UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30323

Report Nos.: 50-369/84-25 and 50-370/84-22

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

Docket Nos.: 50-369 and 50-370 License Nos.: NPF-9 and NPF-17

Facility Name: McGuire Nuclear Station Units 1 and 2

Inspection Conducted: August 20 - September 20, 1984

Inspectors: A. J. Ignatonis for W. Orders 10/29/84 Date Signed

A. J. Ignatonis for R. Pierson 10/29/84 Date Signed

Approved by: H. C. Dance 10/29/84 Date Signed
H. Dance, Section Chief
Division of Reactor Projects

SUMMARY

Scope: This routine, unannounced inspection involved 301 (resident) inspector-hours on site in the areas of operations, safety verification, surveillance testing and maintenance activities.

Results: One violation was identified - failure to report a condition which could have prevented the fulfillment of the safety function of structure/system needed to control release of radioactive material.

REPORT DETAILS

1. Licensee Employees Contacted

   G. Vaughn, Manager Nuclear Plants
   M. McIntosh, Station Manager
   G. Cage, Superintendent of Operations
   E. Estep. Project Engineer
   G. Gilbert, Operations Engineer
   *D. Mendezoff, Licensing Engineer
   T. L. McConnell, superintendent of Technical Services
   *D. Rains, Superintendent of Maintenance
   B. Travis, Operations Engineer

   *Attended exit interview

2. Exit Interview

   The inspection scope and findings were summarized on September 28, 1984, with those persons indicated in paragraph 1 above. The areas of concern and proposed enforcement were discussed with the licensee, who expressed cognizance and concern over the issues. The violation for failure to report is described in paragraph 8.

3. Licensee Action on Previous Enforcement Matters

   This subject was not addressed in the inspection.

4. Unresolved Items

   Unresolved items were not identified during this inspection.

5. Plant Operations (71707)

   The inspector reviewed plant operations throughout the report period, August 20 - September 20, 1984, to verify conformance with regulatory requirements, technical specifications and administrative controls. Control room logs shift supervisors' logs, shift turnover records and equipment removal and restoration records were routinely perused. Interviews were conducted with plant operations, maintenance, chemistry, health physics, and performance personnel on day and night shifts.

   Activities within the control rooms were monitored during all shifts and at shift changes. Actions and/or activities observed were conducted as prescribed in Section 3.1 of the Station Directives The complement of licensed personnel on each shift met or exceeded the minimum required by technical specifications. Operators were responsive to plant annunciator alarms and appeared to be cognizant of plant conditions.

Plant tours were taken throughout the reporting period on a systematic basis. The areas toured included but were not limited to the following: Turbine Building, Auxiliary Building, Units 1 and 2, Electrical Equipment and Cable Spreading Rooms, Station Yard Zone within the protected area and Unit 2 Reactor Building. During the plant tours, ongoing activities, housekeeping, security, equipment status and radiation control practices were observed.

McGuire Unit 1 began the reporting period in Mode 1 operating at 100% reactor power. The unit was maintained at 100% power until 9:49 p.m. on Tuesday, August 21, 1984, when the reactor tripped from 100% power following a loss of offsite power. Loss of offsite 230 kV power occurred, following an inadvertent opening of offsite dispatcher controlled computer operated breakers, resulting in a loss of load to the main generator. Based on first-out annunciator alarms, the reactor tripped on high flux rate. This signal was evaluated as being the result of the loss of power induced voltage transient and was therefore not indicative of an actual high flux rate. Steam generator "C" low-low level was also alarmed at virtually the same time as the high flux rate reactor trip signal. The diesel generators started and loaded in sequence as designed. Prior to and during the time of the Unit 1 reactor trip, Unit 2 was in Mode 3 and receiving steam to its main feedwater pump turbine from Unit 1. Thus when Unit 1 reactor tripped, the operators manually isolated the Unit 1 main steam isolation valves (MSIVs) to prevent overcooling which resulted in water hammer occurring in the secondary system of Unit 1. An unusual event was declared at 10:00 p.m. At 10:27 p.m. after MSIV isolation on Unit 1 and a subsequent delay of providing steam to Unit 2 main feed pump turbines coupled with the opening of the relief valve 2SV-9 below its setpoint, Unit 2 received a reactor trip signal from low-low level in the steam generator. Offsite power was restored at 10:40 p.m., but the unusual event was not secured until after primary and secondary system stabilization of Unit 1 and re-opening of MSIVs. This occurred at 2:45 a.m. on August 22, 1984. Water hammer damage was assessed and following necessary repairs the unit was restarted on Thursday, August 23, 1984.

