UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
101 MARIETTA STREET, N.W.
ATLANTA, GEORGIA 30303

Report Nos.: 50-369/84-21 and 50-370/84-18

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: June 20 - July 20, 1984

Inspectors: for A. J. Ignatonis 9/28/84
W. Orders Date Signed

for A. J. Ignatonis 9/28/84
R. Pierson Date Signed

Approved by: Virgil L. Brownlee 10/1/84
V. L. Brownlee, Section Chief Date Signed
Division of Reactor Projects

SUMMARY

Areas Inspected

This routine unannounced inspection involved 276 resident inspector-hours on site in the areas of operations safety verification, surveillance testing, and maintenance activities.

Results

One violation was identified - two examples of inadequate procedures in which the licensee failed to restore a vent valve to its normally closed position following surveillance test of a check valve and an erroneous lift of a diode lead wire that resulted in a reactor trip.

DETAILS

1. Persons Contacted

Licensee Employees

M. McIntosh, Station Manager
*G. Cage, Superintendent of Operations, Acting Station Manager
D. J. Rains, Superintendent of Maintenance
R. Ruth, Quality Assurance
A. Butts, Quality Assurance
R. White, I&E Supervisor
D. Mendezoff, Licensing Engineering
S. McInnis, Licensing Engineer
R. Phillips, Operations Engineer

Other licensee employees contacted included technicians, operators, and mechanics.

*Attended exit interview

2. Exit Interview

The inspection scope and findings were summarized on July 27, 1984, with those persons indicated in paragraph 1 above. The licensee expressed cognizance and concern over the issues discussed. One violation with two examples as described in paragraph 8, involved use of an inadequate procedure and drawing, respectively, which contributed to an open vent valve in the auxiliary containment spray system and erroneous lift diode lead while troubleshooting, resulting in main steam line isolation valve closure and reactor trip. This violation was discussed in detail.

3. Licensee Action on Previous Inspection Findings

Not inspected.

4. Unresolved Items

Unresolved items were not identified during this inspection.

5. Plant Operations

The inspector reviewed plant operations 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 shifts and at shift changes. Actions and/or activities observed were conducted as prescribed in the Station Directives and/or Operations Management Procedures. The compliment of licensee 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 selective basis. The areas toured include, but were not limited to the following:

Turbine Buildings

Auxiliary Buildings

Units 1 and 2, Electrical Equipment Rooms

Units 1 and 2, Cable Spreading Rooms

Station Yard Zone within the protected area

Cowans Ford Dam, Nuclear Service Water Low Level Intake

During the plant tour, ongoing activities, housekeeping, security, equipment status and radiation control practices were observed.

McGuire Unit 1 began the reporting period operating at, and limited to 95% power due to axial flux deviation considerations as discussed in a previous report. On June 22, 1984, a Technical Specification change was approved permitting power operation at full power. Power was maintained at or about that power level throughout the remainder of the reporting period.

McGuire Unit 2 began the reporting period operating at full power and maintained at or about that power level until 4:00 p.m. on Tuesday July 3, 1984, when a reactor trip occurred due to "C" steam generator low level. The trip, occurred during trouble shooting of an Engineered Safeguards Feature slave relay circuit for Train A, "C" steam generator main steam isolation valve. Details of this event are entailed in paragraph 8. All systems responded as expected, except for one source range channel. When the source range was examined, it was determined that NI-31, one of two source range detectors, was inoperable. The unit was subsequently cooled down and depressurized to allow access to and repair of the detector.

The unit remained shutdown through Sunday, July 8, 1984. Unit startup commenced on Monday, July 9, 1984, with the unit achieving criticality at 00:20 a.m. Monday morning. The generator was subsequently placed on line at 06:20 a.m. that morning and the unit reached 100% power at 1:45 on Tuesday, July 10, 1984. Power was maintained at or about 100% until 6:40 p.m. on July 19, 1984, when a reactor trip occurred during the performance of response time testing of reactor trip breakers. Breaker 2RTA was incorrectly tripped from the control board. This also trips the shunt for the A train bypass breaker and the undervoltage coil of the B train bypass breaker. All safety systems responded normally during and immediately after the reactor trip. The unit completed the report period in Mode 3 at 2235 psig and 557°F.

