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

July 30, 2001

EA-01-161

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 50-269/01-02, 50-270/01-02, AND 50-287/01-02 (EXERCISE OF ENFORCEMENT DISCRETION)

Dear Mr. McCollum:

On June 30, 2001, the NRC completed inspections at your Oconee Nuclear Station. The enclosed report documents the inspection findings which were discussed on July 10, 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 determined that small amounts of leakage from the Unit 1, 2, and 3 reactor vessel head penetrations had been ongoing for an extended period of time. Because Technical Specifications (TS) require that with any reactor coolant pressure boundary leakage, the plant be placed in hot standby within 12 hours, the NRC concluded that a violation of TS occurred. This violation involved equipment failure not avoidable by reasonable quality assurance measures or management controls and is considered to have resulted from matters not within your control. Based on this and after consultation with the Regional Administrator, Region II, and the Director, Office of Enforcement, it has been determined that the exercise of enforcement discretion is warranted in accordance with Section VII.B.6 of the "General Statement of Policy and Procedure for NRC Enforcement Actions - May 1, 2000," NUREG-1600, as amended on November 3, 2000 (65 Federal Register 59274) (Enforcement Policy). Accordingly, a Notice of Violation will not be issued. Additionally, the inspectors also identified three other issues of very low safety significance (Green and No Color).

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/

Loren R. Plisco, Director
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 50-269,270,287/01-02

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E-MAIL COPY?	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO	YES NO

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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:	50-269/01-02, 50-270/01-02, 50-287/01-02
Licensee:	Duke Energy Corporation
Facility:	Oconee Nuclear Station, Units 1, 2, and 3
Location:	7800 Rochester Highway Seneca, SC 29672
Dates:	April 1, 2001 - June 30, 2001
Inspectors:	M. Shannon, Senior Resident Inspector D. Billings, Resident Inspector E. Chrisnot, Resident Inspector S. Freeman, Resident Inspector J. Blake, Senior Project Manager (Sections 1RO8.1, 1RO8.2, 4OA3.3, 4OA3.4) W. Bearden, Reactor Inspector (Sections 1RO8.1, 1RO8.2, 4OA3.3, 4OA3.4) D. Payne, Senior Operations Engineer (Section 1R11.2) G. Salyers, Emergency Preparedness Inspector (Section 1R11.2) E. Testa, Radiation Specialist (Section 2OS2) D. Thompson, Physical Security Inspector (Sections 4OA5.1, 4OA5.2 - in office review)
Approved by:	Malcolm T. Widmann, Acting Chief Reactor Projects Branch 1 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000269,270,287/01-02, on 04/01/2001 - 06/30/2001, Duke Energy Corporation, Oconee Nuclear Station, Units 1, 2, and 3 - Inservice Inspection Activities, Access Control, and Human Performance.

The inspection was conducted by resident inspectors, and regional based inspectors specializing in the areas of health physics, security, maintenance, and operator licensing. The inspection identified two Green findings and two No Color findings (one of which involved an exercise of enforcement discretion and the other a cross-cutting issue). 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: Barrier Integrity

- No Color. A violation of Technical Specifications was identified for exceeding reactor coolant system pressure boundary leakage limits due to cracks in alloy 600 control rod drive mechanism and thermocouple reactor head penetration nozzles. The leakage existed for an extended period of time prior to its discovery; however the licensee’s leak detection practices were adequate and would not have been expected to identify the small amount of leakage during plant operation. Based on the conclusion that the violation was not avoidable by reasonable licensee quality assurance measures and management controls, the NRC is refraining from issuing enforcement action in accordance with section VII.B.6 of the NRC Enforcement Policy.

  There was minimal consequence to this condition because the leak rates were below 1 gallon per minute. The potential safety consequence of circumferential cracking is currently being evaluated by the NRC as a generic problem (Section 1R08.2).

Cornerstone: Physical Protection

- Green. The inspectors identified that the licensee failed on several occasions to detect the contractors conducting tests of the protected area exterior intrusion detection system during an inspection conducted on June 5 - 8, 2000.

  This finding was determined to be of very low significance because no intrusion occurred and there was not two or more similar findings in four quarters (Section 4OA5.1).

- Green. The inspectors identified that in one of four exercises during an inspection conducted on June 5 - 8, 2000, the licensee failed to interdict the intruders before they gained access to vital areas.

This finding was determined to be of very low significance because there was not a loss of a full target set and there was not two or more similar findings in four quarters (Section 4OA5.2).

Cross-Cutting Issues: Human Performance

- No Color. One substantive cross-cutting issue was identified in the area of human performance. From April 13, 2000, through June 30, 2001, lack of attention to detail has resulted in two events, rendered safety-related equipment inoperable five separate times, and resulted in two other instances with the potential to cause events or make safety-related equipment inoperable (Section 4OA4).

B. Licensee Identified Violations

Three violations of very low significance which were identified by the licensee were reviewed by the inspectors. Corrective actions taken or planned by the licensee appear reasonable. The violations are listed in section 4OA7 of this report.

