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

Report Nos.: 50-369/87-30 and 50-370/87-30

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

Facility Name: McGuire Nuclear Station 1 and 2

Docket Nos.: 50-369 and 50-370

License Nos.: NPF-9 and NPF-17

Inspection Conducted: August 1 - 25, 1987

Inspectors: _____ 9/11/87
W. Orders, Senior Resident Inspector — Date Signed

_____ 9/11/87
S. Guenther, Resident Inspector — Date Signed

Accompanying Personnel: D. Nelson

Approved by: _____ 9-11-87
T. A. Peebles, Section Chief — Date Signed
Division of Reactor Projects

SUMMARY

Scope: This routine unannounced inspection involved the areas of operations safety verification, surveillance testing, maintenance activities and followup of previous enforcement actions.

Results: In the areas inspected, no violations or deviations were identified.

REPORT DETAILS

1. Persons Contacted

Licensee Employees

*T. McConnell, Plant Manager
B. Travis, Superintendent of Operations
*D. Rains, Superintendent of Maintenance
*B. Hamilton, Superintendent of Technical Services
N. McCraw, Compliance Engineer
M. Sample, Superintendent of Integrated Scheduling
*N. Atherton, Compliance
*J. Snyder, Performance Engineer
*M. Robson, Design Engineer

Other licensee employees contacted included construction craftsmen, technicians, operators, mechanics, security force members, and office personnel.

*Attended exit interview

2. Exit Interview

The inspection scope and findings were summarized on August 28, 1987, with those persons indicated in paragraph 1 above. One previous Unresolved Item (50-369/87-19-01), involving the potential violation of the Unit 1 rated thermal power limit was discussed; licensee representatives indicated that they were continuing to evaluate whether or not the thermal power limit was violated. The licensee did not identify as proprietary any of the information reviewed by the inspectors during the course of their inspection.

3. Unresolved Items

An unresolved item (UNR) is a matter about which more information is required to determine whether it is acceptable or may involve a violation or deviation.

No new unresolved items were identified in this report.

4. Plant Operations

The inspection staff reviewed plant operations during the report period to verify conformance with applicable regulatory requirements. 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.

Activities within the control room were monitored during shifts and at shift changes. Actions and/or activities observed were conducted as prescribed in applicable station administrative directives. The complement of licensed personnel on each shift met or exceeded the minimum required by Technical Specifications.

Plant tours taken during the reporting period included, but were not limited to, the turbine buildings, the auxiliary building, Units 1 and 2 electrical equipment rooms, Units 1 and 2 cable spreading rooms, and the station yard zone inside the protected area.

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

a. Unit 1 Operations

Unit 1 operated at, or near, rated power from the beginning of the reporting period until the morning of August 14, 1987, when the unit was shutdown to allow the addition of oil to the upper reservoir of the "D" reactor coolant pump. One of the source range nuclear instruments failed to respond properly during the shutdown and, ultimately, required replacement of the channel's detector prior to restarting the unit late on the evening of August 15. At 1:05 a.m., on the morning of August 16, reactor power was at approximately six percent and the operators were making preparations to place the main generator on the line. An apparent "malfunction" in the digital electrohydraulic (DEH) control system caused all the turbine control valves to rapidly open, thereby causing a rapid decrease in main steam line pressure. Although main steam line pressure only decreased to 1011 psi, the rate at which the reduction took place caused the engineered safeguards features actuation circuitry to actuate and initiate a safety injection. The licensee declared a Notification of Unusual Event (NOUE) in accordance with its Emergency Plan Implementing Procedures and partially actuated its technical support center to expedite response to the event. All safety systems appeared to function normally during the transient; however, some of the moisture separator/reheater (MSR) relief valves, which lifted in response to the turbine control valves' opening, failed to reseat fully and resulted in a loss of condenser vacuum. This complicated the recovery from the transient because it eliminated the condenser steam dump valves from use as a heat sink. By 1:45 a.m., the plant was stable at approximately 550 degrees Fahrenheit and 2235 psi, using the steam generator power operated relief valves to maintain temperature.

