitors
- HP/O/1004/058 Revision 27, Calibration Worksheet for Portable Survey Instruments
- HP/O/1000/016 Revision 063, Body Burden Analysis
- SH/O/2001/001 Internal Dose Assessments, Revision 013
- HP/O/1002/049 Revision 003, Operation of The Merlin Gerin DMC 90 Digital Dosimeter
- HP/O/1002/046 Revision 016, Operation of Portable Radiation Survey Instruments

The inspectors reviewed the Advanced Respiratory Protection (SCBA) training plans and file records of completion and interviewed operators for their knowledge of their mask size, location and availability of spectacle inserts. The inspectors reviewed the calibration files for the Effluent Radiation Monitors and observed IP/O/B/0360/033, Revision 018, Sorrento Process Radiation Monitor Low Range Detector Calibration.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS3 Radiological Environmental Monitoring Program

a. Inspection Scope

The inspectors reviewed environmental sample collection procedures, air sampler calibration data, accompanied an environmental sample collection laboratory technician during the collection of air samples. The inspectors reviewed a Self Assessment Audit of the Oconee Nuclear Station Environmental Program performed February 15, and 16, 2000. Seven of forty two TLD locations and six of six air sample locations were verified. The inspectors observed the operation of the meteorological tower, reviewed operational checklist and reviewed files containing calibration and maintenance data for the meteorological system. The Oconee Nuclear Station Units 1, 2, and 3 Annual Radiological Environmental Operating Report 1999 and the Offsite Dose Calculation

Manual Revision 40 were reviewed to evaluate compliance with the Radiological Environmental Monitoring Program (REMP), Offsite Dose Calculation Manual (ODCM), TS, Appendix I to10 CFR Part 50 and 40 CFR 190 requirements.

b. Findings

No findings of significance were identified.

3. SAFEGUARDS

Cornerstone: Physical Protection

3PP1 Access Authorization

a. Inspection Scope

The inspectors reviewed licensee procedures, Fitness For Duty (FFD) reports, and licensee audits and interviewed five representatives of licensee management and five escort personnel concerning their understanding of the behavior observation portion of the personnel screening and FFD program. In interviewing the personnel, the inspectors reviewed the effectiveness of their training and abilities to recognize aberrant behavioral traits. The following are documents and procedures reviewed to evaluate licensee program for maintaining access authorization:

- Fitness for Duty Semi-Annual Report, January through October, 2000
- Nuclear System Directive 218 - Access Authorization Program, Revision 7, dated July 22, 1999
- Fitness-for-Duty Employee Handbook, dated November 2000
- Plant Access Training, dated January, 2001
- Fitness for Duty for Individuals Covered by 10CFR Part 26, dated January 1, 2000
- Alcohol and Drug Use Procedure, dated May 1, 2000
- Annual Audit - Audit Report No. SA-00-07,Nuclear Security Assessment, McGuire, Oconee and Catawba, dated October 5, 2000
- Oconee Human Resources Web Page - Behavioral Observation/Fitness For Duty
- Work Standards Handbook for Duke Nuclear Sites, Revision 1999
- Problem Investigation Process - 0-99-05285, dated December 29, 1999

b. Findings

No findings of significance were identified.

3PP2 Access Control

a. Inspection Scope

The inspectors observed access control activities on January 17 and 18, 2001, and equipment testing conducted on January 18, 2001. In observing the access control activities, the inspectors assessed whether officers could detect contraband prior to being introduced into the protected area. The inspectors also assessed whether the officers were conducting access control equipment testing in accordance with regulatory requirements through observation, review of procedures, and log entries. The lock and key control activity was assessed to determine if locks and keys were changed as required. Preventative and post maintenance procedures were reviewed and observed as performed. The following are documents reviewed to evaluate licensee program for maintaining access control:

- Safeguard Event Logs, 2000-2001
- Video Badging Network Assessment - 99SECO3R2, dated September 30, 1999
- Vital Area Access Assessment, October 20 - December 31, 2000
- Problem Investigation Process, No. 0-99-04412, dated November 9, 2000
- Security Incident Reports, January 2000 to present.
- Entry Turnstile Inspection and Operational Test, SP-304, Rev 10, dated February 8, 2001

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI) Verification

.1 General

Licensee records were reviewed to determine whether submitted PI statistics were calculated in accordance with the guidance contained in NEI 99-02, Regulatory Assessment Performance Indicator Guideline.

