UNITED STATES
NUCLEAR REGU.ATORY COMMISSION
REGION II
101 MARIETTA ST., N.W., SUITE 3100
ATLANTA, GEORGIA 30303

Report Nos. 50-269/80-28, 50-270/80-24, and 50-287/80-21

Licensee: Duke Power Co.
422 South Church Street
Charlotte, NC 28242

Facility Name: Oconee Nuclear Station

Docket Nos. 50-269, 50-270, and 50-287

License Nos. DPR-38, DPR-47, and DPR-55

Inspection at Oconee Nuclear Station near Seneca, South Carolina

Inspectors: RD Martin for — 9/4/80
W. Orders — Date Signed

RD Martin for — 9/4/80
D. Myers — Date Signed

Approved by: RD Martin — 9/4/80
R. D. Martin, Section Chief, RONS Branch — Date Signed

SUMMARY

Inspection on July 1-31, 1980

Areas Inspected

This routine inspection involved 221 resident inspector-hours on site in the areas of plant operation, surveillance testing, test and measuring equipment, maintenance program, physical security, plant modification test witnessing, radioactive waste transport, instructor licensing, LER followup, spent fuel shipment, and offsite contamination.

Results

Of the 11 areas inspected, no items of noncompliance or deviations were identified in 9 areas; 2 items of noncompliance were found in 2 areas (Infraction: failure to maintain traceability and accountability of QA material used on safety-related system; Paragraph 5; Infraction: failure to employ procedure when moving radioactive waste - Paragraph 6.)

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DETAILS

1. Persons Contacted

Licensee Employees

*J. E. Smith, Station Manager
*J. M. Davis, Superintendent of Maintenance
*J. N. Pope, Superintendent of Operations
*T. B. Owen, Superintendent of Technical Services
*R. T. Bond, Licensing and Projects Engineer
*J. Brackett, Senior QA Engineer

Other licensee employees contacted included 10 operations supervisors, 6 technicians, 12 operators, 4 mechanicS, 8 security force members, and 2 office personnel.

*Attended exit interview

2. Exit Interview

The inspection scope and findings were summarized on July 11, 18, 25 and August 1, 1980 with those persons indicated in Paragraph 1 above. This inspection findings were acknowledged without significant comment or rebuttal. Licensee management acknowledged the two items of noncompliance and indicated the events were under investigation.

3. Licensee Action on Previous Inspection Findings

Not inspected.

4. Unresolved Items

Unresolved items are matters about which more information is required to determine whether they are acceptable or may involve noncompliance or deviations. New unresolved items identified during this inspection are discussed in paragraph 21.

5. Maintenance Program

The inspector reviewed selected maintenance work requests and associated procedures to determine the adequacy of management control associated with corrective and preventive maintenance activities on safety related systems.

The inspection revealed that maintenance work requests were not being completed as required by station directive 3.3.5 nor was documentation of materials used in the work performed accomplished in accordance with said directive.

Details of the inspection are delineated below; Paragraph numbers refer to station directive 3.3.5:

3.3.5.3.2 Section II requires independent component verification prior to commencing work on the equipment. On work requests 91903, 48842, 48237 and 48231 the verifications were not documented.

3.3.5.3.2 Section IX states that the signature of the job supervisor indicates the work request has been reviewed and associated procedure complete. Work request 48842 was documented as complete but the associated procedure (MP/O/A/1800/16) was incomplete.

3.3.5.5 requires when a work request is voided, an originating group representative accept the disposition of the request by signing the "accepted by" blank on the form. On voided work request 48061, the required signature was not provided.

With regard to materials used on safety-related systems, Duke Power Quality Assurance Topical Report 1 section 17.2.8 requires materials, parts and components be assigned identifying designation (such as a serial number), in order to provide quality assurance traceability of each item. The program also requires QA designated, subdivided material be identified in accordance with the above requirements. Furthermore, issuance of nuclear safety-related materials, parts and components is required to be controlled and documented in such a manner that quality assurance traceability and inventory accountability is provided.

