



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
REGION II  
101 MARIETTA STREET, N.W., SUITE 2900  
ATLANTA, GEORGIA 30323-0199

Report Nos. 50-369/94-09 and 50-370/94-09

Licensee: Duke Power Company  
422 South Church Street  
Charlotte, NC 28242-1007

Facility Name: McGuire Nuclear Station 1 and 2

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

Inspection Conducted: April 10, 1994 - May 14, 1994

Inspector: M Sinkule for 6/13/94  
G. Maxwell Date Signed  
Senior Resident Inspector  
G. Harris  
Resident Inspector

Accompanying Inspectors: K. Kavanagh, Reactor Engineer Intern, McGuire  
R. Watkins, Project Engineer, RII

Approved by: M Sinkule 6/13/94  
M. Sinkule, Chief Date Signed  
Reactor Projects Branch 3

SUMMARY

Scope: This routine, resident inspection was conducted in the areas of plant operations, surveillance testing, maintenance observations, licensee emergency drill, emergency safety features system walkdown, TS amendment changes, and followup on previous inspection findings. Backshift inspections were performed on April 10, 11, 12, 13, 14, 15, 22, 25, 26, 27, 28, 29, 30, and May 4 and 6, 1994.

Results: In the area of operations, the inspectors identified violation 94-09-01, Inadequate procedure for handling severe weather threats to the facility, paragraph 2.d.

In the area of operations, the inspectors noted good operator response to recent plant challenges, paragraph 2.a.

In the area of operations, the inspectors concluded that the new shift turnover format should improve communications among operations shift personnel, paragraph 2.a.

In the area of maintenance, weaknesses in work planning, communication among station groups, and control room area ventilation arrangement contributed to an event that allowed vapors from a chemical solvent to enter the control room. This will be tracked as Unresolved Item, 94-09-04, paragraph 4.c.

In the area of maintenance, the inspectors concluded that human performance errors continued to contribute to station and equipment unavailability. Personnel, while performing preventive maintenance on ground detection equipment, placed a screwdriver on a shelf that later fell and struck a reverse power trip relay resulting in a reactor trip, paragraph 2.b.

In the area of surveillance, an Unresolved Item, 94-09-03, was identified due to the past inoperability of the leakage detection system. In addition, the engineering group has yet to resolve continuing problems with instability in the station's unidentified leakage calculation, paragraph 4.d.

In the area of plant support, the inspectors concluded that a weakness exists in the facilities's process of technical specification amendment due to lack of procedural guidance, paragraph 5.

In the area of plant support, the processing of vendor information continues to be a problem, paragraph 4.d.

## REPORT DETAILS

### 1. Persons Contacted

#### Licensee Employees

- \*J. Allgood, Safety Review Group
- \*T. Arlow, Safety Review Group
- \*D. Baxter, Support Operations Manager
  - A. Beaver, Operations Manager
  - J. Boyle, Work Control Manager
  - B. Caldwell, Training Manager
- \*R. Cross, Compliance Specialist
  - T. Curtis, System Engineering Manager
- \*R. Deese, Safety Review Group
  - E. Estep, INPO Coordinator
- \*E. Geddie, Station Manager
- \*G. Gilbert, Safety Assurance Manager
- \*B. Hasty, Emergency Planner
  - F. Hayes, Human Resources
- \*P. Herran, Engineering Manager
- \*D. Jamil, Electrical Engineer
- \*R. Jones, Superintendent of Operations
- \*C. McGinnis, Work Process Manager
- \*T. McMeekin, Site Vice President
  - M. Nazar, Instrument & Electrical Maintenance Superintendent
  - R. Ouellette, Systems Engineering
- \*K. Reece, Elec./Mod./SPOC Manager
- \*R. Sharpe, Regulatory Compliance Manager
- \*J. Snyder, Regulatory Compliance Manager
- \*B. Travis, Component Engineering Manager
  - H. Wallace, Mechanical Engineering Supervisor
- \*R. White, Mechanical Maintenance Superintendent

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

#### NRC Resident Inspectors

- \*G. Maxwell, SRI
- \*G. Harris, RI
- K. Kavanagh, Intern
- R. Watkins, RII

\*Attended exit interview

### 2. Plant Operations (71707)

#### a. Observations

The inspection staff evaluated plant operations during the report period to verify conformance with applicable regulatory requirements. Control room logs, shift turnover records and equipment removal and restoration records were routinely reviewed.

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 were conducted as prescribed in applicable station administrative directives. The number of licensed personnel on each shift met or surpassed the minimum required by Technical Specifications (TS).