Report Details

Summary of Plant Status:

Unit 1 began the inspection period at 100 percent power. The unit remained at 100 percent power (except for a brief period of power reduction for control rod and main turbine valve testing) until May 18, 2001, when power was reduced to 85 percent due to a broken sight glass in the cooling water system to a heater drain pump. Following repairs, the unit was returned to 100 percent power on May 19, 2001. On May 21, 2001, the unit was run back to 18 percent power due to problems in the generator seal oil expansion tank. Following repairs, the unit was returned to 100 percent power on May 25, 2001, and remained there through the end of the inspection period.

Unit 2 began the inspection period at 100 percent power. On April 26, 2001, the unit was shutdown for the End-of-Cycle (EOC)18 refueling outage. On April 28, 2001, an inspection of the reactor vessel head revealed that four control rod drive nozzles were leaking. After refueling and nozzle repairs, the unit was taken critical on May 28, 2001. From June 5, 2001, until June 14, 2001, the unit remained slightly below 100 percent power due to discrepancies in core thermal power calculations. Following resolution, the unit was returned to 100 percent power on June 14, 2001, and remained there through the end of the inspection period.

Unit 3 began the inspection period shutdown for control rod drive mechanism nozzle repairs. The unit was taken critical on April 23, 2001, and returned to 100 percent power on April 25, 2001. The unit remained at 100 percent power until June 28, 2001, when power was reduced to 60 percent due to a leak on a main feedwater pump vent line. Following repairs, the unit was returned to 100 percent power on June 29, 2001, and remained there through the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R04 Equipment Alignment

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:

- Unit 1A Motor Driven Emergency Feedwater Pump
- Standby Shutdown Facility (SSF) Ventilation Systems
- Unit 3B High Pressure Injection Train

b. Findings

No findings of significance were identified.

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:

- Units 1, 2, and 3 auxiliary building 6th floor
- Units 1 and 2 cable rooms
- Unit 2 reactor building (during EOC 18 refueling outage)
- Keowee Main Transformer

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed procedure TT/2/A/0150/061, Low Pressure Injection (LPI) Cooler Test - Series Mode, Revision 0 to ensure that the cooler would be able to supply the necessary cooling as described in the Updated Final Safety Analysis Report (UFSAR). The inspection focused on deficiencies that could mask degraded performance of the heat exchangers and/or could result in common cause heat sink performance problems. Also assessed was whether 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.

IR08 Inservice Inspection (ISI) Activities

.1 Review of Outage Activities

a. Inspection Scope

The inspectors observed in-process ISI work activities and reviewed selected ISI records. The observations and records were compared to the Technical Specifications (TS) and the applicable Code [American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 1989 Edition, with no Addenda].

Portions of the following ISI examinations performed during the Unit 2 EOC 18 refueling outage (which was the second outage of the second period of the third 10-year interval) were observed:

- ultrasonic (UT) — 12" dia. x 1.125" wall Safety Injection weld, Identification (ID) No. 2-53A-10-8. (UT Procedure NDE-600)

- liquid penetrant (PT) — 12" dia. x 1.125" wall Safety Injection weld, Identification (ID) No. 2-53A-10-8. (PT Procedure NDE-35)

- eddy current (ET) — Observed 15 to 20 examples of Bobbin and Rotating Coil inspection (data collection) from both tube sheets of each of the Unit 2 once through steam generators (OTSG).

  Observed 10 to 12 examples of the ET data evaluation by resolution analysts assigned to resolve discrepancies between the primary and secondary analyst's calls.

A sample of ISI issues in the licensee's corrective action program were also reviewed. Specifically, Problem Investigation Process report (PIP) O-01-01681, concerning misidentification of ET data. The review included the associated corrective action documentation.

In addition to the above observations and reviews for the current Unit 2 outage, the inspectors also reviewed ASME Section XI repairs and non-destructive examination (NDE) activities associated with leakage on the Unit 3 reactor head control rod drive mechanisms (CRDMs). Activities observed included PT and UT examinations of ongoing weld repairs on CRDM nozzles. Additionally, the inspectors reviewed root cause failure analysis reports and metallurgy reports, as well as interviewed licensee personnel.

b. Findings

No findings of significance were identified.

.2 Leakage Detection

a. Inspection Scope

The inspectors reviewed the licensee's program for leak detection to determine if current leak detection practices should have identified the reactor vessel head penetration leakage experienced on all three units during plant operation. The review included:

- descriptions of the leak detection practices and requirements provided in the TS
- recorded leakage data
- discussions with cognizant plant personnel
- photographs of areas examined for evidence of the reactor vessel head penetration leakage

The procedures and recorded data that were reviewed are listed at the end of this report.

b. Findings

A No Color violation was identified for exceeding reactor coolant system (RCS) pressure boundary leakage limits due to cracks in alloy 600 CRDM and thermocouple reactor head penetration nozzles.

During refueling outages on Units 1 and 2, and an unscheduled outage on Unit 3, between November 2000 and April 2001, plant personnel identified indications of RCS pressure boundary leakage. Visual inspection of reactor pressure vessel (RPV) head penetrations by the licensee found multiple cases of boric acid residue accumulation. Leaks were identified in one CRDM nozzle and five of eight thermocouple nozzles on Unit 1, four CRDM nozzles on Unit 2, and nine CRDM nozzles on Unit 3. The licensee concluded that the leaks likely resulted from primary water stress corrosion cracking (PWSCC) axial cracks, which initiated in the weld between the nozzle and the head. Two of the leaking nozzles on Unit 3 and one on Unit 2 had circumferential cracks above the J-groove welds. In each case the small amount of boric acid crystal deposits indicated that the leakage rates were very low.