Contrary to the above, inspection revealed the following:

- Material was returned to stock without being identified as QA material
- Material was issued without completing the required traceability and inventory control documents.
- Material was subdivided but was not identified as QA material as required. Review of station directives and related maintenance procedures indicates the requirement to identify such subdivided material is inadequately addressed.

These items collectively constitute a noncompliance for inadequate program and procedual controls in the quality assurance implementation.

This item was identified as an infraction and applies to Unit 1 (269/80-28-01).

6. Radioactive Waste Transport

On 21 July 1980 a letdown filter transfer cask was being transported to the Oconee Unit 1 auxiliary building loading door on an electric cart. At approximately 1420 on that day the electric cart was allowed to roll through the open auxiliary building door onto the asphalt loading area outside the auxiliary building. The cart and cask overturned, resulting in a spill of radioactive waste water. No offsite release occurred nor were any personnel contaminated.

The spill was estimated to be approximately a quart in volume, read in excess of 400,000 dpm on initial swipes and 6 mr on contact. The spill was immediately contained, isolated and the area decontaminated.

The resident inspector witnessed the events immediately following the spill, and questioned the maintenance personnel responsible for the activity. It was determined at that time and subsequently confirmed that the procedure associated with transport of the letdown filter, MP/O/B/1600/12, was not at the job site, had not been reviewed by the personnel prior to job performance and was not referenced on the job Work Order.

Procedure MP/O/B/1600/12, step 11.1.7 states to "...not allow electric cart to travel on asphalt pavemen due to chance of tipping cart".

Technical Specification 6.4.1 states that the station shall be operated and maintained in accordance with current written approved procedures with appropriate check-off list and instructions when performing:

- Preventive or corrective maintenance which could affect radiation exposure to personnel
- Radiation control procedures
- Operation of radioactive waste management systems

Licensee failure to employ procedure in performance of task discussed violates Technical Specification 6.4.1.

This item was identified as an infraction and applies to Unit 1 (269/80-28-02)

7. Technical Specification Amendment

The Oconee Unit 2, B HPI pump motor failed at approximately 10:20 p.m. on Sunday evening July 13, 1980. Inspection of the motor following partial disassembly, revealed the most expeditious repair entailed removal of the motor and replacemen with a spare.

During the initial run of the uncoupled spare motor on Wednesday morning a rotor bar connection failed. This required the replacement of the spare motor and use of the original motor following bearing repairs. Test runs, coupling to the pump and performance verification of the original motor were scheduled to be complete by 7 a.m. Thursday, July 17, 1980.

Oconee Technical Specification 3.3.1(c) states when reactor power is greater than 60%, an inoperable HPI pump must be restored to operable status within 72 hours or the reactor power must be reduced below 60% within 12 additional hours.

The licensee, in analyzing grid reserves, determined that Oconee Unit 2 operating at full power was essential for Duke to meet expected peak loads.

A temporary revision to Technical Specification 3.3.1(c) to allow continued operation of Oconee Unit 2 at full rated power for an additional 48 hours while maintenance efforts continued on the "B" High Pressure Injection (HPI) pump was requested of the Office of Nuclear Reactor Regulation.

Compensatory actions to be taken by the licensee during the extension consisted of:

- An operator assigned to assure that valve HP-26, the A HPI injection valve, would open in the event of an engineered safeguard signal
- No maintenance would be initiated on any system that may degrade the reliability of the HPI system
- Reactor Coolant System leakage calculation would be performed every four hours.

The 48-hour extension was granted and Technical Specification amended such that for Oconee Unit No. 2 HPI Pump B, the power reduction requirements of Specification 3.3.1.(c) did not apply until 2220 hours of July 18, 1980.