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

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

The inspectors noted the following operational observations and concerns:

- The inspectors reviewed logs and through interviews determined that operator response during two recent plant challenges was good. For example, on May 10 operators noticed that feedwater regulating valve 2CF32 was opening without a demand signal present because of a 7300 process control card failure. An attentive operator noticed that the valve was opening and took manual control of it. The operator continued to monitor the valve while I&E personnel expedited repairs to the 7300 process control card. In addition, the inspectors noted good response and use of procedures during the May 12th reactor trip.
- The inspectors observed that all the alarms on the Unit 1 annunciator panels, 1AD12 and 1AD13, had alarmed at the same time on at least four occasions since 2/10/94. Work orders 94011073, 94017257, 94031357, and 94031771 were generated by the plant staff to investigate the problem. In each case a failed annunciator card was found. When the bad card was replaced the annunciator panel malfunction was corrected. The inspectors noted that each annunciator panel failure was caused by different failed cards. The licensee has generated a Problem Investigation Process (PIP) form, 1-M94-0507, to determine what caused the different cards to fail.
- During this reporting period, the inspectors observed that operations has implemented changes to its shift turnover procedures. These changes included the shift turnover times and the shift briefings. The shift turnover times were changed from 7:30 a.m. and 7:30 p.m. to 7:00 a.m. and 7:00 p.m. This change was implemented to ensure shift turnover was complete prior to the start of maintenance and testing

of plant equipment. The shift briefings are held in the control room at approximately 7:40 a.m. and 7:40 p.m. with all senior reactor operators (SROs), reactor operators (ROs), and non-licensed operators (NLOs) present. Previously, a separate meeting was held between the SROs and the NLOs in the "NLO kitchen." This change was implemented to ensure that the entire shift was receiving the same information. The inspectors concluded that the new shift turnover format enhances communications between operations personnel.

While performing a walkdown of the Unit 1 motor-driven and turbine-driven auxiliary feedwater (CA) pump rooms, the inspectors attempted to verify that the auxiliary shutdown panel (ASP) was locked. On this occasion it was not locked, and when the inspectors opened the door an annunciator alarmed in the control room. The inspectors notified the control room and the door was promptly locked. Two controlled access doors (CAD) must be entered using a keycard prior to reaching the motor-driven CA pump room. A breakglass station on the panel contained a key to the panel for quick access. The inspectors and the plant staff could not locate any requirements that the ASP door be locked. The licensee usually maintained the ASP locked as an additional protective measure. This incident had occurred previously and was documented on a plant Problem Investigation Report.

b. Unit 1 Operations

Unit 1 experienced a reactor trip on May 12 that was caused by a human performance error. Two I&E technicians were performing routine preventive maintenance on ground detection circuitry in the control room panels. A screwdriver used during maintenance activity, within one of the panels, rolled off a protective shelf and struck the reverse power trip relay, 194XG2, causing a signal to be generated that tripped the exciter output and main generator output breakers. All primary and secondary systems responded normally during the trip with some exceptions. EMF-38 detector was actuated due to the release of particulate to the containment atmosphere. In addition, a low level signal was received for B steam generator because of the excessive opening of its associated power operated relief valve (PORV). This subsequently caused a motor driven auxiliary feedwater pump to start. The B PORV stayed open longer due to the unavailability of SV-7, C S/G PORV.

The inspectors reviewed logs and work orders, and through interviews determined that steam generator power operated relief valve, SV-7, had been out of service a number of times because of slow response times caused by internal binding. Recent machining of the valve was suggested as a possible contributor to the binding problem. The licensee has taken data during response time

testing but has yet to determine a root cause. In addition, a new type of packing that requires fewer packing rings had been installed in the valve. The valve was isolated during the recent full load rejection reactor trip that caused all secondary steam relieving components to actuate. Because SV-7 was out of service during the reactor trip, the B power operated relief valve opened sooner and remained open longer, resulting in a lower than anticipated steam generator level in the associated steam generator. The low steam generator level caused an unnecessary actuation of the auxiliary feedwater system. The licensee is continuing to evaluate the power operated relief valve's deficiencies.

The reactor tripped on a high negative rate signal. The negative rate signal is generated when an instantaneous drop in nuclear power is sensed on two of four power range detectors (i.e., -5% power drop, with a two second time constant). The licensee and the inspectors observed the negative rate trip on at least two other occasions. The inspectors reviewed trip data for the nuclear instrumentation and found that nuclear power had not decreased rapidly enough to cause a negative rate trip. The licensee has previously contacted Westinghouse, who agreed that a negative rate trip signal generated by rods driving into the core at a rapid rate due to a full load rejection was not valid. The inspectors reviewed TS and observed that the negative rate trip had been removed from the TS as part of an earlier amendment. All nuclear instrumentation channels were operable at the time of the trip and fully capable of performing their safety function. The licensee had previously written PIP 2-M93-1339, Reactor Trip Investigation, to evaluate the cause of the negative rate reactor trip. This PIP currently shows that the negative rate trips are being generated when rods are inserted and when electrical background noise or changes in the station grounding schemes induce spikes on all four channels.