The unit entered Mode 2 at 8:50 a.m., the reactor reached criticality at 9:04 a.m. and the unit entered Mode 1 at 9:30 a.m. on the morning of August 23. Power was subsequently increased to 100% and was maintained at this level.

On Wednesday, September 5, 1984 at 12:47 p.m. with Unit 1 operating at 100% power, a resin fill diaphragm type isolation valve of the chemical and volume control system (NV) mixed bed demineralizer failed due to a diaphragm rupture when the cation bed demineralizer was valved in. As a result, approximately 750 gallons of reactor coolant was released into the compartment of the mixed bed demineralizer located in the auxiliary building. Some of this water subsequently contaminated other areas of the auxiliary building, but local contamination levels did not exceed 19,000 - 20,000 dpm with general area radiation levels remaining less than 3 mr/hour. The leakage was isolated from the control room and the valve was subsequently repaired. The auxiliary building was subsequently decontaminated and there was no off-site release.

On Friday, September 7, 1987, at 11:30 p.m. with Unit 1 operating at 100% power a body-to-bonnet leak occurred on 1NV337, the inlet valve to the NV mixed bed demineralizer. The cation demineralizer was again in the process of being valved in. The leak was quickly isolated by bypassing the NV demineralizers from the control room. It was later determined that 1NV359, the inlet valve to the cation demineralizer had a stem which had come loose from its disc and was no longer giving a proper indication of valve position. This subsequently caused back-pressure resulting in the above described incidents. The cation demineralizer will remain out of service until parts are obtained to repair 1NV359. Unit 1 was subsequently maintained in Mode 1 at or about 100% reactor power throughout the duration of the reporting period.

McGuire Unit 2 began the reporting period in a shutdown condition which had existed since July 27, 1987. On Monday, August 20, 1984, at 7:15 a.m. while preparing to restart the Unit 1, a leak developed inside containment while personnel were performing the upper head injection fill and vent procedure required for unit restart. The reactor coolant system (RCS) pressure was at 1800 psi and at a temperature of 520°F. The source of the leak was determined to be a blown-out sight glass in the upper head injection vent line, which could not be immediately isolated. An unusual event was declared at 7:35 a.m. Station personnel succeeded in partially isolating the leak which subsequently reduced the leakage from approximately 50 gpm to 20 gpm. The containment was isolated and the RCS pressure and temperature were steadily decreased. At 11:00 a.m. the leak was isolated and the unusual event was terminated at 11:08 a.m., RCS pressure was at 370 psi and at a temperature of 325°F. There was no off-site release.

Following damage assessment and completion of necessary repairs the unit was again prepared for reactor startup. On the evening of August 21, 1984, when the unit was in Mode 3 with steam being supplied to the main feed pumps from Unit 1, a loss of 230 kV offsite power caused Unit 1 to trip which subsequently caused Unit 2 to trip as described in the Unit 1 plant operating summary. At the time of the trip only the shutdown banks were pulled. All systems responded normally and Unit 2 was again prepared for unit startup.

On August 22, at 3:50 a.m., the shutdown banks were again pulled. The unit entered Mode 2 at 9:00 p.m. and reached criticality at 9:14 p.m. on August 22, 1984. The unit entered Mode 1 at 4:25 a.m. and was placed on the grid at 4:26 a.m. on August 23, 1984.

Unit power was subsequently increased and was maintained at or about 100% reactor power. On August 26, 1984, at 11:00 p.m. a leaking check valve on the "D" steam generator auxiliary feed line caused an over-pressurization of the pump suction piping, causing the suction relief valve to the turbine driven feed pump to lift. The resultant overpressurization caused inoperability of suction line instrumentation which necessitated extensive instrumentation replacement as well as repair of the check valve CA-37. The suction lines to the motor driven auxiliary feed pumps were determined to be unaffected.

Power was maintained at or about 100% reactor power through August 31, 1984, when a reactor trip occurred at 9:14 a.m. The trip resulted when Power Range Channel 43 was out of service for surveillance testing and a technician inadvertently pulled cables from Power Range Channel 42 instead of Power Range Channel 43 resulting in a 2/4 trip logic. All systems responded normally.