The resident inspector verified licensee compliance with compensatory measures stated previously and witnessed satisfactory functional test and return to service of B HPI pump.

The inspector has no further questions on this matter.

8. Apparent Reactivity Anomoly

On the evening of 7/9/80, Oconee 1 was initiating startup following an outage for repair of the 1B1 reactor coolant pump and implementation of control system modifications per NRC Confirmatory Order relating to the 2/26/80 event at Crystal River. At 2235, as control rods were being withdrawn to establish criticality, criticality occurred at 19% withdrawn on control rod group 5--about 1.3% $\Delta K/K$ different from the predicted critical rod position of 39% withdrawn on control rod group 6. The reactor was taken subcritical as required by plant procedures.

After an evaluation of actual core conditions and a reactivity balance procedure revision to reflect actual core reactivity, criticality was re-established at 0656 on 7/10/80. Review of core power distribution data at low power levels confirmed that the apparent reactivity anomaly was due not to unusual core conditions but primarily to inaccuracy in the core physics data. Following a reactor trip from 15% power on the morning of 7/10/80 another criticality was attained at 1800 on 7/10/80 which indicated a slightly lower apparent reactivity anomaly of about 1.2% $\Delta K/K$.

The three startups on 7/9 and 7/10/80 indicated the following reactivity differences:

DATE	TIME	CORE BURNUP	CRITICAL PREDICTED	ROD POSITION ACTUAL	REACTIVITY ANOMALY
7/9/80	2235	97.4 EEPD	Gp 6 @ 39% wd	Gp 5 @ 19% wd	1.28% ΔK/K
7/10/80	0656	97.4 EEPD	Gp 6 @ 64% wd	Gp 5 @ 28% wd	1.30% ΔK/K*
7/10/80	1757	97.4 EEPD	Gp 6 @ 76% wd	Gp 5 @ 52% wd	1.17% ΔK/K*

*The Reactivity Balance Procedure had been revised before these criticalities occurred, such that the actual reactivity differences between actual and predicted were small (~0.1% ΔK/K).

Of the three observed anomalies, the 1.17% ΔK/K anomaly is assigned the highest confidence level due to the strict boron control and sampling measures taken. Therefore, a value of 1.20% ΔK/K is considered to be the best estimate of the reactivity difference.

The sources of the reactivity anomaly have been identified by the licensee as follows:

a.	Inaccuracy in B&W-supplied core excess reactivity data	0.32% ΔK/K
b.	Inaccuracy in core excess reactivity data in the Reactivity Balance Procedure	0.13% ΔK/K
c.	Difference between predicted and measured control rod worths at the critical position	0.36% ΔK/K
d.	Slight burnup dependant differences in boron and samarium worth data in the Reactivity Balance Procedure. (The negative sign means that these differences decreased the apparent anomaly)	-0.10% ΔK/K
e.	Unidentified (but believed to primarily be the burnup dependant change in control rod worth from beginning of cycle)	0.49% ΔK/K
	Total Anomaly	1.20% ΔK/K

Immediate corrective actions were to:

- shut down the reactor
- verify chemistry sample results
- verify that there were no indications of ejected or misaligned control rods or loose parts.

Reactivity balance data was then reviewed for errors. None being found, the Reactivity Balance Procedure was revised based upon the observed critical rod position.

Core power distribution measurements were made at low power levels to confirm no unusual core conditions existed. Core power distribution data at higher power levels were also obtained and reviewed, and reactivity measurements made to verify no core reactivity change had occurred during the outage preceding this incident.

A concerted effort by the Performance Section, General Office Nuclear Fuel Services, and B&W has and is being made to resolve the reactivity anomaly.

This item will be examined during a subsequent inspection.