.2 Safety System Functional Failure Reactor Safety PI

a. Inspection Scope

The inspectors verified the accuracy of the Safety System Functional Failure PI. The inspectors reviewed licensee event reports to determine the number of events or conditions in the previous four quarters that had been reported in Licensee Event Report

(LERs) that prevented, or could have prevented, the fulfillment of a safety function. The following functions were monitored:

- Reactor and Primary Coolant Integrity
- Emergency Core Cooling
- High Pressure Heat Removal
- Residual Heat Removal
- Emergency Boration
- Primary System Safety and Relief
- Main Steam Isolation
- Containment Integrity
- Reactor Protection
- Accident Monitoring
- Emergency AC and DC Power
- Equipment Cooling
- Essential Compressed Gas
- Control Room Emergency Ventilation
- Spent Fuel

b. Findings

No findings of significance were identified.

.3 Occupational Radiation Safety

a. Inspection Scope

For the cornerstone area of Occupational Radiation Safety, the inspectors interviewed cognizant personnel, reviewed shift logs and PIP reports between April 01, 2000, and January 01, 2001, to support the PI verification. Selected PIP's O-C-00-0468, O-C-00-0954, O-C-00-1098, O-C-00-1691, O-C-00-1720, O-C-01-0002, and 0-C-01-0062 were reviewed for assignment of responsibility, licensee evaluation, timely closure and applicability for PI reporting screening criteria.

b. Findings

No findings of significance were identified.

.4 Public Radiation Safety

a. Inspection Scope

For the cornerstone area of Public Radiation Safety the inspectors interviewed cognizant personnel and reviewed PIP reports between April 01, 2000, and January 01, 2001, to support the PI verification. Selected PIPs O-00-03123, O-00-02219, O-00-0035, O-00-01969, O-00-02639, O-00-04371, O-01-00034, O-01-00246 were reviewed for assignment of responsibility, licensee evaluation, timely closure and applicability for PI reporting screening criteria.

b. Findings

No findings of significance were identified.

.5 Physical Protection PIs

a. Inspection Scope

The inspector reviewed Duke Power's programs for gathering and submitting data for the Fitness-for-Duty, Personnel Screening, and Protected Area Security Equipment Performance Indicators. The review included Duke's tracking and trending reports and security event reports for the performance indicator data submitted from the first quarter 2000 to the first quarter of 2001. Licensee records were reviewed to confirm the accuracy and completeness of PI data in accordance with the guidance contained in NEI 99-02, Regulatory Assessment Performance Indicator Guideline.

b. Findings

No findings of significance were identified.

4OA3 Event Followup

.1 (Closed) LER 50-269/00-07-00 Loss of Positive Control of a Radioactive Calibration Source

The inspectors reviewed the circumstances associated with this LER and found that the licensee's discovery of a lost radioactive calibration source containing 0.022 microcuries of Am-241 on November 7, 2000, was promptly reported pursuant to 10 CFR 20.2201 Sections (b)(1) and (2). A licensee's investigation concluded that the source was probably mistaken for a radiological sample in the counting laboratory and disposed of as dry active waste. The licensee's radiological assessment for this lost source concluded that external exposure from the lost source would be negligible. Internal exposure was not considered credible. Corrective actions have been implemented. Independent dose assessment, concluded that the lost source and the circumstances of its loss presented a minimal health and safety risk. Therefore, in accordance with Section IV of the NRC's Enforcement Policy, the inspectors determined that this violation was of minor significance and is not subject to formal enforcement action.