6. Surveillance Testing

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
PT/0/A/4200/02	Standby Shutdown Facility Operability Test
PT/1 and 2/A/4209/09	Standby Make-up Pump Check Valve Test - Shutdown
PT/1 and 2/A/4209/01C	Standby Make-up Pump Flow Periodic Test
TP/1/A/1350/34	Standby Shutdown Facility Essential Equipment Functional Test
TP/2/A/1250/02A	Auxiliary Feedwater System Pre-Hot Functional Test
PT/1/A/4208/02	NS Valve Stroke Timing - Quarterly
PT/0/A/4601/08A	Solid State Protection System (SSPS) Train A Periodic Test Above Reactor Coolant System Pressures of 1955 PSI
IP/0/A/3090/02	Controlling Procedure for Instrument and Electrical Safety Related Maintenance
IP/0/A/3090/19	Implementation of Independent Verification and Temporary Modifications

7. Maintenance Observations

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 procedure 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
85616	Safeguards Test Cabinet
119556	Pressurizer Level Transmitter
85631	2-BB-140 A

8. Procedural Deficiencies

Categorized below are three examples of apparent inadequate procedures and/or accompanying documentation, or failure to follow procedural requirements.

These events are recent examples of what appear to be an illustration of deficient programmatic features and/or inadequate managerial control.

a. Potential Loss of Containment Integrity and Degradation of Auxiliary Containment Spray

On June 27, 1984, Unit 1, containment spray vent valve 1NS-68 was found open during a quarterly valve stroke timing test of the containment spray (NS) system. Approximately 35 gallons of residual heat removal (ND) water, remaining in the line from previous ND pump periodic tests was drained onto the floor in Room 815 of the Auxiliary Building. The vent valve was closed by a Performance Test Supervisor, who was visually verifying the test. At the time of identification, Unit 1 was operating in Mode 1 at 100% power.

Vent valve 1NS-68, the ND pump 1B discharge to NS nozzles containment isolation test connection, was apparently left open on or about April 17, 1984, while preparing to conduct PT/1/A/4700/27, Containment Spray Check Valve Inservice Test. A prerequisite of the check valve test was to ensure that the header was drained. It is theorized that when the header was drained, the reclosing of vent valve 1NS-68 was overlooked. This valve was open to the auxiliary building atmosphere, and 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 for the period between April 17, and June 27, 1984.

The licensee evaluated the consequences of this event with respect to primary containment integrity and degradation of the auxiliary containment spray system. The check valves inside containment are considered to act as automatic isolation valves. Therefore, according to the licensee containment integrity, although jeopardized, was maintained. The auxiliary containment spray system, which is lined up to the residual heat removal system, (ND) is manually initiated and is required only if 1) when the emergency core cooling system is operating in the recirculation mode and 2) either more than 50 minutes to an hour have passed since the initiation of the accident or both trains of the ND system are operating. Duke Design Engineering evaluated the effects on B Train ND to NS flow into containment if NS had been actuated with the peak accident pressure and with 1NS-68 fully open. The vent valve is a 3/4" Kerotest valve connected to an eight inch header. Flow through the vent valve, 1NS-68, to the Auxiliary Building floor was estimated to be 50 gpm. Usable flow to the containment would have been approximately 2220 gpm. Thus, ND to NS Train B was capable of its design function.

The NRC Region II office also evaluated the radiological consequences and system performance during accident conditions with the vent valve being open and found it to have minimal impact on containment integrity and design function of the system.

The inspector reviewed PT/1/A/4700/27 and the completed R&R's from April through June 1984 time frame and could not identify the manipulation of valve 1NS-68 by procedure. Procedure OP/0/A/1600/09, Removal and Restoration (R&R) of Station Equipment is required for the manipulation of any station equipment not covered by an establishment operating procedure. It appears that this procedure was not used for valve 1NS-68. Procedure PT/1/A/4700/27 was inadequate, in that it failed to specifically address the manipulation of valve 1NS-68 and the restoration of it to proper position following its use. Furthermore, this procedure failed to include independent verification on restoration of valve 1NS-68 to ensure that it is returned to its normally closed position following valve manipulation.