9. Liquid Radioactive Waste Spill

At about 6:25 p.m. on July 16, several gallons of evaporator concentrates were spilled onto an outside asphalt surface as the material was being transferred from the radwaste facility to a shipping cask. The release took place when a weld on the hose coupling attached to the cask liner split due to improper seating of the liner in the concrete shield. The concentration of the spilled material was 0.36 microcuries/ml with the following radionuclides predominating: Co-58 (57% of total activity); Cs-137 (14%); and I-131 (10%). The spilled material was contained within an area of approximately 15 square feet. The maximum exposure rate due to the contaminated asphalt was 0.3 mR/hr. There was no offsite release and no personnel were contaminated.

This area will be re-examined during a subsequent inspection.

10. Plant Operations

The inspector reviewed plant operations throughout the report period, to verify conformance with regulatory requirements, technical specifications and administrative directives. The control room logs, shift supervisor's logs, shift turnover records, and the removal and restoration record books for the three units were reviewed. Interviews with plant operations, maintenance, chemistry, health physics and performance personnel were held on the day and night shifts.

Activities within the control rooms were monitored during day and night shifts and at shift changes. The actions and activities were conducted as prescribed in Section 3.08 of the Station Directives. The number of licensed personnel on each shift met or exceeded the minimum required by IEB 79-05C. Operators were responsive to plant annunciator alarms and appeared to be cognizant of plant conditions.

Plant tours were taken during the inspection period as follows:

Turbine Building
Auxiliary Building
Unit 1, 2 and 3 Electrical Equipment Rooms
Unit 1, 2 and 3 Cable Spreading Rooms
Station yard areas within the protected area

During the plant tours, observations were made of ongoing activities, housekeeping, security, equipment status and radiation control practices.

Two items of noncompliance were identified in the area of maintenance activities and are discussed in paragraphs 5 and 6 of this report.

11. Surveillance Testing

The surveillance tests identified below were analyzed and/or witnessed by the inspector to ascertain procedural and performance adequacy.

The completed test procedures examined were analyzed for embodiment of the necessary test prerequisites, preparations, instructions, acceptance criteria, sufficiency of technical content and test results.

The selected tests witnessed were examined to ascertain that current written approved procedures were available and in use, that test equipment in use was calibrated, that test prerequisites were met, system restoration was completed, and that test results were adequate.

Review of the selected completed procedures showed conformance with applicable Technical Specifications and procedural requirements. They appeared to have received the required administrative review and they apparently were performed within the surveillance frequency specified.

Procedure	Title
PT-0-A-0150-15A	RB Isolation Valve Exercise Functional test (OPS)
PT-0-A-0150-15B	RB Isolation Valve Exercise Functional Test (Shutdown)
IP-0-A-0330-03-A	CRD Trip Test
IP-0-A-0330-02-D	CRD Patching and Functional Test
IP-0-A-0305-05-A	RPS Channel-A RB High Pressure Trip
IP-0-A-0305-05-B	RPS Channel-B
IP-0-A-0305-05-C	RPS Channel-C
IP-0-A-0305-05-D	RPS Channel-D

PT-0-A-0202-12	HP Injection System ES Test
PT-0-A-0202-11	HP Inj System Performance Test
PT-0-A-0203-06	LP Inj. System Perf. Test
PT-0-A-0203-08	LP Inj. System ES Test
PT-0-A-0160-03	RB Coolers ES and Performance Test
PT-0-A-0204-09	RB Spray ES Test
PT-0-A-0250-10B	Fire Protection System Test
PT-0-A-0250-15	Annual Fire Protection Equipment Test
PT-0-A-0620-09	Klowce Hydro Start Test
PT-0-A-0204-17	Operability Test of 4160 Breakers
PT-0-A-0204-07	RB Spray System Performance Test

The inspector employed one or more of the following acceptance criteria for evaluating the above items:

10 CFR
ANSI N18.7
Oconee Technical Specifications
Oconee Station Directives

No items of noncompliance or deviations were identified in this area.

12. Test and Measurement Equipment

The inspector examined the licensee's test and measurement equipment control program to verify conformance with regulatory requirements, licensee commitments, industry guides and standards.