Unit 1 unidentified reactor coolant leakage, as determined by a water inventory balance, remained approximately 0.2 gallons per minute (gpm) during the operating cycle. The unidentified leak rate was not unusual and well below the TS limit of 1 gpm. Low boron concentrations in the reactor building normal sump (RBNS) indicated very little leakage from RCS sources. Additionally, review of air sampling data plots for Unit 1 confirmed that air sample counts did not significantly change during the operating cycle.

Unit 3 unidentified reactor coolant leakage also remained approximately 0.2 gpm prior to shutdown in February 2001 for repairs of an unrelated problem. Routine RBNS boron samples were not taken during the operating cycle. However, a single RBNS sample was taken on January 20, 2001. This sample showed low boron concentration, consistent with unidentified leakage calculations.

Technical Specification Limiting Condition for Operation (LCO) 3.4.13.a requires that RCS leakage shall be limited to "No PRESSURE BOUNDARY LEAKAGE," when in Modes 1, 2, 3 and 4. The associated action statement requires that with any pressure boundary leakage, be in Hot Standby within 12 hours and in Cold Shutdown within the following 36 hours. Boric acid residue was discovered during boric acid inspections of the reactor head penetrations while the plant was in Mode 3 and the licensee met the required action upon discovery of the condition. Although it is not possible to determine when the reactor vessel head penetration leakage began, it is clear that it had existed for a time greater than the 12 hours required to be in Hot Standby and therefore, constitutes a violation of the TS.

Based on U.S. and foreign industry operating experience, the licensee performed voluntary inspections to address generic concerns related to reactor head penetration cracking. These inspections were conducted in 1994, 1996, and 1999. They consisted of augmented ET inspections and UT inspections on the Unit 2 CRDMs, and were conducted prior to and in response to NRC Generic Letter 97-01, Degradation of CRDM Nozzle and other Vessel Closure Head Penetrations. These inspections did not identify significant head penetration nozzle cracking. In addition, the licensee's leakage detection methods would not have been expected to identify the small amount of pressure boundary leakage that occurred during the operating cycle, and the licensee continues to conduct enhanced reactor head penetration boric acid visual inspections in response to the recommendations of NRC Generic Letter 88-05, Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components In PWR Plants. Therefore, since the TS violation resulted from equipment failure not avoidable by reasonable quality assurance measures or management controls, discretion is exercised in accordance with section VII.B.6 of the NRC Enforcement Policy and a notice of violation will not be issued. There was minimal consequence to this condition because the leak rates were below 1 gpm. The potential safety consequence of circumferential cracking is being evaluated by the NRC.

1R11 Licensed Operator Requalification

.1 Simulator Training

a. Inspection Scope

The inspectors observed simulator training on June 13, 2001, for reactor operators and senior reactor operators. The inspectors observed simulator training scenarios involving a loss of main feedwater with anticipated transient without scram; a failed closed power operated relief valve; and a stuck open pressurizer safety 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.

.2 Requalification Program

a. Inspection Scope

The inspectors reviewed the facility operating history since the last requalification program inspection for indications of operator weaknesses. The inspectors also reviewed the biennial written examinations for five shift crews and evaluated their effectiveness in providing a basis for assessing operator knowledge of material covered in the requalification training program. Examination quality, licensee effectiveness in integrating industry, plant and student feedback into the requalification training program, and examination development methodology were evaluated as well. The inspectors observed annual dynamic simulator examinations (five scenarios) for four operator teams to assess the adequacy of the licensee’s evaluation of operator knowledge and abilities. During these observations, the inspectors assessed licensee evaluator effectiveness in pinpointing operator performance deficiencies requiring supplemental training. The inspectors also evaluated and observed portions of the walkthrough examination administered during this requalification segment to assess evaluator performance.

The inspectors reviewed and evaluated the licensee’s remedial training program for operator deficiencies identified during the previous year and observed classroom and simulator remedial training administered to three operators. The inspectors also reviewed a sample of on-shift licensed operator qualification records, watchstanding records and medical records to ensure compliance with 10 CFR 55.59, Requalification and 10CFR 55.53, Conditions of License.

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 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:

- high pressure service water (HPSW) jockey pump and valves HPSW-2, HPSW-5, and HPSW-8
- auxiliary building sump pumps
- nuclear instrument system gamma metrics high voltage power supplies
- Unit 3 component cooling pumps
- Unit 2 east penetration room door sweep (ventilation inleakage)
- Unit 2A LPI cooler discharge piping support

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, necessary steps were taken to plan and control the resulting emergent work activities; (4) problems associated with risk assessment and emergent work were adequately identified and resolved.