The unit was returned to 100% power within 24 hours of the trip. The unit operated at essentially 100% power prior to and after the trip without event. The inspectors will continue to investigate the cause of the negative rate trip and conduct a detailed post-trip review of the event.

c. Unit 2 Operations

The unit operated at essentially 100% power throughout the period.

d. Severe Weather Preparations

The inspectors concluded that the Station Procedure RP/0/A/5700/06 Natural Disasters, does not contain adequate instruction to ensure that the facility is protected during severe weather conditions. On March 27, 1994, Mecklenburg county, where the plant is located, was placed under a tornado warning by the National Weather

Service. Although the announcement of the tornado warning warranted implementation of the station procedure, the crew did not enter the procedure because the instructions were confusing. The inspectors reviewed the procedures and determined that the instructions are designed to cover multiple disasters. The procedure does not provide the operator with sufficient detailed instructions to protect the facility. For example, in the event of a tornado threat the procedure did not instruct the shift crew to return important equipment to service, stop fuel handling and activities involving radioactive materials, or stop radioactive releases or realign ventilation systems. Moreover, the procedure did not provide for transition to the Emergency Action Level (EAL) procedures. The procedure also did not provide instructions for the shift crew to perform for a tornado watch. In addition, the inspectors identified some training weaknesses; for example, on-shift manager did not realize the facility was under a tornado warning. The shift control room SRO initially was not knowledgeable of the procedure's existence.

The inspectors reviewed the probabilistic risk assessment for the facility and found that a tornado was the largest contributor to core damage frequency for an external event. This will be identified as Violation 50-369,370/ 94-09-01, Inadequate severe weather preparations. This violation will superseded URI 50-369,370/94-08-02.

One violation was identified.

### 3. Surveillance Testing (61726)

#### Observed Surveillance Tests

Selected surveillance tests were reviewed and/or witnessed by the resident inspectors to assess the adequacy of procedures and performance of, as well as conformance with, the applicable TS.

Selected tests were witnessed to verify that (1) approved procedures were available and in use, (2) test equipment in use was calibrated, (3) test prerequisites were met, (4) system restoration was completed, and (5) acceptance criteria were met.

The following selected tests were reviewed or witnessed in detail:

#### a. Control Room Ventilation Outside Air Pressurization Performance Test

The inspectors observed the performance of the control room outside air pressurization performance test. Technical Specification surveillance requirement 4.7.6.c requires that, at least once per 31 days and on a STAGGERED TEST BASIS, the system be tested by initiating, from the control room, flow through the HEPA filters and charcoal absorbers and verifying that the system

operates for at least 10 hours with the dryimng heaters operating. The control room outside air pressurization is designed to pressurize the control room during the a station blackout or loss of coolant accident. The control room outside air pressurization system consists of two independent pressurization fans with charcoal and high efficiency filter units. The system pressurizes the control room to .6 -.8 inch, water gauge. Technical Specification surveillance requirements required that the system pressurize the control room to a minimum of .125 water gage. The inspectors reviewed the surveillance acceptance criteria to ensure that they were met. The inspectors also reviewed a problem investigation report that revealed during an earlier test that isolation valves VC-1 and VC-2 had deteriorated seals. An engineering review by the licensee revealed that the deteriorated condition of these seals did not prevent the valve from performing its isolation function. The inspectors agreed with the enginnering evaluation.

No discrepancies or abnormalities were observed during the test.

b. Standby Shutdown Facility Operability Test, PT/0/A/4200/02

The inspectors observed the implementation of the Standby Shutdown Facility (SSF) operability test, which demonstrated that the SSF was operable once per 31 days. The test staff utilized procedure PT/0/A/4200/02, Standby Shutdown Facility Operability Test, which incorporated OP/0/B/6350/04 Enclosure 4.1, Startup of SSF Diesel in Test Mode, and Enclosure 4.3, Shutdown of SSF Diesel, as part of the testing procedure. The inspectors observed the test staff perform OP/0/B/6350/04 Enclosure 4.1 for loading the SSF diesel to 700 kW. While evaluating the material condition of the SSF diesel, the inspectors noticed significant coolant leakage accumulating underneath the diesel. This observation was communicated to the test staff and led to the termination of the test. The test staff performed OP/0/B/6350/04 Enclosure 4.3 to shutdown the SSF diesel. Aside from the leakage, no discrepancies were observed by the inspectors. The inspectors noted that maintenance work on the SSF diesel was completed prior to the operability test. The licensee believed that the leakage was coming from the area around the diesel water pump. Work order 94012372-02 was generated to investigate the problem. The leakage was identified as water from the water pump modifications; repairs were made to fix the leak. The operability test was performed again and no leaks were detected. The SSF diesel successfully met the operability requirements.