The program as detailed in Station Directive 2.3.1 and Duke Administrative Policy Manual chapter 2.3, applies to test and measuring equipment affecting proper functioning of station safety-related and control designated structures, systems and components.

The program provides that the test equipment be assigned permanent identifying designations which are etched into, or attached to, the devices or the case containing a device such that the Test and Measuring equipment is conspicuously identified.

The program provides that each piece of test equipment be calibrated at periodic intervals, and/or prior to use with the intent that the desired accuracy and quality level be maintained. Manufacturer's intervals, intervals determined from equipment history, or intervals from other recognized sources are to be utilized. Calibration is accomplished using certified equipment having known valid relationships to nationally recognized standards, or recognized natural physical constants, or through accepted ratio techniques.

The program requires that the test equipment displays a calibration sticker showing the date of calibration, the date the next calibration is due and the initials of the person performing the calibration.

Test equipment which fails to meet the applicable calibration specifications or schedule is to have a HOLD tag attached showing the date of rejection, the reason for rejection and the initials of the person rejecting the equipment.

Items and processes determined to be acceptable based on measurements made with test equipment that is subsequently found to be out of calibration limits are to be reevaluated, within 7 days.

Each piece of test equipment is detailed in an inventory history system delineating the following details on each piece of equipment:

Type of test equipment
Manufacturer
Manfacturer"s serial number
Model number(s)
Calibration frequency and specifications
History of calibrations, repairs, restrictions on use

Examination of licensee's program and witnessing of selected program processes, records, and indices indicates the system to be in conformance with regulatory requirements, licensee commitments and industry guides and standards.

There were no items of noncompliance or deviations identified in this area.

13. Physical Security

During the report period, implementation of the physical security program was observed and selected records were reviewed. Interviews were held with several members of the physical security organization. The areas inspected include, physical security organization, physical barriers, access and badge controls, pat down searches, and communication checks. The guidance and acceptance criteria used for this inspection is provided in 10 CFR 73.55(b), (c), (d), (f), and (g), Regulatory Guide 5.20 and NUREG-0219.

The review of the physical security organization revealed the following:

a. A member of the physical security organization who has the authority to direct the shift activities is required to be present at all times. This item was verified on July 8, shift 1 and 2, July 9, shifts 1 and 2 and July 10, shifts 1, 2, and 3.

b. Shift complement requirements were checked on July 8 to verify the shift was prope·ly manned.

Inspection of physical barriers consisted of walking the protected area fence on July 9 and observing that all gates were locked closed, and the isolation zone was free of objects. Within the buildings, vital area doors were noted to be closed and locked. Throughout the month on all shifts, it was noted that guards were posted at control room doors when the doors were malfunctioning.

Access control was checked each time the inspector entered the protected area. Persons and packages were observed being searched on July 8-12. This review also included observation that all persons within the protected area properly displayed their identification badge.

To determine compliance with escorting procedures, the inspector spot checked visitors and their escorts during the report period. The escort is required to remain with the visitor at all times and keep control of the visitor. Appropriate procedures and requirements were followed during the period of observation.

Records were selectively examined during the month to determine if required communications checks were performed. Checks are required to be done at the beginning of each shift for onsite communications and once per day for offsite equipment. For the period examined all checks were completed as required.

No items of noncompliance or deviations were identified in this area of inspection.

14. Integrated Control System Loss of Power Test

As a result of experiences gained from the Crystal River-3 incident of February 26, 1980, a modification to the power supply for the non-nuclear instrumentation (NNI) has been installed on Oconee unit 1. The modification is described in Nuclear Station Modification 1531, NSM 1531). This NSM provides a redundant source of power to all indications and control loops in the Integrated Control System (ICS), necessary to reach and maintain the reactor at hot shutdown upon loss of the normal power supply to the NNI. The indications and controls needed to maintain the plant at hot shutdown are stated in an April 1, 1980 letter from DPC to NRC. NSM 1531 incorporated these indications and controls.

