UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
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ATLANTA, GEORGIA 30303-8931

January 12, 2001

Duke Energy Corporation
ATTN: Mr. H. B. Barron
Vice President
McGuire Nuclear Station
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

SUBJECT: MCGUIRE NUCLEAR STATION - NRC INTEGRATED INSPECTION REPORT 50-369/00-06 AND 50-370/00-06 AND INDEPENDENT SPENT FUEL STORAGE INSTALLATION INSPECTION REPORT 72-38/00-01

Dear Mr. Barron:

On December 16, 2000, the NRC completed an inspection at your McGuire Units 1 and 2 and the McGuire Independent Spent Fuel Storage Installation. The enclosed report presents the results of this inspection. The enclosed report documents the inspection findings which were discussed on December 19, 2000, with Mr. Jack Peele and other members of your staff.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

No findings of significance were identified.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Robert C. Haag, Chief
Reactor Projects Branch 1
Division of Reactor Projects

Docket Nos. 50-369, 50-370, 72-38
License Nos. NPF-9, NPF-17

Enclosure: NRC Integrated Inspection Report 50-369/00-06, 50-370/00-06, 72-38/00-01
w/Attached NRC's Revised Reactor Oversight Process

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U.S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos:	50-369, 50-370, 72-38
License Nos:	NPF-9, NPF-17
Report No:	50-369/00-06, 50-370/00-06, 72-38/00-01
Licensee:	Duke Energy Corporation
Facility:	McGuire Nuclear Station, Units 1 & 2 McGuire Independent Spent Fuel Storage Installation
Location:	12700 Hagers Ferry Road Huntersville, NC 28078
Dates:	September 17 - December 16, 2000
Inspectors:	S. Shaeffer, Senior Resident Inspector M. Franovich, Resident Inspector R. Carroll, Senior Project Engineer (1R01 and 1R07) E. Lea, Project Engineer (1R06) D. Jones, Radiation Protection Inspector (2OS3 and 4OA1) G. Salyers, Emergency Preparedness Inspector (1EP2, 1EP3, 1EP4, 1EP5 and 4OA1) R. Chow, Reactor Inspector (4OA5) W. Gloersen, Reactor Inspector (4OA5) J. Furia, Senior Radiation Protection Inspector (2PS2)
Approved by:	Robert Haag, Chief, Projects Branch 1 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000369-00-06, IR05000370-00-06, IR 07200038-00-01 on 09/17-12/16/2000, Duke Energy Corporation, McGuire Nuclear Station, Units 1 & 2. Resident Inspector Report.

The inspection was conducted by resident inspectors, regional radiation protection inspectors, a regional emergency preparedness inspector, project engineers, and inspectors reviewing the pre-operational readiness of the licensee's Independent Spent Fuel Storage Installation (ISFSI).

A. <u>Inspector Identified Findings</u>

No findings of significance were identified.

B. <u>Licensee Identified Violations</u>

A violations of very low significance which was identified by the licensee has been reviewed by the inspectors. Corrective actions taken or planned by the licensee appear reasonable. The violation is listed in Section 4OA7 of this report.

Report Details

Summary of Plant Status:

Unit 1 operated at or near 100 percent power for the entire inspection period. Unit 2 began the inspection period in no mode due to End of Cycle 13 (2EOC13) refueling outage. Unit 2 was restarted on October 10, 2000, and reached 100 percent power on October 14. On November 15, Unit 2 was manually tripped from approximately 18 percent power due to an inadvertent turbine runback. The unit was restarted on November 16, 2000, and operated at approximately 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity

1R01 Adverse Weather Protection

a. Inspection Scope

The inspectors assessed the effectiveness of the licensee's cold weather protection program as it related to ensuring that the standby shutdown facility (SSF) and the auxiliary feedwater (CA) system would remain functional and available for plant shutdown when challenged by cold weather. In addition to inspecting the associated freeze protection equipment (e.g., heat tracing, area space heaters, etc.), the inspectors reviewed the following documents:

- Nuclear Station Directive (NSD)317, Freeze Protection Program, Revision 1
- Inspection Procedure (IP)/0/B/3250/059, Preventive Maintenance And Operational Check Of Freeze Protection, Revision 6
- Model Work Order 96074556, Cold Weather Inspection - SSF Diesel Generator
- IP/0/B/3250/059A, Monthly Check Of Freeze Protection, Revision 4
- Operation Procedure (OP)/0/A/6500/013, Service Building Rounds Sheets, Revision 14
- Performance Test (PT)/0/B/4700/070, On Demand Freeze Protection Verification Checklist, Revision 9

b. No findings of significance were identified.

1R04 Equipment Alignment

a. Inspection Scope

For the systems identified below, the inspectors reviewed plant documents to determine correct system lineup, and observed equipment to verify that the systems were correctly aligned by conducting partial walkdowns of:

- Unit 2 residual heat removal (ND) system
- Units 1 and 2 Safety Injection (NI) Train A
- Safe Shutdown Facility

The inspectors assessed conditions such as equipment alignment (i.e., valve positions and breaker alignment) and system operational readiness (i.e., fuel tank levels, water tank levels, and temperature) that could affect operability of these systems.

b. Findings

No findings of significance were identified.

1R05 Fire Protection

a. Inspection Scope

To assess the adequacy of the fire protection program implementation, the inspectors toured the following risk significant areas to assess transient combustible material control, visible material condition and lineup of fire detection and suppressions systems, status of manual fire equipment, and condition of passive fire barriers:

- Unit 2 emergency diesel generator rooms during maintenance activities
- Vital and non-vital battery rooms
- Vital instrumentation power equipment rooms
- Cable spreading rooms

On November 20, 2000, during evening hours, the inspectors observed an unannounced fire drill. The drill was performed to evaluate the capability of the offsite fire agencies and effectiveness of the associated fire brigade coordination, interfaces, and controls between the licensee and the offsite responders. The inspectors reviewed station personnel's use of response procedure (RP)/0/A/5700/022, Spill/Incident Response, and post-drill critique comments.

b. Findings

No findings of significance were identified.