The inspectors verified that the procedures located in the SSF control room were current and that all applicable changes were incorporated using PT/1/A/4700/11, Auxiliary Plant Panels' Document File Verification. No deficiencies were identified. The inspectors performed a walkdown of all components within the SSF.



The inspectors did not observe any problem with the material condition of the SSF.

c. Leakage Detection Systems

The licensee initiated an investigation of the reactor coolant leakage detection system. The system should have been designed to provide an alarm in the control room to signify, within one hour an increase in reactor coolant leakage of greater than 1 gallon per minute.

Technical Specifications state that the following Reactor Coolant Systems shall be operable:

- a. The Containment Atmosphere Gaseous Radioactivity Monitoring System (EMF-39);
- b. The Containment Floor and Equipment Sump Level System or Flow Monitoring System, and
- c. Either the Containment Ventilation Condensate Drain Tank (VUCDT) Level Monitoring System or a Containment Atmosphere Particulate Radioactivity Monitoring System (EMF-38).

The Process and Effluent Radiological Monitoring system monitors primary and secondary system malfunctions. Containment Atmosphere Particulate and Gaseous Detectors, EMF 38 and EMF 39, are provided for each unit to monitor the containment atmosphere for radioactive particulate and gases.

The Containment Floor and Equipment Sumps are a sub-system of the Liquid Waste system. These sumps are located in the containment building. The sumps collect liquid that leaks from systems inside the containment building.

The Ventilation Unit Condensate Drain Tank (VUCDT) is a subsystem of the WL system. The VUCDT of each unit receives water that has been condensed by the Containment Ventilation cooling units and stores the water prior to subsequent release.

The licensing basis for the alarms was determined to be Regulatory Guide 1.45.

Engineering personnel determined on April 6, 1994, that EMF-38 and EMF-39 may have been past inoperable depending upon the background radiation levels in the containment. Typically, the alarm setpoint of these EMFs is three times the background reading of the instrument. This setpoint is chosen to ensure compliance with the requirements of 10CFR20 and the Offsite Dose Calculation Manual. However, the alarm requirements from Regulatory Guide 1.45 are calculated using assumed values for NC system activity. Therefore, with the high background reading on the instruments and

a correspondingly high value for three times the amount of background radiation, the assumed activity would not be sufficient to actuate the alarm within 1 hour for a 1 gpm leak. Further research revealed that the CF&E Sump Level indicating system and the Containment Ventilation Unit Condensate Drain Tank Level indicating system both may have been past inoperable because they were unable to provide a warning of leakage in excess of 1 gallon per minute in less than 1 hour.

In response, operations personnel issued Operations Special Order 94-06. This special order was written in response to the questions surrounding the station's ability to meet the requirements of Regulatory Guide 1.45. This order instructs the operations shift personnel to monitor the level in the CF&E sump once every hour and calculate a volume input into the sumps. Special order 94-06 also provided actions to be taken if EMF-38 became inoperable. This Special Order was deleted and replaced with PT/O/A/4200/40, Reactor Coolant Leakage Detection. The procedure provides specific instructions to the Control Room personnel to monitor various indications for NC leakage.

The inspectors reviewed the FSAR and Technical Specifications which clearly state that the instruments should meet the requirements of Regulatory Guide 1.45.

Capability of the instruments to calculate the input to the CF&E sumps or to the VUDCT was not a part of system design. The level monitoring instrumentation for these portions of the WL system are designed with sufficient sensitivity to discriminate inputs at the 1 GPM range. However, the instrumentation had no capability to calculate input flow.

The inspectors reviewed the DBD and could not find references to the regulatory requirements contained in Regulatory Guide 1.45.

The following concerns have been evaluated previously by the inspectors:

- 1) Previous reports revealed that the past unidentified leakage calculation was in error due to a design deficiency that did not include some inputs in the PRT and RCDT. The inspectors determined that on at least 12 occasions last year unidentified leakage had exceeded TS values. This was identified as Violation 50-369,370/ 94-08-01, Unidentified Leakage Calculation.
- 2) The inspectors previously noted that instability continues to be a concern in the unidentified leakage calculation for Unit 1. The licensee had earlier surmised that the instability was due to Charging Flow Control Valve, NV-238, position. However, subsequent evaluation by engineering disproved this theory. The instability in the calculation could possibly make it difficult to

trend true increases in unidentified leakage rate. The inspectors will continue to follow this issue.