.2 (Closed) LER 50-269/99-09-00: Operation With Unrepaired Steam Generator Tube Ends

This event was resolved through the December 15, 2000, issue of Amendment Numbers 318, 318, and 318 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55. The amendments redefined operating license and technical specification requirements for repair of the Oconee steam generator tube ends.

.3 (Closed) URI 50-269,270/99-08-06: Steam Generator Tube End Anomalies Notice Of Enforcement Discretion (NOED)

The December 15, 2000, issue of Amendments No. 318, 318, and 318 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55 included the definition that: “Axial tube imperfections of any depth observed between the primary side surface of the tube sheet clad and the end of the tube are excluded from this repair limit.” This definition excludes the tube end anomalies (TEAs) from consideration for tube repair or removal from service, therefore supporting the licensee’s conclusion that the TEAs were not part of the primary to secondary pressure boundary.

.4 (Closed) LER 50-269/99-07-(00,01): Emergency Operating Procedure Inadequate Due to Deficient Review and Validation

The licensee identified that under certain single failures, operation of the emergency core cooling system (ECCS), as directed by the existing emergency operating procedures (EOP), would not assure adequate core cooling following an accident. The single failures involved:

- One of the borated water storage tank isolation valves failing to close when transferring LPI pump suction to the containment sump with containment pressure <12 psig causing loss of ECCS recirculation through pump cavitation following a loss of coolant accident.

- Failure to redirect HPI pump minimum flow to the LPI pump’s suction instead of the letdown storage tank when using the HPI pump(s) for high pressure re-circulation following a loss of coolant accident.

- Failure of a containment sump isolation valve to open supporting a train of LPI with a small break loss of coolant accident from a core flood tank piping break affecting the injection capability of the other LPI train.

- Failure to manually operate the cross-connect valve between LPI trains following a failure of one LPI train and a small break loss of coolant accident from a core flood tank piping break affecting the injection capability of the other LPI train.

- Failure of one train’s piggyback valve to open due to being powered from a non-qualified alternate power source and the other train fails following a small break loss of coolant accident.

A senior reactor analyst reviewed the licensee’s corrective action documents (PIPs O-99-3123, O-99-3702, O-99-3703, O-99-3863, O-99-4113) associated with this LER and the licensee’s designated corrective actions in these documents (including a revised ECCS single failure calculation, an extensive re-verification of the EOPs and programmatic changes to the EOP verification and validation process). Also, to determine safety significance a Significance Determination Process Phase III risk evaluation was performed. The inability of the ECCS to carry out its safety function was of very low safety significance, due to the low probability of the accident(s) occurring combined with the unique conditions existing that would cause the single failure coincident with the accident. The apparent cause of the inadequate EOPs was a lack of an engineering analysis of ECCS single failures and weak programmatic controls in the verification and validation of EOPs.

4OA6 Meetings

Exit Meeting Summary

The inspectors presented the inspection results to Mr. M. Nazar, Station Manager, and other members of licensee management at the conclusion of the inspection on April 9, 2001. The licensee acknowledged the findings presented. No proprietary information was identified.

4OA7 Licensee Identified Violations The following finding of low significance was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG -1600 for being dispositioned as a Non-Cited Violation (NCV).

If you deny the NCV, you should provide a response with the basis for your denial, within 30 days of the date of this inspection report, to the 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 facility.

NCV Tracking Number	Requirement Licensee Failed to Meet
NCV 50-269,270,287/00-08-05	10 CFR 50, Appendix B, Criterion III, “Design Control,” requires in part that applicable regulatory requirements and design bases be correctly translated into procedures. 10 CFR 50.46(d) requires an Emergency Core Cooling System that meets the general requirements of Criterion 35 of Appendix A. Appendix A requires an Emergency Core Cooling System capable of withstanding a single failure and still accomplish the system’s safety function. As of September 23, 1999, the operation of the Emergency Core Cooling System as directed by the Emergency Operating Procedures was unable to perform its safety function given certain single failures. These single failures and the licensee corrective actions are more fully described in Licensee Event Report 50/269/99-07 (Section 40A3.4).