1R06 Flood Protection Measures

a. Inspection Scope

The inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and related flood analysis documents to identify what areas, if any, would be affected by internal or external flooding at the McGuire Nuclear Station. Documentation was also reviewed to determine if the licensee had performed an analysis to identify and implement requirements to assure that their flood mitigation plans were consistent with their risk analysis assumption. Procedures were reviewed to determine if those actions required to be performed by an operator to mitigate the consequences of a flood had been identified and proceduralized. The inspector performed walkdowns of selected areas and systems that were identified as risk significant to identify possible sources of flooding. During the walkdown, the inspectors accessed the adequacy of both components and structures that were installed to help mitigate the consequences of a flood. The inspectors also looked for possible sources of internal flooding that are not analyzed. Problem investigation process (PIP) reports were reviewed to determine what problem, associated with flooding, had been identified, and that once the problem was identified each problem was entered into the licensee's corrective action program and effective corrective action has been taken or planned. Documents that were reviewed are listed below:

- PIP M-99-0431 - Auxiliary Building Flood Design Bases and Past Studies Reviewed
- PIP M-99-1343 - Nuclear Service Water (RN) Butterfly Valve Tightened Beyond Closed Seat
- PIP M-00-4441 - Small Pump Under Each EDG have No PM
- PIP M-00-4468 - Motor Lifting Lug Bolts Missing on the 1A and 2A NI Pump Motors
- OP/0/A/6100/017 - Operation of SSF During Significant Sabotage Event That May Cause Plant Flooding, Revision 24
- AP/0/A/5500/44 - Plant Flooding, Revision 0

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance

a. Inspection Scope

This annual inspection, which focused on the testing of the 2A residual heat removal (ND) pump air handling unit (AHU), was conducted to verify that: (1) test acceptance criteria and results appropriately considered differences between testing and design conditions; and (2) frequency of testing was sufficient to detect degradation prior to loss of design heat removal capabilities. The inspectors reviewed the November 28, 2000, performance of PT/2/A/4204/010**,** 2A ND Pump AHU Performance Test, Revision 9, as well as the results from previous performances of this test. Other documents reviewed included:

- Maintenance Procedure (PM)/0/A/7450/003, Safety Related Fans And Air Handling Units Preventive Maintenance (PM)
- Work Order 98262144 (August 9, 2000) and 98052851 (October 12, 1999) PM Task Completion Comments for 2A ND pump AHU
- Calculation MCC-1223.24-00-0065, ND Pump Motor Cooler Operability Evaluation
- PIP report M-99-4521, concerning classification of engineered safeguards system AHUs as a(1) under the Maintenance Rule
- Service Water System Program Manual, (Revision 3: dated January 20, 2000), Section 12.7.12, Residual Heat Removal Pump Room Air Handling Units
- Updated Final Safety Analysis Report, Table 9-8, Nuclear Service Water Flow Requirements

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification

a. Inspection Scope

The inspectors reviewed licensed operator requalification performance, training and associated training documentation to verify that performance deficiencies had been addressed through the requalification training program. Specifically, the inspectors reviewed activities concerning required monitoring of auxiliary feedwater sources during initiating events.

b. Findings

No findings of significance were identified.

1R12 Maintenance Rule Implementation

a. Inspection Scope

For the equipment issues described in the PIPs listed below, the inspectors reviewed the licensee's implementation of the Maintenance Rule (10 CFR 50.65) with respect to the characterization of failures, the appropriateness of the associated a(1) or a(2) classification, and the appropriateness of either the associated a(2) performance criteria or the associated a(1) goals and corrective actions:

PIP Number	Title/Description.
M-00-3909	Failure of valve 2NV-222 during B train ESF test
M-99-5678	SSF D/G and EDG nickel-cadmium battery cell degradation
M-00-0994	Analysis of nickel-cadmium battery cell degradation
M-00-3232	Valve 2NV-151 seat leakage during ECCS pump head curve testing
M-00-4026	Chilled water gasket failure
M-00-4111	2EMF33 failure (loss of flow alarm in control room)
M-00-4476	VA 2B Unfiltered exhaust fan bearings failure

b. Findings

No findings of significance were identified.

1R13 Maintenance Risk Assessments and Emergent Work Evaluation

a. Inspection Scope

The inspectors reviewed the licensee's control of plant risk and configuration through the review of selected structures, systems, and components (SSCs), listed below, within the scope of the maintenance rule or which were otherwise risk-significant. Emphasizing potential high risk configurations and high priority work items, the inspectors evaluated the following: (1) effectiveness of the work prioritization and control; (2) assessment of integrated risk of the work backlog; and (3) safety assessments and/or management activities performed when SSCs are taken out of service. The inspectors also reviewed the licensee's implementation of the Maintenance Rule (10 CFR 50.65) a(4) with respect to risk assessments for work activities.

PIP Number Work Order #	Title/Description
M-00-4110	2CA-57 steam generator injection check valve back leakage and inoperable 2B motor driven CA pump
M-00-4318	Containment isolation valve 2NC-56B failure to close during surveillance test
M-00-4748	SSF diesel-generator loss of radiator water
M-00-4696	Unit 2 high pressure feedwater heater level switch failure and subsequent positive reactivity addition
M-00-4085	Leakage past 1NC-2 pressurizer code safety valve, pressurizer relief tank thermal capacity, and RCS leakage calculation
M-00-4095	Leakage developed on 2B NV pump inboard seal

b. Findings

No findings of significance were identified.

1R14 Personnel Performance During Nonroutine Plant Evolutions

a. Inspection Scope

The inspectors reviewed the operating crews' performance during the following nonroutine evolutions and/or transient conditions to determine if the response was appropriate to the event and in accordance with procedures and training. Operator logs, plant computer data, and associated operator actions were reviewed.

PIP Number	Title/Description
M-00-4132	Improper jumpering on valves SA-48 and SA-49 resulted in ESF activation/start of TDAFW
M-00-4700	Operator response to procedure ES-0.1 regarding identification of total feedwater flow rate to steam generators during Unit 2 manual reactor trip on November 15, 2000

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations

a. Inspection Scope

The inspectors reviewed selected operability evaluations affecting risk significant SSCs listed below, to assess the technical adequacy of the evaluations. Where compensatory measures were involved, the inspectors also determined whether the compensatory measures were in place, would work as intended, and were appropriately controlled.

PIP Number	Title/Description
M-00-3619	Ability to start emergency diesel generator on one starting air bank
M-99-5538	2NS-1B valve stroke test frequency
M-99-5082	1NC-2 Pressurizer safety valve seat leakage and valve operability
M-00-4516	SSF Unit 1 Pressurizer heater capacity given small leakage on code safety 1NC-2
M-00-3232	Valve 2NV-151 seat leakage
M-00-3217	Boron concentration for refueling operations

b. Findings

No findings of significance were identified.

1R16 Operator Workarounds

a. Inspection Scope

The inspectors evaluated the selected risk-significant operator workarounds listed below, for potential affects on the functionality of mitigating systems. The workarounds were reviewed to determine: (1) if the functional capability of the system or human reliability in responding to an initiating event was affected; (2) the effect on the operator's ability to implement abnormal or emergency procedures; and (3) if operator workaround problems were captured in the licensee's corrective action program.

- Operator workaround 00-01 concerning required periodic cycling of valve 1SM-89 due to leakage of valve 1SM-84 (B main steam-line drain)

- Operator workaround 96-13 concerning monitoring of auxiliary feedwater suction sources

b. Findings

No findings of significance were identified.