The inspectors will continue to assess the licensee's corrective actions as they relate to the leakage detection system deficiencies. This will be identified as Unresolved Item 50-369,370/ 94-09-02, Deficiencies of leakage detection systems.

One Unresolved Item was identified.

#### 4. Maintenance Observations (62703)

Resident inspectors reviewed and/or witnessed routine maintenance activities to assess procedural and performance adequacy and conformance with the applicable TS.

The activities were witnessed to verify that, where acceptable, approved procedures were available and in use, prerequisites were met, equipment restoration was completed, and maintenance results were adequate.

The following maintenance activities were reviewed or witnessed in detail:

##### a. Emergency Diesel Generator - Voltage Regulators

Recently a licensee plant located in Region II encountered simultaneous degradation of both of the plant's emergency diesel generators. The degradation occurred when the voltage regulators had their adjustable potentiometers adjusted to settings that were too low. Apparently, the potentiometers were physically located on the front of the regulator cabinets where they could be inadvertently bumped and mispositioned. The inspectors evaluated the type and design of the voltage regulators for the McGuire plant emergency diesel generators for similarity. The inspectors determined that the McGuire voltage regulators did not have external settings that could be accidentally mispositioned.

This other Region II plant also experienced failures of the diesel generators' poppet valves, which resulted in start failures. The valve failures resulted from a slipping cam that provided starting air distribution timing. The diesel generators at the plant have air distributors designed by Fairbanks. McGuire diesels have starting air distributors; however, the distributors are keyed into place.

##### b. Nitrogen Supply Containment Isolation Valve Failure INI-47

The inspectors determined from control room records that on April 19 the Unit 1 nitrogen supply containment isolation valve, INI-47, failed open after the reactor operators filled the cold leg accumulators. Valve INI-47 supplies nitrogen for cover pressure to all four Unit 1 cold leg accumulators. The plant staff tagged

closed INI-45 and INI-202 in compliance with Technical Specification (TS) 3.6.3.c, Containment Isolation Valves, to maintain containment integrity. The licensee discovered that the piping between INI-47 and INI-45 was Duke Class G, which is not seismically qualified. The licensee generated a PIP form, 1-M94-0510, to investigate the acceptability of using Class G components to fulfill containment integrity requirements. The plant staff used the Seismic Qualification Utility Group criteria in order to determine if the piping would resist an earthquake. After a review and walkdown of the applicable piping, the licensee concluded that the piping was operable. The licensee also generated a PIP, 1-M94-0522, to determine the cause of the INI-47 failure.

Upon further discussions with the plant technical staff, the inspectors learned that the licensee had approximately 2.5 days before they had to declare the cold leg accumulators inoperable on low nitrogen cover pressure per TS 3.5.1.1, Accumulators Cold Leg Injection, as a result of nitrogen leakage. One inoperable cold leg accumulator required the plant to be in Hot Standby within 6 hours. The plant staff's preliminary investigation of the INI-47 failure indicated that the gear on the primary mechanical switch was bad. The plant staff replaced the actuator with another actuator, verified correct rotation of the valve, and tested (using VOTES) the valve. The licensee stated that the valve was cycled and proper indication was verified. The inspectors noted that INI-47 was governed by generic letter 89-10.

The inspectors accompanied the plant staff into containment to witness the penetration leak test. The inspectors verified that all applicable procedures were followed and the readings from the leak rate monitor were recorded. The inspectors noted that the leak rate for INI-47 was in the alert range but the readings were not close to the failure limit. The inspectors verified that containment integrity was restored within 20 minutes. After completion of the appropriate documentation, the plant staff declared INI-47 operable and exited TS 3.6.3.c.

The licensee is continuing to investigate the failure mechanism of INI-47. The inspectors noted that the INI-47 actuator was cleared by radiation protection and was in the IAE shop. The inspectors will continue to follow the investigation.

c. Notification of an Unusual Event due to Solvent Vapor Entry into the Control Room

On April 7, 1994, work was being performed on Train A of the control room chiller system. The work was being performed as part of an extensive overhaul of the chiller subsystem. The maintenance workers were using a chemical solvent, R7-K-54, containing methyl isobutyl ketone to remove glue residue from the chiller heat exchanger surfaces after insulation was removed.