- removal of the station auxiliary service water pump from service due to parts problems
- removal of cooling water from reactor building cooling unit (RBCU) 2B
- cell replacement in 1CA battery
- closure of normal cooling water supply to Unit 1 Alterex exciter
- removal of SSF from service following draining of the Unit 2 condenser circulating water header
- removal of 3A LPI system from service
- risk management/control of work orders remaining open when work was delayed and not rescheduled

- removal of instrument air dryers from service for modification

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Non-routine Plant Evolutions

a. Inspection Scope

The inspectors reviewed personnel performance during selected non-routine events and/or transient operations to determined if operators responded in accordance with procedures and training. Documentation was reviewed to determine if personnel performance deficiencies were captured in the licensee's corrective action program. The inspectors reviewed operator logs, plant computer data, or strip charts to determine what occurred and how the operators responded. The non-routine evolutions reviewed during this inspection period included the following:

- Unit 3 reactor startup and power escalation on April 23, 2001, following forced outage
- Unit 2 shutdown for refueling outage on April 26, 2001
- Unit 2 reactor startup and power escalation on May 28, 2001, following refueling outage
- Unit 1 power reduction on May 21, 2001, due to seal oil problems
- Unit 1 power escalation on May 24, 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 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 work as intended, and were appropriately controlled; and (5) where continued operability was considered unjustified, the impact on TS LCO. The inspectors reviewed the operability evaluations described in the following PIPs:

- PIP O-00-01456, Low Pressure Service Water (LPSW) Net Positive Suction Head (NPSH) on Loss of Offsite Power

- PIP O-00-01567, LPSW Suction Cross-connect
- PIP O-00-02085, Effects of High Pressure Service Water (HPSW) Start on LPSW NPSH
- PIP O-00-03835, Minimum Lake Level Needed for Reverse Gravity Flow to LPSW Suction
- PIP O-01-00595, Blowout Panels in East Penetration Room
- PIP O-01-00815, Cable and Penetration Room Doors
- PIP O-01-01941, LPSW Piping to 2A LPI Heat Exchanger Wall Thickness

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 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:

- PT/0/A/0251/010, Auxiliary Service Water Pump Test, Revision 47
- PT/1/A/0600/013, 1B Motor Driven Emergency Feedwater Pump Test, Revision 38
- PT/2/A/0203/006B, LPI Pump Test - Decay Heat, Revision 17
- PT/3/A/0600/13, 3A Motor Driven Emergency Feedwater Pump Test, Revision 36
- TT/2/A/0150/058, 2LP-9 & 10 Differential Pressure Test, Revision 0
- WO 98314085, Replace Backup IA Aftercooler Moisture Separator Drain Traps

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

The inspectors conducted reviews and observations for selected licensee outage activities to ensure that: (1) the licensee considered risk in developing the outage plan; (2) the licensee adhered to the outage plan to control plant configurations based on risk; (3) that mitigation strategies were in place for losses of key safety functions; and (4) the licensee adhered to operating license and TS requirements. Between April 24, 2001, and May 30, 2000, the inspectors reviewed the following activities related to the refueling outage on Unit 2 for conformance to the applicable procedure and witnessed selected activities associated with each evolution:

- reactor shutdown
- reactor cooldown and initiation of decay heat removal
- mid-loop operations to install and remove steam generator nozzle dams
- RCS level and low range pressure instrument calibrations
- refueling interlocks
- containment closure
- electrical power alignments and testing
- condenser waterbox isolation
- fuel movement
- Mode change verification
- reactor startup
- zero power physics testing

The inspectors also walked down the Unit 3 containment during heat-up prior to reactor startup.

b. Issues and 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 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.

- IP/0/A/0305/001M, Reactor Protective System Channel A Reactor Coolant (RC) Pressure Instrument Calibration, Revision 61
- IP/0/A/0310/003B, Engineering Safeguards System Analog Channel A RC Pressure Channel Calibration, Revision 55
- PT/0/A/0620/009, Keowee Hydro Operation , Revision 19
- PT/1/A/0261/010, Essential Siphon System Vacuum Test, Revision 1
- PT/2/A/0610/001J, Emergency Power Switching Logic Functional Test, Revision 19
- PT/2/A/0610/001L, Load Shed Channel Verification, Revision 3

b. Findings

No findings of significance were identified.

1R23 Temporary Modifications

a. Inspection Scope

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:

- MP/0/A/3009/022, Motor AC-100HP and Above - Disassembly, Repair, and Assembly, Revision 4
- Modification ONOE 16123, Remove Air Supply from Unit 1 Alterex Temperature Control Valve and Fail Valve Open

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS2 ALARA Planning and Controls

a. Inspection Scope

The inspectors reviewed the ALARA (As Low As is Reasonably Achievable) training, planning, execution and results of the Unit 3 forced outage reactor vessel head repair work. Radiation worker training, instructor notes, radiation worker training student guides, review questions, student study materials, tests and test results were reviewed for initial radiation worker and vendor radiation worker bypass employees. Radiation Work Permit (RWP) No. 3167 UNIT 3 RX (Reactor) BLDG UNDER REACTOR VESSEL HEAD CRDM INSPECTION AND REPAIR, and bar code sub-tasks were reviewed. Shutdown chemistry crud burst procedure CP/0/B/2002/010, Addition of Hydrogen Peroxide to the Reactor Coolant System, Revision 21 and clean-up results for the shutdown were reviewed by the inspectors. Initial and post decontamination radiation dose surveys were evaluated to determine work area dose reductions. ALARA Briefing Checklists for RWP No. 3167 were evaluated for worker attendance, material content, radiological concerns and ALARA controls. The inspectors evaluated post job critiques, Total Effective Dose Equivalent (TEDE) ALARA Evaluations, Multibadge (extremity badge) dose evaluations, electronic dosimeter placement and individual worker doses, and requests for exposure dose extensions. The inspectors independently evaluated the licensee's Beta dosimetry analysis. The use of a reactor half-hemisphere head mock-up and the worker shielded booth for training and equipment evaluation were observed. The outage job ALARA work plan dose estimates and man-hour revisions were evaluated. The evaluations, observations, and reviews were conducted to determine if licensee activities were performed according to UFSAR, TSs, Selected Licensee Commitments (SLC), and 10 CFR Part 20 requirements. The inspectors interviewed plant management, supervisors, craft workers, and health physics technicians to evaluate their knowledge and input for the ALARA work plan. Inspectors attended daily turnover and management outage planning meetings. Documents reviewed during this inspection included:

- Nuclear Policy Manual Nuclear System Directive 507, Radiation Protection, Dated July 17, 2000
- Radiation Protection Policy Manual, ALARA Policy III-04, Revision 0
- System ALARA Manual Section IV, ALARA Planning, Revision 11
- Procedure CP/0/B/2002/010, Addition of Hydrogen Peroxide to the Reactor Coolant System, Revision 21

- Procedure SH/0/B/2000/001, Operational Beta Program, Revision 000

- Procedure SH/0/B/2000/007, Placement of Personnel Dosimetry for Non-Uniform Radiation Fields, Revision 001

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA3 Event Followup

.1 (Closed) Licensee Event Report (LER) 50-269/00-08-00: Pressurizer Relief Valve Setpoint Found Out-of-Tolerance

This LER addressed a test report which documented that the as-found setpoint for pressurizer code safety valve 1RC-67 was outside the three percent tolerance allowed by TS 3.4.10. The inspectors reviewed the LER and determined that the event had a very low risk significance because licensee analysis determined that pressure would peak below the TS safety limit when using the as-found setpoint. This LER has been entered into the licensee’s corrective action program as PIP O-00-04563. No findings of significance were identified. This event did not constitute a violation of NRC requirements.

.2 (Closed) LER 50-287/01-02-00, 01: Containment Cooler Inoperable Due to Design, Exceeds TS Time

On March 3, 2001, during testing, the 3B reactor building cooling Unit (RBCU) exceeded the operability limit for motor stator temperature and was declared inoperable. The licensee concluded that if voltages were higher than normal, stator temperatures might have exceeded qualification limits for accident conditions; thereby rendering the RBCU inoperable from the time of motor installation on December 19, 1999. Because the associated safety analysis takes credit for two of the three RBCUs to transfer heat in the postulated post-accident environment, this event had a credible impact on safety considering single failure requirements. The licensee’s evaluation of the problem using actual initial conditions (e.g., containment and ultimate heat sink temperatures, cooling coil fouling coefficients, etc..) for the period that the 3B RBCU was considered inoperable, concluded that either one of the remaining RBCUs would have been capable of handling the expected heat load. The root cause was determined to be a vendor design configuration and a larger than normal air gap between the stator and rotor. The licensee replaced the motor and sent a report addressing the RBCU high stator temperatures to the RBCU vendor. The licensee also plans changes to procedures to require measurement of motor stator temperatures during PMT and is evaluating possible changes to TS 3.6.5.

The inspectors evaluated this event using the Containment Integrity SDP. Because one RBCU alone would have performed the safety function of cooling the containment during the period when the 3B RBCU was inoperable (i.e., containment function would not have been affected), the inspectors considered this event to be of very low safety significance (Green). This licensee identified violation of TS 3.6.5, which is captured in the licensee's corrective action program as PIP O-01-00786, is documented in Section 4OA7 of this report.

.3 (Closed) LER 50-269/00-06-00, 01: Reactor Coolant System Pressure Leakage due to Cracks Found in Several Small Bore Reactor Vessel Head Penetrations

The reported condition involved leakage on one CRDM nozzle and five of eight thermocouple nozzles on Unit 1. The leaks were determined to have resulted from cracks (predominately due to PWSCC), which had initiated in the weld material. The licensee captured this issue in their corrective action program as PIP O-00-04134. Repairs to the reactor head penetrations were completed prior to restart from the outage. No other findings or issues of significance were identified. This event constituted a violation of NRC requirements, for which the NRC is refraining from issuing enforcement action in accordance with section VII.B.6 of the NRC Enforcement Policy. This issue was reviewed by the inspectors in Section 1R08.2.

.4 (Open) LER 50-287/01-01-00: Reactor Pressure Vessel Head Leakage due to Stress Corrosion Cracks Found in Nine Control Rod Drive Nozzle Penetrations

The reported condition involved leakage on nine CRDM nozzles on Unit 3. The leaks were determined to have resulted from cracks (predominately due to PWSCC), which had initiated in the weld material. The licensee captured this issue in their corrective action program as PIP O-01-00587. Repairs to the reactor head penetrations were completed prior to restart from the outage.