1R17 Permanent Plant Modifications

a. Inspection Scope

The inspectors reviewed the following modification to: (1) verify that the design bases, licensing bases, and performance capability of risk significant SSCs have not been degraded through the modification; and (2) verify that the modification performed during risk-significant configurations did not place the plant in an unsafe condition.

Modification Number	Title/Description
NSM-22507	Busline relay modification

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (PMT)

a. Inspection Scope

The inspectors reviewed PMT procedures for the equipment below to ensure the equipment was returned to service satisfactorily. The inspectors evaluated the PMT to ensure it properly addressed the work performed.

- 1B RN Pump following repair of outboard bearing race
- EDG 2B Following air distributor tubing replacement (WO# 98265781)
- PIP M-00-4502 Main generator output breaker air pilot valve replacement (WO# 98298885)
- PIP M-00-4162 During PT/0/A/4150/028, reactivity computer picoammeters not responding as expected
- PIP M-00-4110 Restoration of check valve 2CA-57 to operable status
- PIP M-00-4686 Corrective maintenance for auxiliary relay contact problem on valve 2RN-174B

b. Findings

No findings of significance were identified.

1R20 Refueling and Outage Activities

a. Inspection Scope

During the inspection period, the inspectors reviewed refueling and outage-related activities initiated during the previous inspection period documented in Inspection Report 50-369,370/00-05. Refueling and unit startup parameters were monitored during increased risk periods. Control rod drop time test results were reviewed and zero power physics test results and test conditions were also evaluated. The inspectors reviewed the 100 percent core reload video to independently verify fuel assembly reload was conducted in accordance with cyclic-specific reload plan. The inspectors also performed a walkdown of selected portions of the reactor building prior to reactor startup to verify that debris was not present that could affect operability of the containment sump for the emergency core cooling system. The following procedures were also reviewed during the licensee’s restart of Unit 2:

- OP/2/A/6100/001, Controlling Procedure for Unit Startup
- OP/2/A/6100/SU-13, Heatup to 350 °F
- OP/2/A/6100/SU-17, Aligning CA for Standby Readiness
- OP/2/A/6100/SU-19, Heatup to 557 °F
- OP/2/A/6100/SU-20, Mode 1 and 2 Checklist

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing

.1 Routine Surveillance Testing

a. Inspection Scope

The inspectors witnessed surveillance tests and/or reviewed test data of selected risk-significant SSCs listed below, to assess, as appropriate, whether the SSCs met TS requirements, Updated Final Safety Analysis Report (UFSAR), and licensee procedure requirements. The inspectors also determined if the testing effectively demonstrated that the SSCs were operationally ready and capable of performing their intended safety functions. Compensatory measure, where applicable, were also verified.

- PT/0/A/4550/003C, Core Verification

- PT/0/A/4150/033, Total Core Reloading

- Lube oil analysis report for EDGs 1A, 1B, 2A, 2B
- IP/0/3066/002H, Testing motor-operated gate valves using VOTES (2NI-184, containment sump recirculation valve)

- PT/2/A/4208/003 Valve stroke timing for 2NS-1B and 2NS-18A

- PT/0/A/4150/0136, Dynamic rod worth measurement

b. Findings

No findings of significance were identified.

.2 Inservice Surveillance Testing

a. Inspection Scope

The inspectors also evaluated inservice testing of the Unit 2 AFW steam generator check valves (PT/2/A/4252/004, S/G Injection Check Valve Verification for #2 TDCA Pump) to determine the effectiveness of the licensee's American Society of Mechanical Engineers (ASME) Section XI testing program. The inspectors evaluated compliance with ASME code requirements, reviewed test methods and results, acceptance criteria, test instrument range/accuracy, and compliance with TS action statements/reporting requirements. The inspectors also verified that corrective actions were taken as applicable.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the following temporary modification to determine whether system operability and availability were affected, that configuration control was maintained, and that post-installation testing was performed:

Modification Number	Title/Description
MGTM-0189	Disabling 1 of 2 SSF diesel-generator jacket water heaters

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert and Notification System Testing

a. Inspection Scope

The inspectors reviewed the alert and notification system (ANS) design and associated testing commitment, and evaluated the adequacy of the testing program. Reviews were conducted of the ANS (sirens) testing results and related corrective action documentation for the period of January 1, 1999 through October 2000.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization Augmentation (ERO) Testing

a. Inspection Scope

The inspectors reviewed the design of the ERO augmentation system and the maintenance of the licensee's capability to staff emergency response facilities within stated timeliness goals. Records of an August 6, 2000 unannounced, ERO augmentation off-hour drill were reviewed. The drill involved actual travel to the plant by ERO personnel. Follow-up activities for problems identified through augmentation testing were reviewed to determine whether appropriate corrective actions had been implemented.

b. Findings

No findings of significance were identified.

1EP4 Emergency Action Levels (EALs) and Emergency Plan Changes

a. Inspection Scope

The inspector reviewed Revisions 99-03, 00-01, and 00-02 to McGuire's Radiological Emergency Plan (REP), to determine whether any of the changes decreased the effectiveness of the REP. The inspector reviewed the REP changes against the requirements of 10 CFR 50.54(q).

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

a. Inspection Scope

The inspector evaluated the efficacy of licensee programs that addressed weaknesses and deficiencies in emergency preparedness. Documents reviewed included exercise and drill critique reports, Emergency Plan Implementing Procedures, self-assessment reports, and audit reports SA99-24, and the Draft audit for 2000. No emergency declarations had been made since the last NRC inspection of the emergency preparedness program.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation

a. Inspection Scope

The inspectors observed an emergency drill from the control room simulator on December 14, 2000, to assess licensed operators' performance in the area of emergency preparedness. Licensee facilities participating in the drill included the Control Room Simulator (CRS), Technical Support Center (TSC), Operational Support Center (OSC), and Emergency Operations Facility (EOF). The inspectors reviewed operators emergency declaration activities and verified that the correct declaration and associated followup actions had been performed in accordance with the licensee's procedures and regulatory requirements. The observed scenario (a loss of coolant accident evolving to an offsite release) was followed by an exercise of the established severe accident management guidelines (SAMGs) . Following the drill, the inspectors attended the post-exercise critique to evaluate the licensee's self-assessment process.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS3 Radiation Monitoring Instrumentation

a. Inspection Scope

The inspectors evaluated the accuracy and operability of radiation monitoring instruments used for the protection of occupational radiation workers and the adequacy of the program for providing workers with self-contained breathing apparatus (SCBA). The licensee's programs for radiation monitoring and SCBA were evaluated against Technical Specifications (TSs), implementing procedural requirements, and 10 CFR 20.