At about 2:30 a.m., the licensee began to suspect that an approximate 35 - 40 gallon per minute reactor coolant system leak had developed inside containment during the transient. Normal reactor coolant system letdown flow, which had isolated during the safety injection,

had been recently restored and was evaluated as the most likely source of the leak. At about 5:30 a.m. the leak was located on a drain line connected to the letdown header. The leak was isolated shortly thereafter when excess letdown flow was established and normal letdown was secured. Condenser vacuum was reestablished at about 10:40 a.m. when the licensee succeeded in reseating the MSR relief valves. The NOUE was terminated at about 4:15 p.m., after completing two satisfactory reactor coolant system leakage calculations.

The leak emanated from a crack in the heat-affected zone of a weld connecting the 3/4 inch drain line for 1NV-829 to the three inch letdown line. The licensee dye penetrant tested several welds on other letdown line vent and drain valves and on the letdown header itself. Only one other crack indication was identified; it was on a one inch drain line socket weld in the vicinity of the original leak, but had not progressed through-wall to generate another leak. The failed drain lines were removed and shipped to a contractor for analysis to determine the failure mechanism before the unit was restarted. It was determined that the failures were initiated by high-cycle fatigue that occurred early in the plant's history. That high-cycle fatigue had been arrested (presumably by the installation of a variable letdown orifice) and was followed by several low-cycle fatigue events (presumably water hammers) which caused the crack to propagate and eventually fail. The licensee inspected several pipe supports on the letdown line for evidence of a severe water hammer event but failed to find any indication that one had occurred.

Both drain lines were repaired to their original configuration for unit restart, but the licensee is evaluating the possibility of removing those and other unused vent/drain valves on the letdown line during the upcoming Unit 1 outage. The licensee is also continuing to investigate this sequence of events and has scheduled three Abnormal Plant Event (APE) meetings for upper level site management to discuss and review the incident and associated corrective action assignments.

While reestablishing normal letdown flow on August 20, the operators noted that 1NV-459, the variable letdown orifice valve, appeared to be passing an excessive amount of flow even with the valve indicating fully closed. The licensee isolated and disassembled the valve and found apparent erosion induced damage. The valve internals were replaced, the valve was retested and the unit was restarted on the afternoon of August 21. The licensee plans to have the failed 1NV-459 internals evaluated by a contractor to determine the cause for the failure.

The resident inspectors will track the licensee's investigation and corrective actions for the letdown line/valve failures/transients, the apparent DEH malfunction and the MSR relief valve failures as an IFI (50-369/87-30-01).

b. Unit 2 Operations

Unit 2 operated in Mode 1 (power operation) from the beginning of the reporting period until the evening of August 14, 1987, when a controlled unit shutdown was initiated. On the morning of the 14th, the licensee discovered that three bolts, which secure the missile shield blocks that separate the upper and lower containment compartments, were missing. One of those bolts was later determined to be in a critical location with regard to maintaining the operability of the containment divider barrier, so, at about 8:00 p.m. on August 14, a unit shutdown was commenced and a NOUE was declared. The reactor entered Mode 3 (hot shutdown) at 11:45 that evening and was maintained in that Mode until about 4:30 p.m. on the 15th, when, in order to comply with the Technical Specifications, a cooldown to Mode 5 (cold shutdown) was initiated. The cooldown and the NOUE were terminated at 4:15 a.m. on the 16th, when the missing hold-down bolts were replaced. The details concerning this incident will be addressed in NRC special inspection report number 50-369,370/87-35.

A Unit 2 restart was initiated on the evening of August 16, with criticality being achieved at about 6:55 p.m. The unit was at approximately one percent power at 8:35 p.m., with steam generator feedwater being supplied by the auxiliary feedwater (CA) system. Neither steam-driven main feedwater pump was in service at the time because of the limited steam supply. One of the main feed pumps was being started in accordance with the applicable operating procedure, which requires that the trip circuitry be reset and then tested by inserting a manual trip signal. Since the other feed pump was still in the tripped condition, the CA system actuation logic interpreted the trip as a total loss of main feedwater and started the remaining CA pumps. The licensee reported the ESF actuation as required and has initiated action to change the procedure to prevent recurrence. This incident will be tracked as an IFI pending review of the modified procedure (IFI 50-369,370/87-30-02).

After returning to service on August 16, Unit 2 escalated to full power and remained there for the balance of the reporting period.

No violations or deviations were identified.

5. Surveillance Testing

Selected surveillance tests were analyzed and/or witnessed by the inspector to ascertain procedural and performance adequacy and conformance with applicable Technical Specifications.