The unit was again restarted with reactor start-up commencing at 11:45 a.m. on September 1, 1984. Criticality was achieved at 12:05 p.m. and the unit entered Mode 1 at 2:20 p.m. and was paralleled to the grid at 2:51 p.m. Power was escalated and the unit was maintained at or about 100% reactor power until September 15, 1984, when power was reduced to 10^{-8} amps to check and add oil to the main coolant pumps. Maintenance personnel determined that main coolant pump oil levels were in fact satisfactory and that the problem was related to the instrumentation. Following repairs, power was increased and the unit entered Mode 1 at 3:20 p.m. on September 16, 1984. The unit was placed on the grid at 3:30 p.m. and power was increased to 100% reactor power and maintained at or about 100% power for the duration of the reporting period.

6. Surveillance Testing (61726)

The surveillance tests categorized 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, and sufficiency of technical content.

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 completed and test results were adequate.

The selected procedures perused attested 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
OP/1/A/6200/09	Accumulator Operation
OP/1/A/6200/10	Upper Head Injection, Enclosure 4-2
PT/0/A/4600/02B	Incore and Nuclear Instrumentation Systems Recalibration: Leaving Target ΔI Band

PROCEDURE (Continued)	TITLE
TP/2/A/2650/06	Unit Loss of Electrical Load
TP/1/A/2650/06	Unit Loss of Electrical Load
TP/1/A/2650/12	Loss of Offsite Power Test
TP/2/A/2650/12	Loss of Offsite Power Test
OP/0/A/6350/01A	125 VDC/120 VAC Instrument and Control Power, Encl. 4.6 Steps 2.1.1 to 2.1.2 placing EVCS in service for EVCA battery charger
OP/2/A/6350/05	AC Electrical Operations Other Than Normal Lineup; Enc. 4.3 Shifting Power Supplies on 4.16 kV 2ETB for 2ATO to SATB
OP/0/A/6350/08	Operation of Station Breakers; Enc. 4.2, 4.16 kV Essential Switchgear Breakers (2ETB Normal Incoming) (2ETB Standby from SATB)

7. Maintenance Observation (62703)

The maintenance activities categorized below were analyzed and/or witnessed by the resident inspection staff to ascertain procedural and performance adequacy.

The completed procedures examined were analyzed for embodiment of the necessary prerequisites, preparation, instruction, acceptance criteria and sufficiency of technical detail.

The selected activities witnessed were examined to ascertain that where applicable, current written approved procedures were available and in use, that prerequisites were met, equipment restoration completed and maintenance results were adequate.

The selected work requests/maintenance packages perused attested conformance with applicable Technical Specifications and procedural requirements and appeared to have received the required administrative review.

WORK REQUEST	EQUIPMENT
120104	Replace T/D CA Pump Suction Pressure Switches for 2RN 69A, 2CA 116B, 2CA86A and 2RN 162B
120105	Replace T/D CA Pump Suction instruments which were over-ranged following check valve failures
120106	Replace T/D CA Pump suction instruments for 2CA-161C and 2CA-162C
120240	Repair of 2CA22
010146	Recalibration of CS and RHR
010147 010148 010149	Air Handling Units 1A, 1B, 2A and 2B pressure instrumentation

8. Reporting Inadequacy

As reported previously in inspection report number 50-369/84-21, on June 27, 1984, the McGuire Unit 1 containment spray vent valve 1NS-68 was found open during a quarterly valve stroke timing test of the containment spray (NS) system. The root cause of this event appeared to be a disregard for the requirements of Technical Specification 6.8-1 requiring the use of procedures. Specifically, Procedure PT/1/A/4700/27, Containment Spray Check Valve Inservice Test appeared to be inadequate nor was procedure OP/0/A/6100/09, R&R of Station Equipment employed. The latter procedure is required to be used for the manipulation of any station equipment under Operations control when the removal and restoration of that equipment is not covered by an established operating procedure.

Further inspector review of the above event indicated a lack of timely reporting. Code of Federal Regulations Part 50.73 "Licensee Event Report System," part (2)(v)(c) states in part that the licensee shall report within 30 days:

> "Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to control the release of radioactive material."