The inspectors reviewed calibration procedures and records for the most recent calibrations of seven types of radiation monitoring instruments. The instruments included a containment high-range radiation monitor, an NNC Gamma 60 portal monitor, a RM-14 hand held frisker, a RO20 ionization chamber, a small article monitor (SAM), a personnel contamination monitor (PCM), and an electronic dosimeter (ED). The inspectors verified that the calibrations for the instruments were current. The inspectors verified the accuracy of the instrument response for a RM-14, a RO20, a SAM, a PCM, and a ED through the use of selected calibration sources or the licensee's instrument calibration equipment. The inspectors verified that the calibrations were current for several randomly selected RM-14s, RO20s and EDs which were then currently available for use.

The inspectors toured the plant and verified that SCBAs are available at selected locations and that equipment was available for refilling SCBA air bottles. The licensee's training manual for advanced respiratory protection was reviewed by the inspectors and determined to include provisions for training personnel in the use of SCBA, including air bottle change out. The training records for three randomly selected individuals who were on duty in the Control Room were reviewed. The inspectors determined that the selected individuals had been trained and qualified in the use of SCBA in accordance with the training manual.

The effectiveness of characterization and resolution for selected radiation protection related issues identified during April through October, 2000, was evaluated by the inspectors.

The following licensee procedures were reviewed:

- IP/1/A/3005/010, Radiation Monitoring System High Range Area Channel Calibration
- HP/0/B/1005/0027, Calibration of the NNC Gamma 40/60 Portal Monitor
- HP/0/B/1005/006, Calibration of Eberline Radiation Monitor RM-14
- HP/0/B/1005/028, Calibration of the Ionization Chambers
- HP/0/B/1005/052, Calibration of the NE American Small Articles Monitor
- HP/0/B/1005/038, Calibration of Eberline Personnel Contamination Monitor
- HP/0/B/1005/056, Operation and Standardization of the Merlin Gerin CDM 90 Calibrator
- HP/0/B/1008/006, Respiratory Protective Equipment Maintenance and Storage

- HS0113, Advanced Respiratory Protection - SCBA

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2PS2 Radioactive Material Processing and Shipping

a. Inspection Scope

The inspector reviewed the licensee's facilities, processes and programs for the collection, processing, treatment, shipping, storage and disposal of radioactive materials and radwaste. The inspector conducted reviews of the following: in-plant liquid and solid waste systems: waste processing and sampling program; shipment activities and records; assurance of quality, including corrective action reports; and training.

Systems reviews, which included system descriptions, control panel review, facilities tours, and a review of system changes in accordance with 10 CFR 50.59, was conducted for the following systems/subsystems: reactor coolant drain tank subsystem; waste drain tank subsystem; waste evaporator feed tank subsystem; laundry and hot shower tank subsystem; floor drain tank subsystem; ventilation unit condensate drain tank subsystem; radwaste facility subsystem; and, contaminated warehouse subsystem. The inspector also toured abandoned in-place radwaste equipment and facilities, and interim storage locations used for processed radwaste. Highly contaminated and/or high dose rate areas toured included the cubicles containing the following radwaste components: waste evaporator feed tank; waste evaporator; recycle evaporator; and, spent resin tank "B".

The inspector reviewed the licensee's Process Control Program (PCP), including: PCP procedure (Duke Power Company Corporate Process Control Program, Revision 12, and McGuire Nuclear Station Process Control Program, Revision 15); process documentation; scaling factor derivation, sampling type, sampling frequency, and effect of changing plant conditions (McGuire Nuclear Station 10 CFR 61 Manual, Revision 6); and, determination of waste characteristics and waste classification.

The inspector selected five solid radwaste shipping records for detailed review against the requirements contained in 10 CFR Parts 20, 61 and 71, and 49 CFR Parts 100-177. The shipments selected included spent resin, dry active waste, and outage equipment shipments and were Nos. RSR00-9, RSR00-10, RSR00-11, RSR00-12 and RSR00-13. The inspector also conducted direct observations of a receipt of radioactive material and shipment (RSR00-15) of dry active waste on September 27, 2000.

The inspector reviewed the licensee's program for assurance of quality in the radwaste processing and radioactive materials transportation program by reviewing: quality surveillances; departmental self-assessments (RP-SA-00-05 and RP-SA-99-11); and, seven problem investigation reports (PIPs) involving the radwaste and transportation program in 2000 [M-00-0542, M-00-1284, M-00-2043, M-00-3227, M-00-3431, M-00-3769, and G-00-0329].

The inspector reviewed the licensee's program of training for personnel involved in the radwaste and radioactive materials transportation program with regard to the requirements contained in NRC IE Bulletin 79-19 and DOT 49 CFR, Subpart H. Records reviewed included training requirements, course outlines/training modules, test questions, examinations and examination scores. Reviewed records were for licensee personnel in materials handling, radiation protection and radwaste.

b. Findings

No findings of significant were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator (PI)Verification

Emergency Preparedness Cornerstone

Licensee records were reviewed to determine whether the submitted PI statistics were calculated in accordance with the guidance contained in Section 2.4 (Emergency Preparedness Cornerstone) of NEI 99-02, Revision 0, "Regulatory Assessment Performance Indicator Guideline."

.1 Emergency Response Organization Drill/Exercise Performance PI

a. Inspection Scope

The inspectors assessed the accuracy of the PI for ERO drill and exercise performance (DEP) over the past eight quarters through review of drill records for that period. The documentation was reviewed for successes in emergency classifications, notifications, and protective action recommendations and compared to the DEP Performance Indicator.

b. Findings

No findings of significance were identified.

.2 ERO Drill Participation PI

a. Inspection Scope

The inspector assessed the accuracy of the PI for ERO drill participation during the previous eight quarters by comparing the licensee's Key ERO member drill participation tracking list to actual drill participation list in the "Drill Packages" for that period.

b. Findings

No findings of significance were identified.

.3 Alert and Notification System Reliability PI

a. Inspection Scope

The inspector assessed the accuracy of the PI for the alert and notification system reliability through review of the licensee's records of the siren tests for the previous 12 months. A sample of records for the weekly silent, weekly low growl tests, and quarterly full-cycle tests was reviewed.

b. Findings

No findings of significance were identified.

.4 Occupational and Public Radiation Safety PIs

a. Inspection Scope

The inspectors verified the accuracy of the Occupational Exposure Control Effectiveness and the RETS/ODCM Radiological Effluent Occurrence Performance Indicators (PIs) for the Occupational and Public Radiation Safety Cornerstones. To verify the PI data, the inspectors reviewed the April through September, 2000, monthly files generated pursuant to procedure SH/0/B/2006/001 NRC Performance Indicator Data Collection, Validation, Review, and Approval. The inspectors verified that the procedurally specified sources of information for the radiation safety PIs was collected each month and that potential and actual PI occurrences were accurately assessed for reportability.

b. Findings

No findings of significance were identified.