The chemical was applied to the heat exchanger surfaces with saturated rags and spray bottles. Approximately two gallons of the ketone chemical solvent was issued for the job; however, the amount that was actually used could not be readily determined. Members of the maintenance crew using the solvent complained of irritation. As a precaution, the station's safety group evacuated the area in the immediate vicinity of the chiller. Some additional ventilation was installed to reduce the solvent's airborne concentration. In addition, the safety group took air samples in the immediate vicinity of the chiller. The samples showed only a 2-3 ppm concentration of the solvent. The inspectors reviewed the Material Safety Data Sheets and found that the chemical became life threatening at 3000 ppm. The on-coming shift crew complained of irritations from the odor. In response, the control room doors were opened and ventilation fans were put in place to dissipate the odor. The shift supervisor later declared an Unusual Event because portions of the auxiliary building had been evacuated.

The inspectors questioned the transport mechanism for the solvent odor entering the control room. Although, only small quantities of the solvent entered the control room the inspectors were concerned that larger quantities of the ketone solvent or another chemical could possibly enter via the same pathway. The licensee originally postulated that it could have entered the control room from leaking duct work. Leakages in excess of TS values had been discovered earlier through duct work that had not been properly welded since construction. This is documented in LER 93-03.

The inspectors conducted a thorough walkdown of the control room ventilation systems and control room area ventilation systems. The inspectors and the licensee concluded that the most probable path for the solvent odor to reach the control room was through a return plenum vent for the control room area air handling units. The inspectors and licensee concluded that the frequent opening of the electrical penetration door allowed the solvent vapor to enter the control room. Once in the control room the solvent vapors entered a nearby return air duct for the control room air handling system and were recirculated in the control room. The inspectors are continuing to evaluate the vapor control room entry path.

The inspectors reviewed the FSAR and DBD and found that the McGuire control room was pressurized only during in emergency. A control room smoke and purge fan was part of the original design, but a subsequent evaluation revealed that use of the fan may degrade the safety function of the control room emergency pressurization system. The inspectors learned through interviews with system engineers that the smoke and purge fan currently would not have an automatic shutdown feature if the emergency ventilation system were to actuate. The inspectors noted that a breathing apparatuses are provided in the control room in the event of a serious gas release.

The inspectors reviewed work order 94021287, task 01, Removal of Insulation, and found that removing the residual glue had not been included. Although the stripping activities were documented on the work order there was not a pre-planned task to facilitate the removal. The inspectors will continue to evaluate this lack of pre-planning. This item will be tracked as an Unresolved Item, 50-369,370/94-09-03, Control Room Habitability.

d. Inadequate Core Cooling Monitor Failure

The inspectors evaluated the circumstances associated with the local electronics cabinet power supply failure of the Unit 2 train B inadequate core cooling monitor (ICCM). The ICCM consisted of two independent trains that were used to provide reactor vessel level indication, core exit thermocouples (CETCs) temperature, and reactor coolant system subcooling margin.

Part of the inspectors' evaluation included observing the plant staff attempting to reset the ICCM following a failure of the cabinet power supply. The licensee contacted Westinghouse for assistance to correct the problem. Upon further discussions with Westinghouse, the licensee learned that a technical bulletin had been issued regarding the failure of power supplies in ICCMs. The plant staff could not find a copy of the technical bulletin at any of the Duke sites (McGuire, Catawba, or Oconee). The staff at Catawba did have a copy of an information letter sent by Westinghouse in November 1991 about ICCM power supplies. However, McGuire staff had not seen a copy of the letter prior to this power supply failure.

The Westinghouse information letter detailed the potential safety issue involving failures of the electrolytic input capacitors in AC/DC power supplies used in Eagle-21 Process Protection System, Qualified Display Processing System, Reactor Vessel Level Instrumentation System, Inadequate Core Cooling Monitoring System, Auxiliary Shutdown Indication System, Plant Safety Monitoring System, and ATWS Mitigating System Actuation Circuitry. The inspectors questioned the plant staff about not receiving a copy of the Westinghouse information letter since it was addressed to McGuire and Catawba. The licensee stated that at the time they were experiencing problems with the power supply for the plasma display. Therefore, the plasma display output was measured on a quarterly basis with a voltmeter. It was not clear to the technical staff that the information letter also applied to the local electronics cabinet power supply. Since no further action was required, the information letter was closed by the Operating Experience Program (OEP). The technical staff informed the inspectors that the new vendor manual process should prevent this type of occurrence again. In the new process, a Problem Identification Process (PIP) form will be generated to track vendor information received by OEP.

One Unresolved Item was identified.