Following the installation of NSM 1531, a test was performed to verify integrated system performance. This test is described in TI/.'E/320/05, "Integrated Control System Loss Power Test". An NRC order confirming DPC's commitment to perform such a test was issued April 17, 1980. The test was conducted on July 5, 1980.

Prior to running the test, comments on the test procedure were discussed with licensee representatives and resolved. Selected portions of the test were witnessed by the inspector. Several discrepancies identified during testing have been reviewed by the licensee and either resolved or corrected and retested. The inspector concurred with the resolution in each case.

Results of the test have been incorporated into Emergency Procedure EP/1/A/1800/31, "Loss of 1KI Bus (And Control Room Indications Powered From 1KI)".

The inspector discussed the NSM and associated testing with several operators and supervisors to verify their acquaintance with the modifications. The personnel were found to be familiar with the modifications and had received associated training through requalification training.

The inspector had no further questions on this matter.

15. Instructor Licensing

On March 28, 1980, a letter was sent to all licensees concerning Senior Reactor Operator (SRO) license examinations for utility training staff instructors involved in training on systems, integrated response, transients and simulator courses. The letter requested that license applications for those instructors not holding an SRO license be submitted on or before August 1, 1980.

The inspector verified all Oconee instructors except one hold SRO licenses. The unlicensed instructor shall have a license application submitted on or before August 1, 1980 pursuant to the request.

The inspector has no further questions on this matter.

16. Licensee Event Followup

At approximately 1400 on April 16, 1980 the Oconee 1 Reactor Building personnel hatch inner door gasket was discovered leaking. The personnel hatch was declared inoperable at approximately 1430. The inner door gasket hatch was repaired, and the personnel hatch was declared operable at 1137 on April 17 after sucecessful completion of a leak rate test.

The licensee reported the occurrence via LER 80-10 pursuant to Technical Specification 6.6.2.1.b(2).

As corrective actions, the licensee installed shim gaskets and adjusted the latching brackets to provide an adquate seal. The licensee also obligated to leak test the personnel hatch following each cold shutdown if feasible.

The inspector witnessed PT/0/A/150/08A, Reactor Building Personnel Hatch Leak Test performed on Oconee Unit 2 on July 8, 1980 pursuant to LER 80-10 obligation.

The inspector verified that the test equipment was calibrated , the test procedure employed was current, the test prerequisites were satisfied and radiological work practices were followed. Independent verification of test data indicated that the test was satisfactory.

The inspector had no further questions regarding this matter.

17. Broken Holddown Springs

After being notified on May 16, 1980 that Toledo Edison had discovered several broken holddown springs at Davis-Besse I, (a Babcock and Wilcox 177FA plant), Duke Power began inspecting all spent fuel assemblies and core verification films to identify any similar problems at Oconee. Of 1217 assemblies inspected only four were identified as having broken springs. The four affected assemblies are 1D47, 1D17, 1C43, and 3C33.

The cause of the broken springs has not been conclusively defined; however, from preliminary analysis the licensee believes that the four Oconee springs failed due to high cycle fatigue. The licensee and B&W mutually concluded that the spring failures pose three potential concerns: (1) loss of hold-down force; (2) loose parts and (3) interference with normal CRA movement.

At B&W's recommendation, Duke instituted the following precautionary actions on May 23, 1980:

a. Increased frequency of control rod movement test.

b. Verified that adequate monitoring of loose parts monitoring system was being performed.

c. Verified that normal chemical sampling would identify increases in silver, indium or cadmium (or their daughters) in the RCS thereby indicating substantial control rod degradation.

During the report period, B&W has completed a safety evaluation of core operation with broken holddown springs. On the basis of the safety evaluation and information gained from the examination of fuel assemblies and holddown springs, B&W concluded that control rod excerising frequency as currently required by Technical Specifications is an acceptable measure for the demonstration of control rod operability.