.5 Reactor Safety PI

a. Inspection Scope

The inspectors verified the following two Reactor Safety Performance Indicators (PIs) for accuracy:

Cornerstone	PI
Mitigating Systems	Safety System Unavailability, Emergency AC Power System
Barrier Integrity	Reactor Coolant System Leak Rate

To verify the Performance Indicator (PI) data, the inspectors reviewed plant chemistry records, control room logs, Technical Specifications Action Item Log entries, and maintenance rule data. The inspectors also reviewed surveillance activities associated with determining the RCS identified leakage rate.

b. Findings

No findings of significance were identified.

4OA3 Event Followup

.1 Unit 2 Manual Reactor Trip Following a Turbine Runback

a. Inspection Scope

On November 15, 2000, the inspector observed control room operators respond to a Unit 2 turbine runback and subsequent manual trip. The Unit 2 turbine-generator automatically reduced power from 100 percent power to below 20 percent power. The turbine runback was caused by actuation of all four channels of over power delta temperature (OPDT) and over temperature delta temperature (OTDT) runback logic. Subsequent to the turbine runback, the Operations Shift Manager directed reactor operators to manually trip the reactor since turbine load continued to decrease towards full load reject conditions and the cause of the runback was not known. Operators manually tripped the reactor at approximately 18 percent power. Following the reactor trip, the Unit 2 AFW pumps automatically started due to expected low-low steam generator levels. The inspectors verified that plant equipment necessary to safely shutdown the unit operated correctly, plant personnel responded adequately, and that the licensee made timely notifications to the NRC. The inspector also independently reviewed plant parameters, status of mitigating systems, and condition of fission product barriers. After the reactor trip, the licensee discovered that the turbine runback was due to a trip of an non-safety related electrical breaker which supplies power to the logic circuitry. The turbine runback was not initiated by a valid OPDT or OTDT condition. The licensee’s preliminary post-trip recovery activities and the suspect breaker were also observed. Root cause investigation was in progress at the end of the inspection period.

b. Findings

No findings of significance were identified.

.2 (Closed) LER 50-370/00-01, Inadvertent start of the turbine driven auxiliary feedwater (TDCA) pump during testing. The event occurred due to a failure of licensee personnel to properly place electrical jumpers to prevent the TDCA pump from starting during pre-planned testing. This problem constitutes a minor violation of TS 5.4.1 for failure to follow procedure. Based on the inspectors review, the event had no adverse impact on operation of the unit in Mode 3. No new issues were revealed by the LER.

4OA5 Other

.1 Review of Independent Spent Fuel Storage Installation (ISFSI) Construction and Related Plant Modification

a. Inspection Scope

Utilizing Inspection Procedure (IP) 60853, the inspectors examined the cask transportation route and facilities to be used for receiving, transferring, and transporting empty or loaded casks from the commercial truck entering the plant boundary to the concrete storage pad. Areas included the turbine building transferring point (which uses a 200 ton Unit 1 turbine building crane for transferring casks), refueling floor, cask decontamination pit, cask loading pit, and the cask storage pads. The inspectors examined the modifications, reinforcement, or markings performed along the route to reinforce the structures or restrict the transporter from running over unsafe or weak areas and compared observations to design drawings. The inspectors also examined the concrete storage pads and related construction.

The inspectors examined the modifications completed around Unit 2 decon pit and platform for the storage of the lifting beam and long pole and compared observations to design drawings. The elements inspected included the member size and lengths, welding sizes and symbols, anchor bolt diameters, and base plate sizes.

The inspectors reviewed an approved procedure and a calculation of the transporting route. The inspectors reviewed maintenance and inspection records for the last two years for the Unit 1 turbine building 200 ton crane and the Unit 2 fuel building 125 ton crane. The inspectors also reviewed records for the transporter load test performed at the manufacturer’s facility and transporter route test performed at the site by carrying a 125 ton text weight and driving it on the field route. The inspectors selected and reviewed several Problem Investigation Processes (PIPs).

The inspectors reviewed resolutions of variation notices (VNs) and quality control (QC) inspection records for construction of the pad contained in the modification package

NSM MG-42481 Part 2, Concrete pad construction and related activities. The VNs and QC records included reinforcing bar support change, material change, testing of the concrete air contents, slump, temperatures, and compacted soil density. The following documents were reviewed.

- Procedure MP/0/A/7650/186, Receipt of New TN-32A Casks, Revision 001
- Calculation MCC-1151.03-00-0004, Spent Fuel Cask Transport Path Evaluation of Buried Pipes and Components, Revision 5
- Final Scope Document NSM MG-42481/ P2 for Concrete Pad and Related Construction and P4 for Evaluation and Improvement of the Transporting Route
- Certified letter for the Cask Transporter Load Test dated September 9, 1999
- Field Load Test per Procedure TN/0/A/2481/P4/01C, NSM MG-42481/P4-Functional/Load Test of the Dry Cask Storage Transporter and Haul Path, Revision 0
- PIPs M-98-1564, M-99-2974, M-99-5125, M-00-0437, M-00-2340, M-00-2750, and M-00-4269
- Quarterly Maintenance and Inspection Records from 3/7/98 to 8/7/00 for Unit 1- 200 ton Turbine/Generator Bay Crane
- Quarterly Maintenance and Inspection Records from 5/28/98 to 6/27/00 for Unit 2- 125 ton Spent Fuel Building Main Crane
- Drawing No. MC-1220-30, Auxiliary Building Miscellaneous Steel, Sheet 7, Revision 12
- Drawing No. MC-1220-200, Auxiliary Building Miscellaneous Steel, Revision 1
- Drawing No. MC-1220-58, Auxiliary Building Decon Pit Platforms Miscellaneous Steel, Sheet 6, Revision 5
- Drawing No. MC-1220-58.01, Auxiliary Building Decon Pit Platforms Miscellaneous Steel, Revision 0
- Drawing No. MC-1030-10-10, Spent Fuel Cask Haul Path Inspection Drawing, Revision 0
- Drawing No. MC-1030-10-11, Spent Fuel Cask Haul Path Inspection Drawing, Revision 0
- Variation Notices NSM MG-42481 P2J, P2M, P2Z, P2AS, P2AY, P2AZ, P2BA, P2BB, P2BC, and P2BE
- QC Inspection Records QCC-1B, 1D, and M-1A, 1B, 1C, 1D,1G, 1M, and 1S

b. Findings

No findings of significant were identified.

.2 Review of 10 CFR 72.212(b) Evaluation for the Independent Spent Fuel Storage Installation (ISFSI)

Section 72.210 of Title 10 of the Code of Federal Regulations (10 CFR 72.210) grants a general license for the storage of spent fuel in an ISFSI at power reactor sites to any person authorized to possess or operate nuclear power reactors under 10 CFR Part 50. Section 72.212 gives the conditions for this general license and 72.212(b) delineates requirements that the general licensee shall meet. The inspectors reviewed the licensee's 10 CFR 72.212(b) evaluation report and supporting documentation to verify that the Transnuclear TN-32 was suitable for use at the McGuire Nuclear Station under the provisions of the facility's Part 50 license.