With valve 1NS-68 open, the auxiliary building atmosphere check valve 1NS-41 was the only barrier between containment atmosphere and the auxiliary building environment. Check valve 1NS-41 is not tested for leakage thus there was no way to demonstrate containment integrity. In the event of a radiological release in containment subsequent leakage into the auxiliary building is likely. This compromised containment integrity.

Contrary to these requirements the incident was not reported within 30 days as required by CFR 50.73. This is a violation (50-369/87-25-01).

9. Reactor Trip - Operator Error (93702)

On August 31, 1984, the Unit 2 reactor tripped on a two out of four power range (P/R) high flux rate signal. This signal was generated inadvertently by Instrument and Electrical (IAE) personnel while performing test procedure IP/0/A/3207/03B, Nuclear Instrumentation System (NIS) Power Range Rate Circuit (NM-311) and Bistable Relay Drivers (NC-301 and 303) Alignment.

P/R channel 43 was placed in the trip mode in preparation for testing. The power supply cable for P/R channel 42 was then mistakenly unplugged (instead of P/R channel 43), placing P/R channel 42 in the trip mode also. As a result, with two P/R channels placed in the trip mode, the reactor tripped automatically from 100% power.

Prior to the event, one of two IAE technicians removed the instrument fuses on the front of the N/I cabinet for P/R channel 43 per step 10.4 of IP/O/A/3207/03B. Then the IAE technician walked around a row of cabinets to get to the back of the N/I cabinet containing P/R channel 43 to disconnect channel 43's input plugs (per step 10.6 of IP/O/A/3207/03B). (The other IAE technician who was assisting with the test, stayed at the front of the cabinets.) The IAE technician opened the cabinet door for P/R channel 42 instead of the door for channel 43, and disconnected the input plugs on channel 42. This now placed both P/R channels 43 and 42 in the trip mode, and reactor trip was initiated.

The label for P/R channel 43 is on a column between the cabinet doors for channel 43 and 42; had the label been on the door itself, it may have caught the technician's attention and helped him realize that he was opening the wrong door. There are no labels inside the cabinet to identify the instrumentation contained within. Once the incorrect door was opened, it was unlikely that the technician would have realized he was working on the wrong channel.

The test procedure IP/O/A/3207/03B must be completed every 18 months to meet the surveillance requirement of Technical Specification 4.3.1.1 (channel calibration of P/R neutron flux setpoints). This test is completed on one P/R channel at a time, using a generic procedure with procedure steps that apply to any of the four P/R channels. The steps do not refer to the channel being tested; therefore, the IAE technician performing the test must keep in mind which channel is being tested.

Although independent verification (IV) was used to take P/R channel 43 out of service per step 10.5 of IP/O/A/3207/03B, IAE personnel did not perform IV while removing the input plugs per step 10.6 of the same procedure. According to the licensee, IV was not required for step 10.6 because the P/R channel would already be out of service (per step 10.5) and the input plug is verified to be reinstalled upon completion of the test by proper indication of the channel.

It is concluded that two conditions contributed to the event:

a. (1) Use of an inadequate generic procedure in that it failed to specify the P/R channel under test, and

   (2) An inadequate procedure for failing to fully incorporate independent verification so as to ensure that the technicians are at all times working on the correct channel.

b. Personnel error in that the operators failed to ensure they were at all times working on the correct channel.

It appears that the overriding factor was use of an inadequate procedure.

T.S. 6.8.1.a requires written approved procedures be employed in the performance of surveillance tests recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. Regulatory Guide 1.33, February 1978, Appendix A, 3.t identifies testing on the Nuclear Instrumentation System (NIS) as a typical safety-related activity that should be covered by written procedures. Implicit in that requirement is the requirement that the procedure entail sufficient specificity to facilitate the successful completion of the task. This procedure did not entail adequate detail, and as such constitutes a violation of TS 6.8.1.a. However, since this violation is similar to violations on inadequate procedures and inadequate implementation of independent verification identified in Inspection Report numbers 50-369/84-21 and 50-370/84-18, it will not be cited in this report.

10. IE Bulletins (92703)

The following IE Bulletin was reviewed to ensure receipt, evaluation, and appropriate implementation.

(Closed) IEB 84-02, Failure of General Electric Type HFA Relays in Class IE Safety Systems. The licensee has revised their response to this bulletin. Based upon the results of this review, this item is herewith closed.