The licensee concurred with B&W's evaluation and has reduced the frequency of control rod movement tests to comply with Technical Specification surveillance requirements; however, loose parts monitoring and chemical sampling as recommended by B&W will continue.

The licensee and B&W are continuing to evaluate the problem.

18. Spent Fuel Shipment

Irradiated fuel assembly 1D54 which was sent to the Babcock and Wilcox, Lynchburg facility for post irradiation analysis departed Lynchburg at 0300 on July 8, 1980 to be returned to storage at Oconee at 1705 on the same day.

Radiation survey of the cask and transport vehicle verified compliance with 10 CFR 20.205 requirements. Transport and escort vehicles were inspected to verify presence and operability of communication equipment required by 10 CFR 73.37.

The inspector verified that shipment escorts had received required training in accordance with Appendix D of 10 CFR 73.

No items of noncompliance were identified in this area.

19. Low Level Contamination of Waste Oil Vendor Truck and Holding Tank

On June 20, 1980 two of three 55 gallon barrels containing approximately 20 gallons each of contaminated waste oil were inadvertently mixed with approximately 2000 gallons of uncontaminated waste oil. The mixture was shipped offsite using the truck of a waste oil vendor from Greer, SC. The oil was then transferred to the vendor storage tank. The vendor subsequently placed another 2000 gallons of waste oil, from another customer, into the contaminated truck.

The licensee contacted the oil vendor who returned the 4000 gallons of oil to Oconee for onsite storage. The licensee purchased and removed the storage tank from its Greer location and decontaminated the vendor's truck. The activity in the tank at the waste yard was: MN-54: 5.4 E-8 microcuries per cubic centimeter CO-60: 2.6 E-7 microcuries per cubic centimeter, and CS-134: 5.6 E-6 microcuries per cubic centimeter

Prior to the time that Duke made a final decision to remove the vendor's storage tank to the Oconee Station, an independent sample of the diesel oil used to flush the tank was collected by Resident Inspectors. The Region II laboratory results were lower, but in general agreement with, the licensee's analysis of this fluid:

	Cs-137 (μCi/cc)	Cs-134 (μCi/cc)
NRC Results:	1.4E-6	6.0E-7
DPC Results:	2.2E-6	1.0E-6

An indipendent sample of residue in the vendor's storage tank was also analysed by Region II, with the following results:

Cs-137: 4.2E-6 μCi/cc

Cs-134: 2.0E-6 μCi/cc

Independent samples of soil from the vendor's driveway and around the storage tank inlet were taken by the Resident Inspectors. The driveway soil revealed no detectable radioactivity. Low levels of contamination were detected around the tank inlet due to spillage while transferring the waste oil (Cs-137: 1.4E-6 μCi/g; Cs-134: 0.5E-6 μCi/g). These results are in general agreement with similar soil samples taken and analysed by the South Carolina Bureau of Radiological Health. These levels, which did not represent a radiological health hazard, were further diluted by mixing the soil when the storage tank was excavated.

After Duke's decontamination of the vendor's tank truck, NRC inspectors took nine (9) swipes from both inside and outside the tank, the drain valve, and various locations on the truck. These swipes were analyzed in the Region II laboratory. All results indicated no detectable activity, confirming that the decontamination of the tank truck was satisfactory.

The inspector had no further questions on this matter.

20. Unit Three Inverter Trips

On May 7, 1980, Oconee Unit 3 was operating at 100% full power when at 0916 static inverter 3D1B, which supplies power from 125 VDC instrumentation and control panel board 3D1B to AC vital instrumentation power panel board 3KV1B tripped. Power was restored to panel board 3KVIB by manually bypassing static inverter 3DIB and supplying the board with regulated AC.