5. Late Technical Specification Amendment Change (71707)

March 22, 1994, the NRC issued amendments No. 141 to facility operating license NPF-9 and Amendment No. 123 to facility operating license NPF-17 for the McGuire Nuclear Station, Unit 1 and 2. The amendments consist of changes to the TS that administratively reduced the measured reactor coolant system flow from 385,000 gallons per minute to 382,000 gpm. The NRC required that the amendment become effective as of its issuance date and be implemented within 30 days of its issuance.

The following TS were modified to reflect the reduction in reactor coolant system (RCS) flow:

- Figure 2.1-1, Reactor Core Protection Limits-Four Loop Operation
- Figure 3.2-1, Reactor Coolant System Total Flow Rate Versus Rated Thermal Power -Four Loops in Operation, and
- The overtemperature delta temperature (OTDT) and overpower delta temperature (OPDT) setpoint equation constants in Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints

The licensee discovered that these amendments had not been implemented nearly 7 days after the effective due date. I&E personnel were called upon to immediately readjust the OTDT and OPDT setpoints to their new values. The figures had been incorporated into the existing technical specifications.

The inspectors concluded that this incident occurred because the licensee currently does not have an implementation procedure for TS amendments. This is considered a weakness.

6. Engineered Safety Feature System Walkdown (71710)

The inspectors conducted a walkdown of the Auxiliary Feedwater System. In preparation for the inspection, the inspectors reviewed the licensee's system lineup procedure to verify that it matched the plant drawings. The inspectors then walked the system piping to verify that the as-built configuration corresponded with the diagrams and valve positions agreed with those indicated in the system documentation. In addition, the pumps were inspected for leaks, oil level, and general material condition. Motor control centers for the motor-driven auxiliary feedwater pumps were energized, valve positions were correct, and no material condition discrepancies were noted.

The inspectors reviewed and referenced some Risk-based Inspection Guidance (RIG) in their system walkdown. The RIG was prepared specifically for the McGuire Auxiliary Feedwater System by the Idaho National Engineering Laboratory (INEL) and is based upon probabilistic risk assessments assuming the failure of certain system components (e.g.

valves, pumps, and electrical power) and the core damage frequency associated with each failure. The inspectors used a valve checklist contained in the RIG during their system line-up verification and compared it the P&ID and the AFW procedures for normal operation to verify that the checklist corresponded to the actual system alignment.

The inspectors reviewed several problem investigation reports that recently had been written on the auxiliary feedwater system. The inspectors were concerned about the nature and content of the reports and their cumulative effect on the system's ability to perform its function. A special meeting was held with licensee to discuss the reports. The NRC inspectors concluded that the reports were of no consequence to the safety function of the system.

No violations or deviations were identified.

#### 7. Licensee Emergency Drills (71707)

On April 27, 1994, the inspectors participated in the licensee emergency drill by assuming the role of NRC in the Technical Support Center (TSC). The inspectors in the resident's office also maintained a dialogue throughout the scenario to obtain information from the licensee's communicator to the NRC Incident Response Center in Bethesda, Maryland.

The inspectors noted that the site assembly was completed in 29 minutes and 47 seconds. The transition of control from the control room to the TSC was considerably smoother than the transition during the March 16, 1994, emergency drill. The licensee's corrective actions, such as providing sticky note pads at all TSC stations, enhanced communications among the TSC groups. The inspectors also noted that the public address system between the TSC and the OSC failed to operate properly. This failure has occurred during the past two emergency drills.

The inspectors observed that the plant staff could not determine if there was a fire in the "1A" emergency diesel generator room. The NLO was instructed to say that he could not enter the room because of "severe fuel oil fumes." The message relayed to the operations supervisor by the controller was that the halon had actuated. At that point, the operations staff believed that there was a fire in the "1A" emergency diesel generator room. The inspectors observed that the control room annunciator 1AD13-E3, Fire Detection System Alert, did not actuate. This would have confirmed that there was a fire. The confusion surrounding the existence of a fire led to a delay in the dispatch of the fire brigade to the "1A" diesel generator room. The inspectors noted that a fire in the "1A" emergency diesel generator room was not planned in the scenario. The inspectors discussed this controller communications problem with emergency planning and operations personnel.

The inspectors noted during the scenario that the STAs were required to manually check the critical safety function status trees because of the staged inoperability of the operator aid computer. This proved to be a



good method of ensuring that the STAs were able to identify and communicate this critical information to operators.

The inspectors reviewed the licensee's critique of the drill and found that it adequately recognized these and other deficiencies. The licensee proposed corrective actions for the next emergency drill, which is scheduled for June 1994.

In addition, the inspectors determined that poor communications resulted in the report of a false bomb threat to the control room during a separate security exercise.

No violations or deviations were identified.