During this review, the inspectors identified an inconsistency between the estimated crack propagation rate contained in the licensee's root cause determination and observed conditions on Unit 3. The licensee stated that circumferential cracks would only result due to the presence of boric acid on the outer surface of the nozzle. This would result from leakage via axial cracks and collection of boric acid in the annulus between the CRDM nozzle and reactor head. The licensee concluded that circumferential flaws were expected to take between 3.5 to 10 years to grow through-wall. Visual inspections during outages would detect the leakage from the axial cracks prior to the development of the circumferential through-wall crack. The inspector determined the licensee did not find evidence of leakage for Units 1 and 2 reactor vessel head penetrations during previous outages. The licensee stated that there had also been no prior evidence of leakage for those CRDM nozzles located in the peripheral region of the head on Unit 3. Two CRDM nozzles, 50 and 56, located in the peripheral region, contained circumferential flaws which had grown through-wall or near through-wall. This represents an inconsistency in the licensee's conclusion, which would suggest either a larger circumferential crack growth rate or earlier leakage from axial flaws not identified during outage inspections. The potential safety consequence of circumferential cracking is currently being evaluated by the NRC as a pending generic problem, therefore, this LER remains open.

4OA4 Cross-Cutting Issues

a. Inspection Scope

On May 10, 2001, a grounding event occurred on 4160 Volt Switchgear 2TD which damaged one breaker on the bus. The inspectors reviewed this event for concerns related to the areas of human performance that could affect multiple cornerstones. The review included a check of the licensee's cause evaluation for the event plus a look at previous issues, documented in inspection reports, to identify any negative performance trends or patterns of a cross-cutting nature. The inspectors looked for patterns where human error contributed to events, made safety-related equipment inoperable, or affected plant risk and involved more than one cornerstone.

b. Findings

A No Color finding for a negative performance trend was identified in the area of human performance. From April 13, 2000, until completion of this inspection, lack of attention to detail has resulted in two events, rendered safety-related equipment inoperable five separate times, and resulted in two other instances with the potential to cause events or make safety-related equipment inoperable.

Examples supporting this negative performance trend are as follows:

- On May 10, 2001, maintenance personnel neglected to note in the work package that a ground strap had been installed on safety-related bus 2TD. Consequently, it was still in place later when the breaker with the ground strap was closed. This caused damage to the breaker and had the potential to cause other breakers to trip, which would have disabled safety-related equipment (NCV 50-269,270,287/01-02-01).

- On May 14, 2000, operators failed to notice they had the wrong revision of the procedure for placing the building spray system in its emergency alignment. Consequently, when the improper revision was implemented, RCS water was admitted to portions of the LPI system suction piping not designed for RCS pressure. This overpressurized the LPI suction piping and resulted in approximately 140 gallons lost from the RCS (NCV 50-287/00-05-05).

- During implementation of procedures for improved TS, licensee reviewers failed to recognize the addition of at least two new TS requirements. Consequently, later use of the procedures that implement these requirements resulted in two inoperable LPI trains during shutdown of Unit 3 on April 13, 2000, and several inoperable channels of the reactor protection system (RPS) on October 11, 2000 (NCVs 50-287/00-05-08 and 50-269,287/00-07-03).

- On April 19, 2000, while increasing power on Unit 2, operators failed to adhere to procedure limits and cautions, by not calibrating the nuclear instrumentation when readings indicated less than actual power. Consequently, several RPS trip functions were rendered inoperable during the power increase (NCV 50-270/00-05-04).

- Between April 13, 2000, and April 16, 2000, operators failed to verify the operability of the low temperature overpressure protection (LTOP) alarms prior to entry into the TS mode of applicability. Consequently, one train of LTOP was rendered inoperable beyond the TS completion time (NCV 50-287/00-06-04 and LER 50-287/00-03-00).

- On June 26, 2000, maintenance workers delayed work on an emergency power supply and began work on all three units' main exciter systems without notifying work control. Consequently, the workers unknowingly increased the likelihood of a unit trip while one of the emergency power supplies was degraded (NCV 50-269,270,287/00-05-03). Later, because work control was not aware of the delay, problems with testing on the other emergency power supply rendered both emergency power supplies to Unit 3 inoperable.

- On August 10, 2000, operators improperly signed that the valve alignment procedure for siphon seal water header B was completed even though they did not actually verify the position of the valves in the procedure, did not perform the procedure in sequence, and left four valves in a position not called for by the procedure. If left uncorrected this error had the potential to render safety-related equipment inoperable (NCV 50-269,270,287/00-06-02).

- On September 15, 2000, the licensee failed to evaluate and control the application method and amount of a flammable paint thinner in the Unit 2 East Penetration Room, which constituted a degradation in the defense-in-depth strategy to prevent fires. If left uncorrected this error had the potential to both cause an event and render safety-related equipment inoperable (NCV 50-270/00-06-01).

The common link among these findings was lack of attention to detail on the part of licensee personnel. In each example presented above, whether from failure to follow procedure, or from poor procedure preparation, lack of attention to detail resulted in an event, rendered safety-related equipment inoperable, or created the potential to cause an event or make safety-related equipment inoperable.

Previous operating experience (i.e., Information Notice 99-21) has indicated that plant risk is sensitive to changes in human performance probabilities and pointed out the incentive to ensure that human performance remains consistent with management expectations. The inspectors noted that the lack of attention to detail on the part of licensee personnel did not meet licensee management's expectations for human performance. In each example presented, a lack of attention to detail has been the cause. Because of the effect on safety-related equipment and because of the sensitivity of risk to human performance, the inspectors considered this performance trend a substantive cross-cutting issue not captured in individual issues and is characterized as a finding with "No Color."