.3 Certificate of Compliance

a. Inspection Scope

The inspectors reviewed the licensee's evaluation on the conditions set in Certificate of Compliance for the Transnuclear, Inc., TN-32 Dry Storage Cask (No. 1021). The review included an evaluation to determine compliance with the requirements for operation and design limits set in the general condition for the Technical Specification and special conditions. The inspectors reviewed seven out of ten special conditions which included operation procedures, acceptance test and maintenance program, quality assurance, heavy load requirements, approved contents, design features, and changes to the Certificate of Compliance.

b. Observations and Findings

The inspectors verified that the total cask helium leak rate to be less than 1.0 E-5 cc/sec and the helium minimum purity of 99.99% as stated in the procedure MP/0/A/7650/187, Revision 000 were in compliance with Technical Specifications 3.1.3, 3.1.4, and 4.1.4. The licensee evaluated the heavy load requirements for moving the cask from the spent fuel pool area to the concrete storage area by using NUREG-0612, NRC Bulletin 96-02, and ANSI N14.6. The heavy load equipment included the overhead crane, below-the-hook spread beam lifting devices, and the crawler type transporter.

The inspectors noted that the minimum cooling time for the spent fuel assemblies after removed from the core and stored in the spent fuel pool shall be more than 7, 8, 9, and 10 years depending on the initial enrichment and burn-up rate, instead of minimum 7 years as stated in the licensee's evaluation.

The licensee calculated the maximum lifting height for the cask moving on the outside of spent fuel pool area to be 14 inches based on the drop analysis and therefore, limited a lifting height in the operation procedure to be 12 inches maximum from the ground to ensure a margin of safety.

The inspectors also verified the following design features and compared them to the maximum allowed in the Technical Specification 4.2.2:

- Concrete pad thickness was designed for 35.75 inches and was less than 36 inches maximum allowed.

- Nominal concrete compressive strengths were tested to be between 3,000 and 4,900 psi and were less than 6,000 psi maximum allowed.

- Soil effective modulus of elasticity was calculated between 1,000 and 3,500 psi and was less than 32,600 psi maximum allowed.

- Maximum drop height was limited to 12 inches stated in the procedure and was less than 18 inches maximum allowed

The inspectors reviewed the evaluation report for cask sliding and transport. The licensee used steel and concrete static coefficient of friction of 0.35 and calculated to obtain an enough friction resistance force in order to prevent the cask sliding. The licensee tested the coefficient of friction at site in the similar condition as cask metal sitting on the top of concrete and obtained the result to be well above 0.35. The inspectors previously reviewed the cask transporter haul path evaluation and considered it to be adequate.

The following procedures or calculation were established for cask operation by the licensee and a portion of them were reviewed by the inspectors:

- Calculation MCC-1139.01-00-0174, Cask Drop Analysis, Revision 7

- Procedure OP/0/A/6550/027, TN-32A Fuel Assembly Loading/Unloading, Revision 000

- Procedure MP/0/A/7650/186, Receipt of New TN-32A Casks, Revision 001

- Procedure MP/0/A/7650/187, Loading Spent Fuel Assemblies Into TN-32A Casks, Revision 000

- Procedure MP/0/A/7650/188, Operation of Dry Cask Transporter, Revision 001

- Procedure MP/0/A/7650/189, Unloading Spent Fuel Assemblies from TN-32A Casks, Revision 000

- Procedure IP/0/B/3050/033, TN-32A Dry Cask Pressure Monitoring System Pressure Switch Connector Assembly and Bench Calibration, Revision 000

- Procedure PT/0/B/4550/038, TN-32A Dry Cask Pressure Switch Channel Operation Test, Revision 000

- Procedure PT/1/A/4600/003C, Weekly Surveillance Items, 045

- Procedure HP/0/B/1006/025, Radiation Protection Controls for Loading Spent Fuel Assemblies into TN-32A Dry Storage Casks, Revision 000

- Procedure HP/0/B/1006/028, Radiation Protection Controls for Unloading Spent Fuel Assemblies from TN-32A Dry Storage Casks, Revision 000

c. Conclusions

The licensee adequately evaluated the requirements stated in the Certificate of Compliance for the dry storage cask TN-32.

.4 Review of Site Characteristics and Parameters

a. Inspection Scope

The inspectors reviewed the licensee's evaluation for the reactor site characteristics and parameters and compared them to the requirements set in the TN-32 Cask Safety Analysis Report (SAR) and NRC Safety Evaluation Report (SER). The review included site characters of earthquake intensity, tornado, average ambient temperature and temperature extremes, snow and ice loadings, flooding, and lightning. At the time of this inspection, the licensee was in the process of revising the fire and explosion and meteorology sections of the 10 CFR 72.212(b) evaluation report, hence these areas were not part of the inspection scope.

b. Observations and Findings

The design seismic accelerations at the cask center of gravity were 0.24g horizontally and 0.15g vertically for the small pad of 16 feet by 16 feet. The design seismic accelerations were less than maximum allowable of 0.26g horizontally and 0.17g vertically stated in the SAR. For the large pad of 32 feet by 96 feet, the design seismic accelerations at the cask center of the gravity were 0.257g horizontally and 0.181g vertically. The 0.181g value is higher than the maximum allowable of 0.17g stated in the SAR. The licensee and Transnuclear had revised the SAR and submitted it to NRC for review and pending NRC approval. Therefore, the McGuire Nuclear Station was only allowed to store the TN-32 casks on the small pads.

The design tornado wind speed combining the rotational and translational speeds at McGuire site was 360 mph and 3 psi pressure drop which met the requirements stated in the SAR. The site average ambient temperature was 88.8 degrees F during the month of July which was less than the allowable of 100 degrees F stated in the SAR. The site temperature extremes were 104 degrees F and -5 degrees F which were less than the allowable of 115 degrees F and -30 degrees F allowed in the SAR. The maximum monthly snow accumulation on record at the McGuire site was 19.3 inches which translated to about 30 psf uniform load (including 3 inches of ice assumed) which was less than 50 psf allowed in the SAR. The maximum lake water level of elevation 773.9 feet using the McGuire Lake Probable Maximum Flood (PMF) plus six feet of wave height would be less than the embankment of the lake elevation between 775 feet and 780 feet. The lower elevation of the ISFSI is elevation 755.7 feet which is 35 feet higher than the maximum water level of elevation 721 feet in case of the worst condition which the dam will break for 1,000 feet long along the embankment starting from weakest point of the concrete dam area. The site of the ISFSI is approximately 1,500 feet from the concrete dam. An electrical charge due to lightning would be conducted through the body of the cask to ground, and no significant thermal effect would result. The SAR had fully evaluated the lightning effects.