PARTIAL LIST OF PERSONS CONTACTED

Licensee

T. Coutu, Superintendent of Operations
T. Curtis, Mechanical System/Equipment Engineering Manager
M. Nazar, Station Manager
W. Foster, Safety Assurance Manager
B. Hamilton, Engineering Manager
D. Hubbard, Modifications 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

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-269,270,287/00-08-01	URI	Risk Significance of High Temperatures in the Low Pressure Injection and High Pressure Injection Pump Rooms (Section 1R04.3)
50-269,270,287/00-08-02	URI	Risk Significance of Potential Flooding Problem From Non-Safety Related Lines in the Auxiliary Building (Section 1R06.1)
50-269,270,287/00-08-03	URI	Risk Significance of Potential Flooding Problem From Fire Suppression Systems in the Cable Spreading Rooms (Section 1R06.2)
50-269,270,287/00-08-04	URI	Risk Significance of Uncontrolled Design Changes to Penetration Room Blowout Panels (Section 1RO6.3)

Previous Items Closed

50-269/00-07-00	LER	Loss of Positive Control of a Radioactive Calibration Source (Section 4OA3.1)
50-269/99-09-00	LER	Operation With Unrepaired Steam Generator Tube Ends (Section 4OA3.2)
50-269,270/99-08-06	URI	Steam Generator Tube End Anomalies NOED (Section 4OA3.3)
50-269/99-07-(00,01)	LER	Emergency Operating Procedure Inadequate Due to Deficient Review and Validation (Section 4OA3.4)

LIST OF ACRONYMS USED

AC	-	Alternating Current
ACB	-	Air Circuit Breaker
BTP	-	Branch Technical Position
CC	-	Component Cooling
CF	-	Core Flood
CFR	-	Code of Federal Regulations
CRDM	-	Control Rod Drive Mechanism
DBD	-	Design Basis Document
DC	-	Direct Current
DHR	-	Decay Heat Removal
EAL	-	Emergency Action Level
ECCS	-	Emergency Core Cooling System
EOC	-	End-of-Cycle
EOP	-	Emergency Operating Procedure
ET	-	Eddy Current
F	-	Fahrenheit
FSAR	-	Final Safety Analysis Report
HAWT	-	High Activity Waste Tank
HPI	-	High Pressure Injection
HPSW	-	High Pressure Service Water
IP	-	Inspection Procedure
KHS	-	Keowee Hydro Station
LAWT	-	Low Activity Waste Tank
LCO	-	Limiting Conditions for Operation
LER	-	Licensee Event Report
LRT	-	Leak Rate Test
LPI	-	Low Pressure Injection
LPSW	-	Low Pressure Service Water
LSCM	-	Loss of Sub-Cooling Margin
LTOP	-	Low Temperature Overpressure Protection
MCC	-	Motor Control Center
NCV	-	Non-Cited Violation
NRC	-	Nuclear Regulatory Commission
NRR	-	Nuclear Reactor Regulation
NSD	-	Nuclear System Directive
PARS	-	Publicly Available Records
PI	-	Performance Indicator
PIP	-	Problem Investigation Process
PMT	-	Post-Maintenance Testing
PT	-	Penetrant
PWSCC	-	Primary Water Stress Corrosion Cracking
RBCU	-	Reactor Building Cooling Unit
RBS	-	Reactor Building Spray
RCP	-	Reactor Coolant Pump
RCS	-	Reactor Coolant System
RPV	-	Reactor Pressure Vessel

SBLOCA	-	Small Break Loss of Coolant Accident
SDP	-	Significance Determination Process
SER	-	Safety Evaluation Report
SRP	-	Standard Review Plan
SSC	-	Structure, System and Component
SSF	-	Standby Shutdown Facility
T/C	-	Thermocouple
TS	-	Technical Specification
UFSAR	-	Updated Final Safety Analysis Report
URI	-	Unresolved Item
USQ	-	Unresolved Safety Question
UT	-	Ultra-Sonic Testing