4OA5 Other

.1 (Closed) URI 50-269,270,287/00-09-01: Failure to Properly Detect Intrusion Detection System Penetrations at the Perimeter Prior to Individuals Gaining Access to the Protected Area

Paragraph 8.3.1, Revision 9, of the Physical Security Plan and Contingency Plan, requires that "an intrusion detection system (IDS) shall be installed such that detection occurs exterior to the protected area barrier." On several occasions during the inspection conducted on June 5-8, 2000, the licensee failed to detect the contractors conducting the tests exterior to the protected area barrier. Using the Physical Protection SDP, this finding was evaluated as a vulnerability of safeguards systems or plans. Since no intrusion occurred, and there was not two or more similar findings in four quarters, the finding regarding the failure to detect the contractors during intrusion detection alarm testing was determined to be of very low safety significance (Green). The NRC also concluded that this finding did not involve a violation of regulatory requirements.

.2 (Closed) URI 50-260,270,287/00-09-02: Failure to Prevent a Simulated Adversary From Gaining Access to Vital Areas

The provisions of the Physical Security Plan and Contingency Plan, Revision 9, specifically Event 6 of the Contingency Plan, states in the event response under Suspected or Confirmed Intrusion or Sabotage Attempt, that the objective of the Security Force is to "interdict between intruders and vital areas." For one exercise during the inspection conducted on June 5-8, 2000, the licensee failed to interdict the intruders before they gained access to vital areas. Using the Physical Protection SDP, this finding was evaluated as a vulnerability of safeguards systems or plans. Since there was not a loss of a full target set and there was not more than two similar findings in four quarters, the finding regarding the failure to interdict the intruders before they gained access to the vital areas was determined to be of very low safety significance (Green). The NRC also concluded that this finding did not involve a violation of regulatory requirements.

.3 Institute of Nuclear Power Operations (INPO) Report Review

The inspectors reviewed the final report issued by INPO for the evaluation that was conducted at the Oconee facility during the weeks of August 21, 2000, and August 28, 2000. The inspectors did not note any safety issues in the INPO report that either warranted further NRC followup or that had not already been addressed by the NRC.

4OA6 Meetings

.1 Exit Meeting Summary

The inspectors presented the inspection results to Mr. W. McCollum, Site Vice President, and other members of licensee management at the conclusion of the inspection on July 10, 2001. In addition, the results of an in office review (Sections 4OA5.1, 4OA5.2) were discussed between Mr. D. Thompson, NRC Physical Security

Inspector, and Ms. J. Smith, Oconee Licensing, on July 24, 2001. The licensee acknowledged the findings presented. No proprietary information was identified.

.2 Annual Assessment Public Meeting

The NRC Senior Resident Inspector assigned to Oconee, the Division of Reactor Projects Acting Chief, Branch 1, and the Director of the Division of Reactor Safety met on June 27, 2001, with Duke Energy Corporation (DEC) to discuss the NRC's Reactor Oversight Process (ROP) annual assessment of safety performance for Oconee Nuclear Station during the period of April 2, 2000 - March 31, 2001. The major topics addressed were: the NRC 's ROP assessment program, the results of the Oconee Nuclear Station assessment, and the NRC's Agency Action Matrix. Attendees included DEC site management, members of the plant staff, local news media, and several state and local officials.

Following the annual assessment meeting, a brief meeting was held with state and local officials to discuss the ROP and NRC activities involving Oconee Nuclear Station. The Oconee Emergency Preparedness Manager and Safety Assurance Manager also attended the meeting.

Both of the meetings, which were held at the Oconee World of Energy Center, were open to the public. Information used for the discussions of the ROP is available from the NRC's document system (ADAMS) as accession number ML011980088. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADMAS/index.html (the Public Electronic Reading Room).

4OA7 Licensee Identified Violations

The following findings of very low significance were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as NCVs.

If you deny the non-cited violation, 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, 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
50-269,270,287/01-02-01	TS 5.4.1 requires written procedures be established, implemented and maintained covering the procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A , February 1978. Item 1.j. of Regulatory Guide 1.33, Revision 2, Appendix A requires an administrative procedure for jumper control. On May 10, 2001, the

	licensee failed to remove a ground strap from safety-related bus 2TD in violation of Maintenance Directive 4.4.13, ONS Maintenance and Modification Work Practices for Equipment Configuration Control, Revised August 14, 2000, as described in the licensee's corrective action program reference PIP O-01-01721 (Green).
50-287/01-02-02	TS 3.6.5 requires, in part, that three RBCUs be operable in Modes 1,2,3, and 4. The 3B RBCU was inoperable from December 19, 1999, to February 16, 2001. Due to the 3B RBCU being inoperable for greater then the TS completion time of seven days, the licensee was not in compliance with TS LCO 3.6.5. The circumstances involving the RBCU and the licensee's corrective actions are described in LER 50-287/01-02-00, 01 (Section 4OA3.2) (Green).
50-269/01-02-03	TS 5.4.1 requires written procedures be established, implemented and maintained covering the procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A , February 1978. Item 9.a. of Regulatory Guide 1.33, Revision 2, Appendix A requires that maintenance which can affect safety-related equipment be preplanned and performed in accordance with written procedures. Sometime before April 24, 2001, the licensee improperly aligned the 1B motor driven emergency feedwater pump because the procedure did not include instructions to perform axial alignment as described in the licensee's corrective action program reference PIP O-01-01402 (Green).