The following documents were reviewed by the inspectors:

- McGuire Nuclear Site Updated Final Safety Analysis Report, April 15, 1999
- TN-32 Cask Final Safety Analysis Report, Revision 0, January 2000
- TN-32 Cask Generic Technical Specification, effective April 19, 2000
- NRC Safety Evaluation Report on Transnuclear, Inc. TN-32 Dry Storage Cask System, effective April 19, 2000

c. Conclusions

The ISFSI site was constructed inside the expansion of the reactor site security boundary. The evaluation for the ISFSI site was adequate and included the comparison of the reactor site characteristics and design parameters stated in the FSAR to the requirements stated in the TN-32 Cask SAR.

.5 Safeguards Program Review

a. Inspection Scope

The inspectors reviewed applicable parts of the licensee's safeguards program and security plan to verify that spent fuel is protected against the design basis threat of radiological sabotage in accordance with the requirements of 10 CFR 73.55, with the conditions given in 10 CFR 72.212(b)(5)(i) through (v).

b. Observations and Findings

The inspectors observed that the existing protected area was expanded to include the area occupied by the ISFSI. The Duke Power Company Nuclear Security and Contingency Plan was changed to reflect the expansion of the protected area (Revision 11). The inspector verified that the plan was approved for use on November 16, 1999, and a letter was transmitted to the NRC pursuant to 10 CFR 50.54(p) (2).

In Section 2 of the Security Plan, the licensee specified the performance objectives of the physical protection system designed to protect against the design basis threat of radiological sabotages. Searches required by 10 CFR 73.55(d)(1) are conducted at the existing personnel access portal. There were no changes to the existing search requirements as a result of expanding the protected area boundary to include the ISFSI. The observational capability required by 10 CFR 73.55(h)(6) was via closed circuit television coverage of the protected area boundary. The licensee took a partial exception to the requirements of 10 CFR 73.55(h), "Response Requirement." The licensee indicated that responding security personnel will not interpose themselves between the dry casks stored in the area and any adversary attempting entry for the purpose of radiological sabotage and/or theft of the special nuclear material (SNM). It was understood that the design of the dry casks were such that they were considered self-protecting with regard to sabotage and/or theft.

c. Conclusions

Based upon the above review, the inspectors had reasonable assurance that the licensee was prepared to fulfill the requirements of the Security Plan and 10 CFR 72.212(b)(5) relative to the security of the ISFSI.

.6 Review of Programs Impacted by ISFSI Operations- Emergency Plan

a. Inspection Scope

The inspectors reviewed the licensee's evaluation of the ISFSI's impact on the reactor emergency plan.

b. Observations and Findings

The inspectors verified that the licensee's Emergency Plan was revised to include emergency action levels (EALs) for ISFSI malfunction events. The following changes were noted in the EALs: (1) fire/explosion and security events and (2) natural hazards, disasters, and other conditions affecting plant safety. The revised Emergency Plan was approved by the licensee and transmitted to the NRC for review on September 7, 2000. Pursuant to 10 CFR 50, Appendix E, the licensee requested NRC approval prior to implementation of the EALs. After NRC approval is received, the licensee will obtain appropriate State and County approval prior to EAL implementation. At the time of this inspection, the licensee had not received approval from the NRC.

The inspectors also verified that the licensee satisfied the requirements in 10 CFR 72.106(c) which states that the controlled area may be traversed by a highway, railroad, or waterway, so long as appropriate and effective arrangements are made to control traffic and to protect the public health and safety. The inspectors observed that the northern boundary of the controlled area extends into a small part of Lake Norman adjacent to the facility's intake structures, which is a public waterway. This area is accessible by boat and used by fishermen. During a situation which would require the licensee to implement an EAL that would result in recommending predetermined public protective actions, the inspectors verified that the licensee had the capability to control traffic and protect the public health and safety in the waterway. This was accomplished via the State of North Carolina’s Emergency Plan, Annex G, Warning and Notification of Boaters on Lake Norman and the Catawba River. It should be noted that the Sate of North Carolina requires that lake evacuation be performed by certified law enforcement personnel. The use of licensee personnel to perform this function was not an option. In addition, the licensee had installed 30 warning sirens around Lake Norman that provides a warning to boaters that when either a siren is heard and/or red flares are observed, to leave the lake immediately and turn on a radio or television to an emergency broadcast station (EBS) for information and instructions. The location of the signs and the sign inspection program was detailed in procedure PT/O/A/4600/096, Alert and Notification Systems (Notice to Boaters), Revision 6, May 4, 2000.

c. Conclusions

Based upon the above review, the inspectors had reasonable assurance that the licensee was prepared to fulfill the requirements of the Emergency Plan.

.7 Review of Programs Impacted by ISFSI Operations - Radiological Protection Program

a. Inspection Scope

The inspectors reviewed the licensee’s evaluation of the ISFSI’s impact on the reactor radiological protection program.

b. Observations and Findings

The inspectors verified that the licensee had procedures in place to control radiation protection activities associated with the loading, handling, storage, and unloading of the TN-32 Dry Fuel Storage Cask. Radiation Protection guidance for performing these activities was provided by procedure HP/O/B/1006/025, Radiation Protection Controls for ISFSI (Independent Spent Fuel Storage Installation), Revision 0. The inspectors reviewed HP/O/B/1006/025 and noted that the procedure provided adequate guidance for the licensee’s staff to assure that the requirements of 10 CFR 20 would be met and that occupational exposures from ISFSI operations would be as low as reasonably achievable. It should be noted that the dry run exercise and operational experience may result in procedural enhancements.

The inspectors also noted that the licensee had scheduled to perform background radiation level measurements of the ISFSI pad the week of December 18, 2000 before storing spent fuel in the ISFSI. Although information on storage pad background radiation levels was not required, the licensee had decided to measure the radiation levels to establish the baseline level for future ISFSI decommissioning activities.

The inspectors reviewed the licensee’s evaluation of effluents and direct radiation from the ISFSI as required by 10 CFR 72.104. The dose evaluation for ISFSI operations was based on only five fully loaded TN-32 casks. The licensee indicated that the 10 CFR 72.212(b) evaluation report would require an update before greater than five casks are stored on the pad. At the time of this inspection, the licensee was in the process of revising the dose estimates using more conservative parameters. The added conservatism was reasonable and the licensee’s methodology was acceptable. The licensee was also in the process of adding more detail to the dose evaluation section of the report to enhance the conclusions made in the 10 CFR 72.212(b) evaluation report.

In addition, the inspectors reviewed the licensee's dose assessment evaluation for a design basis accident to ensure the requirements of 10 CFR 72.106 would be met. At the time of this inspection, the licensee was in the process of revising this section of the 10 CFR 72.212(b) evaluation report to better enhance the conclusions.

c. Conclusions

Based upon the above review, the inspectors had reasonable assurance that the licensee was prepared to fulfill the requirements of the Radiological Protection Program. However, the licensee was in the process of revising the dose calculations to demonstrate compliance with the requirements specified in 72.104 and 72.106 (b).