DOCUMENTS REVIEWED

The following documents were reviewed during the inspection activities discussed in Section 1RO2 of this report:

Screening Documents

Minor Modification OE-15379, Upgrade EWST Instrumentation Lines, 8/31/00

Minor Modification ONOE-13577 and Procedure TN/0/A/3577/MM/01E, SQUG Rewire KHU-1 Emergency Start Aux Relays and Place in Control Circuit for 43C/1X Relay Contact with 43C/2X Relay Contact in Master Relays 4A and 4B Control Circuit, 11/22/99

Minor Modification ONOE-12475 and Procedure TN/1/A/12475/MM/01E, Replace Operator on Valve 1HP-409, 2/17/00

Minor Modification ONOE-15707, Engineering Instructions Install new Spring Pack in Operator for Valve 1LP-19

Minor Modification ONOE-14521, Replace valve 1HP-64 with a DMV-1228 item, 1/26/00

Minor Modification ONOE-12875, Modify Two Support/Restraints, 3/23/00

Minor Modification ONOE-12963, SQUG Replacement of KHS Transformer Cx Differential Relays, 3/9/99

Station Modification NSM OE-33054, Part AM1, Add/modify Main Steam System Piping Supports, 12/13/99

TN/1/A/4697/MM01E, Replacement of Relays in switchgear 1TC, 4/4/00

Abnormal Procedure (AP)/1/A/1700/011, Loss of Power, rev. 26A

AP/1/A/1700/26, Loss of Decay Heat Removal, 2/26/01

AP/3/A/1700/26, Loss of Decay Heat Removal, rev. 6

AP/2/A/1700/16, Abnormal Reactor Coolant Pump Operation (restricted to repair of 2HP-67), rev. 11A

AP/2/A/1700/19, Loss of Main Feedwater, rev. 11

Evaluation Documents

Minor Modification ONOE-12475 and Procedure TN/1/A/12475/MM/01E, Replace Operator on Valve 1HP-409, 2/17/00

Minor Modification ONOE-15707, Engineering Instructions Install new Spring Pack in Operator for Valve 1LP-19

Nuclear Station Modification (NSM) ON-33056/0, Add Strainers to the Motor Driven EFW Pump Suction Piping, 2/10/00

Revise UFSAR Section 3.4.1.1.1 to Include Additional Information to Clarify Turbine Building Flood Description per PIP 00-1278, 7/11/00

Minor Modification ONOE-12477 and ONOE-12478, Motor Operator Replacement Modifications for Valves 1LP-17 and 1LP-18, 6/22/00

UFSAR Update Package Number 00-75, Revise Net Positive Suction Head for RBS and LPI Pumps, 12/28/00

ONOE-13991, Revise SSF ASW Pump Minimum Flow Requirements, 8/19/99

ONOE-15541, Revise Fire Protection Water Supply Design Basis, 10/12/00

ONOE-14239, Change control Room Computer BWST Emergency Low Statalarm Set Point from 7 to 9 feet, 11/10/99

OE-14877, Replace Valve 1LP-17 and Delete Valve 1LP-150, 5/24/00

UFSAR 9.3.3.2.1 Revision to Include LPI Series Mode of Operation, 8/28/00

UFSAR December 31, 1999, Update, Section 3.10.1, “Seismic Qualification Criteria” and 8.3.4.1.6.1 “Cable Installation”, 1/30/01

UFSAR 6.3.3.2 Revision, Change time Required for LPI Pumps to Reach Full Speed from 8 Seconds to 5 Seconds, 10/31/00

Miscellaneous Documents

Duke Power Nuclear System Directive, NSD-209, 10 CFR 50.59 Evaluation, rev. 8

Oconee Nuclear Station Updated Final Safety Analysis Report (UFSAR), 1999 revision dated June 30, 2000

Duke Power Assessment Report, SA-00–10, 10 CFR 50.59 Review Process Bench Marking (PIP G-00-00165), 4/12/200

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.