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

Detailed below are selected tests which were either reviewed or witnessed:

IP/0/A/3207/02E	Nuclear Instrumentation System (NIS) Intermediate Range Compensation Voltage Adjustment
PT/1/A/4250/23	Venturi Fouling Test
PT/2/A/4600/01	Rod Control Cluster Assembly (RCCA) Movement Test
PT/1/A/4601/02	Protective System Channel 2 Functional Test

No violations or deviations were identified.

6. Maintenance Observations

Routine maintenance activities were reviewed and/or witnessed by the resident inspection staff to ascertain procedural and performance adequacy and conformance with applicable Technical Specifications.

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

No violations or deviations were identified.

7. Followup of Previous Enforcement Actions

(OPEN) UNR 50-369/87-19-01 - Review the Problem Investigation Report (Serial No. 1-M87-0116) regarding possible Unit 1 operation above the rated thermal power limit.

Technical Specification (TS) 1.25 defines rated thermal power as the total core heat transfer rate of 3411 megawatts thermal (MWT). The licensee's interpretation of this TS allows slight variations above 100 percent rated thermal power as a result of instrument inaccuracies/variations and control instabilities. The average power level as indicated by computer heat balance calculations over any eight-hour period should not exceed the full steady state power level of 3411 MWT. This interpretation has been found acceptable to the NRC.

The McGuire operator aid computer (OAC) normally monitors secondary thermal power, by means of a calorimetric calculation, as the best estimate of reactor thermal power output. The normal in-line feedwater flow nozzles/venturies, which provide input to the calorimetric calculation, are prone to fouling during unit operation, so the licensee

has installed more accurate, calibrated nozzles instrumented with precision gages for use in performing periodic venturi fouling tests (PT/1/A/4250/23). These tests yield a correction factor/coefficient, which is applied to the normal in-line calorimetric calculations performed by the OAC.

On June 11, the licensee determined that the venturi fouling test conducted on May 20, 1987, had not taken into account the 1.0076 fouling correction factor from the previous venturi fouling test. This resulted in a 0.98784 baseline feedwater flow correction being used instead of a 0.99535 (i.e., 1.0076 x 0.98784) coefficient and caused a nonconservative calculation of secondary/rated thermal power.

The licensee evaluated the venturi fouling test results and the best estimate reactor thermal power history for the period from May 20 through June 11, 1987. The power history from the nuclear station log data base (the OAC) consisted of hourly, one-minute "snapshots" of the secondary thermal power best estimates, which were then averaged to provide eight-hour power estimates. These averages indicated that the Unit 1 reactor had apparently been operated at eight-hour average power levels in excess of the 100 percent rated value during several eight-hour periods. The highest indicated eight-hour average power level maintained during the 22-day period was approximately 100.2 percent.

These indicated power levels, however, were nonconservative because the calorimetric calculations performed by the OAC did not incorporate the appropriate venturi fouling test correction factors; the actual reactor thermal power level during that period of time appears to have exceeded 100.2 percent during at least one eight-hour period.

The licensee's Problem Investigation Report (PIR No. 1-M87-0116) was initiated on June 11, 1987, and indicated that the incident was reportable to the NRC pursuant to 10 CFR 50.73, Section a.2.i.B (i.e., any operation or condition prohibited by the plant's TS). On June 18, the licensee issued a revised PIR evaluation, indicating that even with the incorrect fouling correction factor in use, Unit 1 power averaged 99.87 percent during the period from May 20 to June 11, and, therefore, was not reportable. Accordingly, the licensee's initial intent to submit a Licensee Event Report (LER) explaining the event was lined out on the revised PIR.

The inspector reviewed the licensee's PIR evaluation, the indicated power history during the period of time in question, and the TSs related to reactor thermal power output. It appeared that the licensee had erred in its evaluation and decision not to report the incident pursuant to 10 CFR 50.73. That conclusion was based on the fact that although the average reactor power for the period may have been below 100 percent of rated, the licensed TS limit of 3411 MWT was apparently exceeded during several eight hour periods during the time in question.

The resident inspector questioned the licensee's Compliance staff regarding the incident. The licensee has since reopened the PIR and is continuing to evaluate whether or not that 3411 MWT rated power limit was, indeed, exceeded. If a positive determination is made, then the licensee intends to submit an LER pursuant to 10 CFR 50.73, and the resident inspectors will reevaluate this UNR with regard to the self-identification criteria of 10 CFR 2, Appendix C.