4OA6 Meetings

The inspectors presented the inspection results to Mr. Jack Peele, Acting Site Vice President, as well as other members of licensee management and staff, at the conclusion of the inspection on December 19, 2000. The licensee acknowledged the findings presented.

The inspectors asked the licensee whether any of the material examined during the inspection should be considered proprietary. No proprietary information was identified.

4OA7 Licensee Identified Violations . The following finding of very low significance was identified by the licensee and is a violation of NRC requirements which meets the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600 for being dispositioned as a Non-Cited Violation (NCV).

NCV Tracking Number	Requirement Licensee Failed to Meet
(1) NCV 370/00006-1	Inadequate procedure (TS 5.4.1) for removal of Unit 2 120VAC vital inverters from service. During plant solid RCS operation in Mode 5, de-energizing the vital inverters resulted in an inoperable Low Temperature Overpressure Protection (LTOP) system required by Technical Specification 3.4.12.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

Barron, B., Vice President, McGuire Nuclear Station
Bradshaw, S., Superintendent, Plant Operations
Byrum, W., Manager, Radiation Protection
Cash, M., Manager, Regulatory Compliance
Dolan, B., Manager, Safety Assurance
Evans W., Security Manager
Geer, T., Manager, Civil/Electrical/Nuclear Systems Engineering
Jamil, D., Station Manager, McGuire Nuclear Station
Patrick, M., Superintendent, Maintenance
Peele, J., Manager, Engineering
Loucks, L., Chemistry Manager
Thomas, K., Superintendent, Work Control
Travis, B., Manager, Mechanical Systems Engineering

NRC

R. Bernhard, Region II Senior Reactor Analyst
W. Rogers, Region II Senior Reactor Analyst

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None

Closed

50-370/00-01	LER	Inadvertent start of the Turbine Driven Auxiliary Feedwater (TDCA) pump during testing.

Discussed

None

LIST OF ACRONYMS USED

AFW	-	Auxiliary Feedwater
AHU	-	Air Handling Unit
ANSI	-	American National Standards Institute
ASME	-	American Society of Mechanical Engineers
CA	-	Auxiliary Feedwater
CFR	-	Code of Federal Regulations
COC	-	Certificate of Compliance
CRS	-	Control Room Simulator

EAL	-	Emergency Action Level
EDG	-	Emergency Diesel Generator
EOF	-	Emergency Operations Facility
ESF	-	Engineered Safeguards Feature
F	-	Fahrenheit
FSAR	-	Final Safety Analysis Report
HRA	-	High Radiation Area
IP	-	Inspection Procedure
IR	-	Inspection Report
ISFSI	-	Independent Spent Fuel Storage Installation
LER	-	Licensee Event Report
MOV	-	Motor-Operated Valve
MP	-	Maintenance Procedure
mph	-	Miles Per Hour
ND	-	Residual Heat Removal
NS	-	Containment Spray
OP	-	Operation Procedure
OPDT	-	Over Power Delta Temperature
OTDT	-	Over Temperature Delta Temperature
OSC	-	Operation Support Center
PARS	-	Protective Action Recommendation
PI	-	Performance Indicator
PIP	-	Problem Investigation Process
PM	-	Preventive Maintenance
PMF	-	Probable Maximum Flood
PMT	-	Post-Maintenance Testing
psi	-	Pounds Per Square Inch
PT	-	Performance Test
QC	-	Quality Control
RCS	-	Reactor Coolant System
RN	-	Nuclear Service Water
RPS	-	Reactor Protection System
RWP	-	Radiation Work Permit
SAR	-	Safety Analysis Report
SDP	-	Significance Determination Process
SER	-	Safety Evaluation Report
SNM	-	Special Nuclear Material
SSC	-	Selected Structures, Systems, and Components
SSF	-	Standby Shutdown Facility
TDCA	-	Turbine Driven Auxiliary Feedwater
TDAFW	-	Turbine Driven Auxiliary Feedwater
TN	-	Transnuclear
TS	-	Technical Specifications
TSC	-	Technical Support Center
UFSAR	-	Updated Final Safety Analysis Report
VHRA	-	Very High Radiation Area
VN	-	Variation Notice

NRC's REVISED REACTOR OVERSIGHT PROCESS

The federal Nuclear Regulatory Commission (NRC) recently revamped its inspection, assessment, and enforcement programs for commercial nuclear power plants. The new process takes into account improvements in the performance of the nuclear industry over the past 25 years and improved approaches of inspecting and assessing safety performance at NRC licensed plants.

The new process monitors licensee performance in three broad areas (called strategic performance areas): reactor safety (avoiding accidents and reducing the consequences of accidents if they occur), radiation safety (protecting plant employees and the public during routine operations), and safeguards (protecting the plant against sabotage or other security threats). The process focuses on licensee performance within each of seven cornerstones of safety in the three areas:

Reactor Safety

- Initiating Events
- Mitigating Systems
- Barrier Integrity
- Emergency Preparedness

Radiation Safety

- Occupational
- Public

Safeguards

- Physical Protection

To monitor these seven cornerstones of safety, the NRC uses two processes that generate information about the safety significance of plant operations: inspections and performance indicators. Inspection findings will be evaluated according to their potential significance for safety, using the Significance Determination Process, and assigned colors of GREEN, WHITE, YELLOW or RED. GREEN findings are indicative of issues that, while they may not be desirable, represent very low safety significance. WHITE findings indicate issues that are of low to moderate safety significance. YELLOW findings are issues that are of substantial safety significance. RED findings represent issues that are of high safety significance with a significant reduction in safety margin.

Performance indicator data will be compared to established criteria for measuring licensee performance in terms of potential safety. Based on prescribed thresholds, the indicators will be classified by color representing varying levels of performance and incremental degradation in safety: GREEN, WHITE, YELLOW, and RED. GREEN indicators represent performance at a level requiring no additional NRC oversight beyond the baseline inspections. WHITE corresponds to performance that may result in increased NRC oversight. YELLOW represents performance that minimally reduces safety margin and requires even more NRC oversight. And RED indicates performance that represents a significant reduction in safety margin but still provides adequate protection to public health and safety.

The assessment process integrates performance indicators and inspection so the agency can reach objective conclusions regarding overall plant performance. The agency will use an Action Matrix to determine in a systematic, predictable manner which regulatory actions should be taken based on a licensee's performance. The NRC's actions in response to the significance (as represented by the color) of issues will be the same for performance indicators as for inspection findings. As a licensee's safety performance degrades, the NRC will take more and increasingly significant action, which can include shutting down a plant, as described in the Action Matrix.

More information can be found at: http://www.nrc.gov/NRR/OVERSIGHT/index.html.