The Duke Power Administrative Policy Manual (APM), Revision 21 addresses the application of independent verification program requirements affecting station procedures and directives. Independent verification is required for equipment which improperly aligned could result in the release of radioactive liquids or gases from the site.

The root cause of this event appears to be inadequate attention to the requirements of Technical Specification 6.8.1 requiring the use of procedures and that they be adequate. Also, this Technical Specification requires that applicable procedures implement the requirements of NUREG-0737. Item I.C.6, Operating Activities, in effect, requires independent verification so as to verify correct performance of operating activities. The above stated requirements appear not to have been implemented and thus constitute a violation of Technical Specification 6.8.1.

In as much that there is another example on use of an inadequate drawing covering work on safety - related equipment detailed in paragraph 8.c., the two examples collectively constitute a violation (50-369/84-21-01; 50-370/84-18-01).

b. Inadvertent Reactor Protection System Trip

On July 2, 1984, at 2:29 p.m., with Unit 2 operating at 100% power, the Train A reactor trip breaker (RTB-A) was inadvertently tripped during the performance of PT/0/A/4601/08A, Solid State Protection System (SSPS), Train A Periodic Test Above Reactor Coolant System Pressure of 1955 PSI.

An IAE technician read a "Note" as an action statement and placed the "Input Error Inhibit" switch in the "Normal" position. This action was completed prior to reading the "CAUTION" statement which required the blocks to be installed on the source, intermediate and power range instruments for Train A. Failure to install the blocks in the correct sequence resulted in the "NIS Hi Flux I/R Reactor Trip" and the "NIS Hi Flux Low Setpoint P/R Reactor Trip" signals, causing the RTB-A to open. The Train A reactor trip bypass breaker was racked into the "Connect" position and remained closed throughout this incident, therefore, opening RTB-A did not result in a reactor trip.

This event appears to have been caused by two distinct deficiencies.

(1) The technician failed to follow the direction afforded in the procedure.

(2) The procedure was deficient in clarity and specificity by containing action statements in a caution statement.

TS 6.8.1.a requires written approved procedures be employed in the performance of surveillance tests. 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.

Pursuant to the provisions of 10 CFR 2, Appendix C, IV.A, a notice of violation will not be issued for this violation in as much as the event meets the criterion therein.

c. Inadvertent Reactor Trip

On July 3, 1984, during performance test of the Engineered Safeguards Features slave relay circuit for the Unit 2 Train A main steam isolation valve (MSIV), a test indicator lamp failed to light. In troubleshooting the cause of this problem, an Instrument and Electrical (IAE) technician erroneously lifted a lead of a diode for the solenoid valve which controls air to the "C" steam generator MSIV. The solenoid de-energized, resulting in the MSIV closure and subsequent reactor trip from 100% power on low-low level in steam generator "C".

The trouble shooting was performed under the general provisions of procedures, IP/O/A/3090/02 - Controlling Procedure for Instrument and Electrical Safety Related Maintenance and IP/O/A/3090/19 - Implementation of Independent Verification and Temporary Modifications. The schematic diagram used, MCM-1399.08-113, was inadequate in that it did not entail the circuity on the output of the terminals joined by the diode which the IAE technician lifted. Furthermore, although the technician used the two procedures listed on the work request, neither of these specifically dealt with trouble shooting the safeguard test cabinet.

The cause of pulling the wrong lead of a diode was personnel error contributed by lack of information (drawings, etc.) to identify the potential consequences. Furthermore, this event appears to be similar to a violation reported in Inspection Report Nos. 50-369/84-11 and 50-370/84-09 where on April 20, 1984, a technician inadvertently de-energized an essential 4160 volt bus during an Engineered Safety Features test when a jumper was installed on the lifted lead rather than on the required terminal. Contributing factor to the event was an erroneous procedure and misleading electrical elementary drawings. This constitutes a violation of Technical Specification 6.8.1 which requires use of adequate procedures/drawings when performing work on safety related equipment, and is another example of use of inadequate instructions as detailed in paragraph 8.a of this report. (50-369/84-21-01; 50-370/84-18-01).