When inverter 3DIB tripped all loads on 3KVIB AC Vital Instrumentation Power Panel were deenergized; these loads included the following:

a. Reactor Protection System (RPS) channel "B"
b. Engineered Safeguards (ES) Channel "B"
c. Reactor Coolant Pump "B" Power Monitor
d. Control Rod Drive Primary Trip Breaker Assembly Unit 11.

Since RPS Channel C had previously been bypassed for testing, the licensee initiated a plant shutdown pursuant to Technical Specification 3.5.1 which requires that a minimum of three of the four RPS channels be available.

At 0930 RPS Channel B as well as the other denerized loads were reenergized, reset and the unit was returned to 100% power by 0950.

On May 8, 1980 at 1204 static inverter 3DIB was returned to service. At 1354 on May 9, 1980, the inverter tripped. Once again a power reduction was iniated but as on May 7, was terminated after the inverter was bypassed and RPS Channel "B" was reset.

On May 16, 1980 after troubleshooting and repair had been performed, a an attempt was made to place inverter 3DIB in service. The inverter immediately tripped, was again manually bypassed but was successfully placed in service later in the same day after the D.C. input power fuse was replaced.

In analyzing the problems encountered with inverter 3DIB it appears the incident on May 7, 1980 was a result of a blown D.C. input power fuse. I&E Technicians replaced the fuse, checked the logic voltage, and inspected the inverter wiring for signs of a problem. The logic voltage checked good, there were no signs of wiring problems. Therefore, the inverter was placed in service. Two days later the same D.C. input power fuse blew. An I&E Assistant Engineer requested that the logic boards in 3DIB inverter be inspected for bad components. One bad transistor was found on one logic board. The licensee feels that this transistor was probably breaking down when the May 7, 1980 incident occurred; when the inverter cooled down, it evidently regained its' characteristics and was able to operate for about two days before it broke down completely.

The transistor was replaced and a dummy load consisting of resistive and inductive components was placed on 3DIB inverter on 5-12-80. The inverter was cycled between low load and full load for four days with no apparent problems. On May 16th, I&E Technicians and an operator placed the inverter back in service.

Both the May 7 and May 9, 1980 events resulted in a unit shutdown being initiated pursuant to Technical Specification 3.5.1 when power to RPS Channel B was lost, since RPS Channel C had already been bypassed for testing. Although the inverter failure caused a loss of the loads from panel board 3KVIB, the remaining three AC vital instrumentation power panelboards were in service, and required instrumentation was available.

This incident, was transmitted as Reportable Occurrence RO-287/80-8 by Duke letter of June 6, 1980.

21. Emergency Power Switching Logic

At 1114 hours on July 10, 1980, the Unit 3 reactor tripped. The resulting auxiliary power transfer caused Emergency Power Switching Logic (EPSL) relay $27NYA_2$ to energize.

At approximately 2040 hours, breaker 3DIA-13 tripped. This breaker supplies control power for the "A" phase of the Normal Voltage Circuit of the EPSL. The breaker tripped due to a fault condition caused by burned out relay $27NYA_2$. The licensee determined the coil clearing contact of $27NYA_2$ did not open causing the relay to burn.

Relay $27NYX_B$, directly above $27NYA_2$ was declared inoperable since it's condition was questionable. This disabled EPSL transfer to start-up and/or stand-by bus.

The licensee decided to repair the EPSL while the unit came up in power.

The circuitry was repaired, tested, and returned to service on July 11, 1980 at 1545 hours.

Investigation reveals that the EPSL is the only safety-related circuitry which ensures that a reliable source of power is available to the 4160 main feeder buses under accident conditions. A cursory review of the accidents analyzed in the Oconee Safety Analysis Report indicates that EPSL is relied upon in the analyzed transients therein.

The inspector discussed the matter with the licensee and will review the concern in greater detail during the next report period.

This matter of a more detailed review of the EPSL is identified as an Unresolved Item (287/80-21-01) pending completion of that review by the inspector.