8. Followup on Previous Inspection Findings and Licensee Event Reports (92701 and TI 2515/112)

The following previously identified items were reviewed to verify that the licensee's responses, where applicable, and actions were in compliance with regulatory requirements and that corrective actions have been implemented. Verification included review of selected records, observations, and discussions with licensee personnel.

- a. (Closed) Inspector Followup Item 50-369,370/93-07-01, Followup of Licensee actions regarding FSAR updating for changing environs. The inspectors evaluated the licensee's emergency plan and the most recent revision to the FSAR, as related to periodic updating/evaluation of population changes for evacuation routes and siren placement. The inspectors noted that the recent revisions to the FSAR did get special emphasis concerning changing environs. The licensee has formalized the reviews and updates of the FSAR to require the FSAR update that follows each refueling outage to address changes in the local environs. The inspectors observed that this formal method was addressed in the Site Regulatory Compliance Manual, Section 3.7.4. The primary sections of the FSAR that are affected by this program are in the FSAR Chapter 2, which consists of five sections. Section 2.1, Geography and Demography, and Section 2.2, Nearby Industrial and Transportation and Military Facilities, are the most likely sections to require changes during the plant life. Therefore, the licensee has required that these two sections be reviewed for changes during each subsequent FSAR update. The inspectors do not have any further questions about this matter. This item is closed.
- b. (Closed) LER 370/92-06, Unit 2 experienced a turbine/reactor trip due to equipment failure and an unknown cause. The inspectors evaluated the Plant Investigation Report details concerning this event and interviewed those individuals who were familiar with the circumstances which resulted in the reactor trip. On April 9, 1992, Unit 2 experienced a transient that resulted in a reactor trip. The trip occurred following the failure of a pressure

transmitter, 2CMPT5190, that provided control signals to condensate cooler bypass valve 2CM-58. This valve regulates the bypass flow around the generator stator water and hydrogen coolers. The transmitter failure resulted in the closure of 2CM-58. The sequence of events then included the following: 1) condensate booster pumps tripped on low suction pressure; 2) the associated main feedwater condensate booster pumps tripped, causing a feedwater transient on the feedwater system; 3) the main turbine generator tripped; and 4) the reactor trip. The transient on the feedwater system was compounded by a condensate valve, 2CM-420, that was designed to cause a main generator load rejection to occur. Valve 2CM-420 apparently did not respond to its demand signal to ensure adequate feedwater flow. The plant staff made repairs to the pressure transmitter and to the controls for valve 2CM-420. The long-term corrective actions have required frequent inspections of 2CM-420 to ensure its operability. For tracking purposes this LER is closed. However, the inspectors will continue to evaluate the impact of failures of non-safety related components and valves on overall plant reliability.

- c. (Closed) URI 369/370 94-G8-02, This URI is closed by violation 94-09-01, contained in this report.

#### 10. Exit Interview (30703)

The inspection scope and findings identified below were summarized on May 17, 1994, with the Station Manager and member of his staff. The following items were discussed in detail:

Violation 50-369,370/ 94-09-01, Inadequate Severe Weather Procedure, paragraph 2.d.

Unresolved Item 50-369,370/ 94-09-02, Deficiencies of Leakage Detection System, paragraph 3.c.

Unresolved Item 50-369,370/94-09-03, Control Room Habitability, paragraph 4.c.

The licensee representatives present offered no dissenting comments, nor did they identify as proprietary any of the information reviewed by the inspectors during the course of their inspection. The licensee was informed by the inspectors that the item discussed in paragraph 7 was closed.

#### 11. Acronyms and Abbreviations

AFW	-	Auxiliary Feedwater
ATWS	-	Anticipated Transient Without Scram
BS	-	Backshift
CETC	-	Core Exit Thermocouples
cfm	-	Cubic Feet per Minute
CPU	-	Central Processing Unit

DBD	-	Design Basis Document
EAL	-	Emergency Action Level
EMF	-	Radiation Monitor Area
FSAR	-	Final Safety Analysis Report
IAE	-	Instrumentation and Electrical
ICCM	-	Inadequate Core Cooling Monitor
IFI	-	Inspector Followup Item
INEL	-	Idaho National Engineering Laboratory
kV	-	kilovolt
kW	-	kilowatt
LCO	-	Limiting Condition for Operation
LEL	-	Lower Explosion Level
LER	-	Licensee Event Report
LOCA	-	Loss of Coolant Accident
LOOP	-	Loss of Off-site Power
MDSS	-	Material Data Safety Sheet
NLO	-	Non-licensed Operator
NOED	-	Notice of Enforcement Discretion
NRC	-	Nuclear Regulatory Commission
NWS	-	National Weather Service
OAC	-	Operator Aid Computer
OEP	-	Operating Experience Program