PARTIAL LIST OF PERSONS CONTACTED

Licensee

C. Boyd, Work Control Superintendent
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

C. Casto, Director, Division of Reactor Safety, Region II
D. LaBarge, Project Manager, NRR
M. Widmann, Acting Chief, Branch 1, Division of Reactor Projects, Region II

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

50-269,270,287/01-02-01	NCV	Failure to Remove a Ground Strap From Safety-Related Bus 2TD as Required by Maintenance Directive 4.4.13, ONS Maintenance and Modification Work Practices for Equipment Configuration Control, Revised August 14, 2000 (Section 4OA7)
50-287/01-02-02	NCV	3B RBCU Was Inoperable Greater Than The Time Allowed by TS 3.6.5 (Section 4OA7)
50-269/01-02-03	NCV	Improper Alignment of The 1B Motor Driven Emergency Feedwater Pump due to Inadequate Procedure (Section 4OA7)

Previous Items Closed

50-269/00-08-00	LER	Pressurizer Relief Valve Setpoint Found Out-of-Tolerance (Section 4OA3.1)
50-287/01-02-00, 01	LER	Containment Cooler Inoperable Due to Design, Exceeds TS Time (Section 4OA3.2)
50-269/00-06-00, 01	LER	Reactor Coolant System Pressure Leakage due to Cracks Found in Several Small Bore Reactor Vessel Head Penetrations (Section 4OA3.3)
50-269,270,287/00-09-01	URI	Failure to Properly Detect Intrusion Detection System Penetrations at the Perimeter Prior to Individuals Gaining Access to the Protected Area (Section 4OA5.1)
50-260,270,287/00-09-02	URI	Failure to Prevent a Simulated Adversary From Gaining Access to Vital Areas (Section 4OA5.2)

Previous Items Discussed

50-287/01-01-00	LER	Reactor Pressure Vessel Head Leakage due to Stress Corrosion Cracks Found in Nine Control Rod Drive Nozzle Penetrations (Section 4OA3.4)

LIST OF ACRONYMS USED

ALARA	-	As Low As Reasonably Achievable
AC	-	Alternating Current
ACB	-	Air Circuit Breaker
ASME	-	American Society of Mechanical Engineers
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
FIN	-	Finding
FSAR	-	Final Safety Analysis Report
GPM	-	Gallons Per Minute
HAWT	-	High Activity Waste Tank
HPI	-	High Pressure Injection
HPSW	-	High Pressure Service Water
IDS	-	Intrusion Detection System
INPO	-	Institute of Nuclear Power Operations
IP	-	Inspection Procedure
KHS	-	Keowee Hydro Station
LAWT	-	Low Activity Waste Tank
LCO	-	Limiting Condition 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
NDE	-	Non-Destructive Examination
NRC	-	Nuclear Regulatory Commission

NRR	-	Nuclear Reactor Regulation
NSD	-	Nuclear System Directive
OTSG	-	Once Through Steam Generator
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
RBNS	-	Reactor Building Normal Sump
RBS	-	Reactor Building Spray
RC	-	Reactor Coolant
RCP	-	Reactor Coolant Pump
RCS	-	Reactor Coolant System
RPS	-	Reactor Protection System
RPV	-	Reactor Pressure Vessel
RWP	-	Radiation Work Permit
Rx	-	Reactor
SBLOCA	-	Small Break Loss of Coolant Accident
SDP	-	Significance Determination Process
SER	-	Safety Evaluation Report
SLC	-	Selected Licensee Commitment
SRP	-	Standard Review Plan
SSC	-	Structure, System and Component
SSF	-	Standby Shutdown Facility
TEDE	-	Total Effective Dose Equivalent
T/C	-	Thermocouple
TS	-	Technical Specification
UFSAR	-	Updated Final Safety Analysis Report
URI	-	Unresolved Item
USQ	-	Unresolved Safety Question
UT	-	Ultra-Sonic

DOCUMENTS REVIEWED (for Section 1R08.2)

Root Cause Failure Analysis Report, Oconee Unit 1 Thermocouple Nozzle Leaks, dated 1/22/01

Root Cause Failure Analysis Report, Oconee Unit 1 CRDM Nozzle Leaks, dated 1/31/01

Duke Engineering and Services (DE&S) Metallurgy Final Lab Report for Unit 1 CRDM 21, dated 1/17/01

BMX Technologies Final Examination Report for Unit 1 Thermocouples, February 2001

DE&S Preliminary Report for Unit 3 CRDM Sample Cracks Morphology, dated 3/12/01

DE&S Preliminary Report for Unit 3 CRDM 56 Boat (Circumferential Crack), dated 4/4/01

DE&S Preliminary Report for Unit 1 Thermocouples, Unit 1 CRDM 21, Unit 3 CRDMs, dated 4/12/01

Procedure PT/1/A/0600/010, Reactor Coolant Leakage, Revision 052

Procedure OP/0/A/1102/023, Reactor Building Tour, Revision 6

SGMEP 105, OTSG Specific Assessments of Potential Degradation Mechanisms Applied to Oconee Unit 2 EOC 18, dated 4/24/01

SGMEP 105, OTSG Specific Assessments of Potential Degradation Mechanisms Applied to Oconee Unit 3 EOC 18, dated 2/29/00

PIP O-01-01681, concerning misidentification of ET data